OHIO STATE UNIVERSITY
RESEARCH REACTOR
LICENSE NO. R-75
DOCKET NO. 50-150

REVISED SAFETY ANALYSIS REPORT
AND TECHNICAL SPECIFICATIONS
DATED 15 DECEMBER 1999

REDACTED VERSION*
IN ACCORDANCE WITH
10 CFR 2.390(d)(1)

*Redacted text and figures blacked out or denoted by brackets

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December 15, 1999

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington D.C. 20555

RE: License Renewal of The Ohio State University Research Reactor (OSURR)
License No. R-75, Docket No. 50-150

This letter accompanies submission of the revised Safety Analysis Report (SAR) and Technical Specifications (TS) for the OSURR. The purpose of this submittal is to renew the OSURR license for a twenty-year period until February 3, 2020.

The SAR and TS for the 500kw, LEU fueled reactor were submitted October 7, 1987. The current submittal updates those documents. The format is the same as the original. A brief chapter (10) on financial qualifications has been added. Many of the changes were the result of the revisions made to 10CFR20, which eliminated the use of Maximum Permissible Concentration (MPC).

The changes to the SAR and TS were reviewed and approved by the Reactor Operations Committee in its meeting of December 8, 1999. Correspondence regarding the OSURR and this license renewal request should be sent to my attention.

Sincerely,

David B. Ashley

David B. Ashley
Dean, College of Engineering and
The John C. Geupel Chair in
Civil Engineering

c. Don W. Miller, Director Nuclear Reactor Laboratory
Richard D. Myser, Associate Director Nuclear Reactor Laboratory
Theodore S. Michaels , U.S. Nuclear Regulatory Commission

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SAFETY ANALYSIS REPORT

and

TECHNICAL SPECIFICATIONS

for

The Ohio State University Research Reactor

License Number R-75

Docket Number 50-150

December, 1999

Columbus, Ohio

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1.0 Introduction

1.1 Purpose

This document will present a description and safety analysis for The Ohio State University Research Reactor (OSURR). This reactor, owned and operated by The Ohio State University, is located on the Columbus Campus of The Ohio State University, within the City of Columbus, in central Ohio. The descriptions and analyses presented in this report will provide sufficient information to show that the reactor can be operated with reasonable assurance that the health and safety of the public will be protected.

The description of the reactor system and its associated components are sufficiently detailed to allow an understanding of the general features, characteristics, and basic operation of the reactor. The safety analysis makes conservative assumptions to allow larger safety margins.

1.2 General Facility Description

The OSURR is a pool-type reactor using light water as a moderator and coolant. The core of the reactor utilizes uranium fuel, enriched to 19.5%, in uranium-silicide (U_3Si_2) form, clad in aluminum. The fuel is in solid flat plate form, commonly called MTR-type fuel. Fuel plates are mechanically joined into fuel assemblies (also called fuel elements), which are stacked to form an approximately symmetric rectangular solid. The fuel assemblies are positioned in a grid plate forming a 5 by 6 rectangular matrix for available fuel assembly positions. The grid plate is bolted to the floor of the reactor pool. A plutonium-beryllium (Pu-Be) startup source provides an initial population of neutrons to the core for controlled reactor startup.

Reactor control is effected by three control rods of boron-stainless steel composition (called shim safety rods), and an additional control rod composed-only of stainless steel (known as the regulating rod). These control rods are positioned by electric motors. The three shim safety rods are held by electromagnets and can be inserted into the core under the influence of gravity by turning off the current to these electromagnets. The rods move within aluminum shrouds and extend into special control rod fuel elements. The active length of the control rods is sufficient to completely cover the active portion of the core. The control rod housings are held by brackets mounted to the sides of the reactor pool.

A number of experimental facilities converge at the reactor core. This allows simultaneous performance of a number of different experiments. These facilities include two beam ports, a pneumatic transfer facility (rabbit), a main graphite thermal column, a smaller graphite thermal column, a central irradiation facility (CIF) that can extend into either a water or graphite-filled flux trap, movable graphite isotope irradiation elements (GIIE), and movable dry tubes.

The reactor pool wall is made of barytes concrete. The interior surface of the pool is coated with a waterproof liner. The reactor pool has a capacity of 5800 gallons, with a minimum water depth of 15 feet maintained above the top of the core for shielding. Water purity is maintained by a process system composed of a circulating pump, demineralizer, and particulate filter. Makeup water is added to the reactor pool from city water supplies, after passing through a resin bed demineralizer unit.

The reactor is licensed to operate at continuously variable thermal power up to a maximum of 500 kilowatts. Operation is limited to steady-state power, with no pulsing capabilities. At maximum steady-state power, the average thermal neutron flux in the core is 4.66×10^{12} neutrons/cm^2/second. In cold, clean critical condition, the core contains approximately 2.6% $\Delta k/k$ excess reactivity. Because of their location, geometry, and composition, the controls rods have a total worth of approximately 8.45% $\Delta k/k$. The shutdown margin is at least 1% $\Delta k/k$ with the regulating rod and the highest-worth shim safety rod fully removed from the core.

The core is cooled by natural convective flow of pool water vertically through the core within the flow channels between the fuel plates. Pool water enters the bottom of the core at an inlet temperature of approximately 20 °C, is heated by the core, and exits the top of the core at approximately 60 °C Heated water enters an aluminum plenum, is withdrawn from the plenum and circulated through a closed-loop heat removal system. Heat rejection is achieved through a two-stage, closed-loop secondary cooling system. The primary-secondary heat exchanger removes primary coolant heat to an ethylene-glycol and water mixture, from which heat is rejected to the outside atmosphere through a fan-forced air circulation cooling unit (also referred to as a dry cooler). An additional secondary-loop heat exchanger provides further cooling of the ethylene-glycol and water coolant by using city water as its heat sink. The total heat removal capacity of the cooling system is sufficient to remove all of the 500 kilowatts of thermal energy generated in the core, and maintain an average equilibrium bulk pool temperature of 20-25 °C under all credible environmental conditions.

1.3 Background Information

The OSURR was first operated in 1961. Its operation is regulated by the U.S. Nuclear Regulatory Commission (NRC), under facility license number R-75, docket number 50-150.

The design of the OSURR is based on the Bulk Shielding Reactor (BSR), which was located at the Oak Ridge National Laboratory (ORNL). This reactor is in a class of reactors generally known as a Materials Testing Reactor (MTR). This class of reactors share various common features, among them are light water moderation and cooling, open pools, and plate-type fuel. The reactor itself was supplied by Lockheed Nuclear Products, then a division of the Lockheed Georgia Company. Lockheed operated a reactor very similar in design to the OSURR, at a power level of 1 megawatt steady-state thermal power, in a forced convection cooling mode. When operated in the natural convection cooling mode at power levels up to 10 kilowatts, the Lockheed reactor was essentially identical in operating characteristics to the OSURR for the first 25 years of OSURR operation.

License R-75 authorized The Ohio State University to operate the OSURR at steady-state thermal power levels up to 10 kilowatts, using natural convection cooling. Originally, up to 8 kilograms of 93% enrichment ^{235}U was permitted to be possessed by the university at the reactor site. This was later lowered to 4.6 kilograms upon removal of the Fission Plate from the Bulk Shielding Facility (BSF) of the OSURR, by License Amendment 7, in 1976. As of the end of 1986, nominal (i.e., without considering fuel burnup) 3575.81 grams of ^{235}U was located at the reactor site in the form of fuel, and approximately 80 grams of plutonium was contained in the startup source. In 1988, LEU fuel was received to replace the HEU fuel, and the HEU fuel was shipped offsite in 1995.

The OSURR is utilized by the university for a variety of instructional, research, and service activities. Reactor use is not confined to persons employed or associated with the university. Past utilization has involved area universities and colleges, as well as local secondary and middle schools. Other individuals and groups in private industry and other state and federal governmental agencies have used the OSURR in a variety of ways. At the end of 1999, the OSURR was the only operating research reactor in the State of Ohio.

1.4 Report Organization

A description and characterization of the OSURR site is presented in the following chapter. Details of the facility design are discussed in Chapter 3. The operating characteristics of the OSURR under normal conditions are presented in Chapter 4. Auxiliary systems and radioactive waste management are discussed in Chapters 5 and 6, respectively. Facility features and operational procedures for radiation protection are the subjects of Chapter 7. Chapter 8 contains the safety analysis for the OSURR. Chapter 9 presents the administrative organization and controls for the reactor facility, and Chapter 10 discusses financial qualifications. Appendix A contains the complete Technical Specifications for the OSURR.

2.0 Site Description and Characterization

2.1 General Location

The OSU research reactor is located on property owned by The Ohio State University, west of the main campus. Aerial photographs are shown in Figures 2.1 and 2.2. A map of the surrounding territory with a 3 mile radius circle centered on the reactor building is shown in Figure 2.3.

2.2 Demographics

2.2.1 Surrounding Population

The map shown in Figure 2.3 shows that the reactor building is completely surrounded by residential dwellings within a three mile radius of the reactor site. Figures 2.4 and 2.5 show the locations of major industrial buildings found within a one mile radius of the site. Most industrial and business activities are located south and east of the reactor building. To the west are primarily residential areas. Some businesses are located to the northwest, but census data indicates that these employ relatively few people compared with those firms located to the east and south within a one mile radius of the reactor site.

Data for both residential and industrial populations are as follows:

Type	Radius	Population
Residential	3 Mile	141,600
Industrial	1 Mile	4,600

2.2.2 Local Activities

As of Autumn Quarter, 1998, The Ohio State University is composed of approximately 65,000 students, staff, and faculty. It is a land grant institution, engaged in teaching, research and public service activities. Various community services are provided, including medical and dental services. The university is active in the performing arts, as well as in athletics, and has recently completed construction of the Wexner Center for the Arts as well as the Schottenstien Arena. Activities in these facilities periodically draw up to 100,000 additional people to the campus area for special events such as performing arts concerts, football, basketball and NHL games, circuses, and rock concerts.

Figure 2.1: Aerial View of the OSURR Site, West-East Perspective

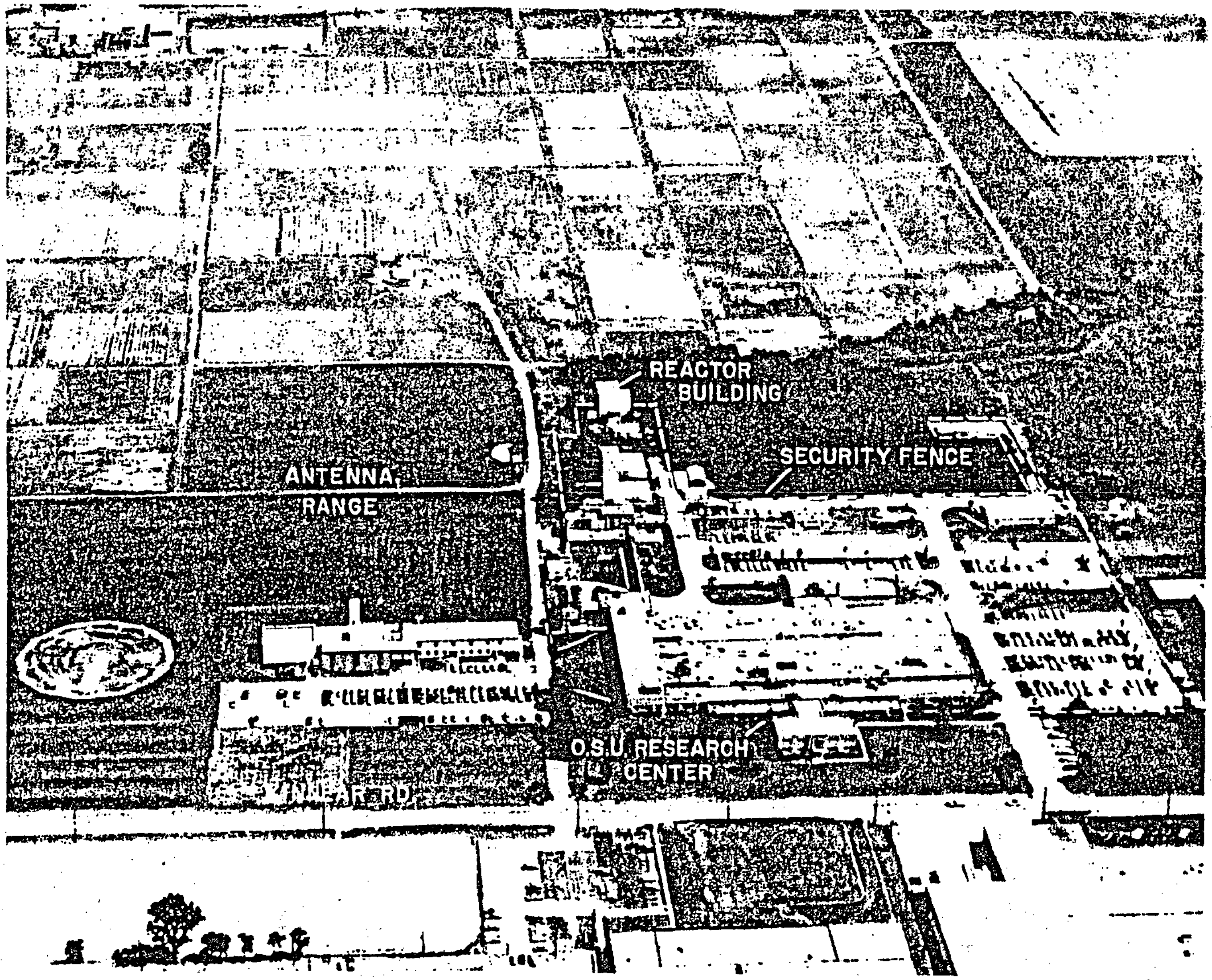

Figure 2.2: Aerial View of the OSURR Site, South-North Perspective

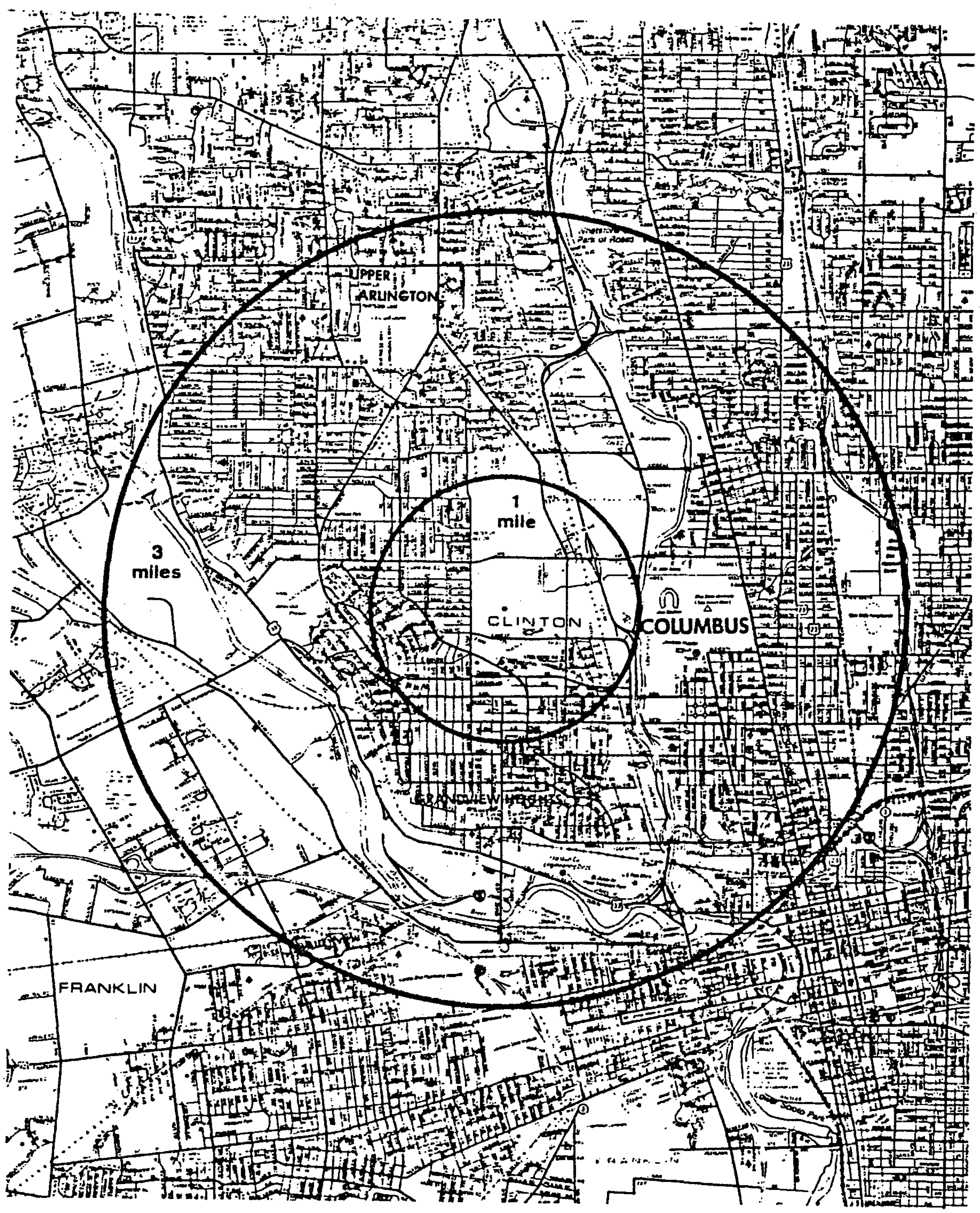

Figure 2.3: Detailed Map of the OSURR Surrounding Areas

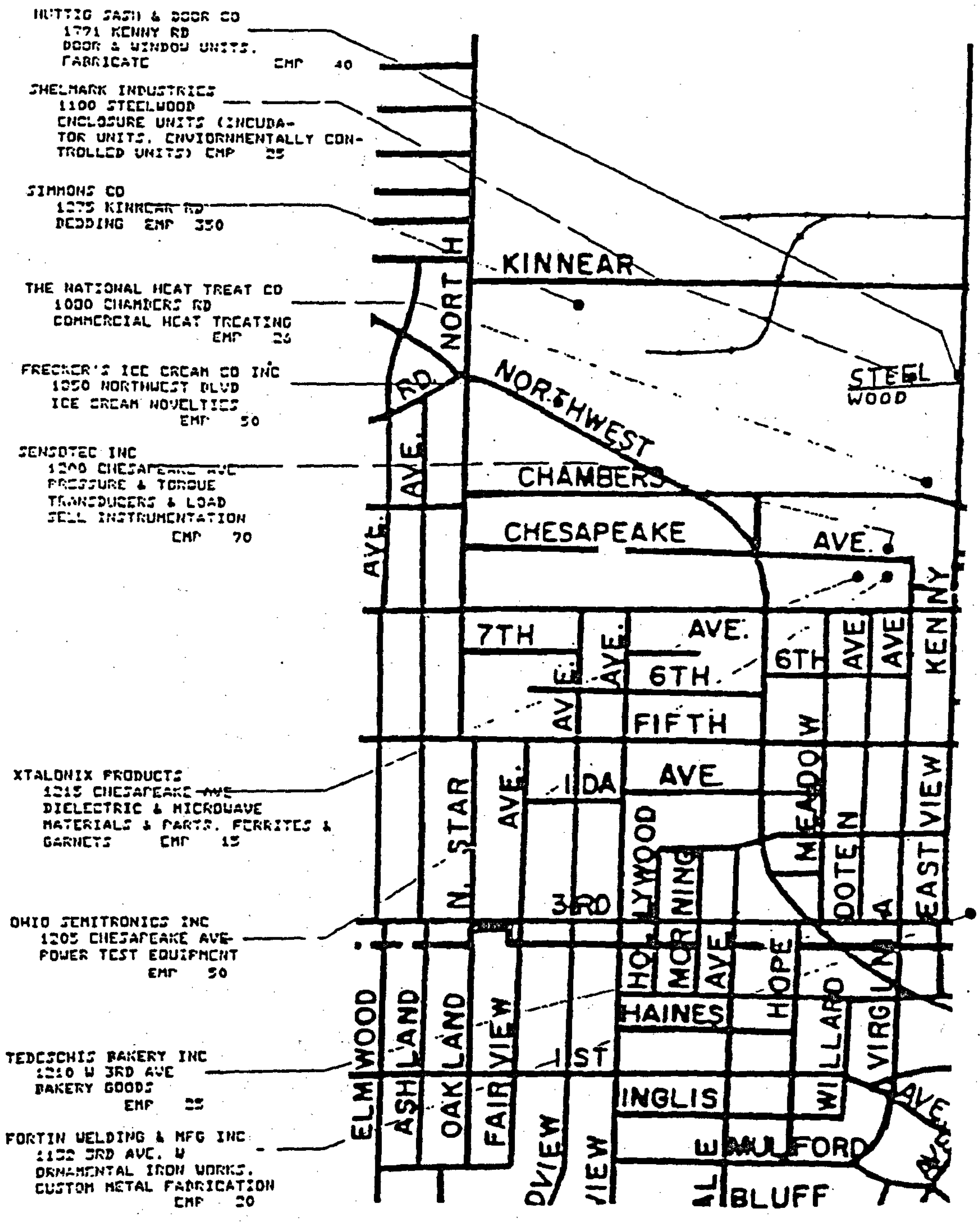

Figure 2.4: Businesses and Industries in the Areas South and East of the Reactor Building and Site

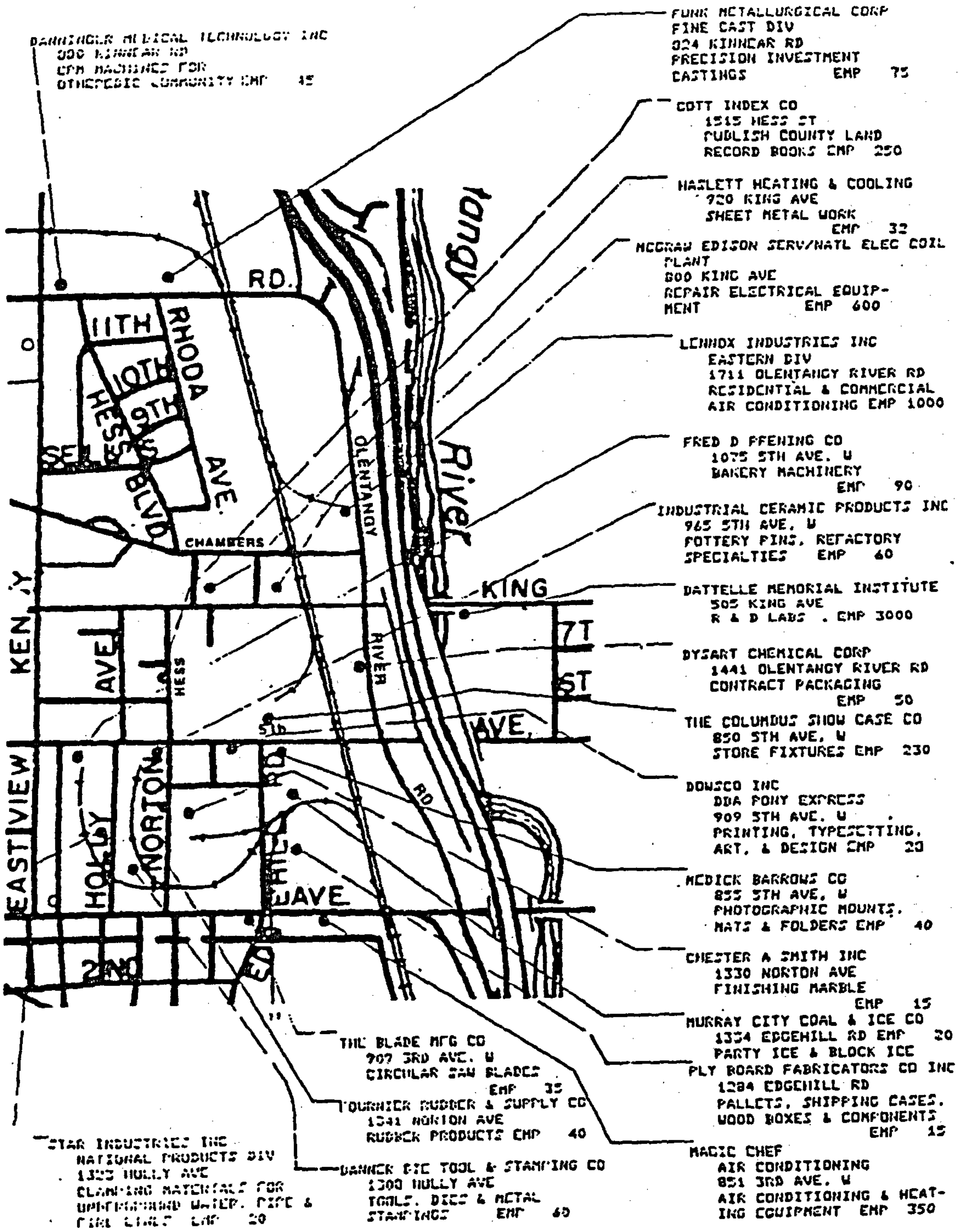

Figure 2.5: Businesses and Industries in the Area East of the Reactor Building and Site

2.3 Topography, Geology, and Seismology

2.3.1 Topography

Most of Ohio includes portions of two physiographic provinces called the Appalachian Plateau and the Central Lowlands. Franklin County is divided into these two sections by a series of north-south scarps and terraces which form a gentle step-like ascent eastward to the Appalachian Plateau. The highest altitude of the county is an elevation of 1130 feet above mean sea level, located in the northeast corner of Plain Township, and the lowest is an elevation of 665 feet above mean sea level at the efflux of the Scioto River from Franklin County. In the northern part and southwestern one-third of the county, the valley floors range in altitude from 780 to 890 feet above mean sea level, hilltops range from 860 to 960 feet above mean sea level, and local relief seldom exceeds 170 feet. The range in altitude of the valley floors in the northeastern and north central parts of the county is 710 to 840 feet. The hilltops range from 900 to 1130 feet above sea level. In the south central and southeastern parts of the county the valley floors range in altitude from 670 to 760 feet and hilltops range generally from 690 to 780 feet (locally they are 840 feet) above mean sea level. Except in the extreme southeast part of Madison Township, local relief does not exceed 50 feet. Columbus is located in the center of the county with a ground elevation of about 812 feet above mean sea level. The reactor site is about 780 feet above mean sea level.

2.3.2 Geology

Columbus lies on the glaciated plains section at the eastern edge of the central lowlands physiographic province. When the Plio-Pleistocene Ice Age began two million years ago, a continental scale ice sheet originating near Hudson Bay moved southward modifying the pre-glacial landscape. The deposits from the most recent glacial advance, the Wisconsin Glaciation, lie directly on the limestone and shale bedrock underlying Columbus. The area was completely buried by glacial till, generally 10 to 30 feet thick, consisting of unsorted clay, silt, sand, pebbles, and boulders (mainly derived from the Canadian Shield) carried south at the base of the ice sheet. Large out-wash deposits in the Scioto and Olentangy Valleys resulted from the great volumes of melt water coming from the kilometer-thick ice sheet (thickening to three kilometers near its source at Hudson Bay). The deposits occur above the present drainage level as gravel terraces and serve as water recharge areas north and south of downtown Columbus, principally on the west side of the Scioto River.

The bedrock immediately underlying the area is composed of the Columbus and Delaware Limestone, and the Olentangy and Ohio Shale of Devonian Age, deposited approximately 350 million years ago when Ohio was at the eastern edge of ancestral North America and was a near-shore marine environment. The Columbus and Delaware Limestone and the

Olentangy Shale represent successive depositional stages from a shallow marine environment to deeper marine environment. All sedimentary rocks underlying Columbus dip gently to the southeast. The pre-Devonian sedimentary rocks were derived from the Cincinnati Arch, a belt of Precambrian crystalline igneous and metamorphic rocks trending north-south along the Ohio-Indiana border. Now confined to the subsurface, this arch controls the linear pattern of Paleozoic deposition in western and central Ohio. The bedrock sediments outcrop in bands extending roughly north-south in Central Ohio. The Ohio Shale outcrops in the eastern part of the Columbus area; the Columbus and Delaware Limestone outcrop on the west side of the Scioto River. Near the close of the Paleozoic Era, the convergence of Africa and North America resulted in the Appalachian Orogeny. The western limits of this major mountain building episode are seen from northeast to south-central Ohio, where the regional southeastern dip of paleozoic strata rapidly steepens and the crustal thickness markedly begins to increase beneath the Appalachian Mountains.

Underlying Devonian strata in the Columbus area are older Paleozoic sedimentary formations. These are in turn underlain by Precambrian igneous and metamorphic basement rocks which make up the Proterozoic Craton extending northward and westward to outcrop in Canada as the Canadian Shield. These basement rocks are roughly 2 kilometers below the surface in central Ohio.

The bedrock at the reactor site is Delaware Limestone, a mixture of argillaceous cherty blue limestones and calcareous brown shales. These strata are covered by glacial drift which is predominantly gravel and clay. A boring analysis taken at a point about 500 feet southwest of the reactor site gave the information shown in Table 2.1.

2.3.3 Seismology

Figure 2.6 is a map of all of the instrumentally recorded earthquakes of Richter Magnitude 4.5 or greater that have occurred in the United States from 1899 through 1990. A belt of seismic activity (the St. Lawrence Seismic Belt) runs through Northwestern Ohio. Although the most persistently seismically active regions in the United States do not lie in Ohio, Ohio is not a-seismic contrary to popular opinion. Figure 2.7 gives the Modified Mercalli Intensity Scale with approximate corresponding values of equivalent Richter Magnitude.

Historically, the most active seismic region in Ohio is near Anna in Shelby County, approximately 60 miles northwest of Columbus. To date this region has had at least 35 earthquakes, including the three largest seismic events instrumentally recorded in Ohio. A map of the historically known earthquakes in Ohio with Richter Magnitude 2.0 or greater is provided in Figure 2.8. Modified Mercalli Intensitiy equivalent of each event is color coded. As of 1999, the first statewide seismic network was established in Ohio became operational (Hansen, 1999). Called "OhioSeis", this network digitally records seismic events of global origin but is intended to significantly

Table 2.1: Rock Types Underlying the Reactor Building Site

Strata	Depth (ft)
Clay	0 to 60
Slab Rock	60 to 63
Hard Clay	63 to 81
Rock	81 to 85
Hard Clay and Gravel	85 to 109
Hard Rock	109 to 115
Clay	115 to 138
Rock	138 to 142
Soft Clay	142 to 158
Limestone	158 to 190

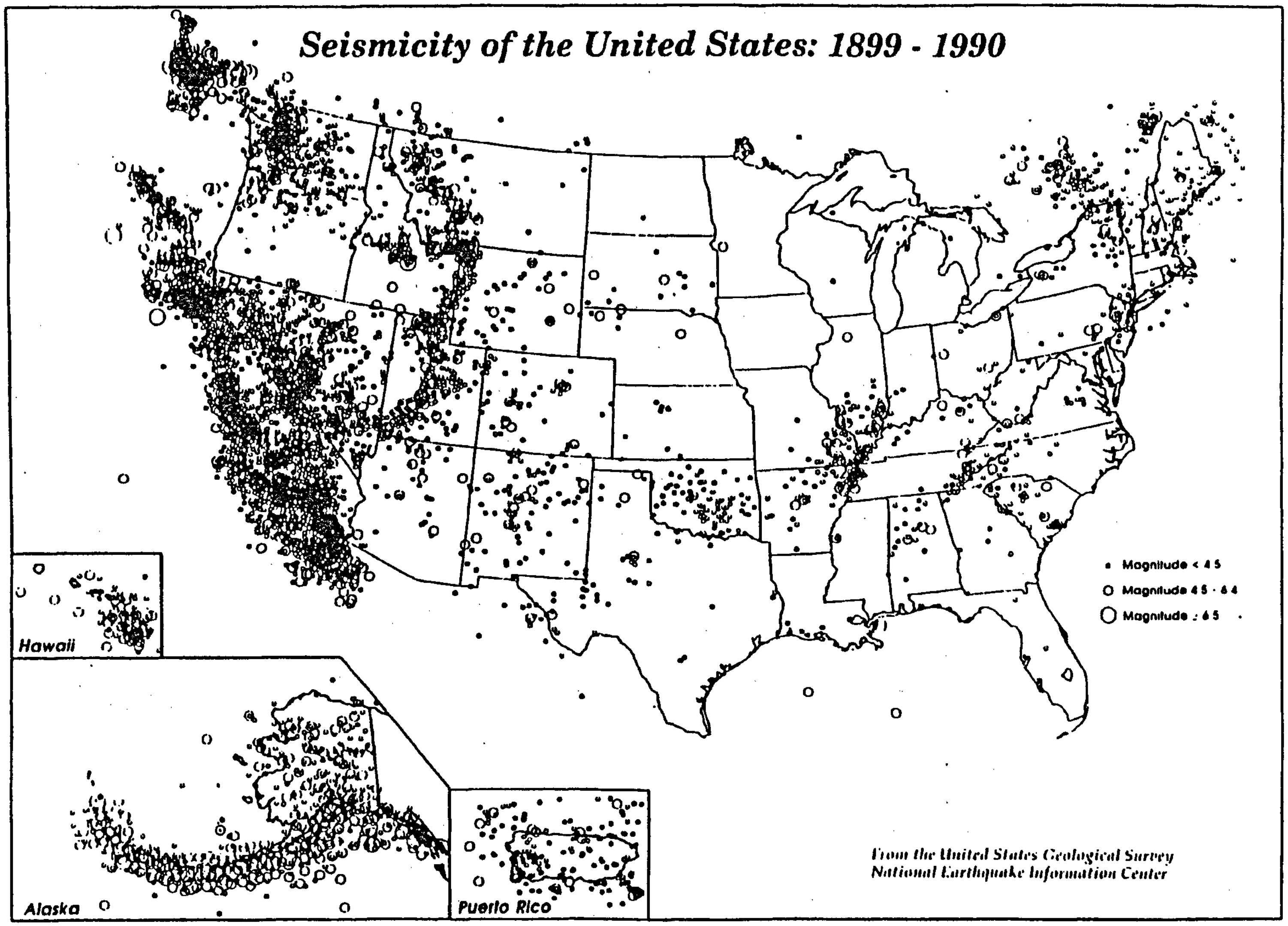

Figure 2.6: Seismicity of the United States: 1899-1990

Modified Mercalli Intensity Scale		Magnitude Scale
I	Detected only by sensitive instruments	1.5
II	Felt by few persons at rest, especially on upper floors; delicately suspended objects may swing	2
III	Felt noticeably indoors, but not always recognized as earthquake; standing autos rock slightly, vibrations like passing truck	2.5
IV	Felt indoors by many, outdoors by few, at night some awaken; dishes, windows, doors disturbed; standing autos rock noticeably	3
V	Felt by most people; some breakage of dishes, windows, and plaster; disturbance of tall objects	3.5, 4
VI	Felt by all, many frightened and run outdoors; falling plaster and chimneys, damage small	4.5
VII	Everybody runs outdoors; damage to buildings varies depending on quality of construction; noticed by drivers of autos	5
VIII	Panel walls thrown out of frames; walls, monuments, chimneys fall; sand and mud ejected; drivers of autos disturbed	5.5
IX	Buildings shifted off foundations, cracked, thrown out of plumb; ground cracked; underground pipes broken	6, 6.5
X	Most masonry and frame structures destroyed; ground cracked, rails bent, landslides	7
XI	Few structures remain standing; bridges destroyed, fissures in ground, pipes broken, landslides, rails bent	7.5
XII	Damage total; waves seen on ground surface, lines of sight and level distorted, objects thrown up into air	8

Figure 2.7: Modified Mercalli Intensity Scale with Approximate Equivalent Richter Magnitude

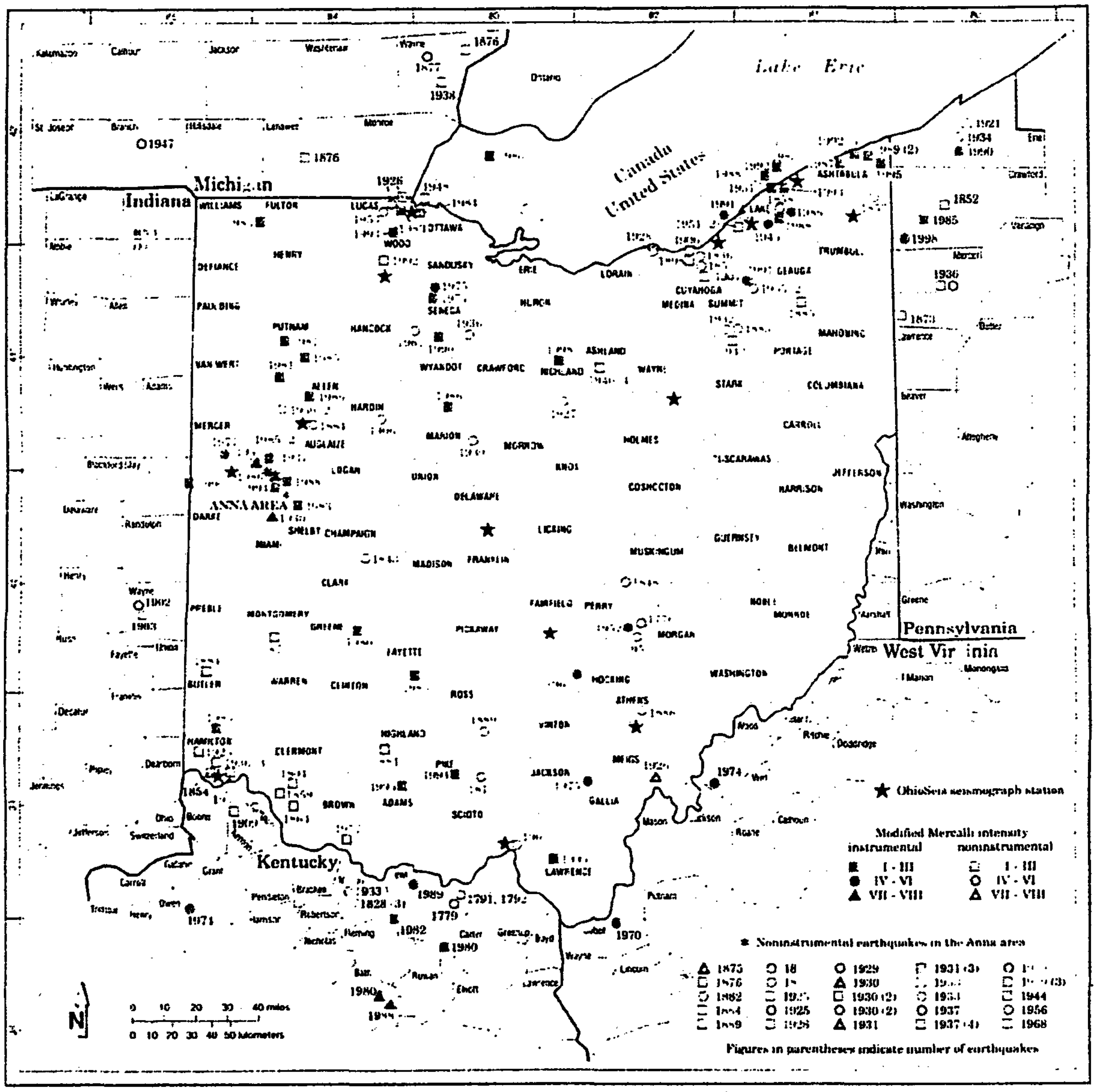

Figure 2.8: Map of Historically Known Earthquakes in Ohio with Richter Magnitude 2.0 or Greater

Notes:

1) Equivalent Modified Mercalli Intensity is color coded.
2) OhioSeis seismic network locations are shown by stars.

enhance the detection, location and magnitude of future seismic events in the mid-continent in general and Ohio in particular. In the near future, the seismic risk assessment to structures of all kinds due to earthquakes originating within and near Ohio's borders will be better understood.

Although very little historical information is available on earthquakes with epicenters in or near Columbus, or about the effects of such earthquakes on the city, it is appropriate to assume that known mid-continent earthquakes have historically had some effects here. Figure 2.9 shows the potential Modified Mercalli Intensity distribution resulting from a Richter Magnitude 8.0 seismic event in the New Madrid area. Damage consistent with MM VIII would be likely in central Ohio.

Figure 2.10 shows the Mid-continent regional distribution of peak lateral ground acceleration (as a percentage of gravitational acceleration = 1 g) having a two percent probability of being exceeded in 50 years. The Anna, Ohio, area discussed above is clearly discernable. More detailed information about this and related seismic risk information is available at the USGS Geological Hazards Web site http://geohazards.cr.usgs.gov/eq/.

The possibility of a Richter Magnitude 7.0 or greater seismic event in the New Madrid area (southeast Missouri - northwest Arkansas - western Tennessee) affecting the Columbus Metropolitan Area 400 miles (600 kilometers) away is unfortunately real. However, The Ohio State University reactor facility is not likely to suffer sufficient damage resulting in a serious hazard. Cracking of the shield would probably be the most serious damage. Should the pool liner be ruptured, pool water would escape from the pool leaving the reactor unshielded in the vertical direction. The reactor would then be sub-critical because of the absence of the moderator.

2.4 Meteorology

Figure 2.11 is a graph of the average wind speed versus wind direction. Tables 2.2 and 2.3 contain summaries of temperature and precipitation for Columbus, Ohio. Table 2.4 contains various other relevant meteorological data. Meteorological information given in Figure 2.11 and Table 2.4 was taken from data published by the National Climatic Center in Asheville, North Carolina. The data in Tables 2.2 and 2.3 was obtained from the WWW page of the State Climate Office for Ohio at http://twister.sbs.ohio-state.edu/climoff.htm.

2.5 Hydrology

2.5.1 Surface Water

Columbus is located in the center of the state and in the drainage area of the Ohio River. Four nearly parallel streams run through or

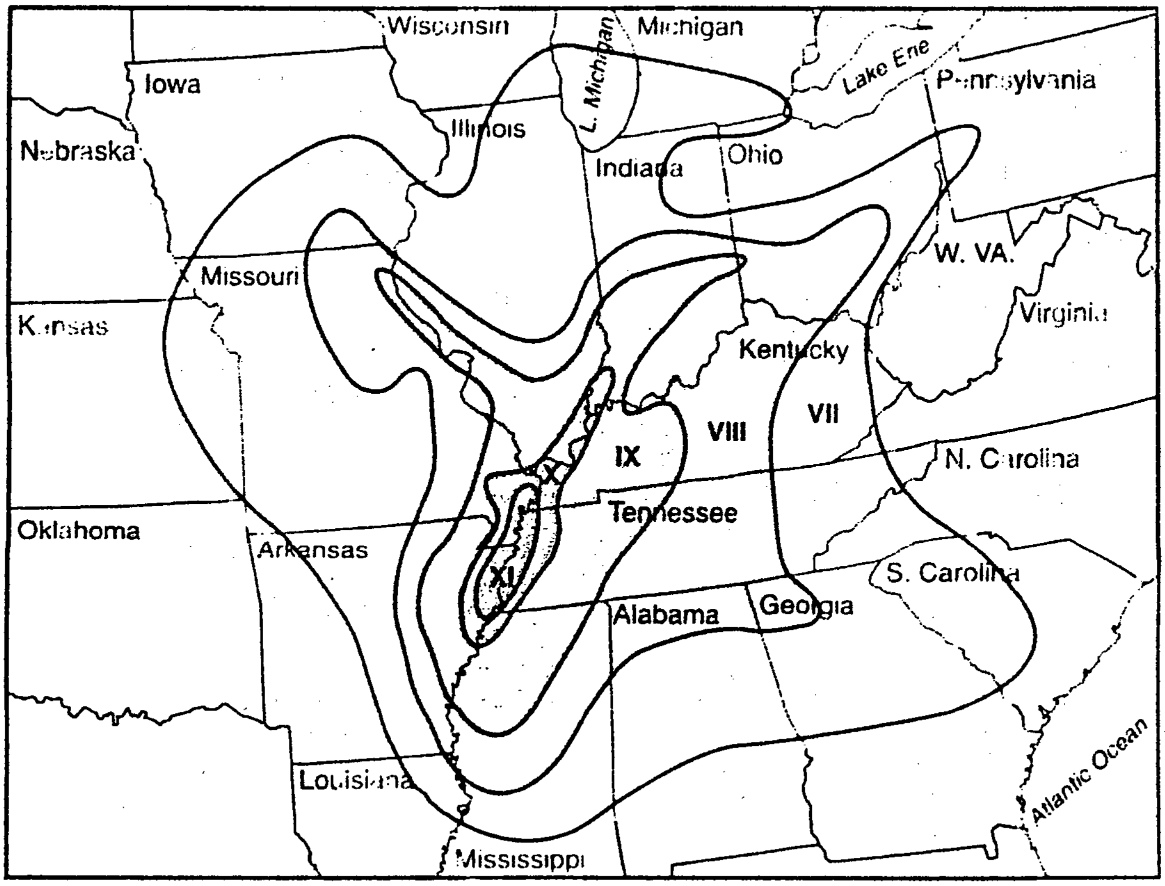

Figure 2.9: Potential Regional Modified Mercalli Intensity Map Projected for a Future Possible Richter Magnitude 8.0 Earthquake in the New Madrid Area

Note: Refer to Figure 2.7 for interpretation of the local severity of projected structural damage in terms of local Modified Mercalli Intensity.

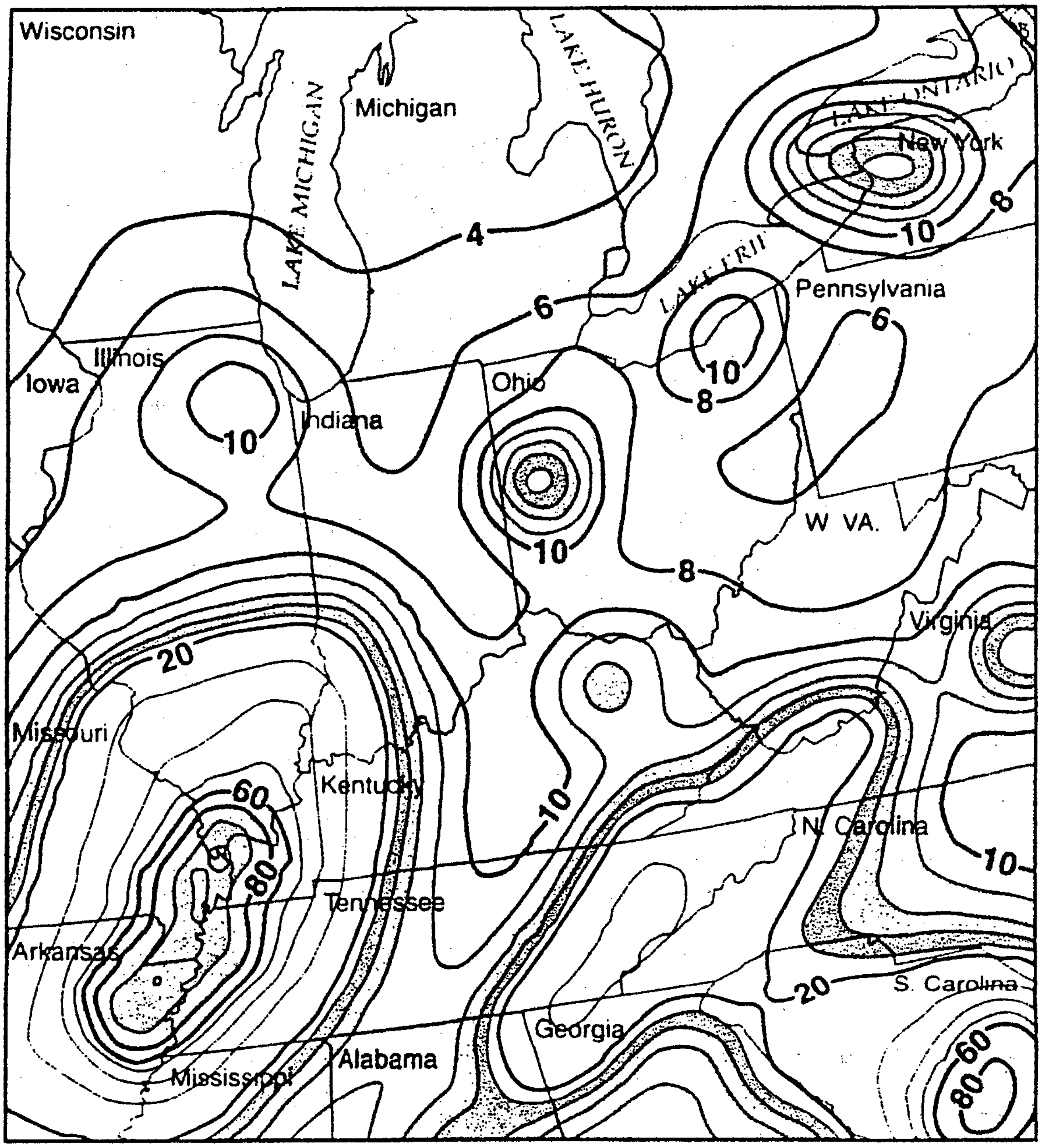

Figure 2.10: Mid-Continent Regional Distribution Map of Peak Lateral Ground Acceleration (as a percent of gravitational acceleration) Having a Two Percent Probability of Being Exceeded in Fifty Years

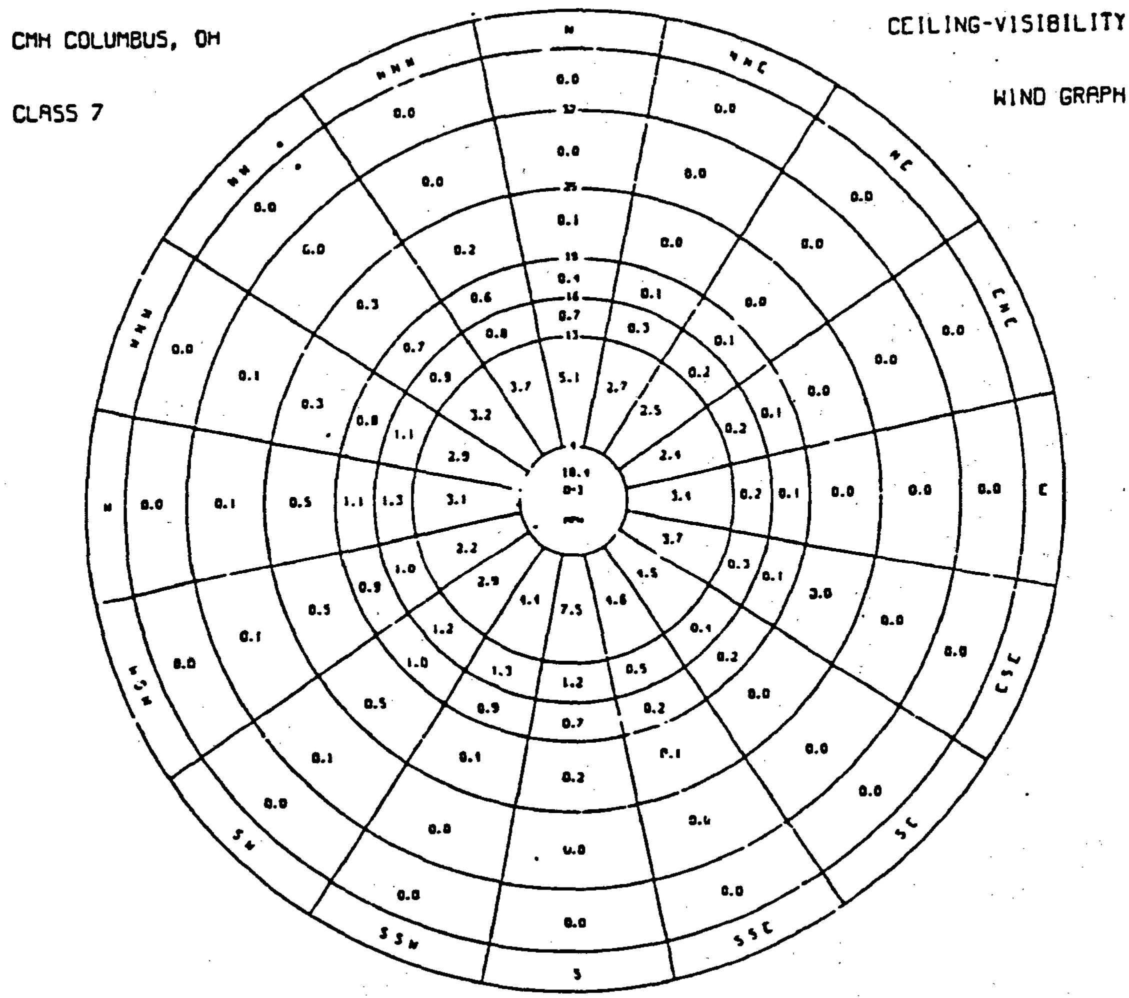

Figure 2.11: Wind Speeds and Directions in the Columbus Area

Table 2.2: Temperature Summary

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*********************** TEMPERATURE SUMMARY *******************************
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Station: (331786) COLUMBUS_WSO_AIRPORT   Missing Data: 0%   NCDC Averages
Averages: 1961-1990   Extremes: 1948-1996

	Averages			Daily Extremes				Mean Extremes				#Day-Max		#Day-Min	
	Max	Min	Mean	High	Date	Low	Date	High	Yr	Low	Yr	=> 90	<= 32	<= 32	<= 0
Ja	34.1	18.5	26.4	74	25/1950	-22	19/1994	39.9	50	11.4	77	0	12	26	2.2
Fe	38.0	21.2	29.6	73	25/1957	-13	02/1951	39.0	54	16.6	78	0	8.0	23	1.2
Ma	50.5	31.2	40.9	82	31/1981	-6	09/1984	50.4	73	28.4	60	0	2.3	18	0.1
Ap	62.0	40.0	51.0	88	23/1960	14	07/1982	57.4	54	45.8	50	0	0.1	6.7	0
Ma	72.3	50.1	61.2	93	30/1953	25	10/1966	70.9	91	55.4	67	0.6	0	0.5	0
Jn	80.4	58.0	69.2	101	25/1988	35	11/1972	75.0	91	63.6	72	4.3	0	0	0
Jl	83.7	62.7	73.2	104	14/1954	43	06/1972	79.0	55	70.5	71	6.8	0	0	0
Au	82.1	60.8	71.5	101	20/1983	39	29/1965	78.4	95	68.3	67	4.8	0	0	0
Se	76.2	54.8	65.5	100	02/1953	31	21/1962	71.0	61	60.4	67	1.6	0	0.1	0
Oc	64.5	42.9	53.7	90	05/1951	17	21/1952	59.9	63	47.4	88	0	0	3.7	0
No	51.4	34.3	42.9	80	01/1950	-4	30/1958	48.2	85	33.9	76	0	1.3	14	0
De	39.2	24.6	31.9	76	03/1982	-17	22/1989	40.8	56	19.8	89	0	8.2	24	0.8
An	61.2	41.6	51.4	104	07/14/54	-22	01/19/94	55.4	91	49.7	76	18	32	116	4.4
Wi	37.1	21.4	29.3	76	12/03/82	-22	01/19/94	36.4	49	20.7	77	0	28	72	4.3
Sp	61.6	40.4	51.0	93	05/30/53	-6	03/09/84	57.0	91	46.6	84	0.6	2.4	26	0.1
Su	82.1	60.5	71.3	104	07/14/54	35	06/11/72	75.9	91	68.6	72	16	0	0	0
Fa	64.0	44.0	54.0	100	09/02/53	-4	11/30/58	57.5	73	47.7	76	1.6	1.3	18	0

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************* Midwestern Climate Center, Champaign IL *********************
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Table 2.3: Precipitation Summary

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************************ PRECIPITATION SUMMARY ***************************
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Station: (331786) COLUMBUS_WSO_AIRPORT Missing Data: 0%

Averages: 1961-1990 Extremes: 1948-1996

	Total Precipitation							Snow			#Days Precip		
	Mean	High--Yr		Low--Yr		1-Day Max		Mean	High--Yr		=>.01	=>.50	=>1.
Ja	2.18	8.29	50	0.65	61	4.79	21/1959	9.2	34.4	78	13.7	1.8	0.4
Fe	2.24	5.15	90	0.31	78	2.15	23/1975	7.0	16.4	79	11.6	1.3	0.2
Ma	3.27	9.60	64	1.01	79	3.40	9/1964	4.4	13.5	62	13.8	1.8	0.3
Ap	3.21	6.39	96	0.67	71	2.03	30/1983	1.1	12.6	87	13.2	2.1	0.6
Ma	3.93	9.11	68	0.95	77	2.12	29/1982	0.0	0.8	89	12.5	2.7	0.7
Jn	4.04	9.75	58	0.71	84	2.55	13/1981	0.0	0.0	49	10.7	2.9	1.0
Jl	4.31	12.36	92	0.99	51	5.13	13/1992	0.0	0.0	49	10.8	3.0	1.1
Au	3.72	8.63	79	0.58	51	3.17	5/1995	0.0	0.0	49	9.4	2.3	0.8
Se	2.96	6.76	79	0.51	63	2.66	14/1979	0.0	0.0	49	8.5	1.9	0.5
Oc	2.15	5.24	54	0.11	63	1.69	3/1986	0.1	4.6	93	9.0	1.1	0.3
No	3.22	10.67	85	0.60	76	2.38	10/1985	1.9	15.2	50	11.6	2.0	0.5
De	2.86	6.99	90	0.46	55	1.74	8/1978	5.3	17.3	60	13.2	1.6	0.3
An	38.09	53.18	90	24.51	63	5.13	7/13/92	29.0	47.5	78	138.8	24.6	6.7
Wi	7.28	14.39	50	3.52	77	4.79	1/21/59	21.9	46.4	78	38.5	4.6	0.8
Sp	10.41	17.91	64	5.02	76	3.40	3/ 9/64	5.6	18.5	87	39.5	6.6	1.6
Su	12.07	22.02	58	6.00	51	5.13	7/13/92	0.0	0.0	49	30.9	8.2	2.9
Fa	8.33	13.78	85	1.42	63	2.66	9/14/79	2.0	15.2	50	29.6	5.2	1.3

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************* Midwestern Climate Center, Champaign IL **********************
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Table 2.4: Other Meteorological Data

COLUMBUS, OH
PORT COLUMBUS INTL AP

ANNUAL

PERIOD OF RECORD 1965-74
29211 OBSERVATIONS

TABLE 10. CEILING, VISIBILITY, AND WEATHER BY WIND DIRECTION (PERCENT FREQUENCY OF OBSERVATIONS)

WIND DIR	CEILING (FEET) 0	100	200 TO 300	400 TO 900	1000 TO 1400	1500 TO 1900	2000 TO 2900	3000 TO 4900	5000 TO 9500	OVER 9500	VISIBILITY (MILES) 0 TO 3/16	1/4 TO 3/8	1/2 TO 3/4	1 TO 2 1/2	3 TO 6	OVER 6	WEATHER: RAIN AND/OR DRZL	FRZ RAIN AND/OR FRZ DRZL	SNOW AND/OR IP	FOG	FOG AND SMOKE	SMOKE AND/OR HAZE	TSTM	HAIL
N	.0	.0	.1	.7	.3	.5	.8	.8	.7	5.5	.0	.0	.1	.6	2.2	6.6	.7	.0	.3	1.3	.0	1.7	.1	
NNE	.0	.0	.0	.2	.2	.2	.2	.4	.3	2.2	.0	.0	.0	.2	.7	2.8	.3	.0	.2	.5	.0	.4	.0	
NE	.0	.0	.6	.2	.1	.1	.2	.3	.4	2.4	.0	.0	.0	.2	.7	2.9	.3	.0	.1	.4		.6	.0	
ENE		.0	.0	.2	.1	.1	.2	.3	.4	2.2		.0	.0	.3	.7	2.8	.3	.0	.1	.4	.0	.7	.0	
E	.0	.0	.0	.2	.1	.2	.3	.5	.7	4.0	.0	.0	.0	.3	1.4	4.1	.5	.0	.1	.8	.0	1.6	.0	
ESE		.0	.0	.3	.1	.2	.3	.6	.8	3.7	.0	.0	.0	.3	1.6	4.0	.7	.0	.1	.9	.0	1.4	.0	
SE	.0	.0	.1	.4	.2	.2	.4	.4	.8	3.4	.0	.0	.0	.3	1.7	3.6	.9	.0	.1	1.0	.0	1.3	.1	
SSE	.0	.0	.1	.2	.1	.1	.3	.6	.8	3.5	.0	.0	.0	.4	1.6	3.7	.6	.0	.0	.6	.0	1.3	.0	
S	.0	.1	.1	.8	.4	.4	.8	1.4	1.6	6.7	.0	.0	.1	1.0	2.3	7.9	1.3	.0	.2	1.6	.1	3.2	.1	
SSW	.0	.0	.1	.3	.3	.2	.5	.8	.7	3.0	.0	.0	.1	.4	1.5	4.0	.6		.1	.6	.0	1.5	.1	
SW		.0	.1	.3	.3	.4	.6	.8	.6	2.6	.0		.0	.3	1.1	4.1	.6		.2	.4	.0	1.1	.1	
WSW		.0	.0	.3	.3	.5	.6	.7	.3	2.0	.0		.0	.3	1.0	3.7	.4	.0	.4	.4	.0	.6	.1	
W			.0	.4	.6	1.0	1.4	1.2	.4	3.6		.0	.0	.5	1.5	6.9	.6	.0	1.0	.6	.0	1.1	.1	.0
WNW	.0		.0	.3	.4	.5	1.0	.8	.3	1.8	.0	.0	.0	.3	.8	4.0	.4	.0	.5	.3		.6	.0	
NW	.0		.1	.3	.3	.4	.4	.6	.3	1.7	.0		.0	.2	.8	3.2	.3	.0	.4	.4		.6	.0	
NNW	.0	.0	.1	.5	.3	.4	.5	.5	.3	2.0	.0	.0	.1	.3	.9	3.3	.4	.0	.4	.6		.7	.0	
CALM	.0	.0	.0	.0	.0	.0	.1	.1	.2	2.8	.1	.0	.0	.2	.9	2.1	.1		.0	.0	.0	.8		
TOT	.1	.2	.9	5.7	4.3	5.5	9.2	11.1	9.7	53.1	.3	.1	.6	6.6	22.4	69.9	9.0	.2	4.3	11.6	.3	19.8	.7	.0

IP = ICE PELLETS (REPLACES SLEET AND SMALL HAIL)

TABLE 11. WIND DIRECTION VS. WIND SPEED (PERCENT FREQUENCY OF OBSERVATIONS)

A. ALL WEATHER

WIND DIR	WIND SPEED (KNOTS) 0-3	4-6	7-10	11-16	17-21	22-27	28-33	34-40	OVER 40	TOT	AVG SPEED
N	.7	4.3	2.9	1.4	.1	.0				9.6	7.1
NNE	.5	1.8	1.1	.4	.0					3.8	6.5
NE	.3	1.9	1.2	.3	.0					3.9	6.3
ENE	.4	1.6	1.2	.3	.0					3.4	6.4
E	.8	3.0	1.9	.4	.0					6.1	6.2
ESE	.6	3.0	2.2	.3	.0					6.1	6.6
SE	.3	2.3	2.3	.9	.0	.0				6.1	7.3
SSE	.3	2.6	2.3	.6	.0					5.8	7.6
S	.3	4.2	5.0	2.5	.2	.0				12.3	8.2
SSW	.1	1.0	2.3	2.0	.3	.0	.0			5.9	10.1
SW	.1	.8	1.9	2.2	.4	.1	.0	.0	.0	5.3	11.0
WSW	.1	.6	1.7	2.1	.3	.1	.0			5.0	11.4
W	.1	1.2	3.1	3.5	.4	.1	.0			8.4	11.0
WNW	.1	.8	1.8	2.1	.3	.1	.0			5.1	10.6
NW	.2	1.0	1.6	1.4	.2	.0				4.3	9.3
NNW	.3	1.6	1.6	1.2	.1	.0				4.7	8.3
CALM	3.3									3.3	
TOT	8.4	31.9	34.4	21.7	3.0	.5	.1	.0	.0	100.0	8.2

ALL WEATHER: ALL WIND OBSERVATIONS

B. IFR

WIND DIR	WIND SPEED (KNOTS) 0-3	4-6	7-10	11-16	17-21	22-27	28-33	34-40	OVER 40	TOT	AVG SPEED
N	.1	.4	.3	.3	.0					1.2	8.4
NNE	.0	.1	.1	.1	.0					.3	7.8
NE	.0	.2	.1	.0	.0					.3	8.7
ENE	.0	.2	.1	.0						.3	8.7
E	.1	.3	.2	.0						.6	8.4
ESE	.0	.3	.2	.1						.7	7.3
SE	.1	.2	.3	.1	.0					.7	7.3
SSE	.0	.3	.2	.1						.6	8.7
S	.1	.6	.6	.2	.0					1.3	7.7
SSW	.0	.1	.3	.2	.0	.0				.7	8.8
SW	.0	.1	.2	.2	.0	.0				.5	9.4
WSW	.0	.1	.2	.2	.0	.0	.0			.5	10.3
W	.0	.1	.2	.2	.1	.0				.7	10.6
WNW	.0	.1	.2	.2	.0	.0				.5	9.7
NW	.0	.1	.2	.2	.0	.0				.5	9.8
NNW	.0	.2	.2	.2	.0	.0				.7	9.4
CALM	.3									.3	
TOT	.8	3.4	3.8	2.3	.3	.0	.0			10.8	8.1

IFR: CEILING < 1000 FT AND/OR VISIBILITY < 3 MI BUT ≥ 200 FT AND ≥ 1/2 MI.

TABLE 12. WEATHER CONDITION BY HOUR (MEAN NO. OF DAYS)

	WEATHER CONDITIONS	HOUR (LST) 01	04	07	10	13	16	19	22
WEATHER TYPE	RAIN AND/OR DRIZZLE	32.3	33.7	32.6	33.0	32.7	35.8	32.1	30.7
	FRZ RAIN AND/OR FRZ DRIZZLE	.9	1.3	1.2	.4	.6	.2	.5	.6
	SNOW AND/OR ICE PELLETS	16.3	18.0	17.3	18.7	17.7	17.6	13.9	16.8
	HAIL								.1
	PRECIPITATION	46.7	50.6	50.5	50.9	49.8	52.6	45.3	49.6
	FOG	39.8	61.0	101.3	43.7	23.1	22.9	26.3	25.8
	FOG AND SMOKE	.1	.8	6.1	.8	.3	.2	.2	.1
	SMOKE AND/OR HAZE	39.4	48.8	118.6	129.8	82.7	69.0	58.4	62.9
	OBSTRUCTIONS TO VISION	60.7	89.6	103.6	159.6	103.7	65.4	78.6	82.6
	THUNDERSTORM	2.9	1.9	1.3	.7	3.4	3.4	4.8	3.0
WIND SPEED (KNOTS)	CALM	22.3	29.4	21.3	4.3	3.2	.8	2.5	13.2
	1 – 6	176.8	178.1	169.3	100.7	73.6	72.6	130.9	171.6
	7 – 10	110.0	104.2	112.1	140.3	134.9	133.8	164.6	120.2
	11 – 16	48.3	44.8	30.0	99.9	127.3	133.9	68.4	52.6
	17 – 21	6.5	5.7	3.3	12.0	21.6	20.6	8.8	6.7
	22 – 27	.9	.8	.8	3.2	4.9	3.1	1.7	.9
	28 – 33	.1	.1	.1	.1	.5	.4	.2	.2
	OVER 33					.2			
VISIBILITY (MILES)	0 – 3/16	1.1	3.0	4.3	.8	.2			.3
	1/4 – 3/8	.6	1.1	1.6	.4	.3	.1	.2	.2
	1/2 – 3/4	.9	1.8	4.6	4.2	1.3	2.5	.9	.7
	1 – 2 1/2	9.7	14.2	33.8	40.6	23.3	21.6	15.3	7.2
	3 – 6	46.8	82.4	86.4	78.1	34.6	45.3	30.0	46.9
	OVER 6	304.0	282.8	214.6	233.2	284.5	295.7	298.7	309.9
TEMPERATURE (°F)	ZERO OR LOWER	.6	2.3	2.4	1.1	.3	.2	.4	.3
	1 – 32	80.6	84.0	89.6	60.8	47.7	49.3	58.2	71.2
	33 – 44	72.8	76.5	73.6	63.6	63.3	60.3	64.6	68.8
	45 – 64	137.7	143.5	137.9	102.2	92.7	93.2	99.4	123.8
	65 – 79	73.4	50.4	61.6	130.2	150.4	158.9	142.1	98.9
	80 – 89					6.8	7.4	.4	
	OVER 89								

VALUES ARE ROUNDED TO NEAREST TENTH, BUT NOT ADJUSTED TO MAKE THEIR SUMS EXACTLY EQUAL TO COLUMN OR ROW TOTALS.

THESE VALUES ARE BASED ON 3-HOURLY OBSERVATIONS.

adjacent to the city. The Scioto River, located to the west of the city and of the reactor site, is the principal stream and flows from the northwest into the center of the city and then flows straight south toward the Ohio River. The Olentangy River, which is located to the east of the reactor site, runs almost due south and empties into the Scioto just west of the business district. Two minor streams, Alum Creek and Big Walnut Creek, run through portions of Columbus and skirt the eastern and southern fringes of the area. Alum Creek empties into the Big Walnut southeast of the city and the Big Walnut empties into the Scioto a few miles downstream. The Scioto and Olentangy feature gorge-like formations with very little flood plain and the two creeks have only a little more flood plain or bottomland.

2.5.2 Ground Water

Infiltration from the reactor site enters the groundwater table which is approximately 45 to 50 feet below the surface. The groundwater flows toward the Olentangy River which is 1.1 miles to the east. The Olentangy River joins the Scioto River at a point about 2.5 miles south of the site, and flows past the city of Columbus in a southern direction. The Dublin Road Water Treatment Plant for the city of Columbus is located on the Scioto River 2.2 miles south by southeast of the reactor site. This location is upstream of the confluence of the Scioto and Olentangy Rivers and uses water collected from the Scioto River basin. Thus, contamination of the city of Columbus' water supply by infiltration of the reactor pool water is virtually impossible. The next major town using water from the Scioto River is Circleville, located 30 miles south of Columbus.

2.6 Reactor Building Description

2.6.1 Reactor Building Design and Construction

The reactor building is a steel framed structure with insulated metal wall panels and built up roof. The ground floor is a concrete slab on grade. The second floor slabs are concrete supported on steel beams. Elevated platforms are checkered plate supported on steel beams. Interior partitions are plasterboard. Floor drains provide drainage.

2.6.2 Layout

The overall exterior ground floor dimensions of the reactor building are 62 feet by 48 feet. The reactor building is divided into three 48 foot sections, extending from west to east (see Figure 2.12). The western section extends 16 feet into the building. It is a single level section with an internal floor to ceiling elevation of 11 feet. The center section, known as the bay area, houses the pools, reactor, and reactor facilities. It stretches 29 feet 3 inches from the west section with a ceiling height of 35 feet. The eastern section of the building is a two level section with an overall floor to ceiling height of 22 feet. On both the first and second levels there is a walkway extending a distance of 3 feet 9 inches east from the bay area

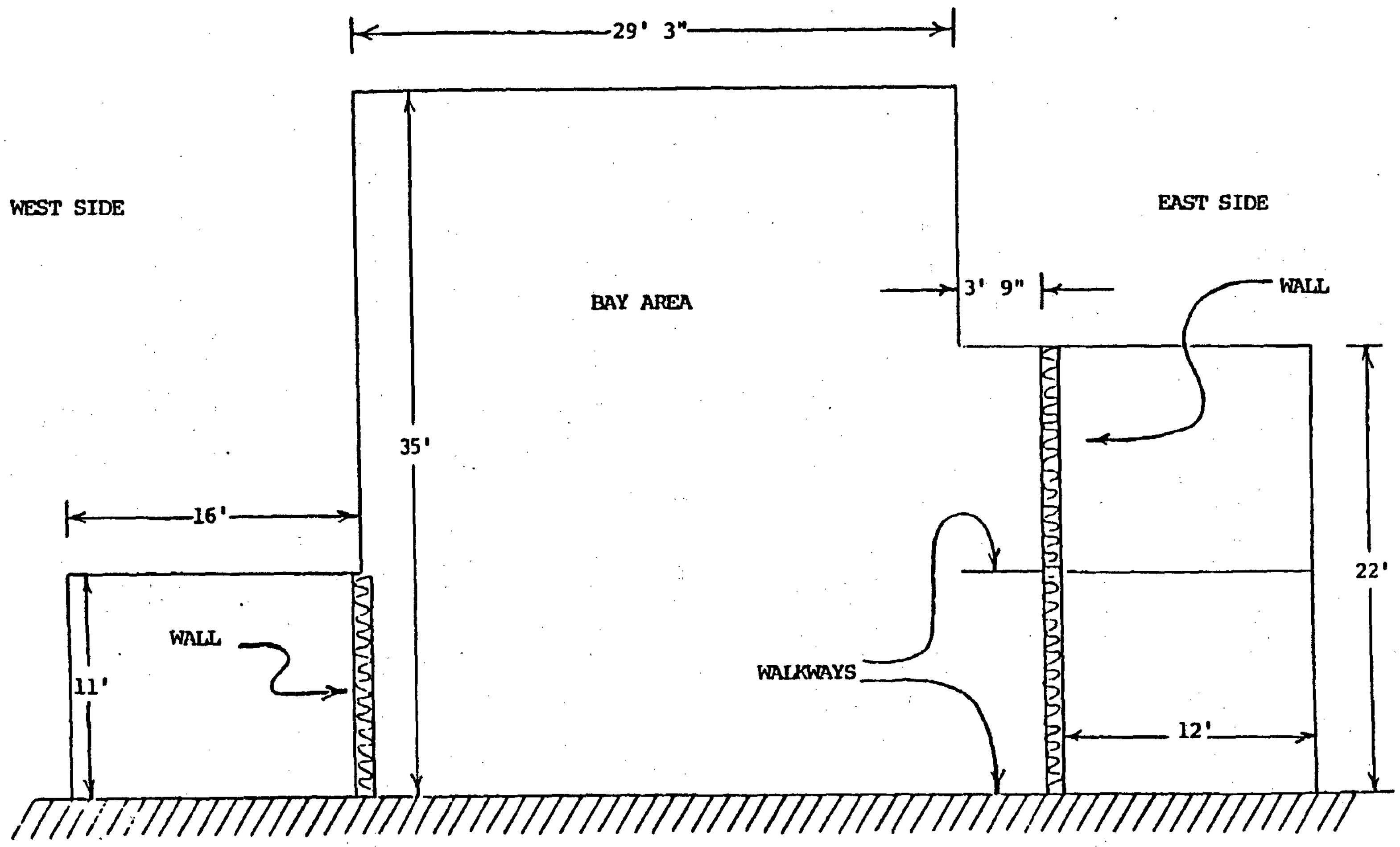

Figure 2.12: Dimensions of the Reactor Building Shown in a Cross-Sectional View

to an interior wall. From here, the building continues a distance of 12 feet to the eastern exterior wall of the Building.

2.6.3 General Features

The reactor building area is serviced electrically by the Columbus and Southern Ohio Electric Company via 13,800 volt-ampere lines and has an electric failure probability of 0.2 failures per year. The in-house electric service is 120/240 volts, three wire, single phase; 240 volts, three phase; and 120/208 volts, four wire, solid neutral for lighting and power.

Water is supplied by the city of Columbus via piping distribution from its treatment and pumping plant located 2.2 miles south by southeast of the reactor site.

Natural gas service to the reactor building is provided by the Columbia Gas System of Ohio. The building also contains service air at a pressure of about 70 pounds per square inch. A forced warm air system, using gas-fired furnaces, heats the building. The building air conditioning system is tied into the ductwork for the furnace system heating the east side of the building. Window air conditioner units are used in certain rooms for additional cooling capacity (e.g., control room).

2.7 Chapter 2 Bibliography

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3.0 Reactor Facility Description

3.1 Reactor Core

3.1.1 Core Structures

3.1.1.1 Grid Plate and Support

The Reactor Core Grid Plate is located at the west end of the reactor pool. The grid plate, shown in Figure 3.1, is a 15.66 inch by 18.70 inch by 5.03 inch A356-T6 aluminum alloy plate, cast with a 5 by 6 array of 2.48 inch diameter through holes. The grid plate holds the core elements in place by fitting the lower end boxes of the elements into the through holes. A 0.25 inch diameter dowel pin fixes the rotational position about the vertical axis of each element. The grid plate is positioned in the pool so that the east and west sides have five holes, and the north and south have six.

The grid plate is attached to a frame by eight 0.5 inch hex head screws. The grid plate frame, shown in Figure 3.2, is made of aluminum extrusion having four feet resting on pedestals mounted to the reactor pool floor with four threaded studs that penetrate the pedestals and frame feet. Leveling shims are located between the frame feet and pedestals. The grid plate and frame assemblies are positioned so that the distance from the reactor pool floor and the center of the core is 3 feet 6 inches. The core is centered between the north and south walls, which are 1 foot 10.25 inches from the center of the core. The inner sides of the east and west walls are a distance of 8 feet 9 inches and 1 foot 10 inches respectively, from the center of the core.

3.1.1.2 Fuel

The OSURR is fueled with uranium enriched to 19.5%, in the form of uranium-silicide (U_3Si_2) and aluminum. This fuel type was developed as part of the Reduced Enrichment for Research and Test Reactors (RERTR) fuel development program of the Argonne National Laboratory (ANL). It is the fuel type of choice by the Department of Energy, in consultation with the National Organization of Test, Research, and Training Reactor Operators (TRTR), in which the OSURR staff is active. The fuel is supplied by the Babcock & Wilcox Company under the terms of the DOE Fuel Assistance Program, managed by EG&G Idaho, Inc.

The uranium-silicide and aluminum fuel exhibits superior properties under a variety of conditions, including high burnup and elevated temperatures. The fuel shows a strong resistance to mechanical degradation and breakaway swelling for burnup densities considerably in excess of the burnup expected in the OSURR fuel cycle length. It is expected that the fuel lifetime in the OSURR core will be reactivity limited, rather than burnup limited. Blister tests conducted in the RERTR program on high burnup fuel have provided data which can be used to show that no fission product releases will occur from the fuel in

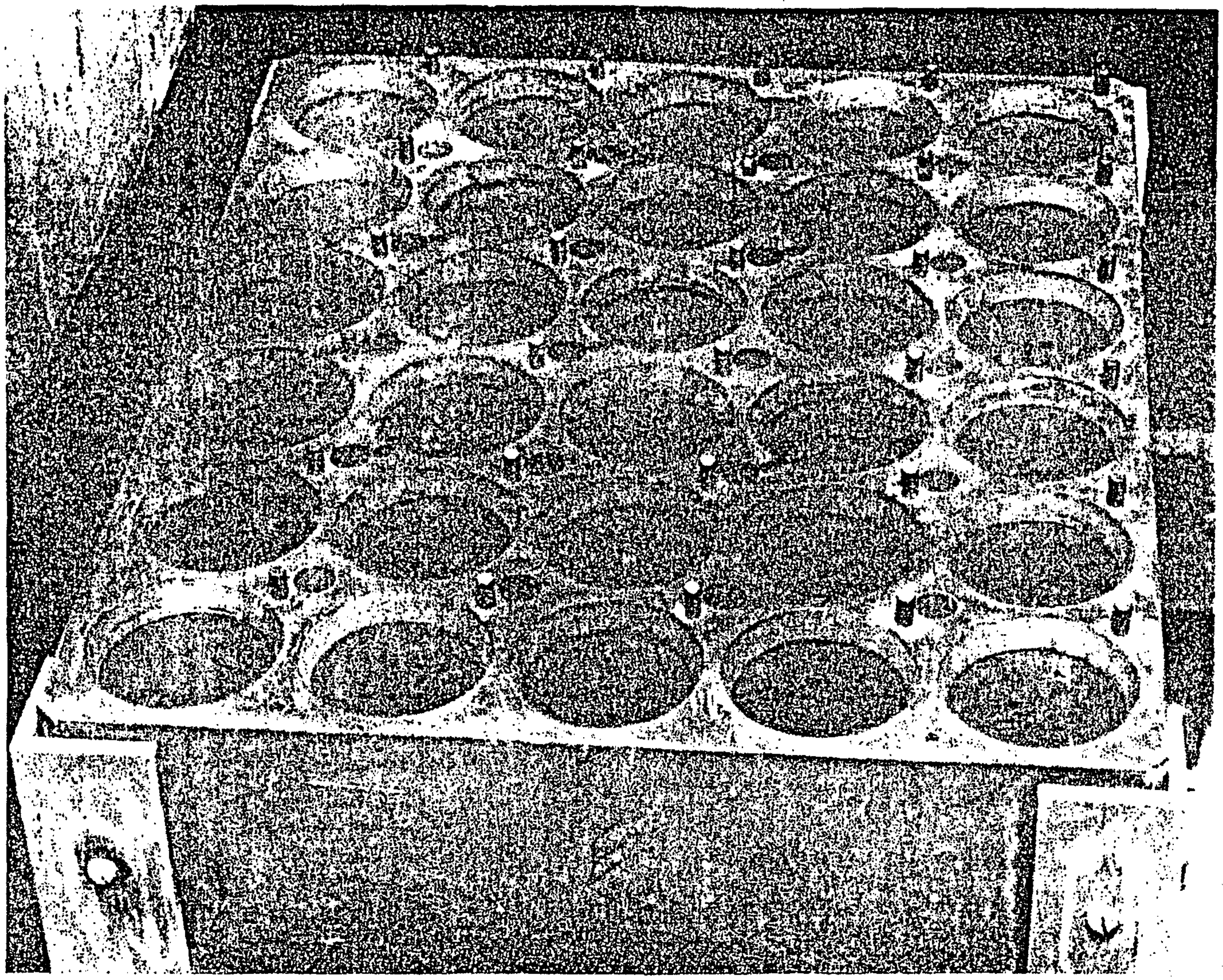


Figure 3.1: OSURR Grid Plate

Figure 3.2: OSURR Grid Plate Frame

Note: Startup source is at the[redacted]] the OSURR core even under maximum credible accident conditions that do not involve direct mechanical damage to the fuel plates. These accident scenarios are discussed in more detail in Chapter 8 of this report.

The active portion of the fuel, commonly referred to as the "meat", is contained within flat aluminum plates 0.050 inches thick. The meat thickness is 0.020 inches. Fuel plates are joined to aluminum side plates to form either standard, partial, or control rod fuel elements (also called fuel assemblies). Each fuel assembly contains both fueled and unfueled (or "dummy") plates, the unfueled plates being made of pure aluminum with no U_3Si_2 content, but otherwise identical to fueled plates. Each fuel assembly has an upper and lower end box. The upper end box consists of a bracket and handle assembly to allow handling of the fuel element with a special handling tool. The lower end box is formed from a tapered aluminum cylinder which inserts into the sockets in the grid plate. A positioning dowel pin is located near the top of the tapered cylindrical section. This pin inserts into a corresponding hole in the grid plate to allow reproducible positioning of the fuel assemblies on the grid plate.

Standard fuel assemblies have a total of 18 plates per fuel element. Of these 18 plates, 16 contain uranium fuel, and two are unfueled, pure aluminum, or dummy plates. The dummy plates are the two outer plates of each fuel element. Use of two dummy fuel plates allows a lower loading of uranium in each standard fuel assembly, permitting the core of the OSURR to be physically larger than it would be if all plates of a standard fuel element contained uranium fuel. If dummy plates were not used, either a smaller core would result for a given excess reactivity, or a larger core could be loaded with a higher excess reactivity. In either case, undesirable operational and/or safety consequences would result, given the design and operational characteristics of the OSURR. A total of [redacted] standard fuel elements are available for the OSURR. Given the reactivity limit[redacted] OSURR, a cold, clean critical core size was predicted using [redacted] standard fuel assemblies, along with 4 control rod fuel elements. Details of a standard fuel assembly are shown in Figure 3.3. Chapter 4 provides information on predicted core size.

The control rod fuel elements are similar to the standard elements in that they utilize the standard fuel plates and a similar side plate. Several of the inner fuel plates have been completely removed to allow room for the control rod to pass through the element. On either side of this gap for the control rod are pure aluminum guide plates. The outer two fuel plates in each control rod fuel assembly contain uranium fuel. The upper end box of the control rod fuel elements have a modified grappling section to allow handling with the special handling tool used for manipulation of control rod fuel elements. An aluminum guide plate for the control rod is also mounted at the top of the assembly. Figure 3.4 shows a control rod fuel element in more detail.

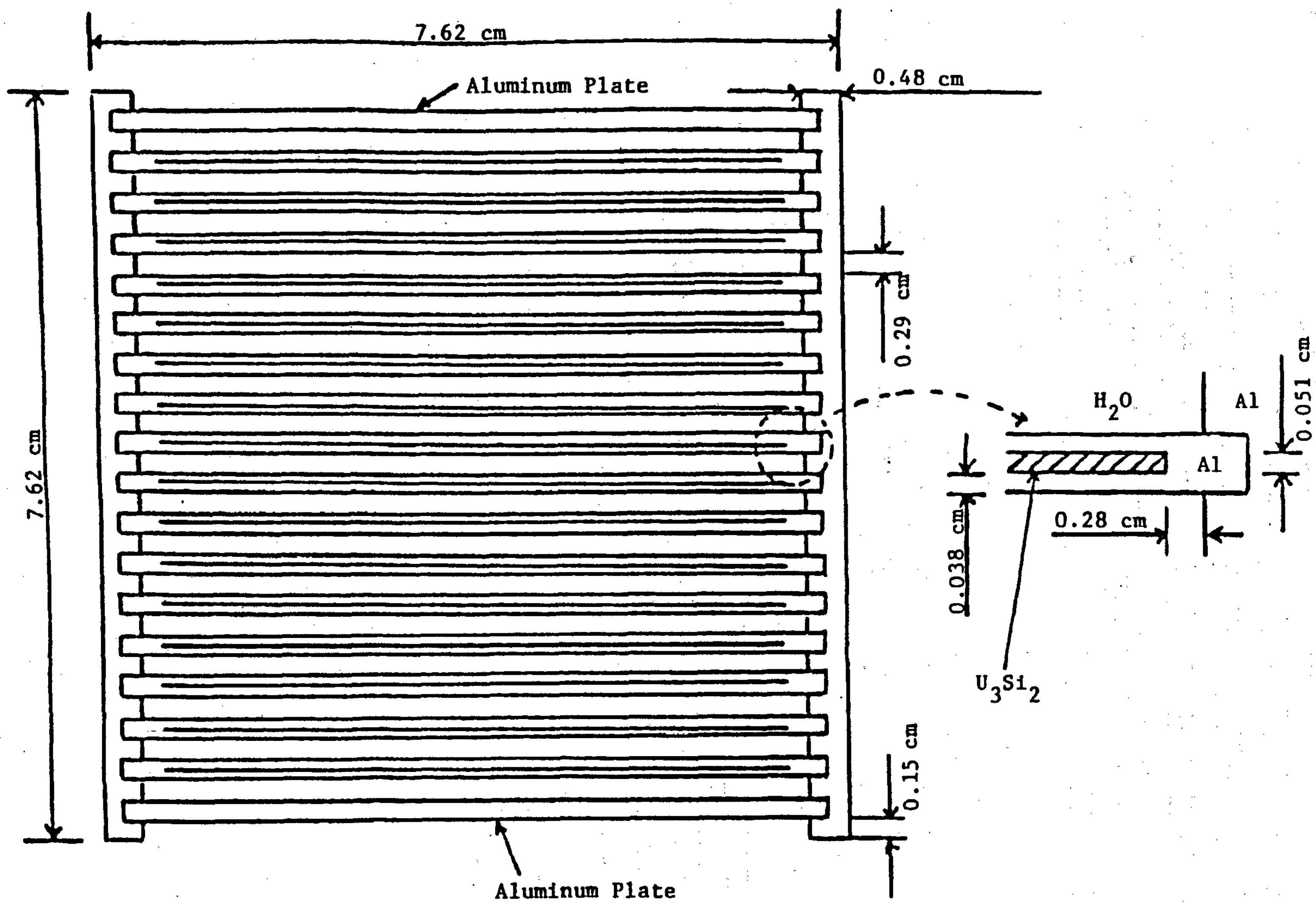


Figure 3.3: OSURR Standard Fuel Element

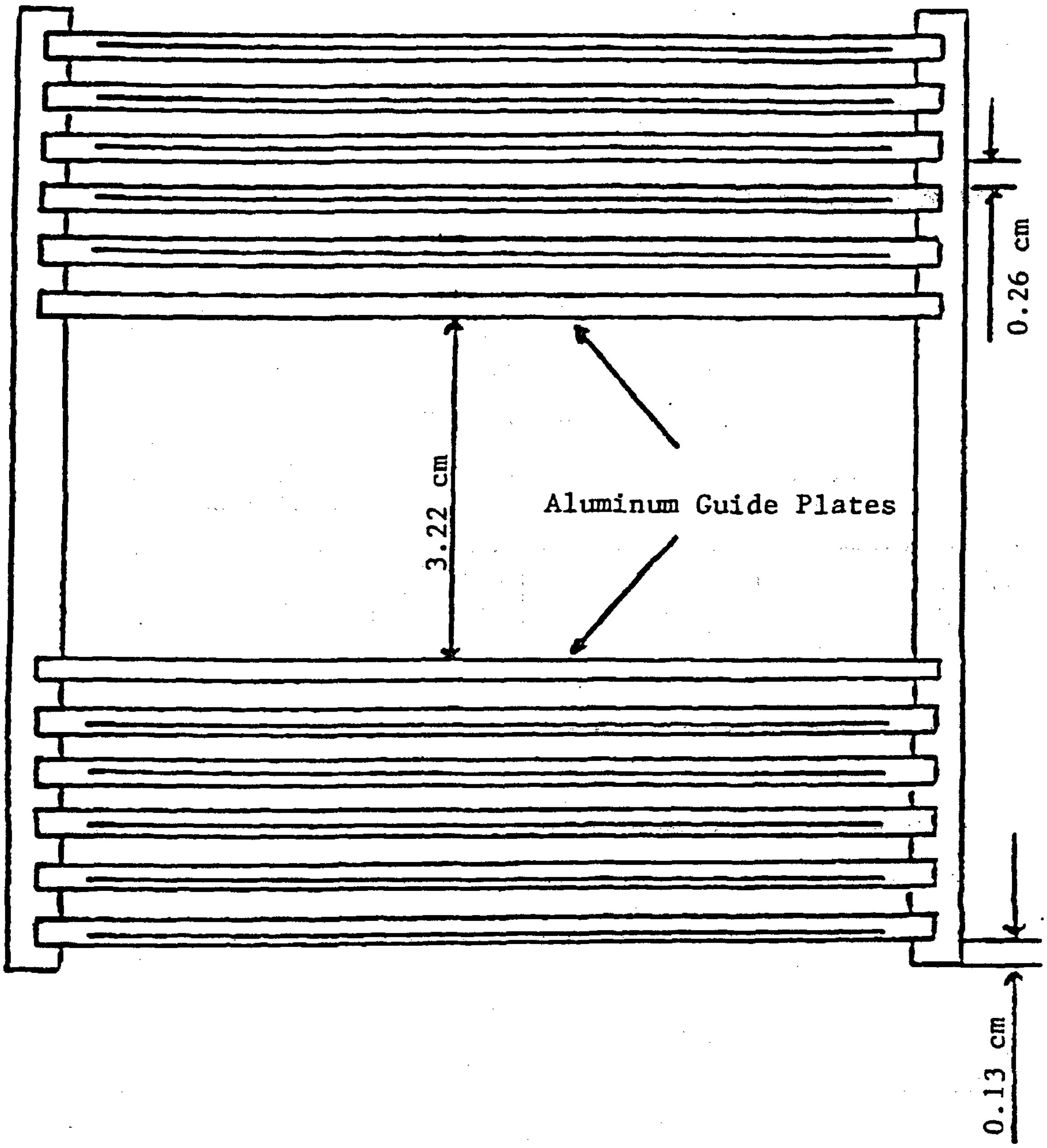


Figure 3.4: OSURR Control Rod Fuel Element

The partial fuel elements are physically identical to standard elements with the exception of their uranium loading. They utilize standard fuel plates, but some of the fuel plates have been removed and replaced with pure aluminum dummy plates. Partial elements are available with 25, 37.5, 50, and 62.5 percent of the nominal uranium loading of a standard element. Use of partial fuel elements allows precise adjustment of the excess reactivity of the OSURR core.

All standard, partial, and control rod fuel assemblies are stamped on both side plates with an identifying alphanumeric symbol unique to the assembly. This identifier, formed of 2-inch high letters and numbers, can be read underwater at a depth of at least 3 feet. These identifiers are used in verifying core loading and in material accounting procedures.

When loaded in the core, fuel elements are positioned on the grid plate. When not in use in the core, or when being stored as spare or spent fuel assemblies, fuel elements are kept in the fuel storage pit at the east end of the reactor pool. A single fuel assembly may be in transit at any one time between the core grid plate and the fuel storage pit. Exceptions to these storage procedures require approval by the Reactor Operations Committee and the Nuclear Regulatory Commission.

3.1.1.3 Core Arrangements

Up to 30 positions on the core grid plate are available for use as fuel element positions. Control rod fuel elements occupy 4 of these positions, and one is reserved for the Central Irradiation Facility (CIF) flux trap (water or graphite-filled). Also, a total of 5 graphite isotope irradiation elements (GIIE) are available for positioning on the grid plate, although not all GIIE assemblies need be mounted. Specially fabricated hollow, water-filled, plug-type elements of pure aluminum will also be made available for use in the grid plate. Allowing for the 4 control rod fuel elements and one CIF flux trap position, a total of 25 positions are available for use as fuel or GIIE assembly positions. More information on both the GIIEs and CIF is given in the following section concerning experimental facilities.

Maintaining a uniform, well-behaved, and predictable flow of coolant through the core is achieved by assuring that all grid plate positions are occupied by some type of assembly. In this way, no holes are left in the grid plate which could result in anomalous flow patterns through the core. The type of assembly occupying a given grid plate position will depend in part on experimental requirements, fuel burnup, and reactivity limitations. In general, however, as symmetric a core as achievable, within these limits, is desirable. Control rod elements should be surrounded with fuel, in as much as possible given reactivity and burnup constraints, so as not to reduce the effective worth of the control rod. A regular pattern of control rod element positions relative to the overall core geometry is desirable to maintain predictable control rod worth.

A cold, clean critical core size is predicted using [redacted] standard fuel elements, 4 control rod fuel elements, and a single, central position for the CIF. This leaves 2 or 3 positions to be occupied by other kinds of assemblies, either GIIE elements or plugs. Thermal hydraulic and reactivity requirements and/or limitations will determine which will be used and their locations on the grid plate.

3.1.1.4 Control Rod Poison Sections

Those sections of the control rods which actually effect control of the neutron population in the core, known as poison sections, penetrate into the core boundary through gaps in the control rod fuel elements. A total of 4 control rod poison sections penetrate the active boundary of the core. Three of the rods are shim safety rods and one is a regulating rod. The poison sections of the shim safety rods are 26 inches long, while that of the regulating rod is 24 inches long. They travel through a maximum stroke of 61 centimeters (24 inches). More detail on the control rod system is presented in section 3.3.8 of this chapter.

3.1.2 Associated Core Structures

3.1.2.1 Core Reflection

The core of the OSURR is reflected by reactor-grade graphite and high-purity water. Solid graphite reflectors, each encased in its own aluminum shell, are mounted to the reactor pool walls and reflect the west and south sides of the core. The GIIE assemblies, if used, provide reflection at the locations where they are placed on the grid plate. A detailed description of the solid graphite reflectors, which also form the extensions for experimental thermal column facilities, is presented in Section 3.1.3.1, while the GIIE assemblies are described in Section 3.8.6 of this chapter.

Where graphite is not present, most reflection is achieved by the high-purity, light water filling the reactor pool. Water provides the primary neutron reflection effects at the top and bottom and along the north face of the core. Some of the reflecting power of the water at these positions is lost by the presence of other structures and facilities, such as the grid plate below the core, and the experimental facilities located along the north face of the core (beam ports, rabbit).

3.1.2.2 Startup Source and Source Drive

A plutonium-beryllium (PuBe) source provides an initial source of neutrons to the OSURR core for controlled startup of the reactor. The initial source activity was [redacted] curies, which provides a neutron emission rate of 8.95×10^6 neutrons/second. A minimum of 10^6

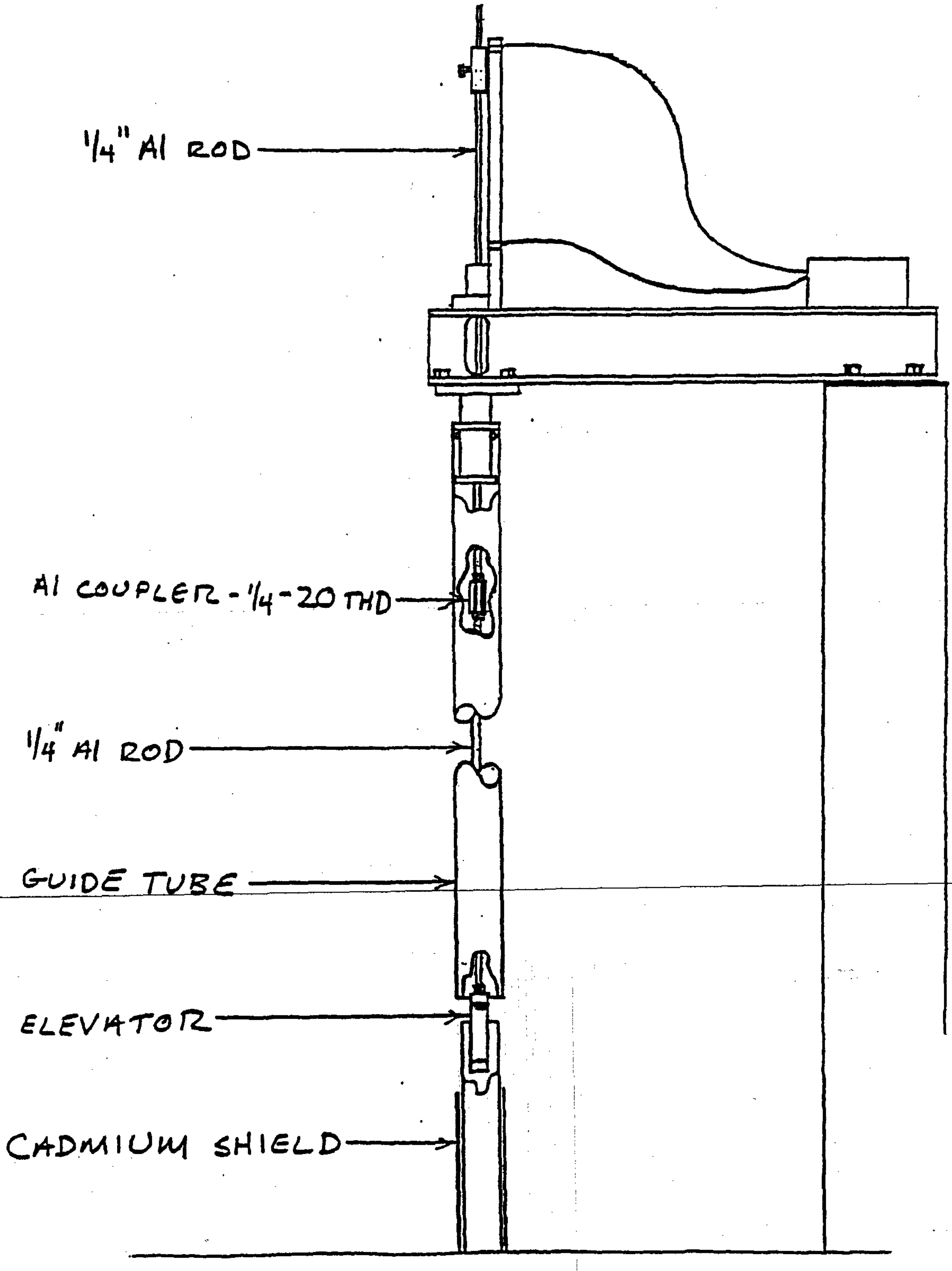


Figure 3.5: OSURR Source Drive Assembly

neutrons/second is required. The source activity has slowly increased as a result of buildup of alpha-emitting decay products of plutonium. Total source activity at the end of 1986 was such that the neutron emission rate was slightly above $1x10^7$ neutrons/second. The source is located in a fixed housing assembly, located [redacted] slightly offset from the vertical centerline of the core. The bottom of the housing is mounted to a positioning bracket attached to the grid plate pedestal, while the top of the drive housing mounts to brackets attached to the pool walls. The lower section of the source housing is wrapped with a cadmium sheet of sufficient thickness to be "black" to thermal neutrons.

Within the source housing is a movable platform on which the source rests. The position of this platform can be raised and lowered as selected from a switch in the control room. The source drives through a maximum stroke of 95 centimeters. The source is normally in its raised position for reactor startup, in which position the neutron emitted by the source are available to induce fissions in the fuel. When raised, the startup source is located between the planes defined by the grid plate and the tops of the fuel elements. After the reactor has achieved a critical or supercritical condition, the source is typically lowered into its shielded position, wherein neither the core nor the startup source is exposed to thermal neutrons from each other. Source shielding consists of cadmium sheet wrapped around the lower section of the source housing. When the reactor is shutdown, the source may be in either its raised or lowered position.

The electromechanical source drive system is operated from the control console. The electric drive motor controller and position encoder is located on the south edge of the top of the reactor pool wall. The source drive motor is mounted on the beam supporting the source drive housing tube, extending outward from the edge of the pool wall, above the surface of the pool. The shroud forming the source drive housing tube, which runs down the length of the reactor pool to the vicinity of the core, is made of aluminum. More detail on the operation of the source drive system, including its safety system interlock features, is contained in a following section on reactor instrumentation and control.

3.1.2.3 Control Rod Housings

The control rods move within aluminum housings. These housings join to the tops of the four control rod fuel elements. The control rod poison sections move upward into these housings above the core when being withdrawn from the core. The aluminum shrouds forming the rod housings are made from 2.75 inch outer-diameter tubing, with 0.125 inch wall thickness. The housings are supported at the top of the reactor pool with brackets attached to the walls of the pool. Holes are cut in the lower sections of the aluminum housings to allow coolant flow and water ejection during sudden rod insertion movements (reactor SCRAM, or rod drop experiments). More information on the control rod system, including information on the rod housings and associated apparatus, is contained in Section 3.3.8 of this chapter.

3.1.3 Reactor and Shielding Pools

3.1.3.1 Reactor Pool

The reactor pool is built on a 15 inch thick regular concrete foundation. The pool has inner dimensions of 3 feet 8.5 inches by 10 feet 7 inches by 20 feet deep, with a capacity of 5800 gallons of water. The walls are constructed of barytes concrete to an elevation of 13 feet 6 inches above the foundation. The remaining 6 feet 6 inches of the walls are made with regular concrete. A scupper is located along the top edge of the east wall. Figure 3.6 shows the layout and dimensions for the reactor pool with scupper. The pool is completely lined with a layer of fiberglass-reinforced epoxy paint coating to prevent leakage, leaching of the materials in the concrete by the pool water, and to facilitate decontamination and repair of the walls.

Constructed at the east end of the reactor pool is the Fuel Storage Pit (refer to Figure 3.7). The sides of the pit are made with 7 inch thick regular concrete, the lid section of which is coated with a 0.25 inch layer of the same sealant used for the reactor pool. The dimensions of the lid section are 16.75 inches by 3 feet 5.75 inches with a depth of 8 inches. The pit dimensions are 12.75 inches by 3 feet 3 inches with a depth of 4 feet 1.25 inches extending from the bottom of the lid section. There are two Fuel Storage Pit Plugs that cover the pit. These plugs are identical except for the positions of the eyebolts used to lift them out of the well. The plugs are made of type 3003-H112 0.25 inch aluminum welded at the joints, filled with lead brick. The dimensions for the plugs are shown in Figure 3.8.

A Fuel Element Storage Rack, capable of storing a complete core loading in a 10 by 3 array exists in the Fuel Storage Pit. Figure 3.9 shows detail on the storage rack. The rack is made with 0.25 inch 3003-HI12 aluminum sides and 0.25 inch boral spacers that span the width of the rack and act as a safeguard against inadvertent criticality. The effective multiplication of the fully loaded Fuel Storage Pit, when filled with water, is 0.68 or less. The floor is a 0.75 inch 1100 aluminum plate, which rests on four 0.5 inch 1100 aluminum bars that are welded to the sides of the rack. The bottom side of the rack floor is elevated 9 inches above the Fuel Storage Pit floor. The plate forming the rack floor has 0.25 inch deep slots into which the boral spacers fit, as well as tapered holes at each fuel element position. The tapered holes act as sockets into which insert the lower end boxes of the fuel elements. The tops of the boral spacers are supported by two 0.25 inch by 0.75 inch 3003-H112 aluminum bars that extend lengthwise across the top of the rack. These bars are welded to the two sides of the rack and have 0.25 inch-width slots for the boral spacers. The secondary purpose of the bars is to hold the top ends of the stored fuel elements in place.

Figure 3.6: Overall Reactor Dimensions

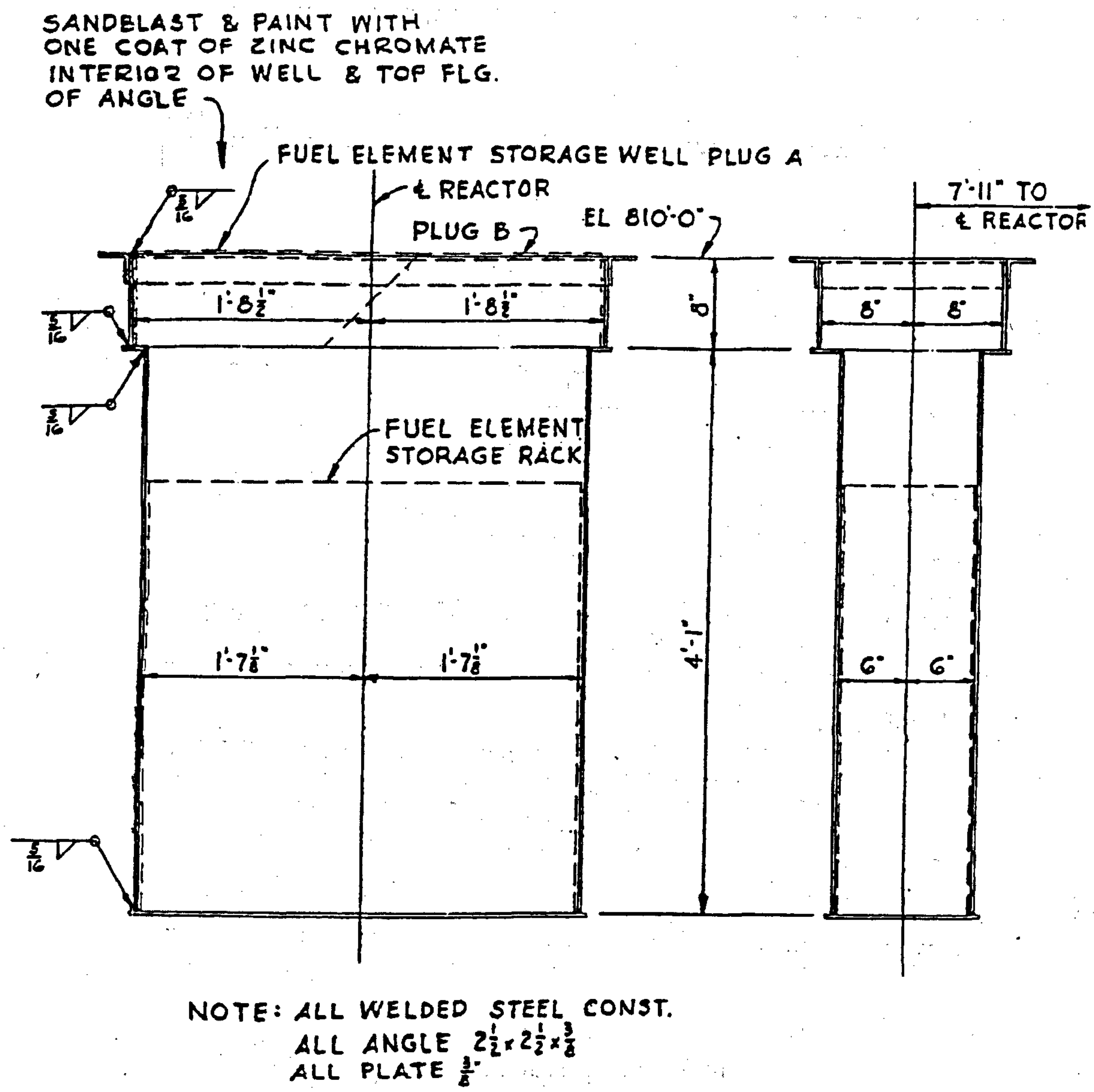


Figure 3.7: Fuel Storage Pit

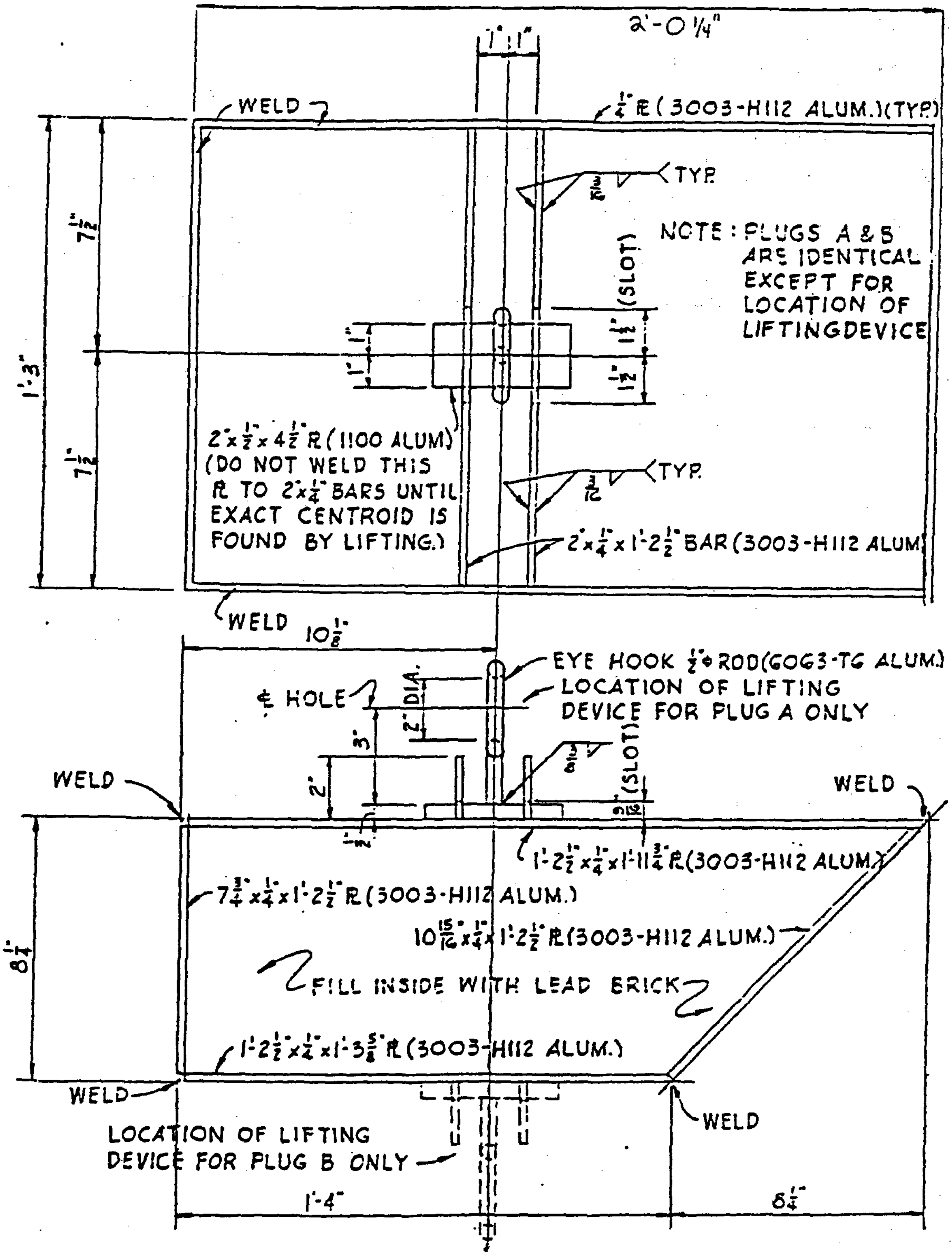


Figure 3.8: Fuel Storage Pit Plugs

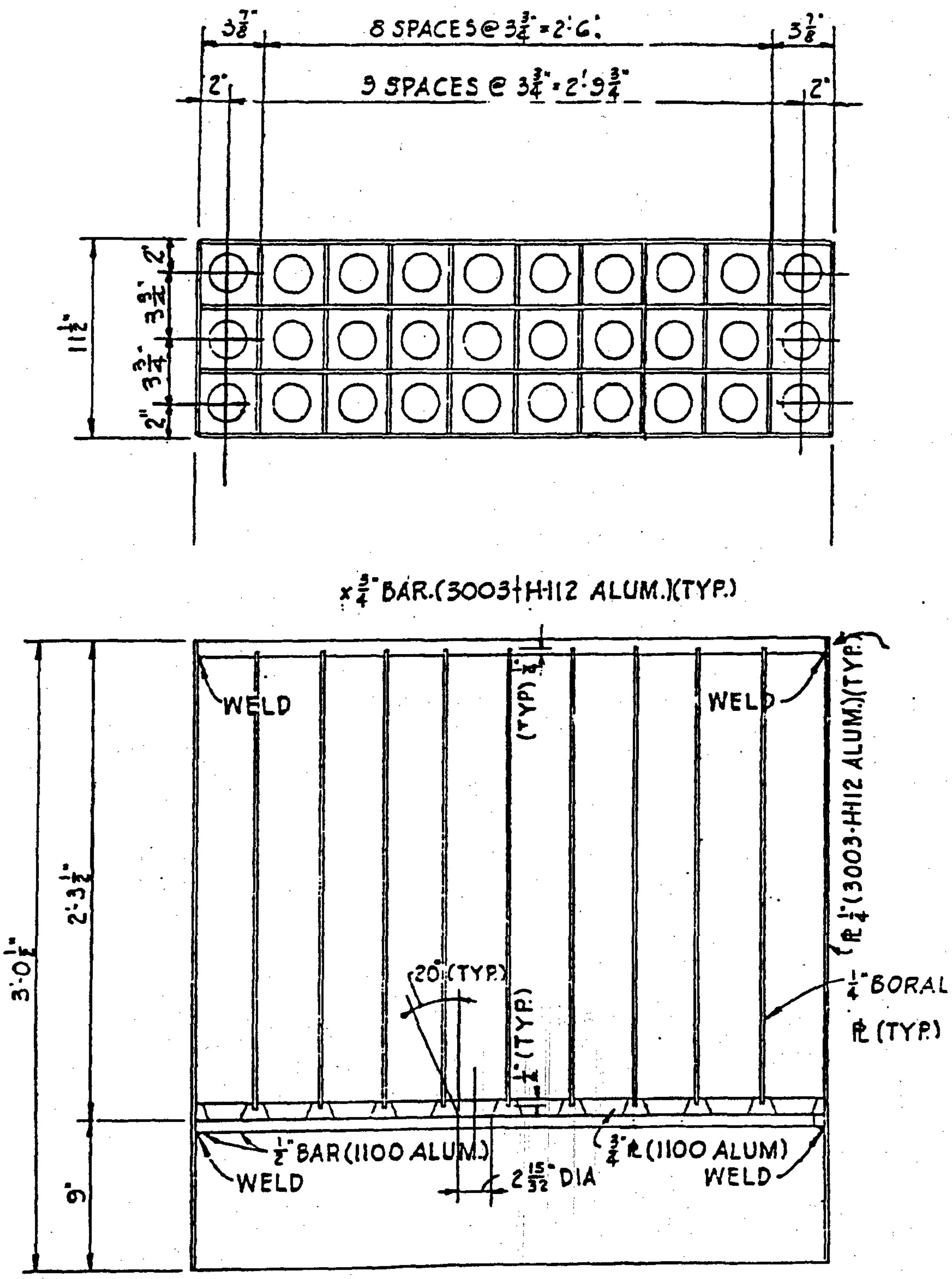


Figure 3.9: Fuel Storage Pit Rack

The reactor core is surrounded on the south and west sides by Thermal Column Extensions. These extensions are made of 0.25 inch 3003-H112 aluminum plates welded at the joints and filled with reactor-grade graphite. They mount to their respective pool walls via slotted plates that are welded to the two sides of the extensions and fit over studs in the walls. Figure 3.10 shows more detail on these extensions.

Behind both of the thermal column extensions are 0.125 inch thick stainless steel plates which lead to the Thermal Columns. The Thermal Columns are graphite filled cavities in the reactor pool walls. These will be discussed further in Section 3.8. In addition to the Thermal Columns, other penetrations of the pool walls include the beam ports and rabbit facility, all of which are located on the north side of the reactor pool. These will be discussed in more detail in Section 3.8.

Two movable, 400 Watt pool lights are suspended in the reactor pool. These lights can be positioned so that they illuminate any desired portion of the pool. The power ON/OFF switch for the lights is located on the control panel in the control room.

3.1.3.2 Bulk Shielding Facility Pool

To the south of the reactor pool lies the Bulk Shielding Facility (BSF) Pool, which is built on the same 15 inch thick Portland concrete foundation that the reactor pool is built on. The inner dimensions of the BSF pool are 8 feet by 12 feet by 15 feet 8 inches deep. It holds 11200 gallons of water. The walls of the pool are made of regular concrete with the exception of the reactor pool side, which is made of barytes concrete from ground level to elevation 13 feet 6 inches. The BSF pool is completely lined with a layer of the same fiberglass reinforced epoxy based paint used on the reactor pool.

As of the end of 1999, the BSF pool contained a ^{60}Co source, licensed separately from the OSURR, and (currently) unused equipment storage racks.

[redacted]

[redacted]

[redacted]

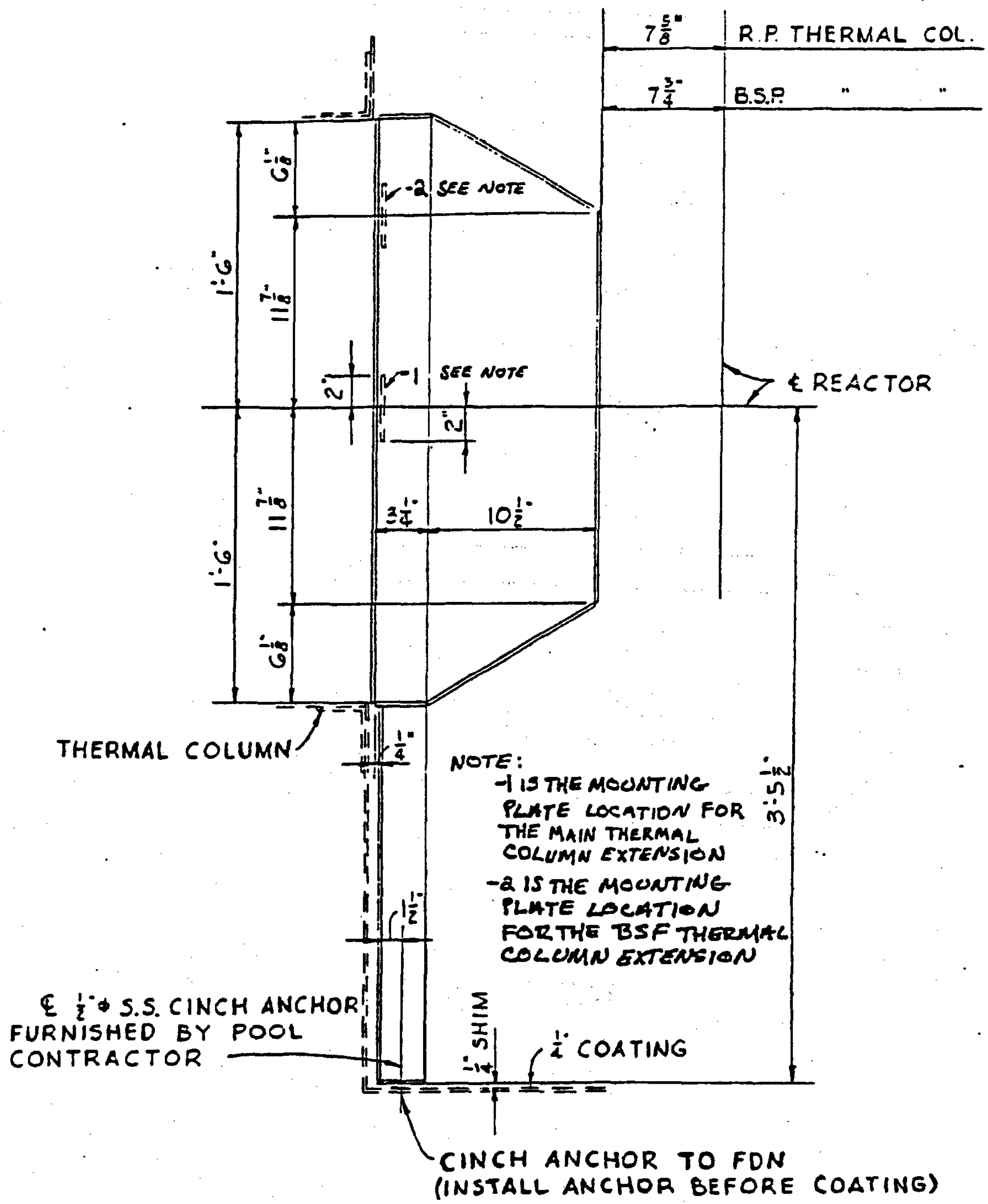


Figure 3.10: Thermal Column Extensions

[redacted]

[redacted]

[redacted]

3.1.4 In-Pool Instrumentation

3.1.4.1 Startup Channel and Drive System

A fission chamber used in the Startup Channel is located in the reactor pool above the northwest corner of the reactor core. A protective guide tube is mounted to the ionization chamber bracket, which in turn is mounted to the west wall of the reactor pool. The tube has a guide running down the inside of the tube which fits into a slot at the top of the fission chamber canister. This allows the canister to move up and down within the guide tube without rotating. The Fission Chamber Drive Motor is attached to the top of the west reactor pool wall and uses a teleflex cable to position the fission chamber. Section 3.3.14 has a more detailed discussion of the Startup Channel.

3.1.4.2 Linear Power Monitoring Channel Detector

The Linear Power Monitoring Channel Detector is a compensated ion chamber (CIC) that measures the neutron flux in the reactor over the full operating range from startup to full power. The chamber

[redacted]

compensates for gamma interaction to produce a signal that is proportional to neutron interaction. It is located above the west thermal column extension, and is supported laterally by a rack mounted to the pool wall above the extension. The chamber is supported vertically by a holder mounted to the top of the reactor pool wall. The cable pipe that extends from the detector canister to the top of the pool fits through the holder and is adjusted so that the detector sits in the desired vertical position. A clamp is attached to the pipe to keep it from sliding through the holder. Figure 3.13 shows the location of the Linear Power Monitoring Channel Detector more precisely. A more detailed discussion is found in Section 3.3.13.

3.1.4.3 Logarithmic Power Monitoring Channel Detector

The Logarithmic Power Monitoring Channel Detector is a CIC identical to the Linear Level Power Monitoring Detector described above, except that the instrumentation is such that the output is in logarithmic rather than linear form. The detector is located next to the Linear Power Monitoring Channel Detector, and is held in place in the same manner. Figure 3.13 shows the precise location. Section 3.3.12 contains a more detailed description of the Logarithmic Power Monitoring Detector.

3.1.4.4 Power Level Safety Channel Detectors

There are two uncompensated ion chambers (UICs), known as the Power Level Safety Channel Detectors, that are located next to the compensated ion chambers as shown on Figure 3.13. The uncompensated ion chambers do not compensate for gamma interactions. Therefore the signals produced by the UICs have components resulting from both gamma and neutron interaction. The UIC detectors are held in place by the same system that positions the CICs. Further discussion of the Power Level Safety Channel Detectors is located in Section 3.3.16.

3.1.4.5 Temperature Sensors and Locations

Temperature of the pool water at various locations throughout the volume of the reactor pool is measured by thermocouples. These thermocouples indicate water temperatures at the inlet (bottom) and outlet (top) of the core. Additionally, thermocouples monitor the temperature at the inlet to and outlet from the cooling system. A thermocouple is also used to monitor the temperature of the secondary coolant. Digital panel meters provide temperature information display.

3.1.4.6 Water Level Sensors and Locations

The water level sensing float switch is used with the reactor safety system. This sensor initiates a reactor trip if the water level in the reactor pool should fall to the lip of the scupper at the east end of the pool. This provides assurance that the requirements

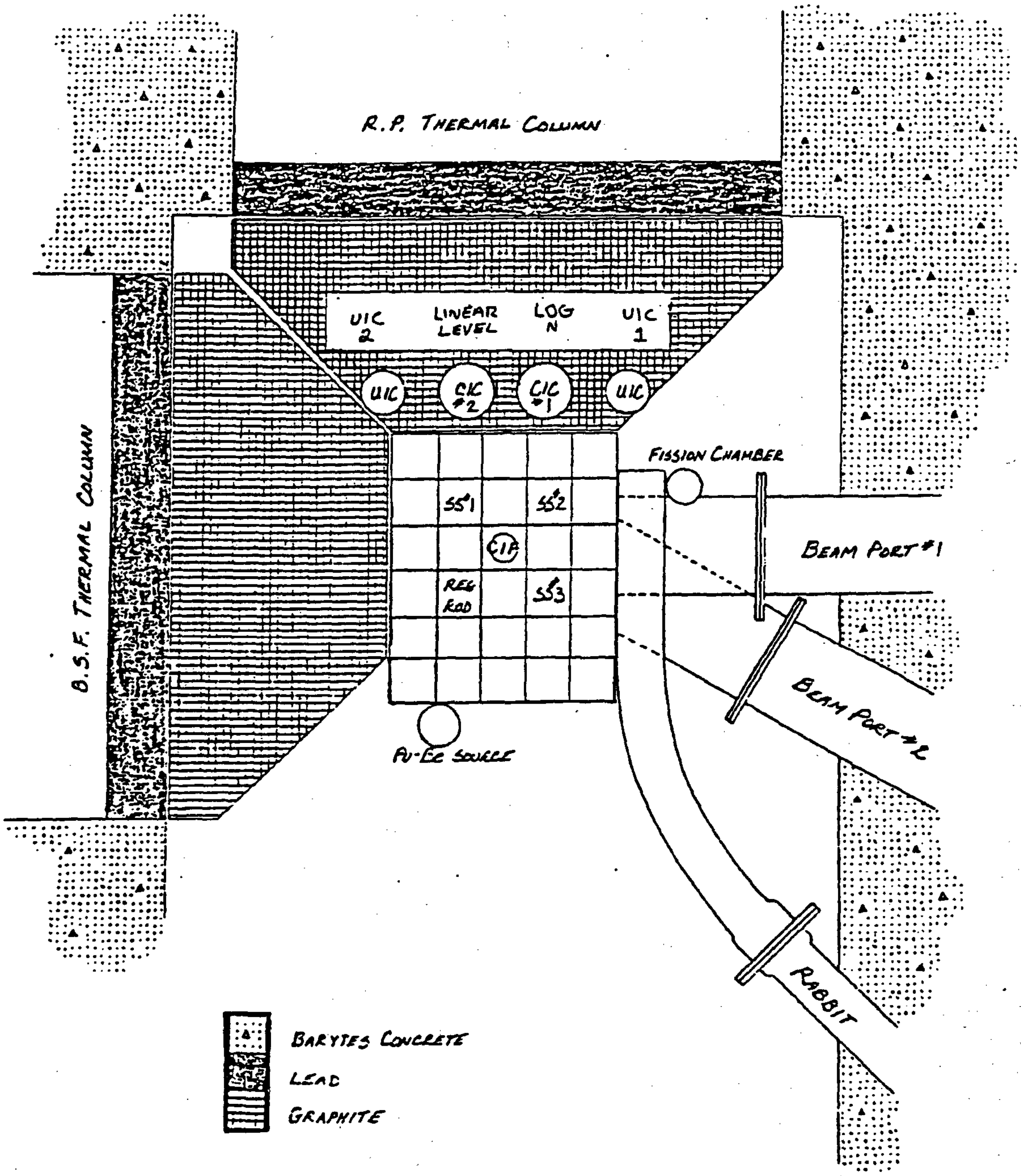


Figure 3.13: Locations of Instruments and Facilities In and Around the OSURR Core

for minimum water depth in the reactor pool, as specified in the OSURR Technical Specifications, are met.

3.2 Reactor Cooling and Water Processing Systems

3.2.1 General Features

Light water in the reactor pool serves as the primary coolant. This water passes through both a heat removal system and a water processing system. These systems are independently controlled and operated. The heat removal system serves to remove reactor-generated heat from the pool water and transfer it to a secondary heat sink (either the outside atmosphere or city water). A schematic diagram of the cooling system is shown in Figure 3.14. The water processing system removes impurities from the pool water to meet required limits on water purity. Both the cooling system and water processing system are designed to meet their required operational parameters with an additional margin of safety.

3.2.2 Cooling System

3.2.2.1 Primary Coolant Loop

Natural convective cooling is the primary means of heat removal from the OSURR core. Water enters the core at the bottom, flows upward between the fuel plates through the flow channels in the fuel elements, and is heated by the warm surfaces of the fuel plates. The heated water rises as result of buoyancy forces, and enters a plenum at the top of the core. This plenum, made of 6061-T6 aluminum plate, is essentially a box surrounding the core, with a sloped top (or cap) above the core, in the shape of a truncated pyramid. The sides of the plenum surrounding the core serve to limit bypass coolant flow between the fuel assemblies, while the plenum cap helps to confine the heated water (which also contains some ^{16}N isotope) to the area immediately above the core, from which it is withdrawn by the suction from the cooling system. A hole at the top of the plenum cap allows natural convection flow of the water in the event that the cooling system pump is turned off or fails. The size of the plenum cap is such that natural convective flow through the core is not significantly disturbed. The plenum can be manipulated from the top of the reactor pool. It is mounted in a way that helps preclude the possibility of flow blockage due to plenum misalignment, and the chance of damaging the tops of the fuel assemblies by mechanical impact.

Water is withdrawn from the plenum cap by the cooling system. Withdrawing water near the top of the core allows introduction of relatively warm water into the cooling system heat exchanger, and subsequently maximizes the temperature difference between the primary and secondary coolant. Flow rate is adjusted so that the natural convection cooling flow through the core is not significantly disturbed by the suction from the cooling system. Suction is taken

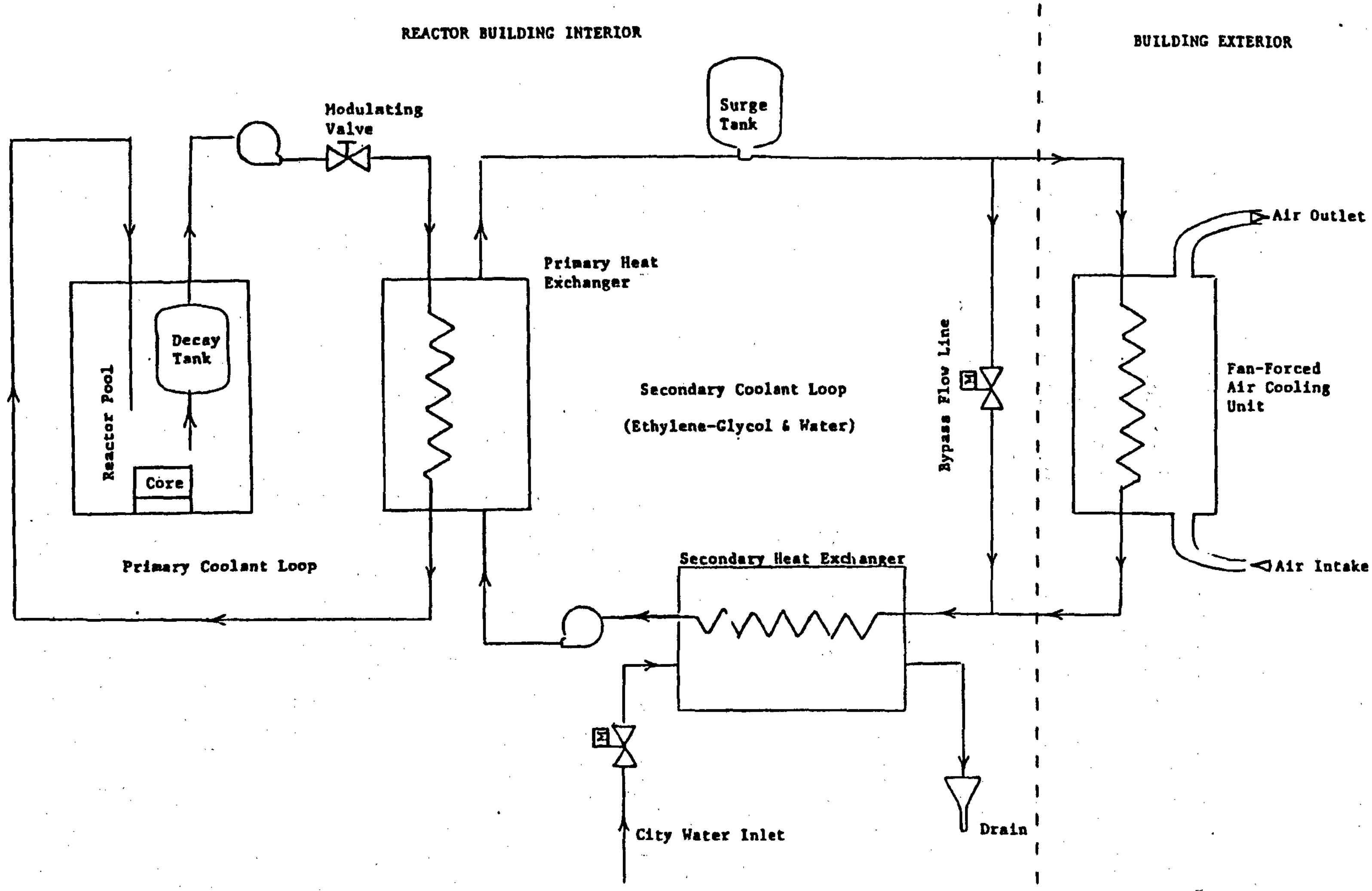


Figure 3.14: Schematic Diagram of the OSURR Cooling System

from two sides of the plenum to help equalize the water temperature distribution above the top of the core.

Primary coolant then passes into a decay, or holdup tank, vertically mounted in the reactor pool, near the southeast corner. The delay time for the water in this tank is equal to about 11 half-lives of ^{16}N, which significantly reduces ^{16}N concentration in the primary coolant as it passes into the downstream components of the primary coolant loop. The decay tank is made of type 304 stainless steel, and has an overall length of 20 feet 11 inches. It has an outer diameter of 10.75 inches, and a wall thickness of 0.109 inches. The tank has a baseplate made of type 304 stainless steel which allows it to rest on the bottom of the reactor pool. Piping runs from the bottom of the decay tank to the core plenum, and out of the top of the decay tank. All piping and decay tank surfaces exposed to the bulk pool water are insulated to prevent heat loss into the bulk pool water. Pipe runs do not penetrate the reactor pool wall, so its shielding capabilities are not compromised. Where possible, cooling system piping runs parallel with existing water processing system piping.

Outlet from the decay tank passes to the primary coolant pump. The primary coolant pump is a centrifugal pump, with the pump made of PVDF. A siphon breaker protects against accidental siphoning of the pool water through the inlet leg. The primary pump does not have to re-prime on each startup, since it is self-priming as long as the pump head is immersed, which prevents air from being introduced into the primary heat exchanger (which could reduce its effectiveness). The pump is protected from running dry in that if the water level drops low enough to expose the pump head, the reactor will have already tripped, blocking operation of the primary pump (if the reactor is tripped, the cooling system need not be operating, since the thermal capacity of the reactor pool is sufficient to remove core decay heat). The primary coolant pump is driven by a 3 HP electric motor. A modulating valve is available in the primary coolant leg to adjust flow rates. This feature allows adjustment of the primary coolant flow rate to match that required to maintain natural convective flow through the core. The primary coolant pump provides the driving head for coolant flow through the entire primary coolant loop.

Water is pumped into the primary heat exchanger from the primary coolant pump. This heat exchanger removes heat contained in the primary coolant (pool water) to a secondary cooling fluid, which is a mixture of water and ethylene glycol. The heat exchanger is a plate-type design (sometimes called a plate-and-frame type), with stainless steel surfaces where contact with the primary coolant occurs. [redacted]

The heat exchanger outlet passes through a flow sensing device to indicate coolant flow conditions. A siphon breaker is added to the [redacted] of the primary coolant loop. This siphon breaker is a small hole and prevents a siphon problem in the return leg in the event of a significant leak in the primary coolant loop. Since the volume flow rate of water through the siphon breaker tube will be small during operation, it will not significantly disturb the overall flow characteristics of the primary loop.

The cooled primary water is returned to the pool at a point above the core outlet plenum. The return flow is directed in a manner that provides dispersion of any core outlet flow that was not drawn off by the primary coolant loop suction, while not causing a disturbance of the convective flow pattern through the reactor core. This dispersion technique commonly used in TRIGA-type reactor designs, prevents buildup of ^{16}N-rich water at or near the surface of the reactor pool.

3.2.2.2 Secondary Coolant Loop

The secondary coolant loop removes heat transferred from the primary coolant to the secondary coolant. This coolant, a mixture of ethylene glycol and water, passes through two separate heat exchangers, which removes heat in the secondary coolant to an ultimate heat sink (the outside atmosphere, or, under certain environmental conditions, a supply of city water).

The secondary coolant mixture is prepared in a way that minimizes the chance of buildup of activation products in the pool water should a leak occur from the secondary loop into the primary coolant. Commercially available ethylene glycol solutions generally contain corrosion inhibitors that may contain activatable elements. Use of a coolant mixture making use of buffers such as lithium tetraborate helps reduce activation product production. Borate compounds assure that any possible secondary coolant leakage into the primary coolant results in a negative reactivity insertion.

Primary to secondary coolant heat exchange occurs in the primary heat ~~exchanger. This unit was described in the previous section. Secondary~~ coolant outlet from the primary heat exchanger then flows into a bypass control valve. This valve is an electrically actuated three-way device and provides a means to control cooling system heat removal capacity. A valve position signal is generated from a secondary coolant temperature sensor located at the inlet to the secondary cooling loop pump. The bypass control valve has 2 inch diameter inlet-outlet dimensions, and has an ANSI class VI leakage rating.

Outlet from the primary heat exchanger can be directed into either the outdoor, fan-forced cooling unit or the bypass leg. The amount of fluid flowing through each of these legs is a function of the hydraulic resistance of the fan-forced cooling unit and the position of the bypass control valve. Coolant directed through the outside cooling unit is cooled by fan-forced air flowing through the device, while that flowing through the bypass leg is not cooled. This arrangement allows for adjustment of total system heat removal capacity in the event of low outside air temperatures, and it limits the possibility of overcooling the primary coolant.

Fluid flow directed through the bypass leg is monitored by a remote-indicating flow meter. Flow indications are used to provide information on total secondary coolant flow rates through both the bypass and outside cooling unit flow paths.

The outside cooling unit is a fan-forced drycooler, sized to provide a total heat rejection capacity sufficient to remove all heat generated by 500 kilowatt operation of the core, if the outside air temperature is 78 °F or less. A total of eight electrically operated fans provide air flow through the unit to cool the secondary coolant. These fans, banked in groups of four, are remotely actuated by two separate (one for each bank) 24 volt starting relays. The unit operates on 230 VAC 3-phase power, and is mounted on an 8 foot by 14 foot concrete pad located on the east side of the reactor building. This unit does not have a "fan cycling" option. Signals to indicate fan activation and flow are provided, with readouts in the control room.

A common flow point is provided for bypass leg outlet and the outlet from the outdoor fan-forced air cooling unit. This common flow point is also a common point for inlet to a surge tank and the return from a secondary pump bypass leg.

The surge tank accommodates volumetric changes in the secondary coolant resulting from temperature changes and leakage. The tank also serves as a charging port for the secondary coolant system. It is placed at an elevated location relative to the remainder of the system, and can serve as an inlet for makeup coolant and/or flushing fluid for secondary system blowdown procedures.

The secondary coolant pump is a positive displacement device with variable flow capability. The variable flow capability provides flexibility in adjusting secondary heat removal capacity and enhances system efficiency both for daily operations and in the initial system startup and optimization. Use of a positive displacement pump in this loop reduces some of the concerns associated with pumps (such as centrifugal pumps) which are sensitive to system head and coolant viscosity. Positive displacement pumps provide an inherently stable flow rate. The secondary pump is driven by a 10 HP variable speed electric motor.

The small bypass flow leg around the secondary coolant pump has a filter which removes accumulated corrosion products. This filter is a stainless steel filter sump with replaceable filter cartridges. Flow through the filter is controlled by a small throttling valve. A flow meter is provided to monitor filter performance.

Outlet from the secondary coolant pump passes through a remotely indicating flow meter to provide an indication of total secondary system coolant flow rate. The coolant then enters a heat exchanger which uses city water as its heat sink. This unit provides additional cooling capacity for the secondary coolant in the event that the outside fan-forced air cooling unit does not provide a sufficiently low secondary coolant temperature. This heat exchanger, like the primary heat exchanger, is of the plate-and-frame design and is located on the main floor of the reactor building. Outlet from this heat exchanger passes back into the secondary side of the primary heat exchanger, which completes the secondary coolant system loop.

City water flow through the secondary side of the city water heat exchanger (sometimes called the tertiary loop) provides additional cooling capacity for the secondary coolant in the event that its temperature is not lowered sufficiently by the fan-forced outdoor air cooling unit. This unit is sized to provide necessary heat removal capacity while not exceeding local limits on discharge water temperature.

City water enters the tertiary loop and passes through a backflow preventer. This device, which operates on a reduced pressure principle, prevents introduction of non-potable water into the city water supply main. It is installed in Room 102 of the reactor building, near the service water line penetration. In the event of control valve closure, the pipe volume between the city water flow rate control valve and the backflow preventer is restricted. A pressure relief valve is installed in this length of pipe to limit pressure buildup and thereby protect the valve seats.

Flow rate of city water through the tertiary loop is regulated by a remotely controlled 2 inch, two-way electrically actuated valve. This device, similar to the bypass leg flow control valve, operates on 110 VAC power, and has an ANSI class VI leakage rating. City water flow rate is determined by secondary coolant temperature and the position of the bypass control valve.

Valve outlet flows through the city water heat exchanger on the secondary side. This heat exchanger was described previously. City water flow rate is monitored by a remotely-indicating flow meter. Discharge from the tertiary loop is to a floor drain on the main floor of the reactor building. Driving head for tertiary loop circulation is provided by the normal city water service pressure, thus eliminating the need for a circulation pump.

3.2.2.3 Cooling System Instrumentation and Control Systems

The reactor cooling system has a variety of sensing devices to indicate the condition and performance of the system. These sensors include temperature, pressure, and flow measurement devices, as well as "flow" or "no-flow", pump and fan actuation, and valve positioning indicator lights.

The following temperature indications are provided:

Primary Loop Inlet Temperature
Primary Loop Outlet Temperature
Secondary Loop Coolant Temperature
City Water Inlet Temperature
City Water Outlet Temperature
Outdoor Air Temperature

The following pressure indications are provided:

Primary Loop Inlet-Outlet Pressure Drop
Secondary Loop Pressure Drop Across Heat Exchanger

The following flow indications are provided:

Primary Loop Flow
Secondary Loop Bypass Leg Flow Rate
Secondary Pump Bypass Flow Rate
Secondary Pump Outlet Flow Rate
City Water Flow Rate In The Tertiary Loop

The following valve position indications are provided:

Bypass Leg Control Valve
City Water Flow Control Valve

The following actuation indications are provided:

Primary Coolant Pump Off-On
Secondary Coolant Pump Off-On
Air Cooling Unit Fan Bank Off-On (Two Banks)
Air Cooling Unit Fans Off-On (Eight Fans)

The information provided by the cooling system instrumentation system is sufficient to indicate system conditions and performance under normal operating conditions. Sensor readouts are available in the control room or at the location of the sensor.

The following cooling system controls are available:

Primary Pump Start/Stop
Secondary Pump Start/Stop
Pump Actuation Control (Ganged or Separate)
Air Cooling Unit Fan Bank 1 Start/Stop
Air Cooling Unit Fan Bank 2 Start/Stop
Air Cooling Unit Fan Actuation (Ganged or Separate)
Bypass Control Valve Control Mode (Auto/Manual)
Bypass Control Valve Manual Positioning (if in Manual)
City Water Control Valve Mode (Auto/Manual)
City Water Control Valve Manual Positioning (in Manual)

These control systems allow flexible operation and control of the cooling system, while not imposing an inordinate amount of control requirements on the operator.

3.2.3 Regenerable Demineralizer (Makeup Water System)

To assure availability of deionized water for the reactor pool, makeup water is taken from city water supply lines and passed through a series of filters, then a carbon filter, and finally through two mixed-bed ion exchange cartridges. A valve can be opened to allow the makeup water to be added to the reactor pool as needed.

3.2.4 Water Processing System

A water processing system provides for the water purification of reactor and bulk shielding facilities. Incorporated in the system are particulate and ion-exchange filters which serve to retard fuel cladding corrosion and capture associated corrosion products. A flow diagram of the assembly and associated components is provided in Figures 3.15 and 3.16.

Water from the reactor and bulk shielding pools recirculates through separate particulate and ion-exchange filters of the process assembly, on an approximate 5 hour per day duty cycle. Monitoring instruments, which are a part of the process assembly, are utilized to assess whether operating values remain within their specified ranges, as required by OSURR Technical Specifications. These monitors provide temperature and conductivity indications at various points throughout the system.

3.3 Reactor Instrumentation and Control

3.3.1 General Features

The reactor instrumentation and control systems provide a means to monitor the condition of the reactor and control its operation. The monitoring instrumentation provides readings on a variety of process variables important to safe and reliable reactor operation such as reactor power level, rate of power change (reactor period), water temperatures, and others. The control system allows for flexible and reliable control of reactor power and for safe reactor shutdown under a variety of conditions.

The basic design of the OSURR instrumentation is similar to that of the Bulk Shielding Reactor and Tower Shielding Reactor, both of which were located at Oak Ridge National Laboratory, and the Breazeale Reactor at the Pennsylvania State University. All of these reactors have operated safely for many years.

Similarly, the essential features of the OSURR control system resemble those of the control system used in similar US research reactors.

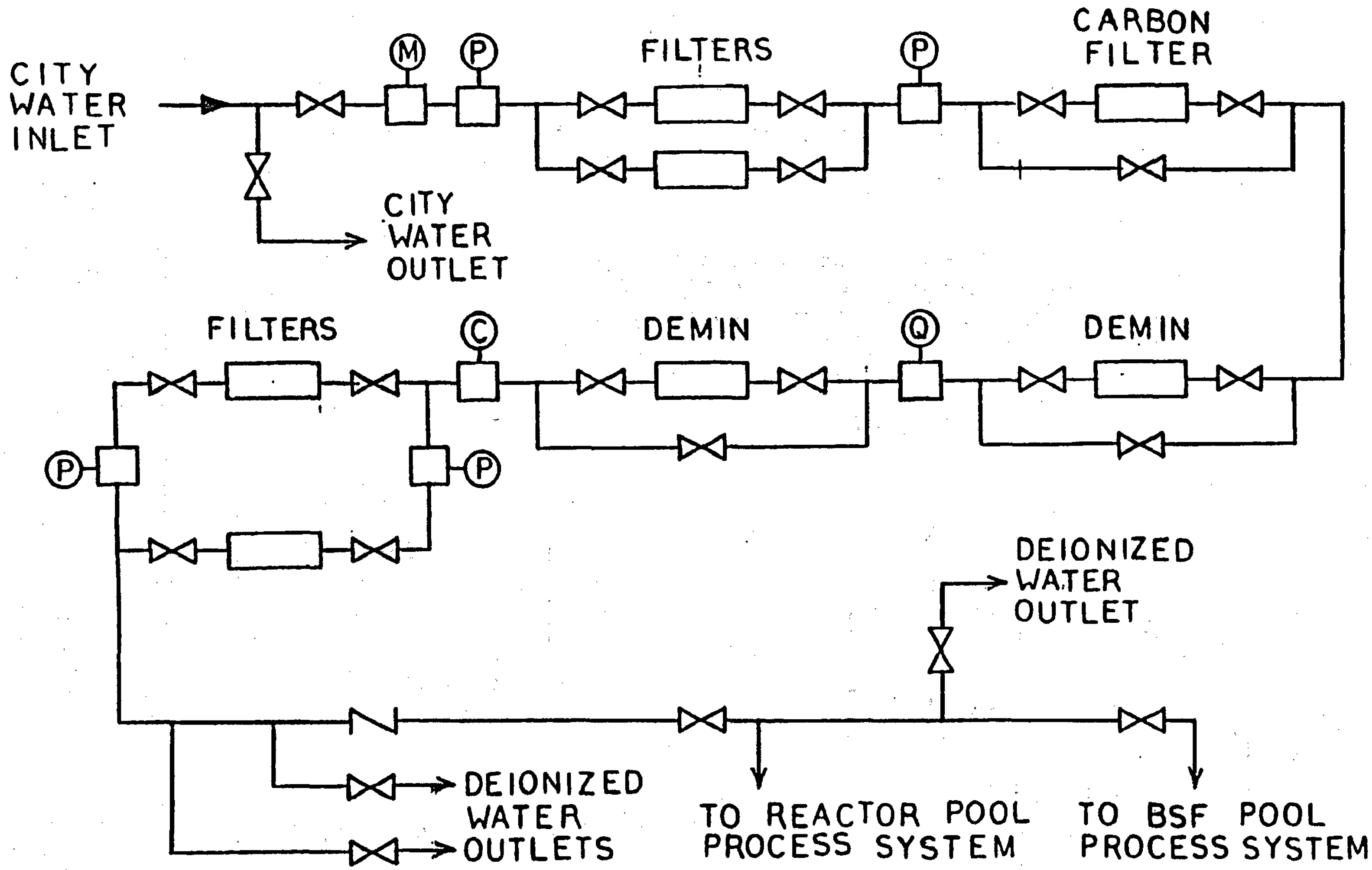


Figure 3.15: Flow Diagram of the Makeup Water System

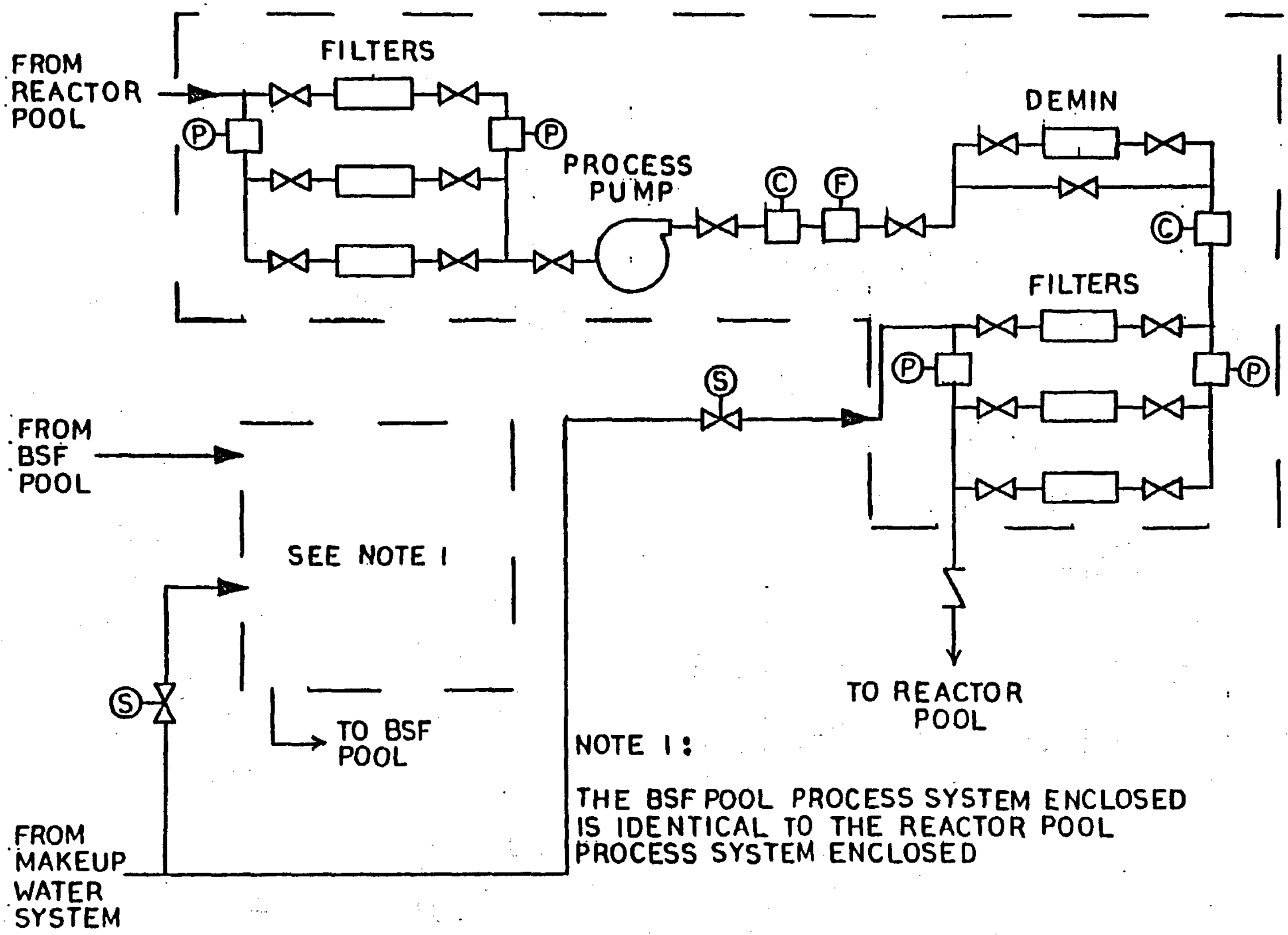


Figure 3.16: Flow Diagram of the Water Process System

Again, these reactors have operated for many years without major difficulties.

3.3.2 Control Room and Operator's Console

All instrumentation and control systems for the OSURR have readouts and control points in the control room (Room 205 in the reactor building). This arrangement allows reactor monitoring and control from a central point. The control room also houses various interlock and permit switches which control operation of various experimental facilities located throughout the reactor building. Building communications systems (intercom, telephone dialcom, and public address) allow contact between the control room and all points within the reactor building.

Figure 3.17 shows a floor plan for the control room. The control room is heated and cooled by the reactor building HVAC system. Additional cooling and ventilation is available from a window air conditioner and vent fan. Reactor instrumentation and control system cables penetrate the floor of the control room below the control console. The door to the control room can be locked to limit access to the room. Windows along the west wall of the control room permit visual observation of activities in most of the central room (reactor room) of the building. A telephone is located in the control room for offsite communications.

The control console has three main panels mounted in 50 inch-height, 19 inch-width cabinets. The cabinets are mounted to the floor of the control room. The central panel contains controls and readouts for positioning the four control rods, as well as the control rod withdrawal interlock system gang-lowering controls. Positioning readouts and controls for the startup source and startup channel fission ion chamber are also located on the central panel. To the left of the central panel is the reactor safety system annunciator panel. A series of indicator lights and pushbutton controls for safety system readout and operation are the main features of this panel. To the right of the central control panel are the readouts and controls for auxiliary reactor systems such as permits and interlocks for experimental facilities, water processing system pump actuation, and pool lighting switches. The linear power monitoring channel range and local readout is also located in this panel. A desk-type shelf is provided at the console at which the reactor operator sits during reactor operations. The three control panels at the console slope back and away from the operator to allow convenient access to the controls and comfortable viewing of the indicators.

Above the central panel of the control console is a 19 inch-width, 24 inch-height vertically mounted rack containing additional controls and readouts. Located here are remote readouts (analog meters) for the logarithmic power monitoring channel, linear power monitoring channel, and reactor period. These meters provide the operator with convenient readouts of these parameters in a location easily readable when seated at the central console.

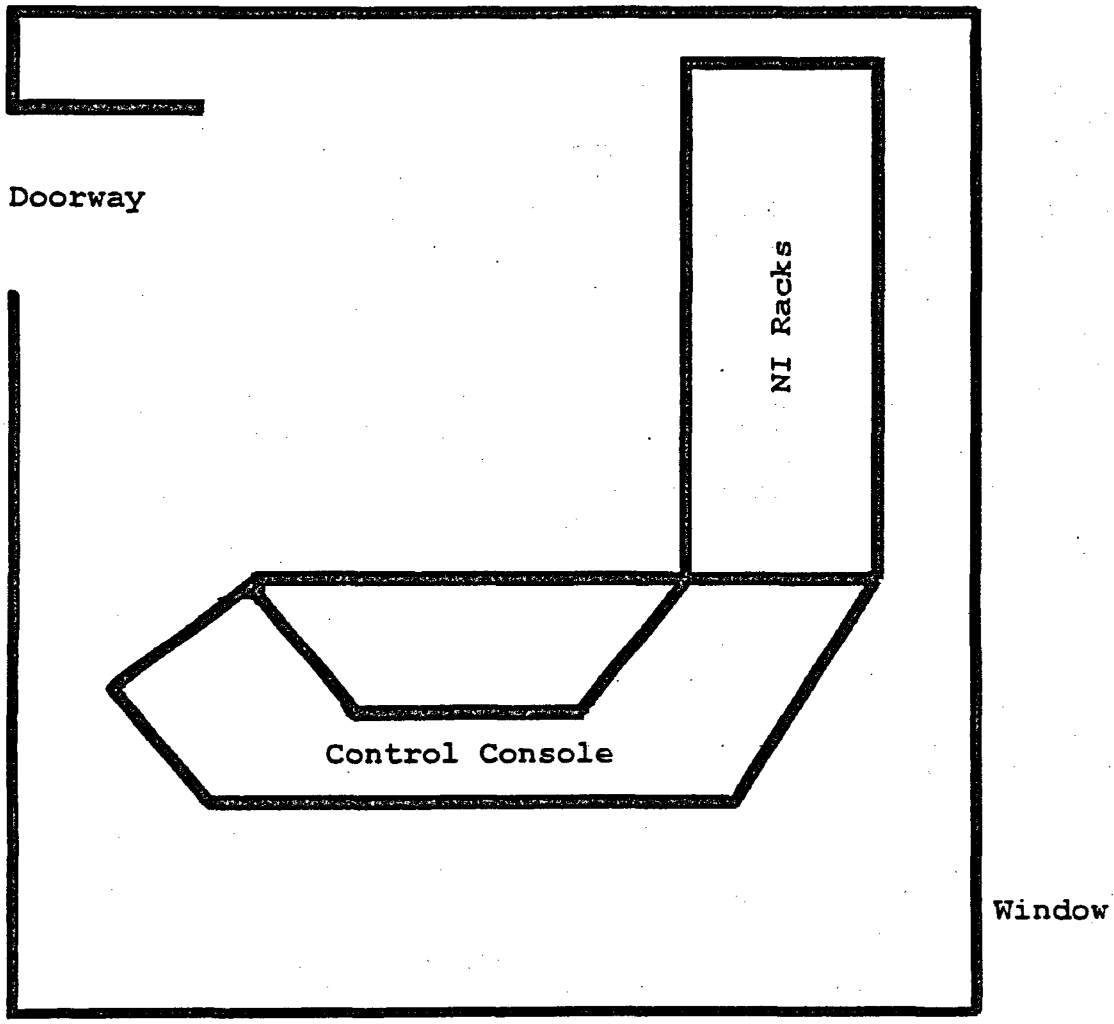


Figure 3.17: Control Room Layout

3.3.3 Nuclear Instrumentation (NI) Racks

To the left of the control console are the four NI racks housing the majority of the NI system components. These racks are 72 inches in height, and 19 inches in width. They are mounted to the floor of the control room, parallel to the east wall of the room. Access to the rear and interior of each rack is available.

Along the top of each rack are mounted the strip-chart recorders for various instrumentation systems. The remainder of each rack contains specific system components. These systems and the details of their components and operation will be discussed in following sections.

3.3.4 Control System Inputs

Control system signal inputs include detector outputs, switch contacts (e.g., from manual scram switches), control rod position indications, startup source and startup channel fission chamber position signals, and safety system inputs related to instrumentation and control system performance. Auxiliary systems also provide informational displays in the control room, including conditions of experimental facilities (e.g., beam port open/closed indications), area radiation monitor outputs, and stack and effluent monitoring system readouts.

Neutron detectors used in the power level safety system, the linear power monitoring channel, and the logarithmic power monitoring channel (from which period information is derived), provide a DC signal as input to their associated signal processing electronics. The magnitude of this DC signal and its time behavior contain information necessary for these systems. The fission ion chamber (FIC) used in the startup channel provides a pulse train (one pulse for each detected neutron event in the chamber) whose pulse output rate and time rate of change of the pulse rate are used to derive reactor power and backup period indications.

Position signals for the control rods, startup source, and startup channel fission ion chamber position are generated by servo positioning transmitters. These signals are used to align position pointers (dials) on the central panel of the control console.

3.3.5 Control System Outputs

The reactor control system output signals control a variety of systems located in the main reactor room. In general, the primary control system outputs concern the position of the four control rods. Positioning from the control room involves actuation of electric drive motors mounted at the top of the control rod drive housings. The control system also provides magnet current to hold the three shim safety rod magnet armatures (one for each rod) to the electromagnet attached to the drive rods. This magnet current can be turned off by the control (safety) system to cause a sudden reactor shutdown (scram) by allowing the shim safety rods to drop into the core.

Control system outputs also include positioning commands for the startup source and startup channel fission ion chamber location. A teleflex cable driven by an electric motor provides positioning capability for the fission chamber, while a linear electric motor (electric cylinder) positions the startup source.

Bias voltages for the various NI channel detectors are provided by the instrumentation and control system. DC power for the startup channel fission ion chamber preamplifier is also provided from the control room (the preamplifier is located in the area of the pool top).

3.3.6 Signal Paths and Cable Runs

Signals from the NI channels are passed by coaxial, or, in some cases, triaxial cables. For neutron sensors, the sensor itself is electrically isolated from ground. Sensor output is connected to the central conductor of the coaxial or triaxial cable. NI channel ground is established at the NI racks, which serve as a single-point, local ground in the control room. Low current, low bias voltage transmissions are made using typical signal cable, such as RG-58/U. Higher bias voltages are passed by appropriately rated cables such as RG-59/U. In general, signal cables use BNC-type connectors, while high voltage connections are made with MHV or SHV connectors.

Other signals such as temperature indications, servo motor position transmitter signals, switch contacts, etc., are passed by multiconductor cables or twisted wire pairs.

Many of the NI and control cables feed into a cable tray running the length of the reactor pool top, down the southeast corner of the reactor shielding pool, and overhead on the main floor of the reactor building. The cable tray enters room 103A and runs along the ceiling to a point below the control console. The cables then run through a ceiling penetration into the cabinets of the control console located in the room above. Cable routing to the NI racks is achieved through penetrations between the console cabinets and the NI racks.

Where cable trays are not used for cable traverses, wires and cables are shielded by conduit. System interconnections are made, where necessary, within appropriately labeled terminal boxes. These boxes, located at various points in the reactor building, are securely mounted to the walls of the building or pool.

3.3.7 NI and Control System Power

Building electrical service provides the initial source of electrical power for the NI and control systems. Separate constant voltage isolation transformers provide electrical service to the control console and instrument systems. The console transformer is located on the main floor, while the instrument power transformer is housed in the control room.

Within the NI and control systems, circuitry is provided to supply the necessary voltages and currents. The various subsystems in the NI and control systems require voltages ranging from indicator and control voltage levels on the order of volts up to high voltage detector biasing and B+ bias levels for vacuum tube-based amplifiers. Circuit capacity is appropriate for the system load under normal conditions. Circuit breakers and fuses are provided for safety and protection of equipment.

No auxiliary backup power system is provided. Loss of offsite power results in deactivation of the NI and control systems, which includes the electromagnet power supplies. Loss of magnet power results in a reactor shutdown from the dropped shim safety control rods. In this sense, the control system is failsafe with respect to a loss of building power event. OSURR operating procedures dictate a minimum warmup time of 1 hour for all vacuum tube-based systems.

3.3.8 Control Rod System

3.3.8.1 Position Control

Control rods are positioned in the core by electromechanical (electromagnet-motor-lead screw) devices controlled from the main console in the control room. A toggle switch (raise-neutral-lower) is provided for each of the four control rods. The toggle switch is spring loaded to return to the neutral position upon release. In addition, prior to the rod being raised (except for the regulating rod) its associated "raise" switch must be depressed. The rod drive speed is the same in the raise or lower direction.

3.3.8.2 Indicators

Several lamp indicators and two dial indicators on the lower part of the main console give control rod position in the core. A description of the indicator lamps and their function is shown in Table 3.1.

Two dial position indicators are provided on each control rod instrument module. One provides fine position readout (1 centimeter travel in 0.02 centimeter increments) and the other indicates coarse position of the rod from 0 centimeters to its upper limit of rod travel in 1 centimeter increments.

3.3.8.3 Interlock Systems

The main console has eight interlocking pushbutton switches in the switch plate. The left bank of four black pushbutton switches initiates downward motion of the control rods. Each switch can be pushed independently or in conjunction with any of the other pushbutton switches. Once depressed, the control rod specified on the

Table 3.1: Control Rod Positioning System Indicator Lights

Jam	Indicates control rod malfunction (lamp color:red).
Up	Indicates control rod lead screw at upper limit of travel (lamp color:orange).
Intermediate	Indicated control rod lead screw is near the middle range of travel (lamp color:yellow).
Down	Indicates control rod lead screw at lower limit of travel (lamp color:green).
Bottom	Indicates control rod at bottom of the core (lamp color:green). This light is not on the regulating rod.
Engage	Indicates control rod engaged to the electromagnet assembly (lamp color:white). This light is not on the regulating rod.
Servo Permit	Indicates power within 5% of servo setpoint, as set on the Linear Level Recorder (lamp color:red). This light on regulating rod only.
Servo On	Indicates servo activated and controlling regulating rod to maintain reactor power within 2% of setpoint (lamp color:blue). This light on regulating rod only.

face of the switch drives is driven into the core. A blue light adjacent to the switch informs the operator that the switch has been depressed. The right bank of four red pushbutton switches underneath the column marked "raise", controls movement of the three shim safety rods and the startup channel fission chamber. A mechanical interlock prevents more than one switch from being depressed at a time. Therefore, only one shim safety rod or the fission chamber can be moved away from the core. This prevents excessive reactivity additions to the core or loss of startup channel information.

3.3.9 Control Rod Magnet Systems

3.3.9.1 Magnet Power and Actuation

Magnet power is supplied to each shim safety control rod by a magnet current amplifier module. Current is adjusted from a potentiometer located on the front of each amplifier. The current is normally set for 150% of the minimum holding current or 10 milliamperes greater than the minimum rod pickup current (as determined during quarterly rod parameter testing).

3.3.9.2 Indicators

Magnet currents for all three shim safety rods are displayed on the magnet current indicating module. Also, lights on the control rod instrument panels give indication of shim safety rod engaged or disengaged.

3.3.9.3 Interlocks

Interlocks are provided to prevent application of magnet current to the electromagnets until certain conditions have been met. The safety system can turn off relay coil current and initiate a slow scram under a variety of conditions, as described in Section 3.6 later in this chapter. Other interlocks include the magnet power key switch, magnet current on/off switches on the magnet control modules, and the magnet power permit pushbutton switch on the scram activation and annunciator panel.

3.3.10 Startup Channel Detector Positioning

A schematic diagram of the startup channel is shown in Figure 3.18. The remainder of this section will discuss the startup channel positioning system in more detail.

3.3.10.1 Position Control

The fission chamber of the startup channel is moved away from the reactor core as power increases. This prevents saturation of the channel at high power levels. The fission chamber position is controlled from the fission chamber instrument module. An electric

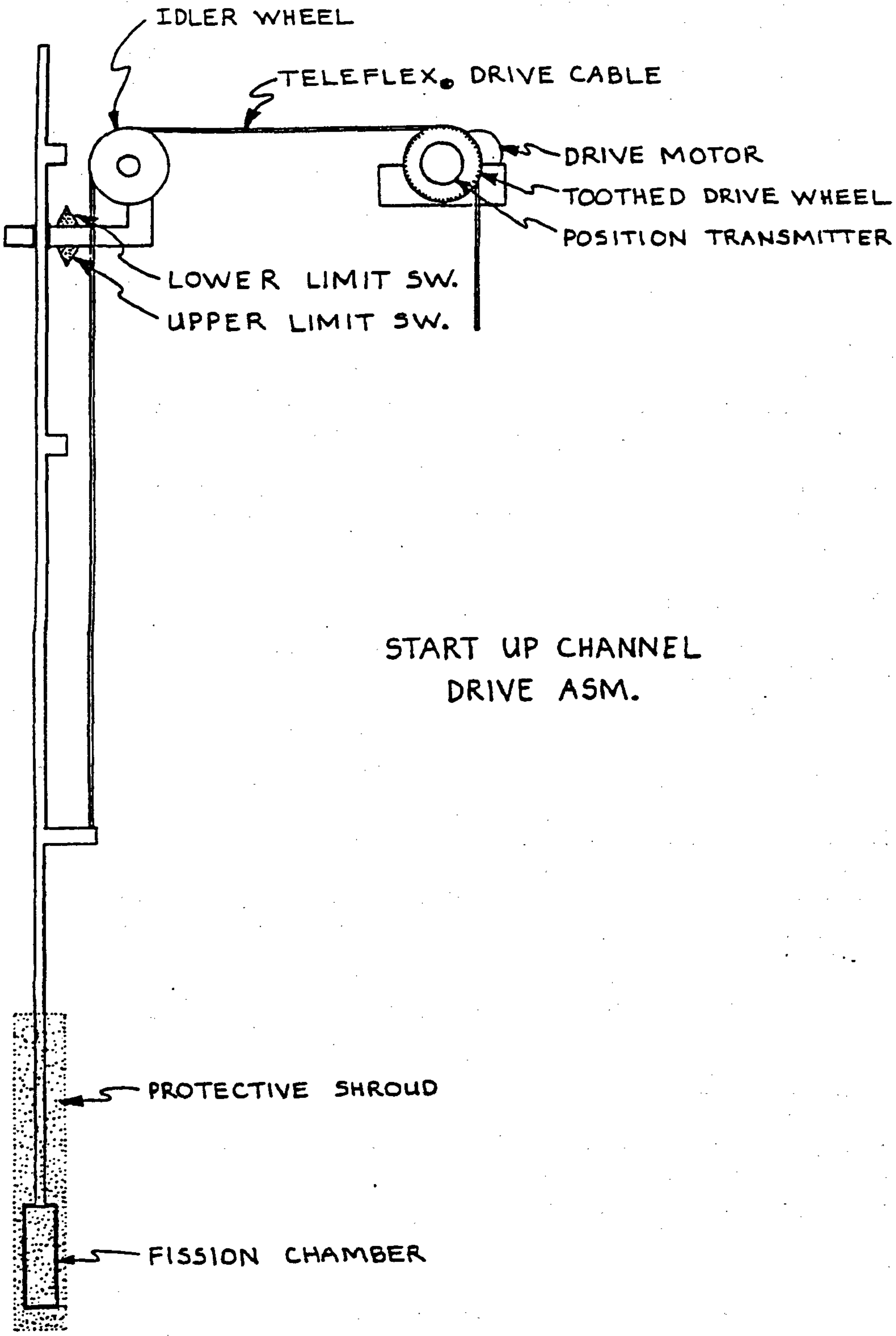


Figure 3.18: Diagram of the OSURR Startup Channel

motor mounted at the top of the reactor pool drives a gear which is linked to a teleflex cable. A position transmitter indicates chamber position.

3.3.10.2 Indicators

Indicators are provided on the instrument panel to indicate the position of the fission chamber relative to a position above the core. A dial type indicator, with 1 centimeter increments, gives overall position (0-160 centimeters), and two lights are provided to give indication of reaching the lower limit (green) and upper limit (125 centimeters - blue) of travel.

3.3.10.3 Interlock

An interlock is provided on the master switch plate to prevent outward motion of the fission chamber while any shim safety rod is being withdrawn, or when any of the other rods are being lowered. This prevents adding reactivity while at the same time reducing power level the detector would see. The purpose is to provide reliable information concerning power indication while pulling rods in the core. No interlock prevents moving the fission chamber inward while simultaneously withdrawing any of the control rods.

3.3.11 Startup Source Positioning

3.3.11.1 Position Control

The startup source is positioned adjacent to the reactor prior to startup to provide an initial source of neutrons. As the reactor creates its own neutrons, the source is driven back into a protective cadmium shield to prevent burnout. The source is positioned with a toggle switch on the source drive instrument panel. No interlocks are provided.

3.3.11.2 Indicators

Indicators are provided on the instrument panel to indicate position of the source relative to its cadmium storage cask. Two lights are provided to give indication of lower limit (in the cask) and upper limit (95 centimeters) of travel.

3.3.12 Logarithmic Power Monitoring Channel

The Logarithmic Power Monitoring Channel (Log-N) provides continuous indication of reactor power covering 8 decades from 10^{-12} to 10^{-04} amperes. The Log-N uses a compensated ion chamber located above the graphite reflector at the west end of the reactor pool.

The Log-N channel consists of a detector, high voltage and compensating voltage power supply, logarithmic and period amplifier, local and remote meters, and a recorder.

The detector is a compensated ion chamber manufactured by Reuter-Stokes, model number RSN-15A. Operating voltages are provided by a power supply module. Positive high voltage and negative compensating voltage are displayed on a digital panel meter located on the front of the power supply module. Compensating voltage is adjusted by a 10-turn potentiometer located next to the meter on the front panel.

The Log-N amplifier, a Keithley model 25012A, receives its input signal from the detector and provides outputs to local and remote current meters, period safety amplifier, and the Log-N recorder. Additionally, an internal signal is provided for the period amplifier portion of the Log-N amplifier (see Section 3.3.15).

Local and remote analog panel meters are provided for monitoring of the reactor power level. The remote meter is located on the center of the main console with various other meters so that a centralized meter location is available for monitoring reactor power. A logarithmic recorder gives a continual, permanent record of the power history of the reactor. Its range is 8 decades, covering power levels from startup to the full operating power of the OSURR.

Although no reactor trip signals are generated directly by the power level signal from the Log-N channel, a continuity circuit is provided such that the front panel switches must be in their "normal" positions in order for the reactor to be operated. As noted above, however, one output of the Log-N amplifier is used as an input to the period safety amplifier system, which can generate a reactor trip. Also, the internal connection to the period amplifier section of the Log-N amplifier provides a signal from which the reactor period signal is derived, and from which an independent reactor trip function may be generated. The Log-N channel stripchart recorder also has a trip function associated with its "on-off" switch, to provide a trip if the device is turned off.

3.3.13 Linear Power Monitoring Channel

The linear level power monitoring channel provides indication of reactor power in discrete decades of current/power covering a range from 10^{-11} amperes through 10^{-03} amperes full-scale deflection. This channel uses a compensated ion chamber located adjacent to the reactor, above the graphite reflector at the west end of the pool (see Figure 13).

The linear level power monitoring channel consists of a detector, power supply, amplifier, local and remote meters, and a recorder. The detector is an RSN-15A compensated ion chamber manufactured by Reuter-Stokes. Positive high voltage and negative compensating voltage are provided by a power supply module. Compensating voltage is adjusted by a potentiometer located on the front panel of the power supply. Both voltages are monitored locally by 3½ digit panel meters.

The linear current amplifier receives an input signal from the RSN-15A detector and amplifies it according to the feedback resistance selected by the selector switch located on the right panel of the main console. The amplifier provides outputs to local and remote meters, as well as a chart recorder. All meters and the recorder read 0-150%, with 100% corresponding to full range current of the decade selected. Also, the recorder has an adjustable setpoint for the Servo System.

The linear level system provides three "slow" scram signals. The positive high voltage power supply has a low voltage trip, the recorder has a trip at 120% indicated power (according to the range selected), and the recorder also has a trip for the "on-off" switch in the "off" position.

3.3.14 Startup Channel

The startup channel provides information about reactor power at initial and low levels of power operation. It also provides information at high power levels, but the detector must be repositioned away from the reactor core to prevent saturation of the channel sensor.

The startup channel uses a fission chamber, Reuter-Stokes model number RSN-10A, as the detector. This detector uses a ^{235}U coated tube to capture neutrons. The detector is positioned above the reactor core on the NNW side. The detector travels through a stroke of 154 centimeters.

The detector position is controlled by a toggle switch on the fission chamber instrument control module in the main console. An interlock is provided so that the shim safety rods cannot be withdrawn while the fission chamber is moved out, and vice-versa.

The positioning system consists of a synchronous motor connected to a position indicator transmitter and a toothed wheel (see Figure 18). By moving the "in - none - out" toggle switch to the "in" or "out" position, the motor rotates causing the toothed wheel to turn in a clockwise or counterclockwise direction. A teleflex cable attached to the top of the detector housing rides on this wheel and moves the fission chamber up or down (i.e., away from or towards the reactor core).

Two lamps are provided to indicate upper and lower limit, blue and green, respectively, of travel of the fission chamber. Micro switches are placed at the top and bottom of the support stanchion and are activated when the chamber is positioned at the top or bottom of its travel.

Additional startup channel instrumentation consists of a high voltage power supply, linear amplifier, discriminator, digital timer/counter, linear-log ratemeter, and a recorder.

The high voltage power supply provides a positive voltage to the fission chamber. An analog display on the front panel of the power supply indicates the magnitude of detector voltage (0-2.0 KV).

Two slow scrams are associated with the startup channel. The stripchart recorder "on-off" switch must be in the "on" position, and the recorder must indicate a count rate of greater than 2 counts per second. The latter requirement insures a detectable, measurable population of neutrons is present in and near the reactor core, which provides a reliable startup.

Periodically, it is necessary (e.g., during core loading experiments) to withdraw control rods when counts (as indicated on the SU recorder) are < 2 counts per second. A bypass key, locked in the key locker, is provided to allow the slow scram alarm protection function to be bypassed. However, the alarm lamp annunciator still lights to indicate a scram condition has occurred. Additionally, a bypass lamp, labeled "source", is illuminated to give the operator indication that the SU system is in a bypassed condition.

3.3.15 Period Monitoring Channel

The signal for the period monitoring channel recorder comes from the period amplifier section of the Log-N/Period amplifier (see Section 3.3.12). The output of the logarithmic amplifier is applied to a differentiating circuit and amplified.

The system is composed of the Keithley model 25012A Log-N/Period amplifier, Leeds and Northrup "Speedomax" type G recorder, and remote indicator. Analog meters are provided for local and remote indications of reactor period. A remote meter is located on the main console instrument panel. Both local and remote meters are analog devices covering a range of period from -30 seconds to +3 seconds.

The recorder provides period indications and a slow scram function. The recorder is set up so that at a period of 5 seconds causes a slow scram reactor trip. This scram function prevents generating reactor power transients which could be difficult to conveniently control.

3.3.16 Power Level Safety Channels

Two uncompensated ionization chambers (UICs), Reuter-Stokes model number RSN-36A, are positioned at the west end of the reactor pool above the graphite reflector. These detectors, along with a high voltage power supply, amplifier, and output meter, comprise the power level safety channels. The channels are independent and redundant. A signal from a UIC is applied to an amplifier which provides an output voltage proportional to input currents between 0 and $2x10^{-05}$ amperes, corresponding to 0 to 200% reactor power. A 3½ digit panel meter, Datel DM-500, is provided to monitor reactor power. Limitations of the data display device restrict actual reactor power readout on the local meter to 0.001 to 200 percent reactor power.

Outputs are provided to the slow scram and fast scram reactor protection systems when alarm conditions occur. Additionally, LEDs give local status to all alarm conditions of the Safety amplifier.

A system continuity circuit is installed in each level safety channel to provide a signal to the Slow Scram system on loss of the +12 volt supply voltage, or when the instrument is removed from its NIM bin. A condition of low output voltage from the high voltage power supply (high voltage to bias the detector) trips a slow scram relay causing a reactor scram. Also, the ground connector to the detector is monitored to detect when the UIC is not connected properly or removed from the system, and trips a relay in the slow scram system.

A "Fast Scram" occurs when reactor power reaches 150% of full power. Section 3.6.2 covers the scram mechanisms in more detail.

3.3.17 Period Safety Channel

The input signal to the Period Safety system amplifier comes from the Log-N amplifier, model 25012A, "Log Analog Out" voltage jack. The signal is differentiated in the period safety amplifier and the outputs sent to a comparator amplifier and external meters and recorders.

The period is displayed locally by an LED display indicating reactor periods from ∞ to 1 second. Also, LEDs are provided to give the status of alarm conditions occurring within the period safety amplifier.

The period safety channel has a system continuity circuit installed that trips a slow scram relay upon loss of the +12 volt supply voltage, or removal of the amplifier from the NIM bin.

A fast scram output occurs when reactor period reaches 1 second. This output is applied to the gate of the current-controlling field effect transistor (FET) of the Magnet Current amplifier (see Section 3.6.2, and the basic schematic diagram of the slow and fast scram systems).

3.4 Cooling System Controls

The control panel for the heat removal system is located above the control console, above the panel containing the safety system annunciators. Controls and indicators are mounted in a vertical rack of 19 inches width. Indicators can be viewed from a seated or standing position.

Section 3.2.2.3 discussed the various monitoring instruments and control available for the cooling system. System operation is controlled from the cooling system panel rack. Various pushbutton and/or toggle switches allow actuation of the various system components such as pumps, valves, and fans. The system is designed so that various subsystems (e.g., primary water circulation loop) can be controlled independently or in conjunction with associated systems.

Cooling system indicators include displays of temperature pressure, flow rates, valve positions, component actuation and operation, and yes/no indications of flow. Numerical data is displayed by either digital or analog meters, while simple on/off indicators are provided by LED displays.

The cooling system instrumentation and controls are interfaced with the reactor safety system. Cooling system operation must be effective under specified reactor power conditions. Primary and secondary cooling loop pumps must be on and flow indications received from both primary and secondary loops If the reactor is operated above a specified power level to avoid a scram. Additionally, loss of cooling system capacity (as reflected in elevated outlet water temperatures) results in a reactor trip condition. Sudden changes in outlet temperature occurring over short times results in a warning annunciator indication to alert the operator to anticipate a reactivity effect.

3.5 Auxiliary Controls

Operation of the circulating pumps for the water processing system is controlled from the control room. Local pump control is available at the pumps themselves through actuation of pump motor contactors and a timer manual override switch. Manual pump control allows pump startup at any time, while automatic control allows the timer to activate the pumps. All valve positions are manually set at the valve location, with the exception of the makeup water inlet valve, which is a switch-activated solenoid valve.

Ultimate control of certain experimental facilities is maintained in the control room in the form of permit switches (interlocks). Operation of systems with control room permit interlocks is not possible until the system interlock switch is activated. Permit switches are provided for rabbit system operation and use of the removable reactivity oscillator.

Those experimental facilities not directly controlled from the control room generally have remotely-reading indications of their condition. For example, opening the gamma shield shutters in the beam ports, or moving the main thermal column shield door away from its fully closed position causes a lamp to be lit on the auxiliary systems control panel.

3.6 Safety System (Reactor Protection System)

3.6.1 General Features

The safety system protects the reactor in the event of an occurrence that could result in operation outside the allowable power ranges. The Reactor Protection System (RPS) shuts the reactor down by allowing the shim safety rods to be inserted into the reactor core under the force of gravity. The system is designed to release the shim safety rods within 50 milliseconds of a scram signal input to the fast or slow scram systems. Additionally, shim safety rods are designed to be inserted into the reactor core within 600 milliseconds upon receipt of this signal.

Alarms are also provided to alert the operator to the system or systems that have sent an output scram signal to the fast or slow scram systems. In addition, provisions are provided such that expected alarms can be pre-acknowledged and the "Low Source" alarm can be bypassed.

3.6.2 Types of Scrams

3.6.2.1 Slow Scram (Relay Scram)

The slow scram system initiates a reactor scram by turning off the current to the electromagnets holding the shim safety control rods to the rod drive assemblies. A set of relay contacts are opened to turn off the power supply generating the current through the electromagnets, as shown in Figure 3.19. With the electromagnet current turned off, the control rods drop into the reactor core.

3.6.2.2 Fast Scram (Electronic Scram)

The fast scram system turns off shim safety rod electromagnet current by biasing a current controlling element in the magnet control modules which reduces the current flow through the electromagnets. When current is reduced below the minimum holding current, the control rod drops into the reactor core.

The fast scram system is somewhat quicker than the slow scram system since there is no lag time associated with the opening of a relay. Also, both systems are fail-safe (initiate a reactor scram) with respect to loss of voltage to the system or removal of a safety device from the instrumentation system.

3.6.3 Scram Functions and Setpoints

The safety system functions and their associated trip setpoints are shown in Table 3.2.

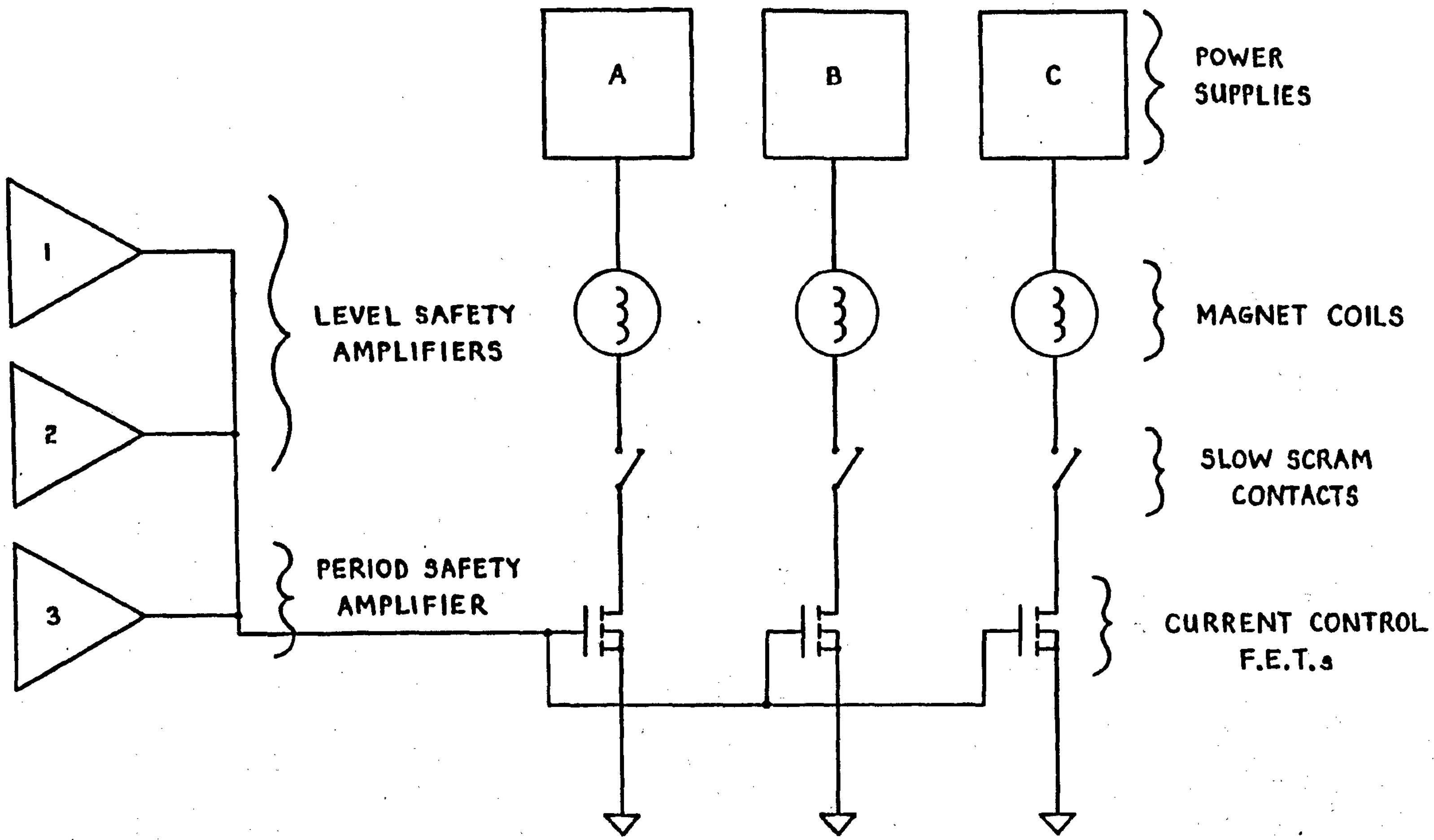


Figure 3.19: Simplified Diagram of the OSURR Reactor Trip System

Table 3.2: Scram and Alarm Functions of the OSURR Safety System

System Conditions	Signal Origin	Setpoint Ranges	System Action	Bypass Provisions
Low Count Rate	Startup Channel Fission Chamber	$\geq$ 2 counts/sec	slow scram & alarm	bypass only if $K_{eff} \leq 0.9$
Fast Reactor Period	Log-N/Period CIC	$\geq$ .5 seconds	slow scram & alarm (recorder switch)	None
Fast Reactor Period	Log-N/Period CIC	$\geq$ 1 second	fast scram & alarm (sigma bus amplifiers)	None
Excessive Servo Demand	Linear Level CIC	$\leq \pm$ 7% servo demand	inhibit servo operation	None
Reactor Overpower	UIC in Level Safety Channel 1, 2	$\leq$ 120% full power	fast scram & alarm	None
Reactor Power Above Setpoint	Linear Level CIC	$\geq$ 10% full scale of linear level recorder	inhibit servo operation	None
Reactor Overpower	Linear Level CIC	$\leq$ 80% full scale on linear level recorder	slow scram & alarm (recorder switch)	None
Manual Scram Switch at console, pool top, BSF area, thermal column, or beam ports	Switch Contacts	Manual Actuation	slow scram & alarm	None
Magnet Power Key Not In The "On" Position	Switch Contacts	Manual Actuation	slow scram	None

Table 3.2: Scram and Alarm Functions of the OSURR Safety System (continued)

System Conditions	Signal Origin	Setpoint Ranges	System Action	Bypass Provisions
Startup Channel LCRM Not In "Use" Function	Switch Contacts	Manual Actuation	slow scram & alarm	None
Log-N/Period Amp. operate switch not in "Operate" Position	Switch Contacts	Manual Actuation	slow scram & alarm	None
Indicated Dose Rate Above Setpoint	ARM System Detectors	$\leq$ 10 mr/hr	local & remote alarm	Temporary Replacement By Equivalent Portable Monitor
High Core Inlet Temperature	Temperature Monitor Below The Core	$\leq$ 40 deg. C.	slow scram & alarm	None
Reactor Power Above Setpoint With No Primary Coolant Flow	Log-N/Period CIC and Flow/No-Flow Monitor	$\leq$ 100 KW	slow scram & alarm	None
Core Inlet Temp. Below Setpoint	Temperature Monitor Below The Core	$\geq$ 20 deg. C.	alarm	None
Safety System Continuity	Safety System Continuity Bus	Manual Actuation	slow scram & alarm	None
High Voltage Failure On CICs or UICs	Monitor Circuit In Bias Supplies	Variable	slow scram & alarm	None

3.6.4 Scram Bypass

Provisions are made to bypass the reactor trip initiated by low counts (less than 2 counts per second) on the source range channel. This allows shim safety control rod withdrawal during core loading operations. The bypass is controlled by a [redacted] of the slow scram panel. A lamp provides indication that this safety feature has been overridden. Activation of this bypass also defeats the scram function associated with turning off the stripchart recorder in the startup channel monitoring system.

The procedure for implementing the bypass are contained in the OSURR operating procedures. This procedure is used only when the effective neutron multiplication factor of the core is not greater than 0.9, source counts are less than 2 counts per second, and shim safety control rod withdrawal is necessary.

3.6.5 Alarm and Annunciator System

Closely integrated with the slow scram system are the alarm and annunciator system. The alarm system provides a visual alarm prior to slow scram or fast scram actuation. The annunciator system alerts the operator to an alarm condition, slow scram occurrence, or a fast scram trip by emitting an audible signal.

An annunciator lamp switch flashes to indicate the presence of a slow or fast scram condition (a separate switch indicates an alarm condition), and is cleared by depressing the switch. Additionally, a time delay relay has been provided such that in the event the alarm or scram condition is not acknowledged within 15 seconds, the building evacuation alarm will sound.

3.6.6 Building Evacuation System (Ventilation Control)

The building evacuation system is activated by two switches. The first switch, located on the slow scram console in the control room, causes a building evacuate klaxon horn to sound. At this signal, all personnel present in the reactor building leave the premises. This action removes people to a safe, alternate location away from the building so that appropriate equipment and personnel can have unimpeded access to the area. The other switch, located underneath the wall-mounted telephone in the control room, turns off all ventilation fans exhausting to the outside of the building. Effluent discharge to the outside environment is thus prevented.

As a precaution, the reactor operator will scram the reactor prior to activating either the building evacuation alarm or ventilation system cutoff, and will announce over the building PA system what is about to occur. Applicable emergency procedures as outlined in the NRL Operating Procedures Handbook are followed to insure safety of all personnel and protection of property.

3.7 Area Radiation Monitors

3.7.1 General Features and Purpose

The Area Radiation Monitors (ARM) provide information on radiation levels around the reactor building. Their purpose is to alert the reactor operator and people in the surrounding areas if radiation dose rates above a specified setpoint exist. These conditions, if not an anticipated part of an experiment, can then be corrected to minimize possible radiation doses to personnel.

3.7.2 Detectors and Location

The ARM system uses four Geiger-Mueller detectors. They are located above the reactor pool, opposite the thermal column and beam ports, near the primary coolant loop heat exchanger, and next to the water processing system.

3.7.3 Readouts, Indicators, and Alarms

Duplicate instrumentation is provided at both local (near the detector position) and remote (control room) locations. Analog meters indicate radiation levels from 0.1 to 1,000 millirem per hour at the local position, while digital ratemeters provide control room readouts. Additionally, a test button is provided, simulating high radiation levels, to test the instrumentation. Amber and red colored lamps are also provided to indicate high background and alarm conditions respectively. An alarm annunciator is located in each of the local ARM meters, and provides a local warning to personnel in the area.

The setpoints for the background and alarm levels are adjusted on the instrument modules (remote) located in the control room. No provisions are made for silencing an alarm except adjusting the setpoint at a higher level (nominally set at 10-30 millirem per hour).

3.8 Experimental Facilities

3.8.1 Central Irradiation Facility

The Central Irradiation Facility (CIF) consists of a 1.5 inch outer-diameter 6061-T6 aluminum tube that extends from the top of the reactor pool down into the central array position of the core. The tube is 270 inches long, with a 0.06-inch wall thickness. An aluminum plug is welded to the bottom. Additionally, a lockable cast aluminum cap is attached to the top. The core end of the CIF tube is inserted into a liner which has an end box attached to it and fits into the grid plate in a similar fashion to the fuel elements. The liner most commonly used is a water filled, 3 inch x 3 inch x 27.30 inch box, made of 0.125 inch aluminum plate, welded at the edges. The end box is welded to the bottom ends of the aluminum plates. The CIF tube is laterally supported by a square collar that fits into the top of the liner. Vertical support is accomplished by a clamp that is attached at the I-beams supporting the control rod drive structure at the top of the reactor pool.

3.8.2 Beam Ports

Two portholes, known as beam ports, penetrate the north wall of the reactor pool. A drawing of one of the two beam ports is shown in Figure 3.20. Each beam port consists of a shutter assembly, beam port tube, and beam port cap. The shutter mechanisms are housed in 0.1875 inch carbon steel boxes. Any voids are filled with barytes concrete. These shutter assembly housings fit into liners in the pool wall. The outside ends of the shutter assemblies are 7.125 inch inner-diameter, type 304 stainless steel tubes, with 0.1875 inch thick walls. These tubes have beveled edges that lead to the actual shutter mechanisms, which decrease the inner diameters of the beam ports to 6.125 inches. The shutter mechanisms are lead-filled cylinders, positioned horizontally across the path of the beam port with their axes perpendicular to the path. These cylinders have 6.125 inch inner-diameter through holes, which line up with the beam port tube in the open position. With the cylinder rotated 90 degrees and the opening is vertically oriented, the shutter is closed. A 6.125 inch inner-diameter, type 3003 H14 aluminum alloy tube with welded flange is screwed to the shutter assembly liner. A 0.062 inch sheet lead gasket is located between the liner and flange. The tubes penetrate the inner pool wall, and are welded to a 0.25 inch aluminum support plate which is embedded in the wall. A 6061-T6 aluminum alloy cap with flange is bolted to each tube. Between the caps and tubes are 0.062 inch natural rubber gaskets.

3.8.3 Rabbit Facility

The rabbit facility makes use of a vacuum system for insertion and extraction of samples to be irradiated. The vacuum system is used instead of a pressure system because the evacuated air used for operation can be expelled away from the place of operation rather than at the place of operation. This is desirable since a small amount of ^{41}Ar is produced in the rabbit facility.

A pipe line diagram is provided in Figure 3.22. When sending a sample to the reactor core end of the facility the vacuum causes air to flow into the system through an air filter into the manifold. The manifold has a series of four 2-position valves, operated by a solenoid. Between each pair of valves is an airway whose flow is controlled by the corresponding pair of valves on either side of the airway. The valves are opened and closed so as to allow air to flow down the rabbit tube toward the reactor core. The rabbit tube is a 2.25 inch diameter 2S aluminum tube, perforated at the end. A 3 inch diameter 2S aluminum tube sheathes the rabbit tube. Thus, the air directed down the rabbit tube flows through the perforations and into the sheathing tube. The sheathing tube is completely sealed around the rabbit tube except for an outlet tube running back to the manifold, where the valves on either side of the tube force the air into the tube that

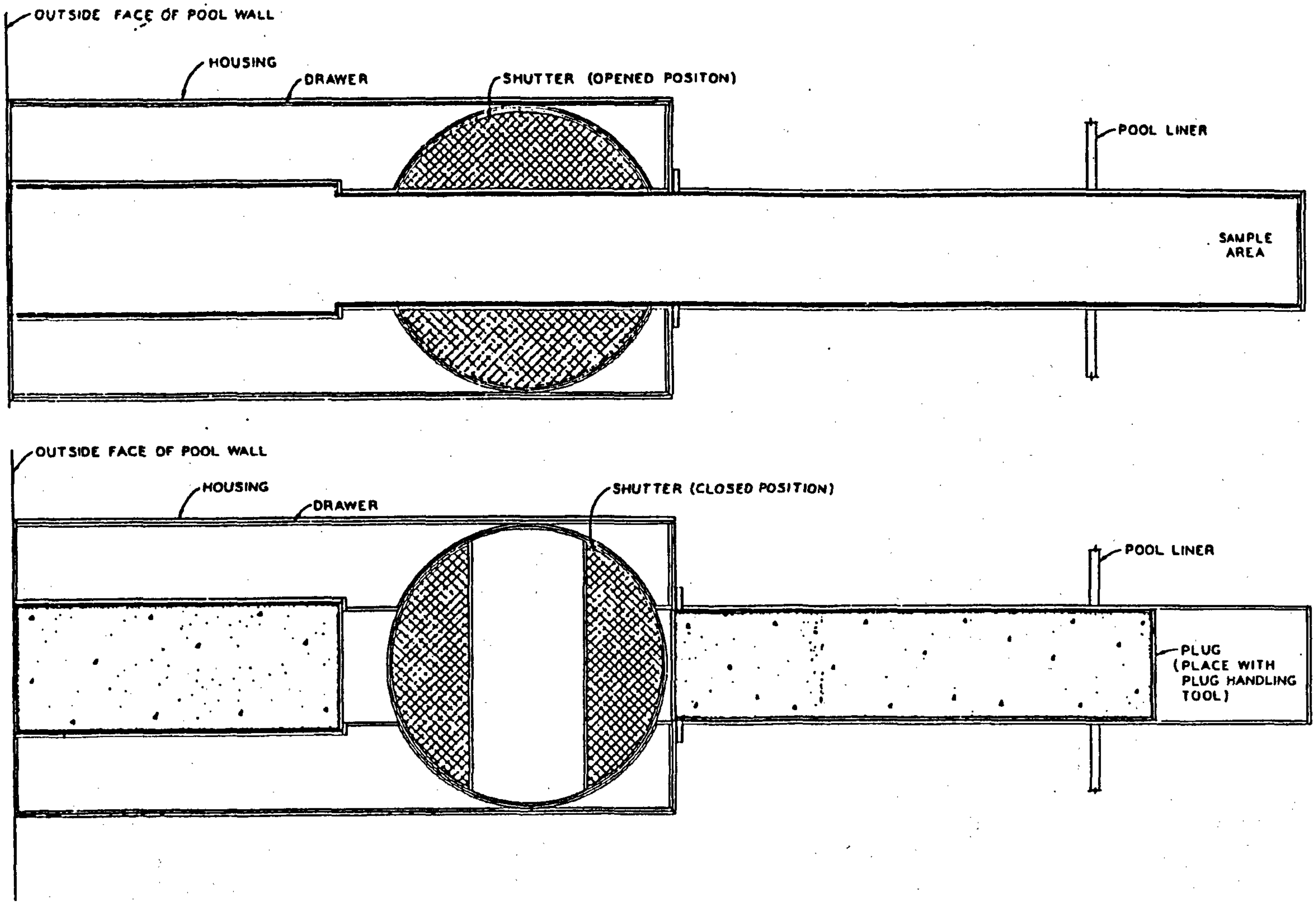


Figure 3.20: OSURR Beam Ports

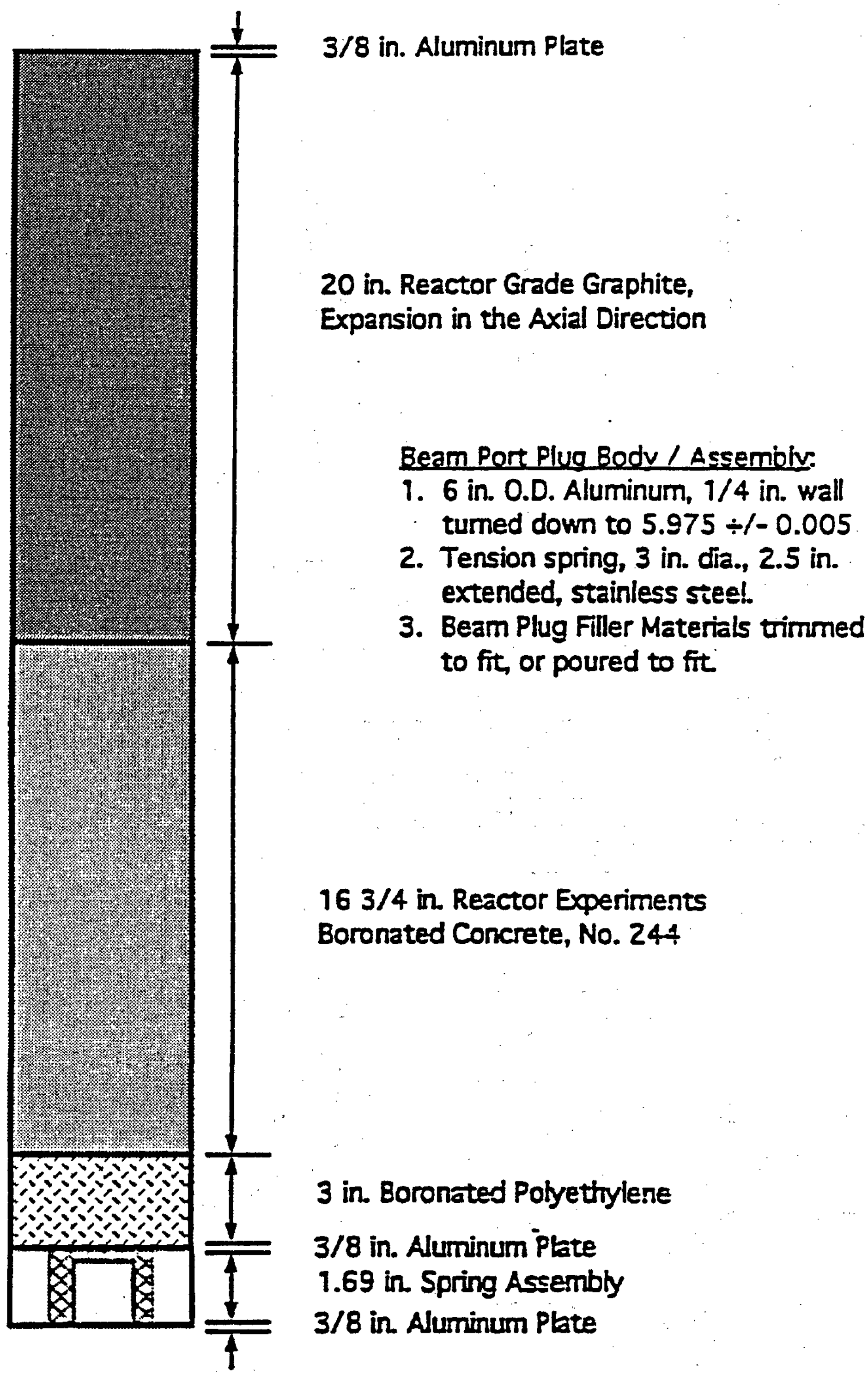


Figure 3.21: OSURR Beam Port Plug

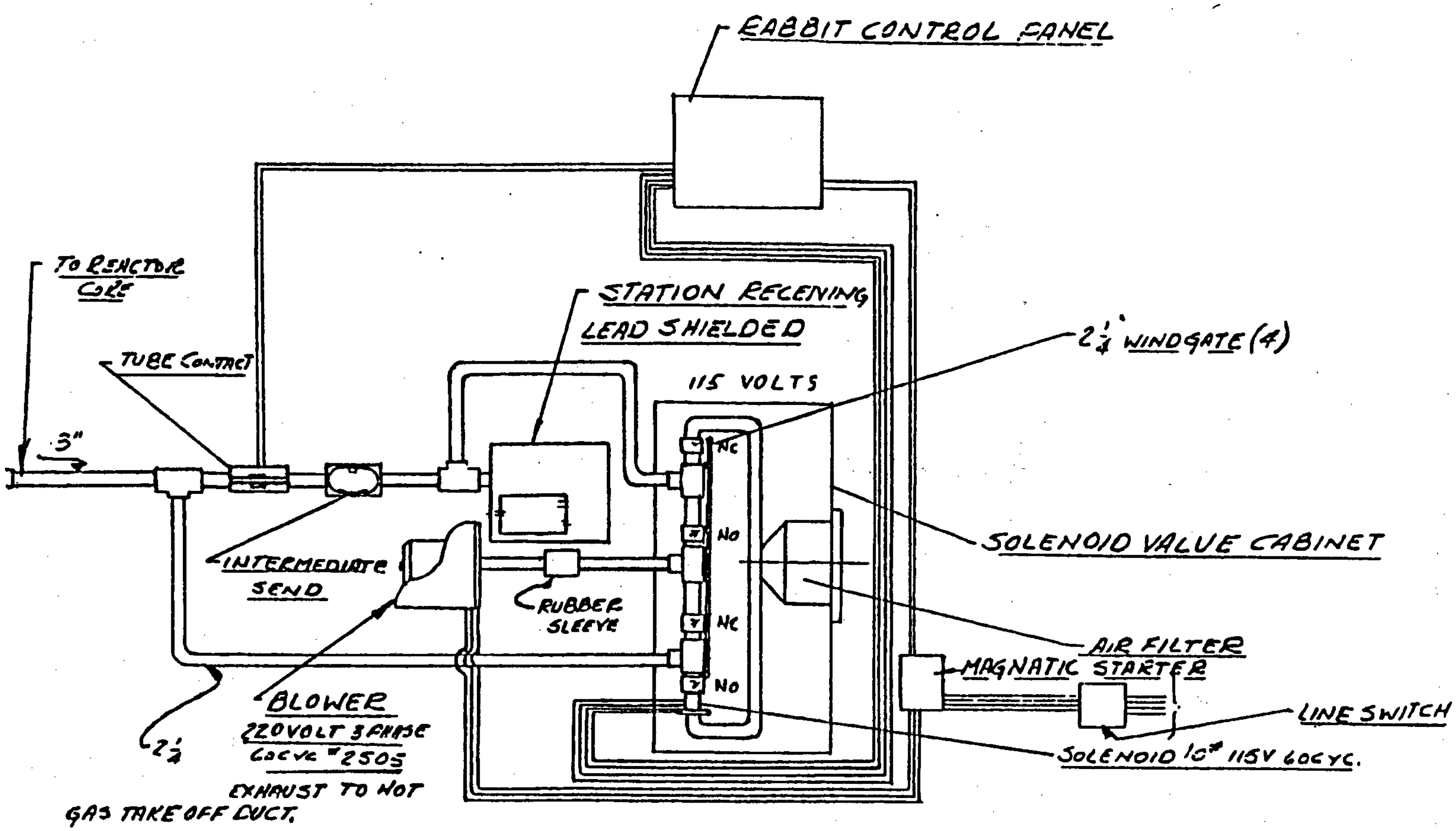


Figure 3.22: OSURR Rabbit System Piping Diagram

leads to the vacuum. Air expelled by the vacuum is piped to the vicinity of the building exhaust fan where it is expelled from the system and removed from the building.

When extracting a sample from the reactor core, the valves on the manifold change state to alter the airflow. Air flows into the manifold through the filter, but is forced into the tube that leads to the sheath around the rabbit tube. Once inside the sheath, airflow is directed through the perforations in the rabbit tube and down the tube toward the manifold. The air is once again forced into the tube that leads to the vacuum and is exhausted.

The solenoid that controls the valves on the manifold can be controlled by either a manual push button or an automatic mechanism actuated by a timer. In the event of an error in timer setting or unintentional release of a sample, the manual control can be used to override the automatic timer control. A permit switch in the control room provides overall control of the rabbit system from the operator's console. Figure 3.23 shows additional details of the rabbit system.

3.8.4 Main Graphite Thermal Column

A cavity exists in the reactor pool wall behind each of the thermal column extensions. The main thermal column, which is the cavity in the west wall of the reactor pool, is stepped twice to help prevent radiation streaming. Figures 3.24 (top cutaway) and 3.25 (side cutaway) show the details of thermal column configuration. The thermal column has a support liner made from 3/8 inch steel plate, which is supported from the foundation by an iron angle structure imbedded in the reactor pool with concrete. An 1/8 inch stainless steel plate is welded to the reactor core end of the liner to provide a waterproof seal. A layer of lead brick is stacked against the inside of this plate. The remaining cavity is filled with 4 inch x 4 inch graphite bars, thirteen of which are removable for foil or sample insertion. The front face of the thermal column is sealed with a removable 4 foot 11 inch x 4 foot 11 inch, ¼ inch 3003-H112 aluminum face plate. Two 0.25 inch thick boral plates are riveted, side by side, to the inside of the face plate for shielding purposes. A 2 inch-width, 0.1875 inch-thick rubber gasket is cemented around the outer edge of the aluminum plate to form an airtight seal around the thermal column opening.

Additional shielding is provided by the main thermal column shielding door. Made of barytes concrete and attached to a "railroad car" type chassis, this door moves on two rails set in the floor of the building. It is supported by an angle iron frame around the edges, and evenly-spaced vertical and horizontal reinforcing rods welded to this frame. The overall dimensions of the shielding door are 96 inches by 31 inches, with a height of 78 inches. The door is moved along the rails by a hand-operated crank mounted at the southwest corner of the door. The hand crank drives a compound sprocket-and-chain system that directly turns the wheels at the south edge of the shielding door.

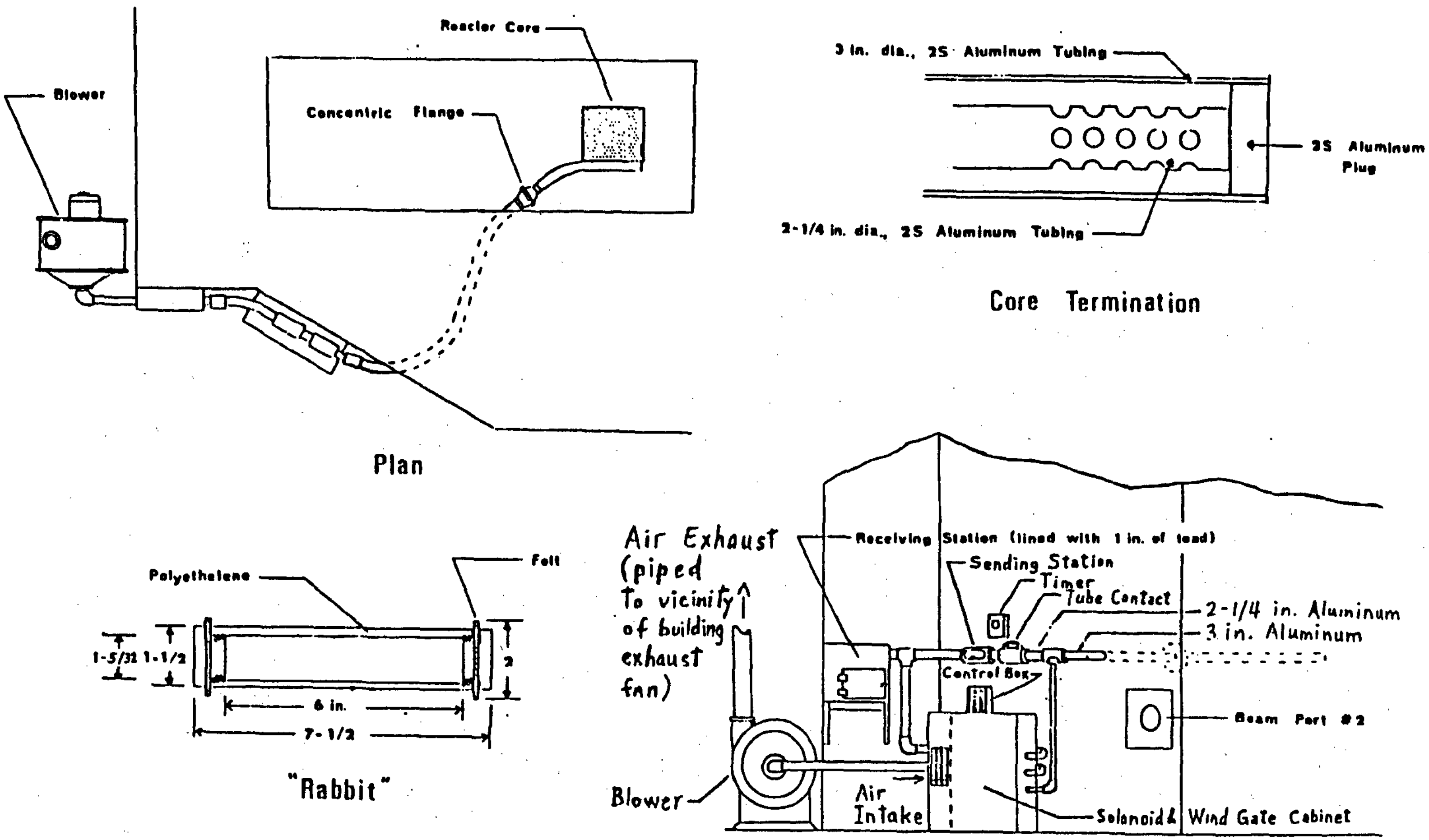


Figure 3.23: Details of the OSURR Rabbit System

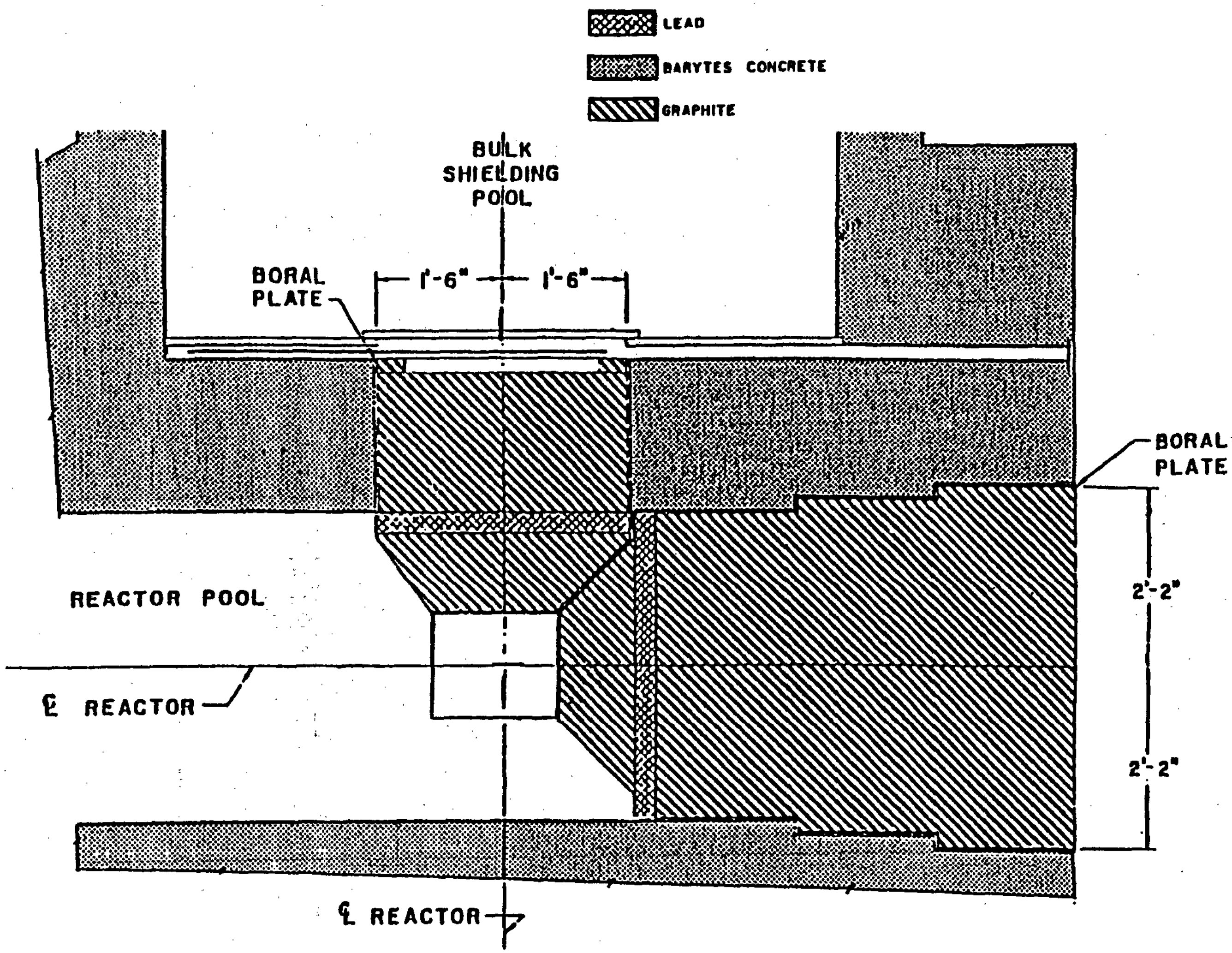


Figure 3.24: Top Cutaway View of the Main and BSF Thermal Columns

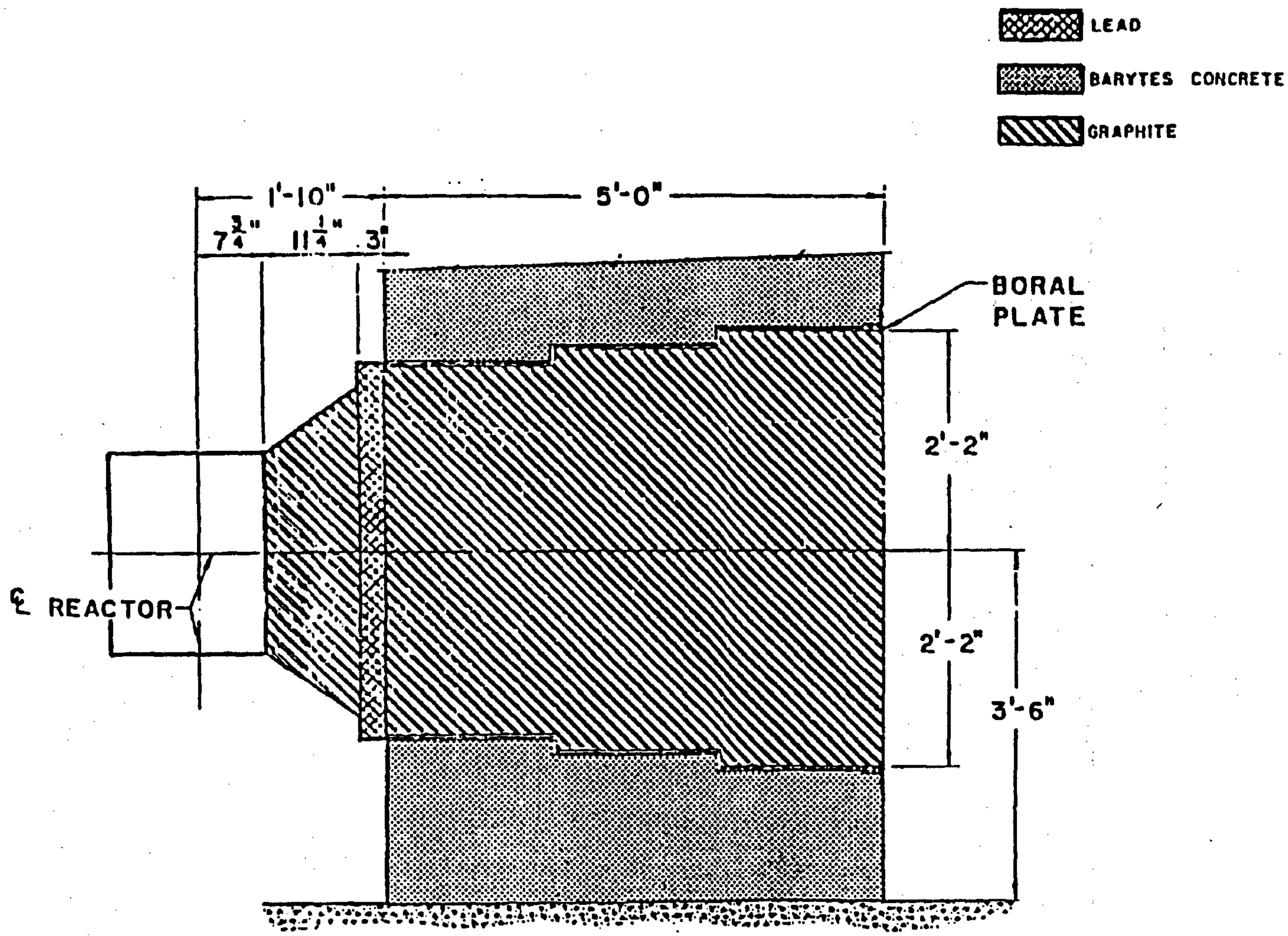


Figure 3.25: Side Cutaway View of the Main Graphite Thermal Column

3.8.5 Bulk Shielding Facility Thermal Column

The Bulk Shield Facility (BSF) thermal column is located between the reactor and BSF pools, behind the thermal column extension on the south wall of the reactor pool. The BSF thermal column, shown in a side cutaway view in Figure 3.26, is an unstepped cavity, lined on the top, bottom, and two sides with 0.375 inch steel plate. The reactor pool end is capped with a 0.125 inch stainless steel plate. A layer of lead brick is stacked along the inner edge of this plate. The remainder of the cavity is filled with removable blocks of graphite. A shutter assembly exists between the graphite thermal column and the BSF pool. The shutter is made up of two ¼ inch plates, one made of boral and the other of 6061-T6 aluminum, that are set side by side on a slide rail. Only one plate is partitioned behind the thermal column at a time. On the BSF side of the shutter, the hole in the BSF wall is capped with a support plate upon which is mounted an end plate. At one time, the support plate housed a heater coil and fission plate, both of which have since been removed.

3.8.6 Graphite Isotope Irradiation Elements

Graphite isotope irradiation elements (GIIEs) can be mounted in any grid plate position. Each of them consists of a graphite element with a one inch diameter hole that runs vertically through the center, and a graphite reflector plug that fits into the hole. The outer dimensions of the graphite elements are identical to those of the fuel elements. Attached diagonally across the tops of the elements are lifting bails which can be engaged by the standard fuel element handling tool. With this tool, the elements can be moved to any desired position in the core.

When an element is not being used for sample irradiation, the graphite plug is inserted into the vertical hole. The plugs have handles on the tops of them that can also be engaged by the standard fuel element handling tool. For sample insertion, the graphite plug is removed and an isotope production tube with sample holder is inserted in its place. Features of the GIIEs, tubes, and plugs are shown in detail in Figure 3.27.

3.8.7 Movable Dry Tubes

Two movable dry tubes are normally stored in the reactor pool. These tubes can be used to place experiments at or near the core boundary. Although a fixed mounting point is provided on the I-beam supporting the startup source drive mechanism above the pool surface, the tubes can be vertically mounted almost anywhere within the volume of the reactor pool. The tubes are locked when not in use. Both dry tubes are sealed at one end and partially filled with lead shot to maintain neutral buoyancy and a low center of gravity for vertical stability when the tubes are unsupported.

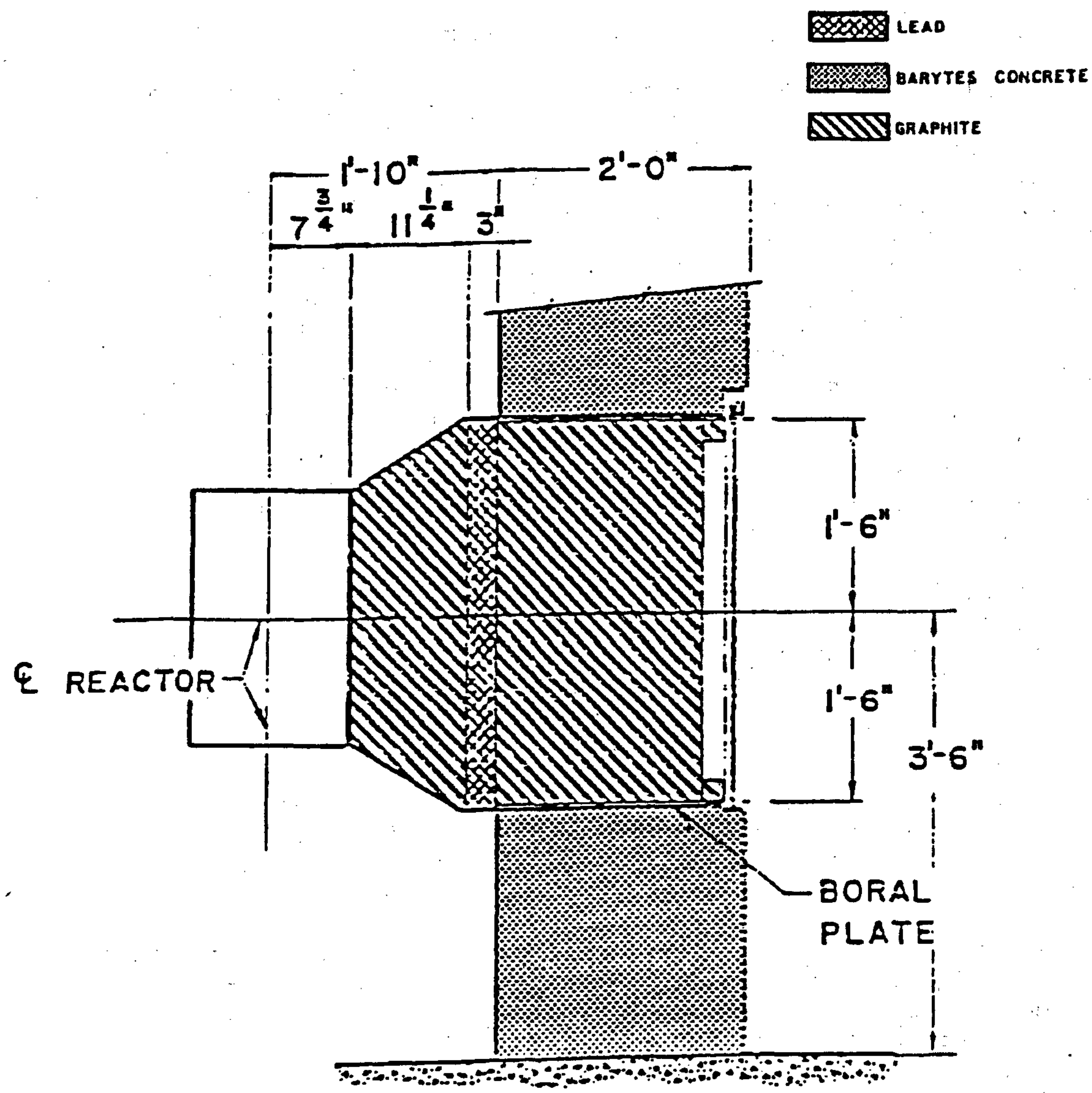


Figure 3.26: Side Cutaway View of the BSF Thermal Column

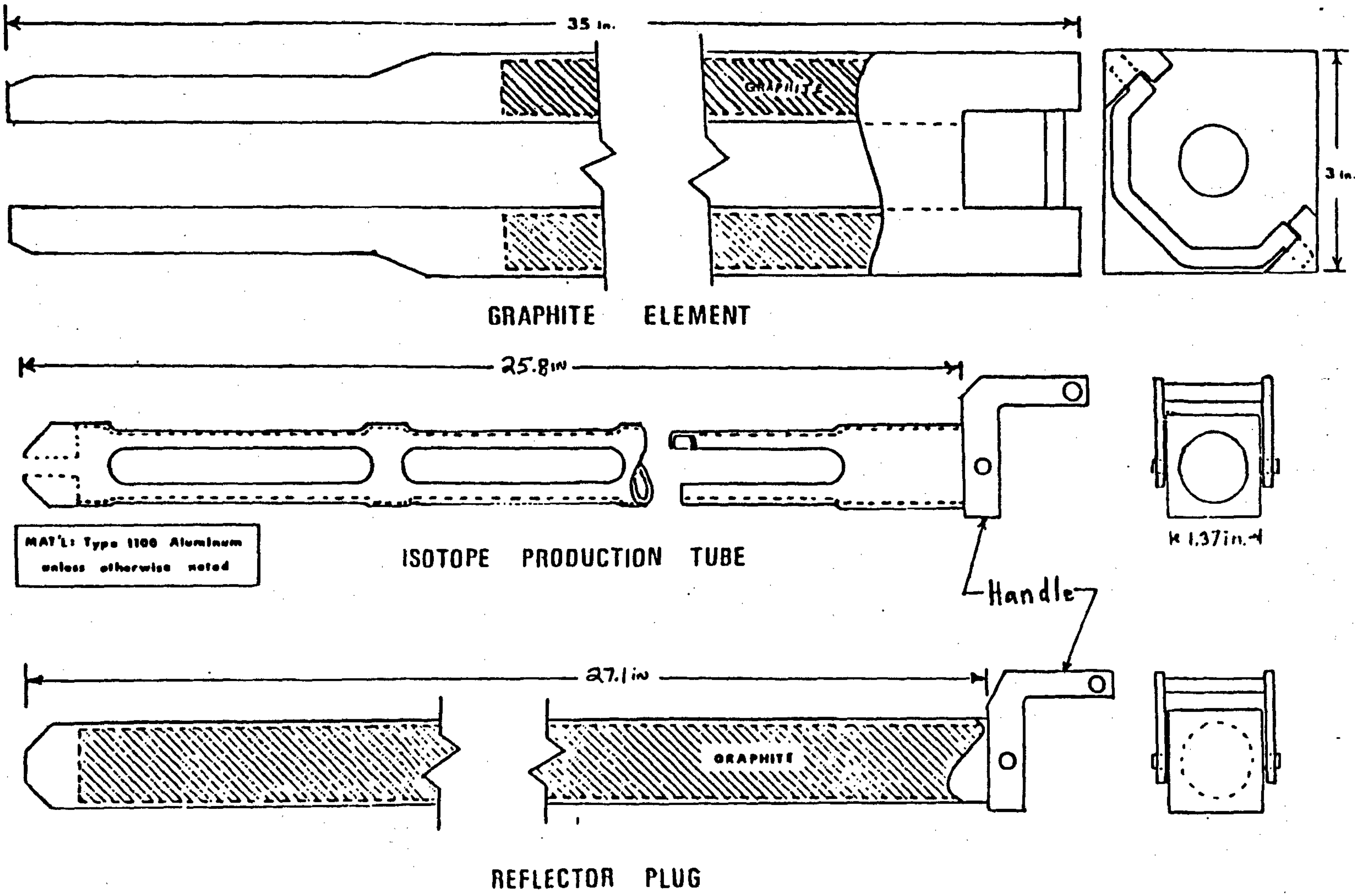


Figure 3.27: Details of the Graphite Irradiation Elements, Tubes, & Plugs

The license for the OSURR allows construction of special dry tubes and other experimental facilities and apparatus. These devices must be approved by the Reactor Operations Committee and deemed not to constitute an unreviewed safety question prior to use. If the committee deems that these devices do constitute an unreviewed safety question, review and approval by the appropriate office within the Nuclear Regulatory Commission is required prior to their use in the OSURR.

4.0 Normal Operating Characteristics

4.1 Core Loading and Critical Mass

Analysis of the OSURR core performance under normal conditions required use of various computer codes. Input data for the computer codes was based on 25 years of actual operating experience with the HEU-fueled OSURR core, and on LEU fuel characteristics as supplied by the RERTR program. The LEU fuel used in the OSURR core was described in Chapter 3.

Neutronic calculations considered, among others, the following design objectives:

1) maintain at least 1% dk/k shutdown margin with the highest-worth control rod and the regulating rod completely withdrawn to their highest position,

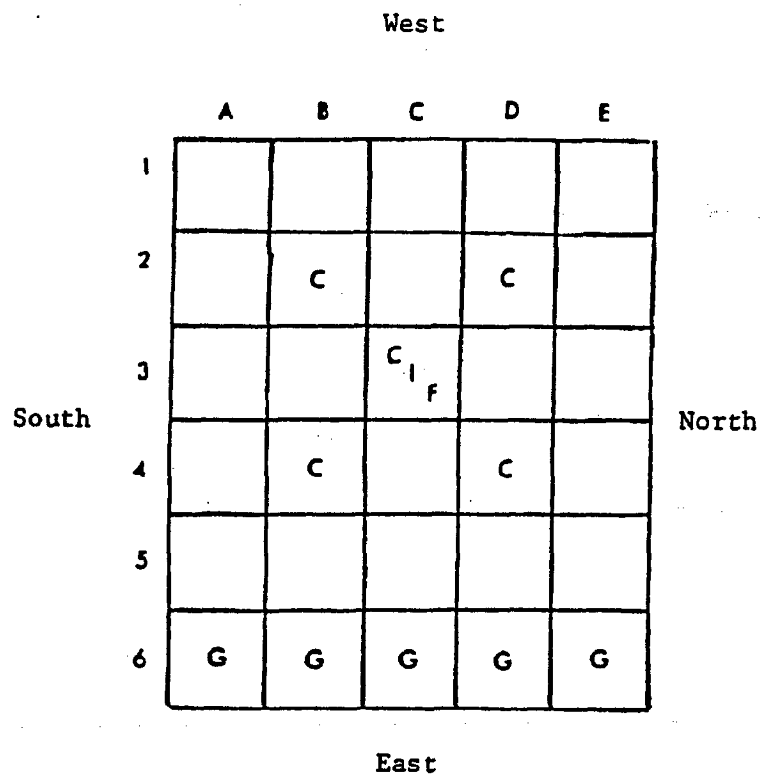
2) for a given power level, minimize the heat generated in the hot channel and

3) for a given power level, maximize the thermal neutron flux at the Central Irradiation Position (i.e., the central flux trap) and at the Beam Ports.

Utilizing the information on the standardized DOE fuel plate, and the specific design of the OSURR standard and control rod fuel assemblies, various core configurations were analyzed to meet the requirements noted above. These core configurations differ in the number and position of the standard fuel assemblies on the grid plate, as well as the number and position of the graphite irradiation elements (sometimes called graphite reflector blocks) on the grid plate. Figure 4.1 shows the numbering scheme for grid plate position, and will be utilized in specifying assembly positions on the grid plate in the remainder of this chapter.

The neutronics analysis used the LEOPARD code for the generation of four-group neutron diffusion parameters on a unit cell basis. The 2DB code was used for reactivity and neutron flux calculations. The computational predictions were tested (benchmarked) with data from the HEU-fueled OSURR core measurements and independent calculations utilizing the EPRI-CELL, DIF3D, and VIM codes.

Analytical results indicate that the OSURR core will contain between [redacted] standard fuel elements and four control rod fuel elements. This yields a nominal ^{235}U core loading of between [redacted] respectively. Possible core configurations are shown in Figures 4.2 through 4.4, which represent grid plate positions and the elements positioned in the available locations. The X's in each of these figures denote positions on the grid plate without either a fuel assembly or a graphite irradiation element (as discussed in Chapter 3,



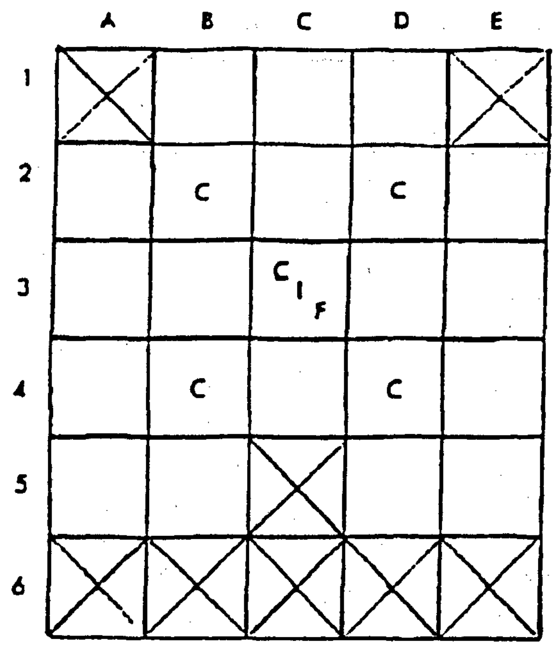
Notes: (1) The control rod positions are shown by the letter "C", and are fixed.

(2) The CIF position (water or graphite-filled central flux trap) is normally fixed.

(3) Graphite irradiation element positions (letter "G") can be removed. They are shown installed in the standard HEU core position.

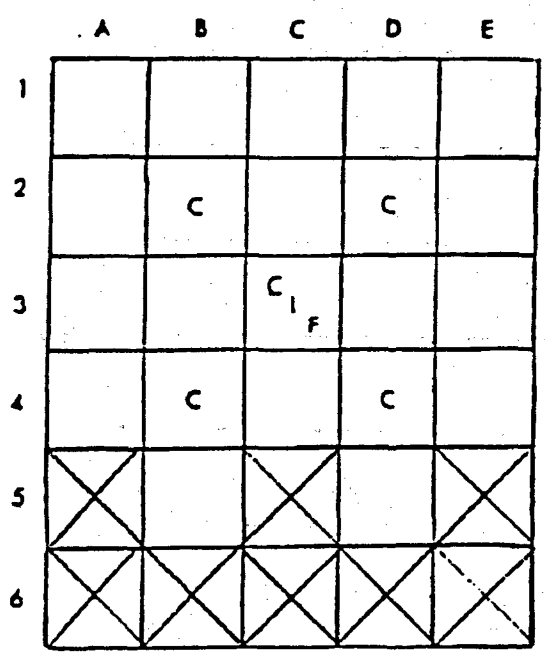
(4) The core is reflected along the west and south edges by the extension pieces for the main and BSF thermal columns (graphite reflectors).

Figure 4.1: Numbering System for Denoting Positions on the Grid Plate of the OSURR



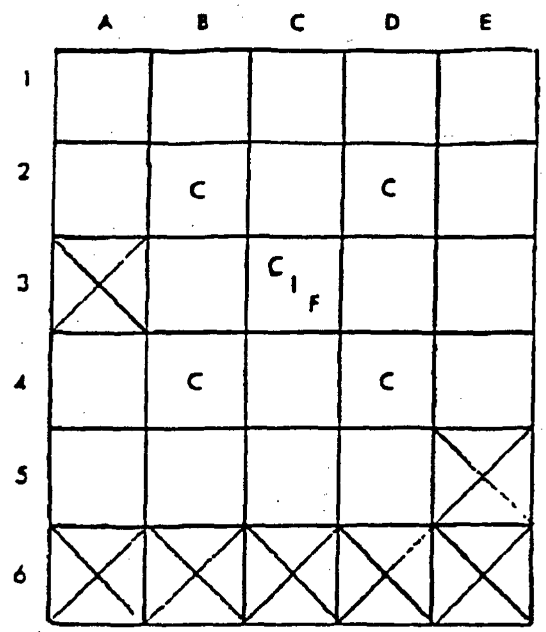
LEU Core F

X's denote vacant grid positions



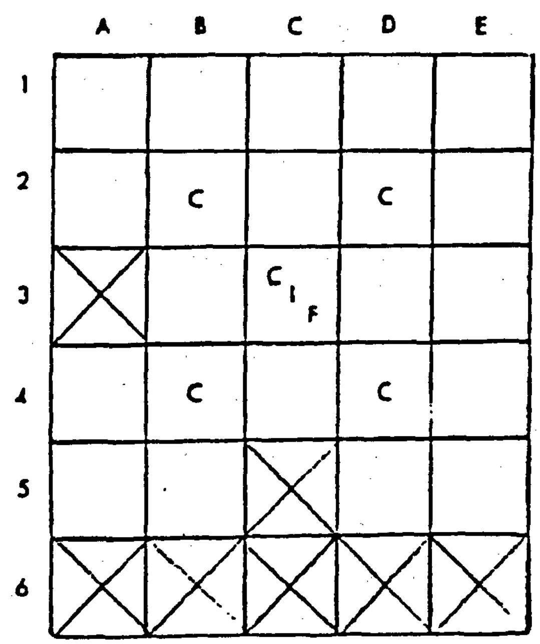
LEU Core G

Figure 4.2: Possible OSURR Core Configurations, LEU Cores F and G



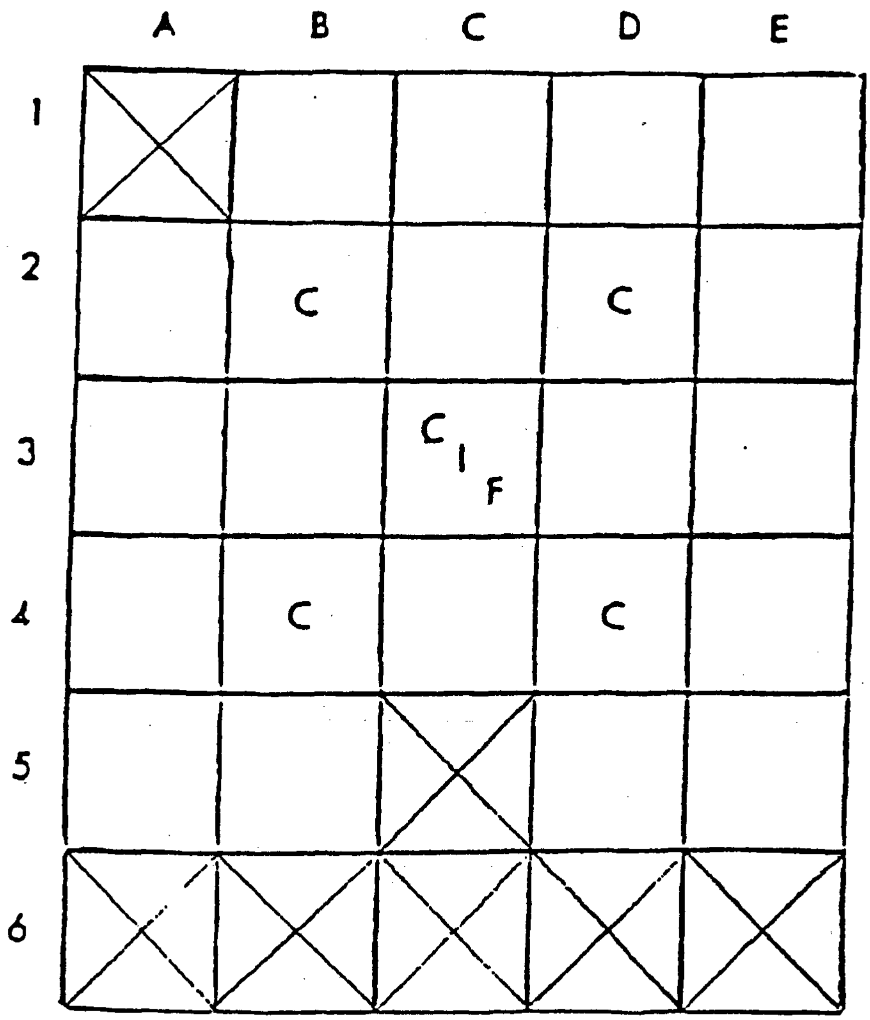
LEU Core J

X's denote
vacant grid
positions



LEU Core P

Figure 4.3: Possible OSURR Core Configurations, LEU Cores J and P



X's denote vacant grid positions

LEU Core I

Figure 4.4: Reference LEU Core for the OSURR (Core I)

aluminum plugs may be inserted at these positions). Note that these configurations all involve removal of all graphite irradiation elements from grid plate row 6, and either two or three standard fuel elements removed from various grid plate locations. Core configurations were also analyzed involving replacement of some of the removed standard fuel assemblies with graphite irradiation elements. Results indicate increased core reactivity effects, which can be useful in regaining reactivity loss resulting from fuel burnup or reconfiguration of reflectors around the core (e.g., removal of shielding plugs from the beam ports, which have some reflector worth). In all cases, the four control rods are positioned at their standard core locations (positions 2B, 2D, 4B, and 4D, for shim safety rod 1, shim safety rod 2, regulating rod, and shim safety rod 3, respectively), and the central irradiation facility (CIF) is placed at grid location 3C.

Of the core geometries shown in Figures 4.2 through 4.4, LEU core I (capital letter i) is taken to be the "reference core" for neutronic calculations to be discussed in later sections. This case represents a geometry which comes closest to the design objectives noted earlier.

Initial loading of the OSURR core with the LEU fuel elements described in Chapter 3 leads to a smaller core than the HEU-fueled core. The higher element loading of ^{235}U in the LEU fuel elements results in higher element reactivity worth. Loading 20 standard LEU elements in the core with four control rod fuel elements and the central flux trap would result in an excess reactivity of 6.9% dk/k for the OSURR core. Since for a fixed configuration and metal-to-water ratio, it is well known that control rod worth will decrease in converting from HEU to LEU fuel (resulting from increased resonance absorption in the fuel and subsequent neutron spectrum hardening), this excess reactivity would likely lead to an inadequate shutdown margin and possibly exceed the total control rod worth.

Removing the graphite irradiation elements from grid locations 6A through 6E results in an excess reactivity reduction of 0.85% dk/k. Removal of selected fuel assemblies reduces the total excess reactivity further. In specifying positions from which fuel elements would be removed, only grid plate positions 1A, 1E, 3A, 5A, 5C, and 5E were considered to prevent "uncovering" of control rods and subsequent reduction in their effective reactivity worth. Fine adjustments of excess reactivity can be made using partially loaded fuel assemblies, or reinserting graphite irradiation elements at selected positions.

Initial criticality of the LEU-fueled OSURR was attained in December, 1988. The initial critical configuration, designated as core LEU-1, had a loading of [redacted] of ^{235}U. The excess reactivity associated with this configuration was quite small, and impractical from an operational viewpoint. However, it did allow estimate of the absolute minimum critical mass. From this, the minimum critical core loading was determined to be approximately [redacted] of ^{235}U. This compares reasonable well with the predicted minimum critical mass of [redacted] of ^{235}U. As of December, 1999, the OSURR core geometry, denoted as core LEU-4, has a loading of [redacted] of ^{235}U, which provides a reasonable excess reactivity and meets the Technical Specifications for minimum shutdown margin. This configuration consists of [redacted] standard fuel assemblies and four control rod fuel assemblies.

4.2 Reactivity Requirements

4.2.1 General Considerations

Consideration of the reactivity requirements for the OSURR included the following:

1) xenon poisoning,
2) temperature feedback,
3) experiment requirements
4) fuel burnup, and
5) control margin to generate reasonable reactor periods.

These effects will be considered in detail in the following paragraphs.

4.2.2 Operational Requirements

The excess reactivity of the OSURR core must be sufficient to compensate for buildup of xenon poisons during extended operation, increases in fuel and moderator temperature, burnup of the fuel over the course of an operating cycle, and control requirements for generation of reasonable rates of power increase under various conditions (i.e., water temperature and xenon inventory).

Analysis of several LEU core configurations for the OSURR indicates that about 1.2% dk/k excess reactivity will be required to compensate for xenon poisoning and temperature feedback. This calculation assumes an operation at 500 KW power of sufficient duration to attain a saturation level of xenon inventory and an equilibrium bulk temperature for the moderator, based on cooling system heat removal capacity under normal conditions. The assumption of xenon poison equilibrium is conservative in that most reactor operations will not approach the total energy production necessary to attain a maximum equilibrium xenon inventory. The assumption of normal cooling system heat removal to attain an equilibrium temperature is also reasonable since cooling system capacity has been "oversized" for most environmental conditions, and that bulk water temperatures will rise slowly and probably not attain a final equilibrium at an elevated temperature before the reactor operation ceases.

Burnup calculations require an assumption for operating power history of the reactor. In recent years, average yearly energy production for the OSURR has been about 2000 kilowatt-hours, which translates to 200 full-power equivalent hours per year. Conservative assumptions for

4.3 Neutron and Gamma Environments

4.3.1 Neutron Flux Calculations

Given a core geometry and operating power, estimates of the neutron flux distribution in two dimensions were made. The operating power was fixed at 500 kilowatts, and the geometry of the core varied slightly to gauge the geometry effects. The lateral flux profile across the core, starting at grid location 3A and proceeding to grid location 3E, was obtained for various core geometries. These are shown in Figures 4.5 through 4.10, with Figures 4.9 and 4.10 showing the reference LEU core. Note that Figures 4.5 through 4.8 show the thermal neutron flux on a linear-linear plot, while Figures 4.9 and 4.10 are shown in a semi-logarithmic scale.

In Figures 4.5 through 4.8, the horizontal flux profile is shown, centered on the standard CIF location (grid plate position 3C). The profile is for grid plate row 3, moving across the grid plate position column from A through E, in a south-north (left to right on the figures) direction. In all cases, the flux peaks in the CIF position (as desired), with a peak thermal neutron flux on the order of 1.4×10^{13} nV. The average thermal neutron flux in the core region is about $4\text{-}5 \times 10^{12}$ nV. For LEU cores J and P (Figures 4.7 and 4.8 respectively), a pronounced secondary flux peak appears at grid plate position 3A, which in these geometries is a vacant (essentially waterfilled) location. The peak results from increased thermalization of fast neutrons in the water at this position. All four figures show the flux peaking in the water reflector region at the north edge of the core, at a location of about +7 or +8 inches from the CIF position.

Figures 4.9 and 4.10 show the reference LEU core (core I) in more detail. Figure 4.9 is similar to the earlier figures in that the flux profile is centered on the CIF, moving along grid plate row 3 in a south-north direction. Figure 4.10 is also centered on the CIF, but moves in an east-west direction along grid plate column C. The flux peaking in the vacant grid plate location 5C, at about -6 inches from the CIF position, is clearly shown. Essentially, this vacant position acts as another water-filled flux trap, with a slightly lower peak flux than the central position, since location 5C is at the edge of the core. Again, these figures show the logarithm of the thermal neutron flux on the ordinate of these graphs, while the abscissa shows position on a linear scale.

Neutron flux profiles were obtained for three other neutron energy groups in addition to the thermal neutron flux estimates presented above. These other energy groups included epithermal and two "fast" neutron groups. As an example, Figure 4.11 shows the flux profile of one of the two fast neutron groups for LEU core 1, centered on the CIF position, moving south to north along grid plate row 3. The depression in the fast flux at the CIF location resulting from thermalization of fast neutrons in the water-filled flux trap is clearly shown. While

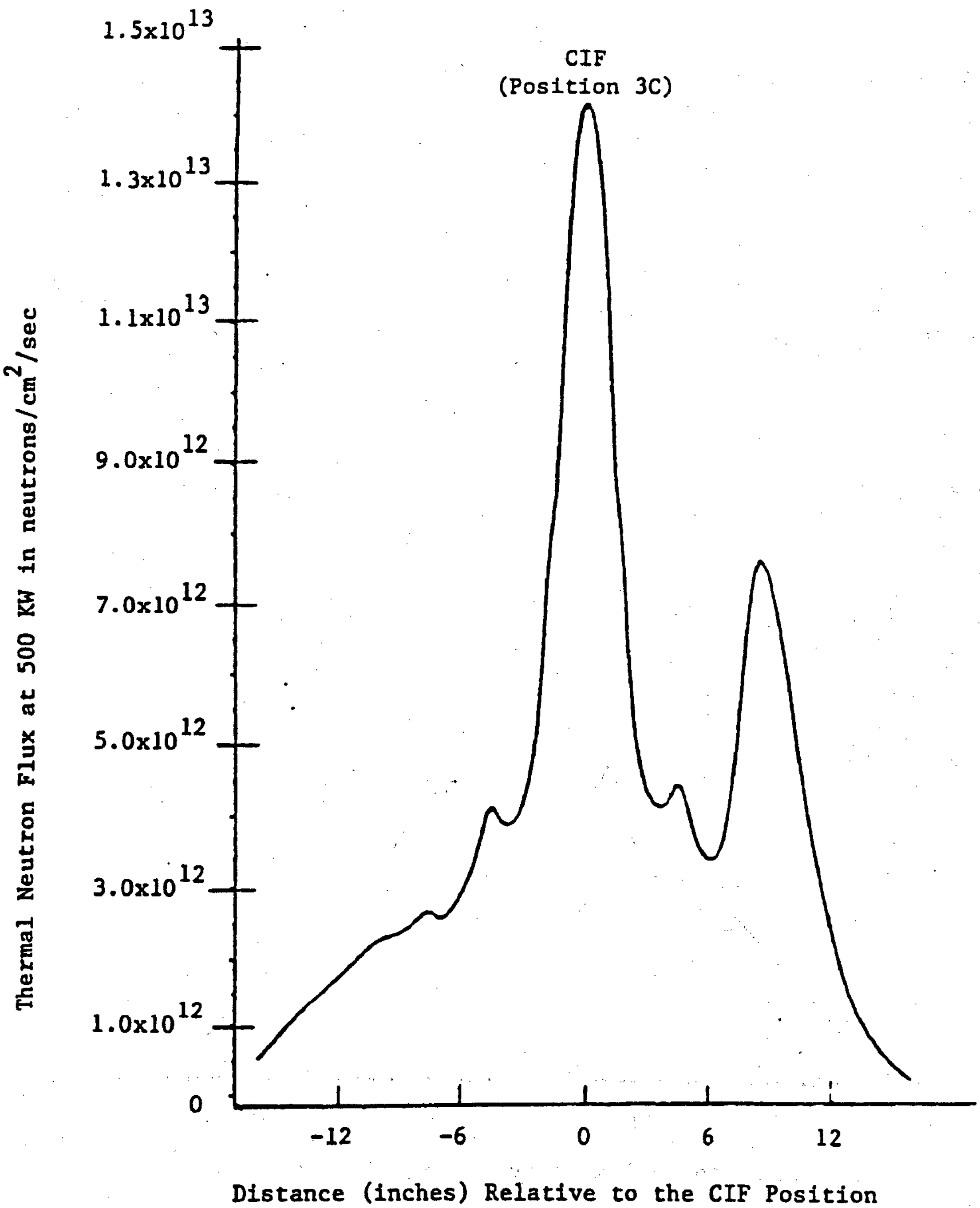


Figure 4.5: Thermal Neutron Flux Profile for LEU Core F at 500 kW Operating Power

Note: The flux profile moves from south (left) to north (right) along grid plate row 3.

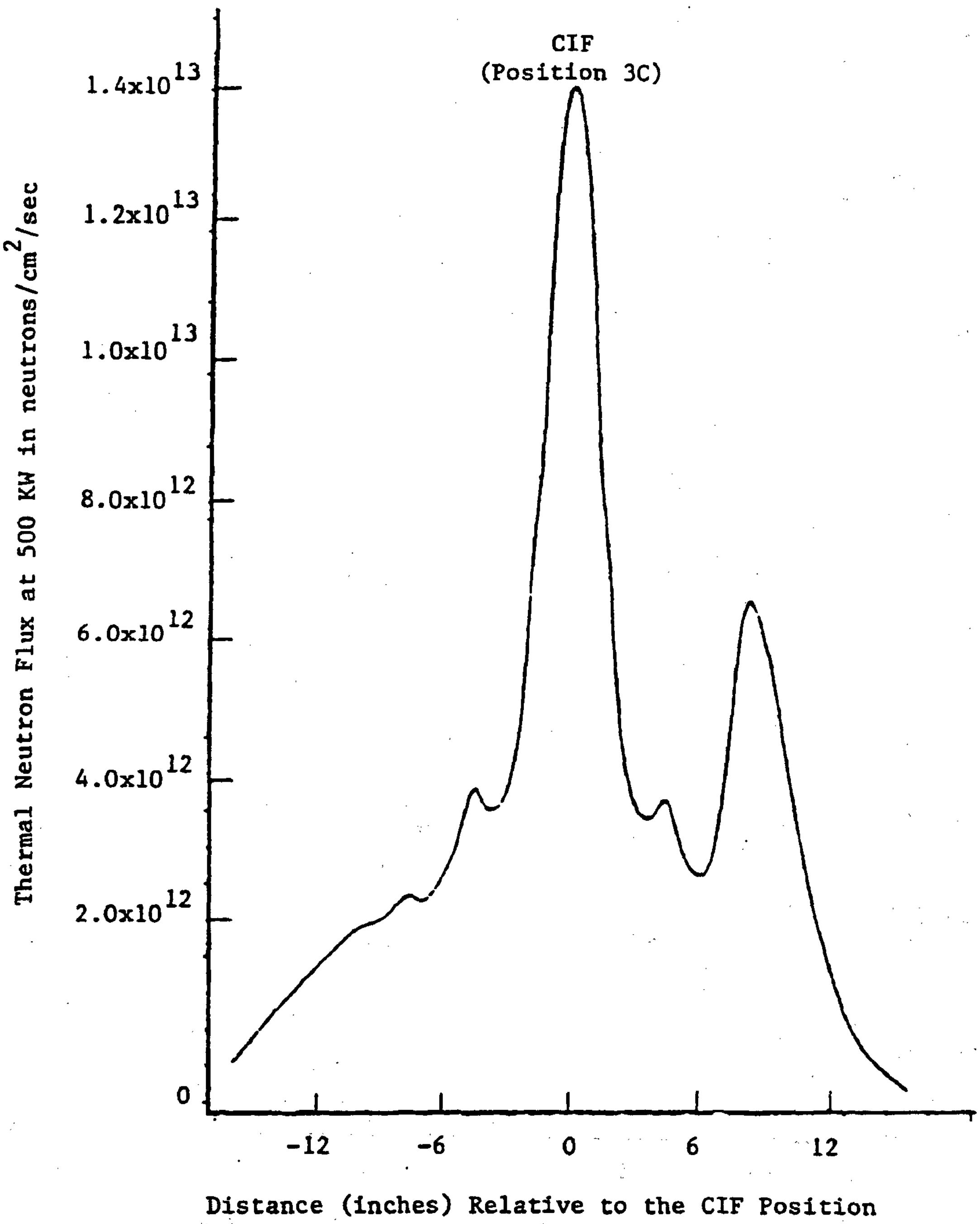


Figure 4.6: Thermal Neutron Flux Profile for LEU Core G at 500 kW Operating Power

Note: The flux profile moves from south (left) to north (right) along grid plate row 3.

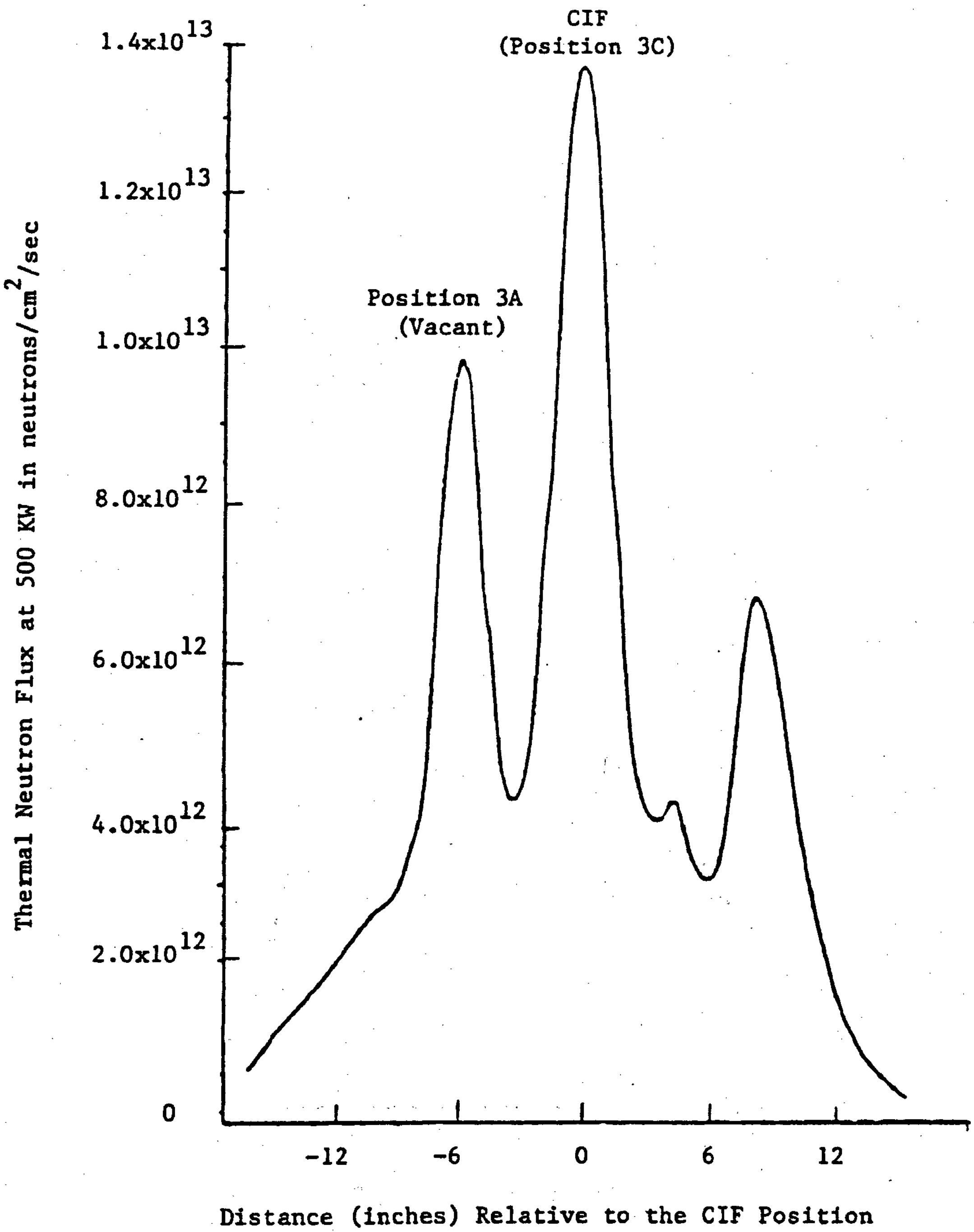


Figure 4.7: Thermal Neutron Flux Profile for LEU Core J at 500 kW Operating Power

Note: The flux profile moves from south (left) to north (right) along grid plate row 3.

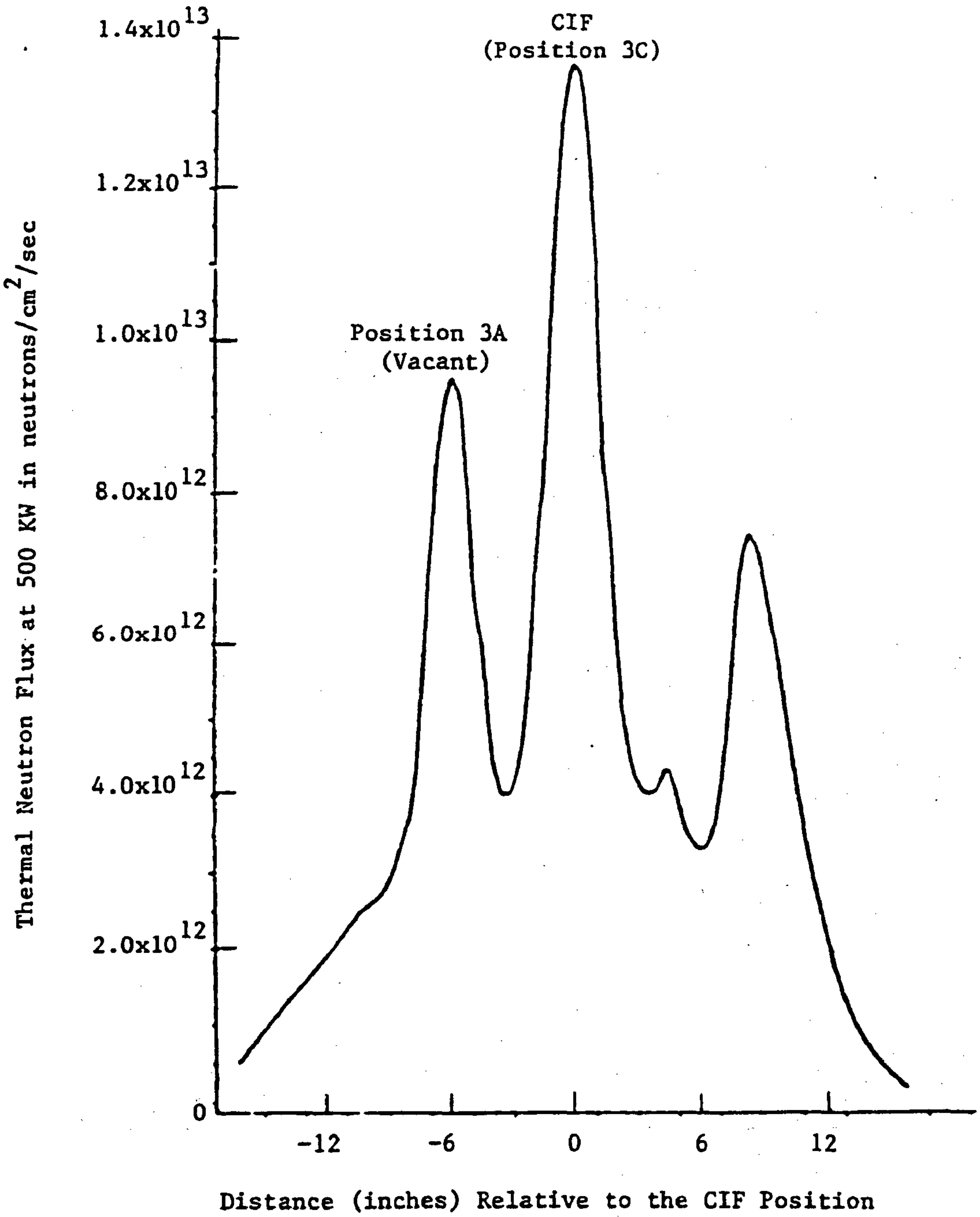


Figure 4.8: Thermal Neutron Flux Profile for LEU Core P at 500 kW Operating Power

Note: The flux profile moves from south (left) to north (right) along grid plate row 3.

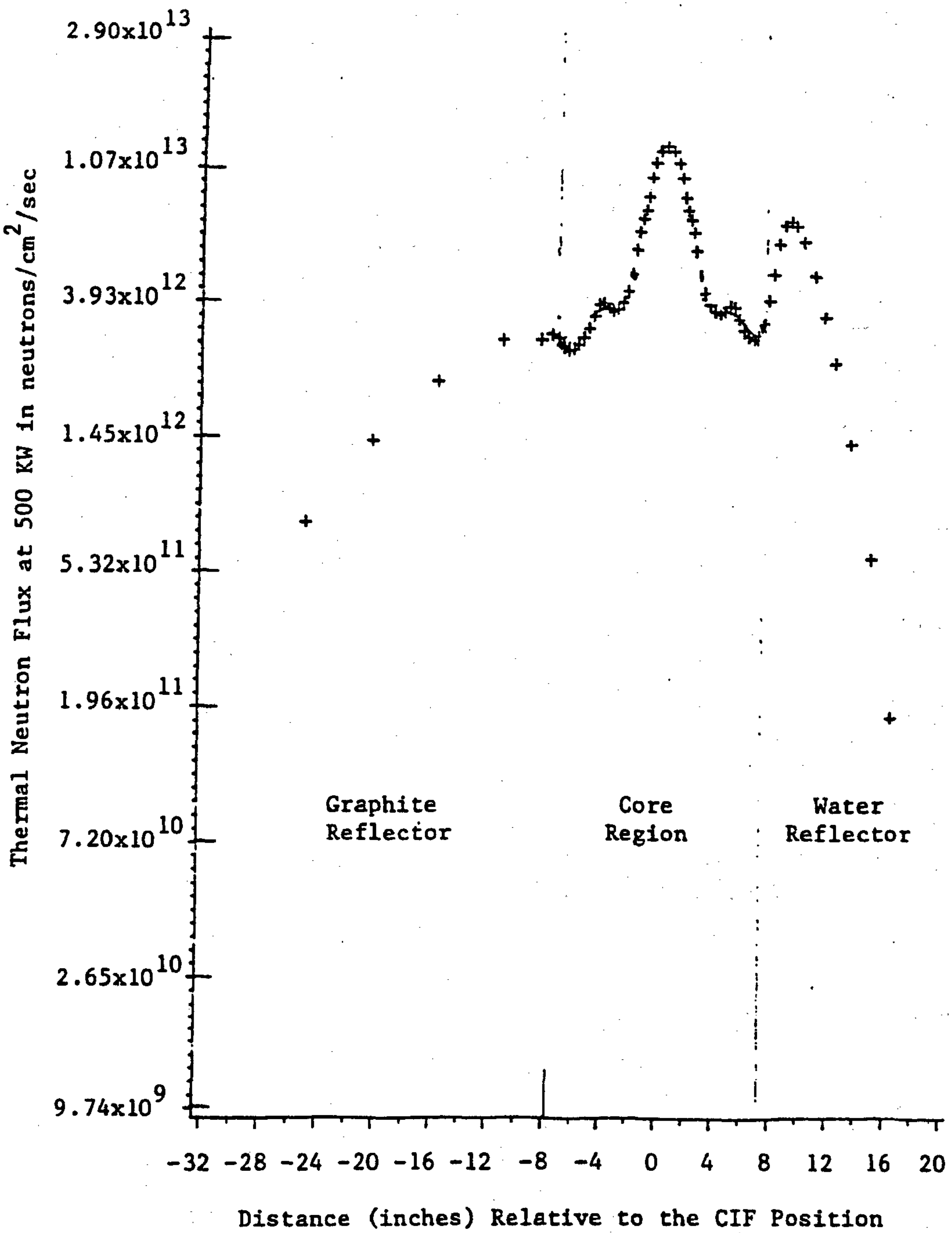


Figure 4.9: Thermal Neutron Flux Profile for LEU Core I at 500 kW Operating Power

Note: The flux profile moves from south (left) to north (right) along grid plate row 3.

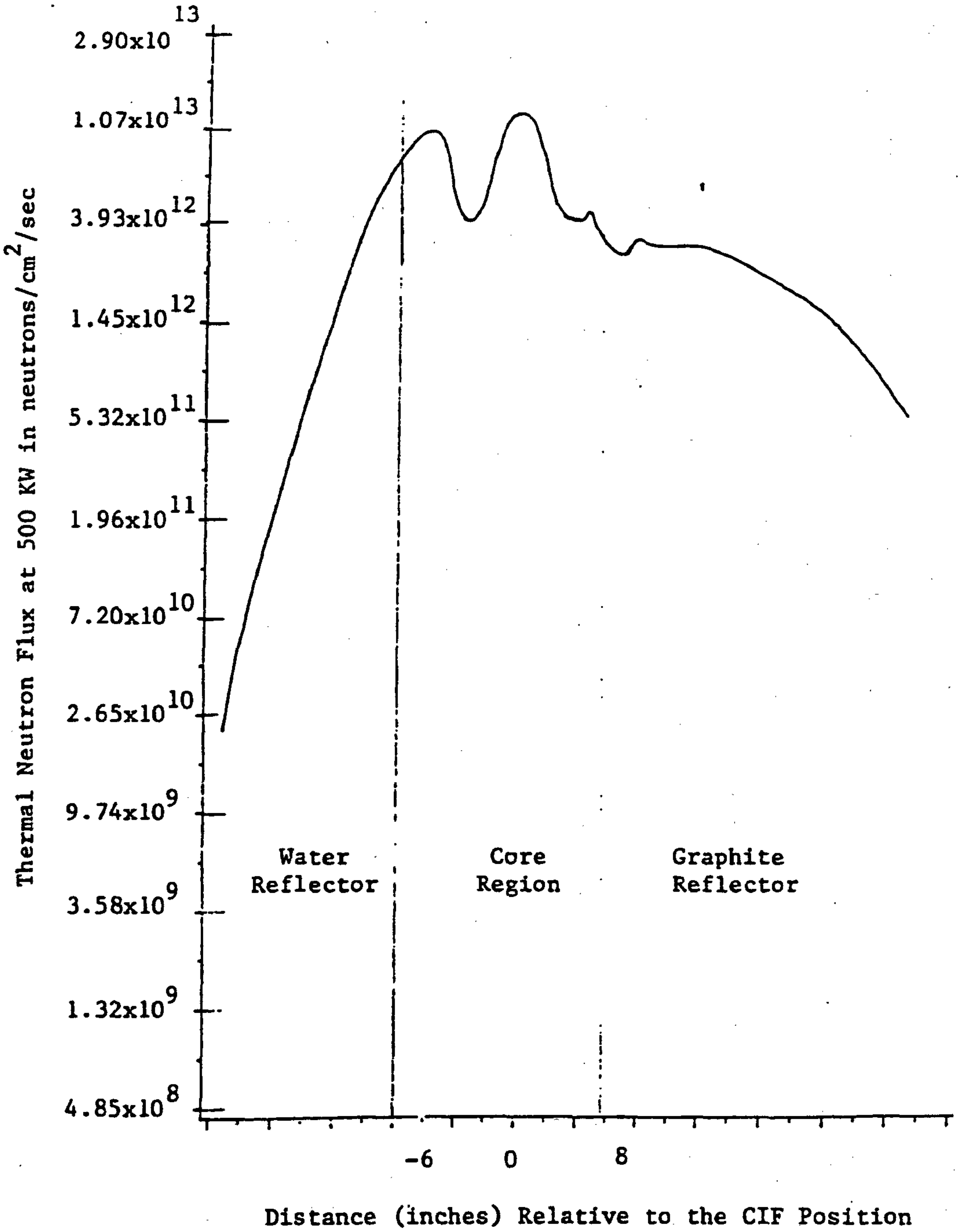


Figure 4.10: Thermal Neutron Flux Profile for LEU Core I at 500 kW Operating Power

Note: The flux profile moves from east (left) to west (right) along grid plate column C.

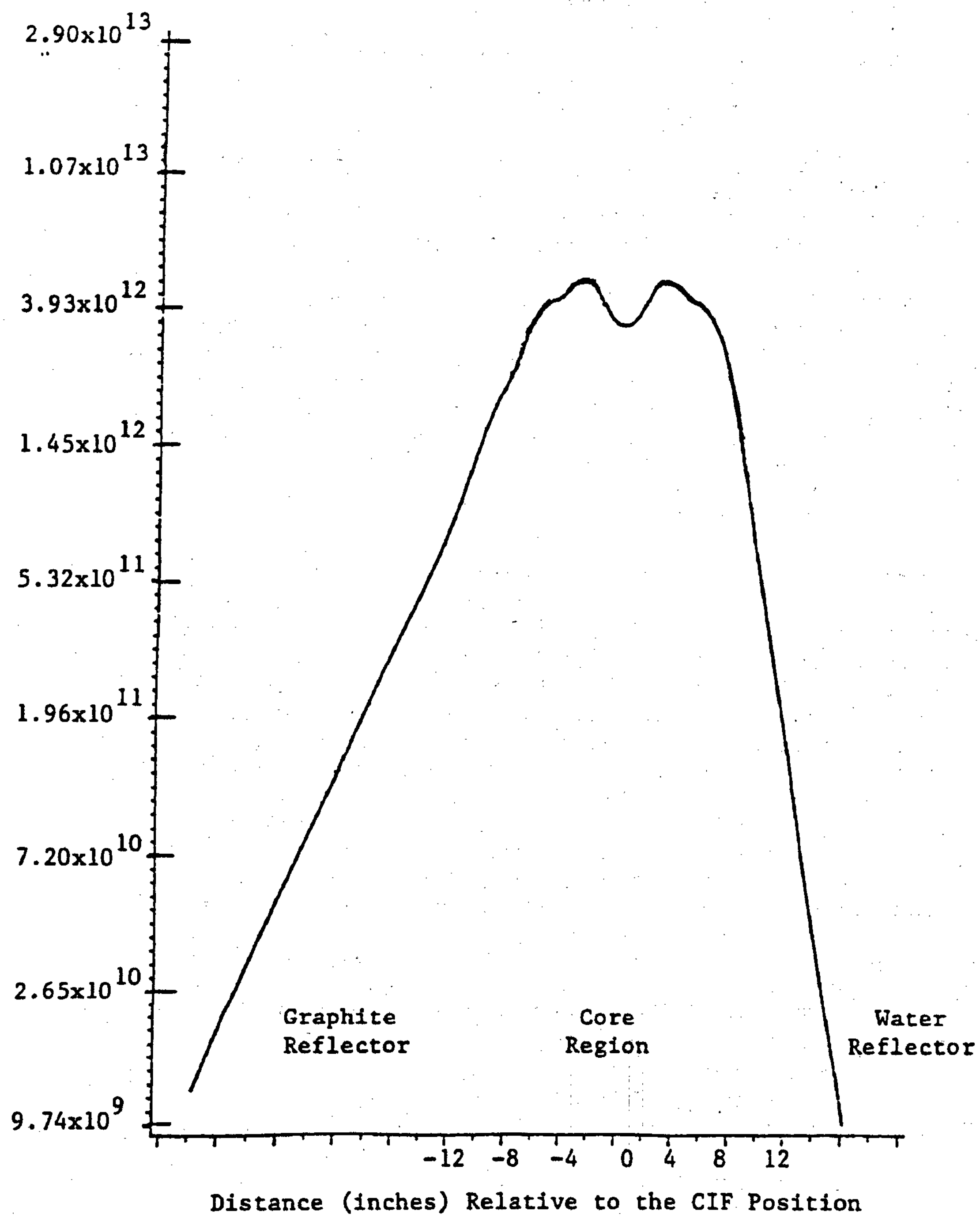


Figure 4.11: Neutron Flux Profile for One of the Two Fast Neutron Groups for LEU Core I at 500 kW Operating Power

Note: The flux profile moves from south (left) to north (right) along grid plate row 3.

the peak fast neutron flux is less than the peak thermal neutron flux, the average fast neutron flux in the core region, excluding the CIF position, is somewhat larger than the average thermal neutron flux. This observation is consistent with physical expectations since higher resonance absorption in LEU increases the fast-to-thermal flux ratio.

In the vertical direction, the flux follows a cosine distribution, but is skewed by absorption effects of the control rods. Measurements of the vertical flux profile for the HEU-fueled OSURR core verify this behavior, and it is expected to hold true for the LEU-fueled core.

Actual measurement of the neutron flux within the boundaries of the fuel elements is not a simple experiment, since it is difficult to place dosimeter materials between the narrowly-separated fuel plates. However, the neutron energy distribution has been measured in the experimental facilities. In the CIF at full power, the thermal neutron flux is about 1.2×10^{13} nv, while in the Rabbit it is about 2.0×10^{12} nv, as of December 1999, using core LEU-4.

4.3.2 Gamma Environment Estimation

The gamma environment in and around the core of the OSURR will be a function of the operating power and power history of the core. In general, measurements made on the HEU-fueled core show that during operations at power levels above about 10 watts, the gamma field component from prompt fission of ^{235}U will dominate that from the shutdown field. Because use of LEU fuel will produce a larger inventory of fission products in the core, the shutdown gamma field in the core be a somewhat larger percentage of the overall gamma field than was the case with HEU fuel. However, assuming that the prompt gamma dose rate is proportional to operating power, a gamma dose will increase by about a factor of 50 in and around the core. Indirect measurements of the gamma dose rate in the CIF at 10 KW (HEU core) show a dose rate of about 10^6 R/hr, so at 500 KW, the dose rate in the CIF will be about $5x10^7$ R/hr. Measurements in Beam Port 1 at a point closest to the core indicate a dose rate of about $5x10^5$ R/hr for the HEU-fueled OSURR operating at 10 KW, so a dose rate of approximately $2.5x10^7$ R/hr is expected at the core end of Beam Port 1 when operating at 500 KW. At the core end of stringer position G-7 (central stringer) of the main graphite thermal column, a gamma dose rate of about $2.2x10^4$ R/hr has been measured for the HEU-fueled core at 10 KW. A dose rate of about $1.1x10^6$ R/hr is expected at 500 KW power at this location.

We have estimated the gamma dose rate in the CIF at full power, using extrapolations of the measured values at lower operating power. The full-power gamma dose rate in the CIF is thus estimated to be about 100 megarads per hour, as of December, 1999, using core LEU-4.

4.4 Reactivity Control Systems

4.4.1 Control Rod System

The total worth of all control rods is estimated to be at least 8.45 % dk/k. This is considered to be the minimum rod worth.

The maximum rate of reactivity addition from the control rod system is estimated to be about 0.02 % dk/k/second. This estimate assumes simultaneous withdrawal of the regulating rod and the highest-worth shim safety rod over a region that includes the highest-worth portion of the differential rod worth curve. Withdrawal speed for the regulating rod and shim rod is based on measured values for the control rod drive motors moving the control rods under normal conditions.

4.4.2 Cooling System Control

Activation of the cooling system during reactor operation can result in the introduction of cooled water to the reactor pool. The negative temperature coefficient of reactivity for the OSURR core results in a net positive reactivity insertion under these conditions, which can lead to an increase in reactor power. Therefore, the cooling system controls are designed to limit the rate at which cooled water is initially added to the reactor pool, thereby limiting the magnitude of the reactivity insertion from the addition of cooled water to the reactor pool.

Reactivity effects resulting from water (moderator) temperature changes occur over relatively long time periods. Thus, any possible effects on reactor power will exhibit somewhat long reactor periods. Under normal operating conditions, the rate of reactivity insertion and its absolute magnitude will be within the range of the reactor control system when operated in either manual or servo (automatic) control modes.

4.4.3 Control Rod Worths

For the reference core configuration, the following control rod worths are estimated from full-core Monte Carlo calculations using the VIM code:

Most Reactive Safety Rod	3.42 % dk/k
All Shim Safety Rods	7.97 % dk/k
Regulating Rod	0.48 % dk/k

Maintenance of a shutdown margin of at least 1% dk/k will require that the total rod worth of the second and third-most reactive shim safety rods be at least 1% dk/k. The maximum worth of the regulating rod will not exceed 0.70 % dk/k.

As of December, 1999, the measured total worth of all controls rods was 8.73 % dk/k for core LEU-4. The worth of the regulating rod was estimated to be 0.57% dk/k, while the total worth of the second and third-most reactivity control rods was 4.81 % dk/k. Using the control rod worth calibration curve for the most reactivity control rod, and that of the regulating rod, the maximum rate of reactivity addition, as of December, 1999, for core LEU-4, is estimated to be approximately 0.17% dk/k/second.

4.5 Temperature and Void Coefficients of Reactivity

The reference LEU core was analyzed to estimate the amount of reactivity inserted under conditions of elevated core and moderator temperature and the formation of voids in the active core region. As expected, both reactivity effects are negative, which will tend to induce a decrease in reactor power.

The temperature coefficient of reactivity is estimated to be approximately -6.3×10^{-3} % $\Delta k/k$ per degree Centigrade of temperature rise. Thus, as the reactor is operated at power levels which will induce significant changes in temperature, reactivity must be added to compensate for the negative reactivity effects of the temperature rise. Given the temperature coefficient of reactivity estimated above, and the worth of the regulating rod shown in Section 4.4.3, the regulating rod alone can compensate for a temperature rise of about 76 °C, which is far greater than the temperature rises expected under normal operating conditions.

The reactivity effects of void formation were also analyzed. However, depending on the assumptions made in the analysis, a range of possible void coefficients of reactivity were obtained for the reference LEU core. Therefore, an approach similar to a sensitivity analysis was taken for analyzing core performance under normal (and accident) conditions, where the highest and lowest estimates for void coefficient were used. The lowest void coefficient of reactivity estimate is -0.18 % $\Delta k/k$/%void, and the highest is -0.45 % $\Delta k/k$/%void. As indicated by the negative quantities, void formation will tend to reduce reactor power. Under normal conditions, however, void formation is not expected to occur anywhere within the active core boundary. For comparison, the THOR reactor had a void coefficient measured to be about -0.22 % $\Delta k/k$/%void.

Initial measurements of the LEU-fueled OSURR indicated a void coefficient of -7.92×10^{-3}/1% void across the active core. The moderator temperature coefficient of reactivity was estimated to be about -6.2×10^{-5} dk/k/°C.

4.6 Neutronics

Estimates for delayed neutron fraction and prompt neutron lifetime were made. The delayed neutron fraction for the reference LEU core is estimated to be 0.766%, and the prompt neutron lifetime was calculated to be 6.6×10^{-5} seconds. These quantities have been measured for the HEU-fueled OSURR as 0.638% for the delayed neutron fraction and 8.12×10^{-5} seconds for the prompt neutron lifetime. The differences between the HEU-fueled OSURR measured values and the LEU-fueled OSURR predicted values are expected as a result of changing the enrichment of the uranium and the loading of the fuel elements.

Reactors of similar design exhibit prompt neutron lifetimes comparable to this estimate. For example, the Borax I core had a measured neutron lifetime of 6.5×10^{-5} seconds.

Following the loading of the initial usable LEU core geometry, the transfer function of the OSURR was measured using a reactivity oscillator. This device allows measurement of the β/ℓ cutoff frequency. Using this with an assumed value for β of 0.007, we estimate the prompt neutron lifetime to be 70 microseconds.

4.7 Effects of Core Geometry

Materials and structures forming and surrounding the core can have a reactivity effect. The simplest example of a core geometry change which has a reactivity effect is the addition or removal of a fuel element. Other examples of core geometry changes causing reactivity effects include addition or removal of a graphite irradiation element, changes in experimental facilities (e.g., flooding of a beam port or the CIF, or moving a dry tube near the core), and movement of the control rod locations. These effects will be further discussed in the following paragraphs.

To estimate the effect of adding or removing a fuel element from the reference LEU core (Figure 4.4), calculations were performed to determine the effect on excess reactivity of removing a standard fuel element from position 5A. The effect of adding a fuel element to the reference LEU core, in position 5C, was also estimated. The results are as follows:

Reference LEU Core	3.27% $\Delta k/k$ excess reactivity
Adding Element At 5C	5.01% $\Delta k/k$ excess reactivity
Removing Element From 5A	2.32% $\Delta k/k$ excess reactivity

In the above analysis, note that position 5C is a relatively higher worth position than position 5A, since 5C is between two control rods, and 5A is at a corner of the core. Thus, the two cases are not precisely equivalent. However, a general indication of the effect of the fuel elements can be noted.

Section 4.1 noted that the reactivity effect of removing all graphite Irradiation elements from positions 6A-6E is about -0.85% Δk/k. Thus, for positions along grid plate row 6, each graphite irradiation assembly has an average reactivity worth of 0.2% Δk/k. However, the absolute reactivity worth of a given graphite irradiation assembly will depend on its position on the grid plate. To estimate the effect of a graphite irradiation element at other positions on the grid plate, several cases were analyzed involving replacement of fuel elements with graphite irradiation elements. The results of one case are as follows:

Baseline Case:

Vacant Positions: 6A-6E, 1A, SC
Graphite Element Positions: None
Excess Reactivity: 3.27% Δk/k

Case M:

Vacant Positions: 6A-6E, 1A
Graphite Element Positions: 5C
Excess Reactivity: 3.62% Δk/k

Thus, the reactivity effect of adding a graphite irradiation element to previously vacant position 5C is about 0.35% Δk/k. Again, position 5C, being between two control rods, is a relatively high worth position, and the effects of adding graphite irradiation elements at various positions on the grid plate may be more or less than this estimate, depending on the position and the surrounding structures and materials.

It should be noted that reactivity insertions resulting from movement of fuel elements or graphite irradiation elements will not occur during reactor operations, since administrative controls impose limits on the times at which core modifications may be made. These administrative limits require that the reactor be shutdown and all control rods inserted prior to making changes in the core geometry.

Reactivity effects of the conditions of experimental facilities have been measured for the HEU-fueled OSURR, with the core configured in the "Standard Core No. V geometry. This configuration has all graphite elements in place in positions 6A-6E, standard HEU fuel assemblies in all other positions, except position 3C, which has a water-filled flux trap and CIF assembly in place, and positions 2B, 2D, 4D, and 4B, which have in their positions shim safety control rods 1 through 3, and the regulating rod, respectively. Reactivity effects of different conditions in Beam Port 1 were measured as follows:

Standard Plug in Place	0.645% $\Delta k/k$ excess reactivity
Polyethylene Plug	0.585% $\Delta k/k$ excess reactivity
No Plug in Place	0.185% $\Delta k/k$ excess reactivity

Thus, the insertion of a standard beam port plug (aluminum, masonite, and concrete) in Beam Port 1 adds about 0.46% $\Delta k/k$ reactivity. Since the reference LEU core will have somewhat different geometry (fewer fuel elements), the reactivity effects of different beam port configurations will vary from these estimates.

It is unlikely that reactivity insertions from beam port shielding plug changes will occur during operation of the reactor, since administrative controls on experiments would likely preclude such changes because of the possibility of high dose rates existing near the exits of the beam ports.

4.8 Heat Transfer Characteristics

4.8.1 Coolant Outlet Temperature

Assuming that power generation is uniform throughout the core, an initial estimate of coolant outlet temperature can be obtained. For a given flow channel, the available coolant volume is obtained from:

$$V = Lwt,$$

where
L = active fuel length (24", or 60.96 cm)
w = effective coolant channel width (2.624", or 6.66496 cm)
t = channel thickness (0.116", or 0.29464 cm).

The fuel length specified above is the nominal length of the active portion of the fuel (i.e., the meat). The effective coolant channel width takes into account the loss of heat transfer surface area at the edges of the fuel where the fuel plate is swaged into the side plates.

Using these data, a channel fluid volume of 119.71 cm^3 is estimated. A typical OSURR reference core would have [redacted] standard fuel elements and 4 control rod fuel elements. The standard fuel assemblies have 17 effective coolant channels, while the control rod fuel elements have 10 channels. These numbers account for the dummy plates used in each type of element, within which no power generation occurs. A total of 329 active coolant channels gives a total coolant volume of 3.938×10^4 cm^3. Using a power density given by:

$$D = P/V$$

with
$P = 5 \times 10^5$ watts
$V = 3.938 \times 10^4$ cm

a power density of 12.696 watts per cc of coolant is obtained, or, on an individual coolant channel basis, a power density of 1.52 kilowatts per channel.

The mass flow rate of coolant through the core can be estimated from

$$m = \rho v A,$$

where m = mass flow rate
ρ = coolant density
v = average coolant velocity through the core, and
A = effective flow area

Analysis indicates that an average coolant velocity of 6.5 cm/sec is reasonable for the OSURR core operating in a natural convection cooling mode at 500 kilowatts of thermal power. For a single coolant channel, the effective flow area is the product of the channel gap (0.116 inches) and the effective channel width (2.624 inches). Thus, A is taken to be 0.3044 in^2, or 1.964 cm^2. Using the mass flow rate equation above, a total mass flow rate of water through the core of 12.76 grams/sec., or 101.3 pounds mass per hour is estimated.

Using the heat generation in a coolant channel of 1.52 KW/channel estimated earlier, a channel heat generation of 5186.24 BTU/hr per channel is estimated (using a conversion factor of 3412 BTU/hr/KW). In terms of temperature differences:

$$Q = mC_p\ (T_2 - T_1)$$

where Q = channel heat generation (5186.24 BTU/hr/KW)
m = mass flow rate (101.3 pounds mass/hr)
C_p = 1 BTU/lbm/°F
T_1 = Coolant Temperature at Core Inlet (assume 75 °F)
T_2 = Coolant Temperature at Core Outlet

Substituting appropriate values in this equation and solving it for T_2 yields a coolant temperature of 126.2 °F (52.3 °C) at the outlet (top) of the core.

This estimate is an average temperature for the core as a whole. The coolant temperature at the outlet of each individual coolant channel will vary somewhat as a result of neutron flux and coolant velocity distributions through the core, but in no case will onset of nucleate boiling occur anywhere in the core.

Operational history of the OSURR up to December, 1999, indicates that the core outlet temperature generally varies in the range from 120 °F. to 130 °F., depending on core inlet temperature at the time of the run. Variations in outlet temperature, as measured by the core outlet temperature thermocouple, are thought to be a result of turbulent flow near the exit of the core shroud, as induced by the suction flow of the cooling system circulation pump. The core outlet temperature rarely exceeds the 130 °F. upper observed value, and at no time has any steam void formation been observed in or around the OSURR core. Core inlet temperature is limited by engineered safety features (reactor trip) to values less than 95 °F., so it is unlikely that the core outlet temperature would ever significantly exceed 145 °F., and would likely be less than this under most operational conditions. These are significantly less than the boiling point of light water.

4.8.2 Fuel Plate Surface Temperature

The temperature of the fuel plate surface (cladding temperature) can be estimated from the temperature of the water leaving the core. First, the volume of the active fuel in a single plate is given by:

$$V = Lwt$$

where
- V = active fuel volume
- L = length of active fuel (24" = 60.96 cm)
- t = thickness of fuel meat (0.020" = 0.0508 cm)
- w = width of fuel meat (2.47" = 6.2738 cm)

Substituting known values yields a plate fuel volume of 19.43 cm^3. Assuming that the OSURR core has [redacted] standard fuel elements, each with 16 fueled plates, and 4 control rod fuel elements, each with 10 fueled plates, a total fuel volume in the core of 6062.7 cm^3 is estimated.

~~The average power density in the fuel at 500 kilowatt operation is~~ given by:

$$q''' = P / V$$

with
- q''' = power density in watts/cm^3
- P = total power generated in the core (500 kilowatts)
- V = fuel volume in the core (6062.7 cm^3)

Using this equation with the appropriate data yields a power density of 82.47 watts/cm^3. Now, the difference in temperature between the coolant fluid and cladding surface temperature can be estimated from:

$$T_S - T_{f1} = (r_f^2 q''')/(2h_S\,[r_f+t_C])$$

where
- q''' = power density (82.47 watts/cm^3)
- t_c = cladding thickness (0.15" = 0.0381 cm)
- r_f = fuel radius (0.01" = 0.0254 cm)
- h_s = conductive constant of coolant fluid (see below)
- T_s = surface temperature of the fuel plate

T_{fl} = coolant fluid temperature (126.2 °F = 52.3 °C)

In the above equation, a value for h_s must be assumed. Analysis indicates that h_s for water can vary from 0.05 watts/cm²/°K up to 3 or 4 watts/cm²/°K, depending on conditions of the fluid. For conservatism, a value of 0.05 watts/cm²/°K is assumed, since this will yield the higher cladding temperature. Using these data, a temperature difference of 8.4 degrees is obtained. Solving the equation for T_s yields a cladding surface temperature of 56.5 °C (133.8 °F). This indicates that the average fuel plate surface temperature will be far below that required to induce onset of nucleate boiling.

4.9 Fluid Dynamics

4.9.1 Detailed Analysis

Section 4.8.1 considered the effects on coolant outlet temperature based on energy produced in the core and the total core flow rate. This approach, based on a fairly simple conservation of energy analysis, leads to initial estimates of the coolant temperature leaving the core. A more detailed analysis takes into account the balance of forces resulting from buoyancy of heated water and frictional losses, as well as power distribution through the core and differences resulting from the unheated, half-heated, and fully heated flow channels.

In this analysis, the following assumptions are made:

(1) water is removed from the plenum above the core at exactly the rate required to prevent zero net flow out of the top of the core (i.e., the suction rate from the primary coolant pump matches the core flow rate),

(2) the bulk temperature of water in the plenum above the core is constant,

(3) all pressure drops are negligible except those at the entrance to the core, through the core, and at the core exit, and

(4) the bulk pool temperature is 77 °F at all locations in the pool except in the core and plenum.

Also, assume that the active length of the core is 24 inches, with an extrapolation length of 4 inches. Now, the heat distribution over the length of the core can be estimated by:

$$q' \propto \sin(1.208(x + 0.3))]$$

where x = vertical position along the core. Thus, we can write:

$$P_i = C\int_0^2 \sin[1.208(x + 0.3)]dx$$
$$= C[(-1/1.208)\cos(1.208(x + 0.3))]$$

Solving this for C:

$$C = P_i/1.548$$

Therefore,

$$q' = (P_i/1.548)\sin[1.208(x + 0.3).$$

This leads to an estimate of the axial power peaking factor:

$$q'_{max} = q'(1) = P_i/1.548$$

The average heat flux is just:

$$q'_{ave} = P_i/2$$

So,

$$q'_{max}/q'_{ave} = (P_i/1.548)/[P_i/2]$$
$$= 2/1.548$$
$$= 1.292$$

Now, we also assume that the flow through the core is laminar flow. Given that the OSURR core will be primarily cooled by natural convection, this is a reasonable assumption. Entrance effects at the core inlet cause a sudden constriction in the coolant flow. The ratio

$$A_2/A_1 = (0.116)/(0.116+0.050) = 0.699$$

This leads to the concept of the equivalent flow diameter:

$$D_2 / D_1 \approx \sqrt{A_2 / A_1} = 0.836$$

The losses at the entrance can be calculated as:

$$\Delta P \approx (0.12)\,(\dot{m}_i^2) / (\rho A^2 2g)$$

At the core inlet, the temperature is assumed to be 77 °F, so the density is 62.24 lbm/ft^3, and thus

$$\Delta P \approx 2.31\times10^{-12}(\dot{m}_i^2/ A^2)$$

This calculation can be done for each channel, including unheated, and partially and fully heated channels.

At the core exit, losses are also encountered. The estimate of these losses is from:

$$\Delta P = (1 - [A_1 / A_2])\,(V^2 / 2g)$$

$$= (3.59\text{x}10^{-10})\,(\dot{m}_i^2 / A^2)$$

Again, this calculation must be done for all flow channels.

In order to use a force balance approach, the temperature of the fluid as a function of vertical position must be estimated. This can be obtained from:

$$T(x) = T_{inlet} + \int_0^x q'\,du / \dot{m}$$

with

$$q' = (P_i/1.548)\sin(1.208(x+0.3)).$$

Performing this integration gives:

$$T(x) = T_{inlet} + (P_i / \dot{m}_i)\,[0.5 - \cos(1.208(x + 0.3))/1.87].$$

Using the temperature distribution, the viscosity and density properties of the water along each coolant flow channel can be calculated. Thus, losses along each channel, including unheated, partially heated, and fully heated channels can be calculated. Using tabulated data, and taking temperature increments of 10 °F from 70 °F through 140 °F, a table was constructed of water viscosity as a function of temperature. The data were fit to a quadratic equation in T, which yields the following curve:

$$v = 2.5\text{x}10^{-6}T^2 - 8.0\text{x}10^{-4}T + 8.17\text{x}10^{-2}$$

This fit yields very small deviations for v with tabulated values over the range of temperature specified.

Estimation of losses in unheated flow channels is relatively straightforward, since

$$\Delta P = 0.1085\,\dot{m}\,2v$$

where v is the value for water viscosity at 77 °F (core inlet temperature). This yields

$$\Delta P = 0.00758\dot{m}.$$

For fully or partially heated channels, the calculation is more complex, since

$$\Delta P = 0.1458\dot{m}\int_0^2 v\,dx$$

and the function

$$v = 2.5\times10^{-6}T^2 - 8.0\times10^{-4}T + 8.17\times10^{-2}$$

must be substituted for $\dot{v}$, and

$$T = (T_{inlet} + P_i/2\dot{m}_i) - (P_i/1.87m_i)\cos(1.28(x + 0.3))$$

must be substituted for T. This leads to cumbersome expressions for channel losses, but these can be computed numerically for unheated, and partially and fully heated channels. An additional refinement considered the fully heated channels to be divided into "hot" and "cool" fully heated channels, with the hotter ones being at or near the peak of the heat flux.

Doing these calculations using a force balance method leads to estimates for core outlet temperatures both with and without a plenum above the core. The following summarizes these results:

Outlet Temperature (°F)	Mass Flow Rate (lbs/hr)	Plenum Height (feet)
133.2	30399.3	None
116.8	42899.5	1
117.5	42144.1	1

These results clearly show the "chimney" effect of adding a plenum above the core, i.e., the flow rate increases, with correspondingly lower outlet temperatures. Based on these calculations, a cooling system was designed assuming a system inlet temperature of 122 °F, with a return temperature of 86 °F. The simplest case, that with no plenum above the core, leads to a temperature increase of 56.2 °F across the core, with a mass flow rate of about 30400 pounds per hour.

4.9.2 Simplified Analysis

A simplified approach to fluid flow through the core considers the average properties of the coolant (viscosity, density) when flowing through the core. The analysis shown in Section 4.9.1 breaks the problem down to individual groups of coolant channels, which leads to a more complex but potentially more accurate analysis. This simplified approach begins with considering that:

$$\frac{\Delta P}{L} = \frac{3\mu v}{B^2}$$

where ΔP = pressure drop across the core,
L = height (length) of coolant channel,
μ = coolant (water) viscosity,
v = coolant (water) velocity, and
B = one-half the width of the flow channel.

Knowing that the reference LEU core will have [redacted] standard fuel elements, each containing an effective 17 heated channels, and 4 control rod fuel assemblies each containing 10 heated channels, a total of 329 heated flow channels are available. Using 1,707,215.0 BTU/hr as the energy production rate, an average of 5189 BTU/hr/channel is generated.

Now, begin by assuming that the temperature increase across the core is approximately 56 °F, which leads to a mass flow rate of:

$$\dot{m} = (5189 \text{ BTU/hr/channel})/(56\ ^\circ\text{F})$$

$$= 92.66 \text{ pounds/hr.}$$

Also, assuming that the core inlet temperature is 77 °F, average temperature is thus

$$T_{ave} = (77 + 56/2)\ ^\circ\text{F} = 105\ ^\circ\text{F}.$$

The density of water at this average temperature is 61.93 lbs/ft^3, and its viscosity is 1.354×10^{-5} lbf-sec/ft^2. The flow area of a coolant channel is:

$$A = wt/L$$

where w = channel width (2.625"),
t = channel thickness (0.116"), and
L = channel length (24").

The flow area is thus calculated to be 0.00211 square feet.

The volumetric flow rate is given by

$$Q = \dot{m}/\rho$$

$$= (92.66/61.93)\ ft^3/hr$$

$$= 1.496\ ft^3/hr.$$

The coolant velocity is estimated from:

$$v = Q/A = (1.496/0.00211)\ ft/hr$$

$$= 709.1\ ft/hr = 0.197\ ft/sec.$$

Now, using

$$\Delta P = (3\mu v L/B^2)$$

and the appropriate constants given or estimated above, the pressure drop across the core is calculated:

$$\Delta P = (3)(1.354x10^{-5})(0.197)(2)/(0.00483^2)\ lbf/\ ft^2$$

$$= 0.685\ lbf/ft^2$$

Now, we invoke the force balance principle, since the buoyancy forces can be taken as:

$$\gamma_{inlet} = 62.24\ lbf/ft^3 \text{ (core inlet), and}$$

$$\gamma_{ave} = 61.93\ lbf/ft^3 \text{ (average across core).}$$

Since the height of the core is 2 feet, we can estimate the pressure drop across the core from the net buoyancy force as:

$$\Delta P_b = 2(62.24-61.93)\ lbf/ft^2$$

$$= 0.620\ lbf/ft^2$$

where ΔP_b is the net buoyancy force from water heated in the core. Note that this different from the earlier estimate of 0.685 lbf/ft^2 by 0.065 lbf/ft^2. This Implies that the averaging of properties of water and temperature difference across the core did not give a net balance of forces. The problem can be approached iteratively by assuming another value for the temperature change. Assuming a ΔT of 58 °F, and re-working the calculations above leads to a force imbalance of only 0.015 lbf/ft^2. This allows a linear extrapolation of temperature difference required to achieve a force balance. This extrapolation yields a ΔT of 58.6 °F to attain a force balance. Use of linear extrapolation is not exact since the coolant properties do not vary linearly with temperature. However, over a small range of temperatures in which the properties of water are well-behaved, a linear estimation yields a reasonably accurate result.

The estimate of a 58.6 °F temperature rise across the core for this simplified approach compares well with the 56.2 °F rise estimated with the more complex model developed in Section 4.9.1, thus giving confidence in these estimates. Assuming the higher value for temperature rise still leaves the core outlet temperature far below that required to initiate nucleate boiling in the hot channel.

4.10 Chapter 4 Bibliography

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Table 4.1: Summary of Reactor Data

<u>Reactor Owner</u>	The Ohio State University
<u>Reactor Location</u>	Columbus, Ohio
<u>Reactor Type</u>	Open Pool, MTR
<u>Reactor Materials</u>	
Fuel	Uranium (19.5% ^{235}U)
Moderator	High Purity Light Water
Reflector Materials	Reactor-Grade Graphite and Water
Primary Coolant	High Purity Light Water
Secondary Coolant	Ethylene-Glycol and Water
Shim Safety Control Rods (3)	Boron-Stainless Steel
Regulating Rod (1)	Stainless Steel
Shielding	Barytes Concrete and Water
<u>Core Dimensions</u>	
Active Core Cross Section	15″ by 15″
Active Fuel Length	24″
Grid Plate	5 x 6 Array (3-Inch Positions)
~~<u>Thermal Characteristics</u> [redacted] Standard Elements, 4 Control Elements)~~	
Maximum Rated Thermal Power	500 kilowatts
Pulsed Power	Not Permitted
Total Power Peaking Factor	1.8
Maximum Heat Flux	43182 W/m^2
Specific Power (Clean, Cold Core)	122 watts/gram ^{235}U
Core Coolant Flow Velocity	5.0 cm/sec (average channel) 6.1 cm/sec (hot channel)
Total Core Flow	3285 cc/sec

Table 4.1: Summary of Reactor Data (continued)

Maximum Water Temperature in the Hot Channel	62 °C
Maximum Cladding Temperature in the Hot Channel (Full Power)	94.52 °C
Core Water Inlet Temperature	20 °C
Core Water Outlet Temperature	52.6 °C (average channel) 58.1 °C (hot channel)
Margin To Nucleate Boiling ($T_{superheat}-T_{surface}$)	20 °C
Core Pressure Drop	30 pa
<u>Nuclear Characteristics At Maximum Rated Thermal Power</u>	
Maximum Thermal Neutron Flux (0.025-0.625 eV)	1.28×10^{13} nV
Average Thermal Neutron Flux	4.66×10^{12} nV
Maximum Epithermal Neutron Flux (0.625-5.53 keV)	3.95×10^{12} nV
Average Epithermal Neutron Flux	2.89×10^{12} nV
Maximum Fast Neutron Flux (> 5.53 keV)	9.13×10^{12} nV
Average Fast Neutron Flux	6.36×10^{12} nV
Minimum Critical Mass	3.57 kg. ^{235}U
Typical Core Loading (22 Elements)	4.1 kg. ^{235}U
Total Control Rod Worth	8.45 % $\Delta k/k$
Minimum Shutdown Margin	1.00% $\Delta k/k$
Temperature Coefficient	-6.3×10^{-3}% $\Delta k/k/°C$
Core Average Void Coefficient	-0.18% $\Delta k/k$/1% void (min.) -0.45% $\Delta k/k$/1% void (max.)

Table 4.1: Summary of Reactor Data (continued)

Prompt Neutron Lifetime	6.6×10^{-5} seconds
Delayed Neutron Fraction	0.766%
<u>Reactivity Requirements</u>	
Equilibrium Xenon	1.00% $\Delta k/k$
Temperature Feedback	0.20% $\Delta k/k$
Burnup, Fission Product Buildup	0.50% $\Delta k/k$
Experiments	0.70% $\Delta k/k$
Reactor Control	0.20% $\Delta k/k$
TOTAL EXCESS REACTIVITY	2.60% $\Delta k/k$
<u>Specific Design and Operating Characteristics</u>	
Void and Temperature Coefficients	Negative
Overpower Trip Limits (Lowest)	120%
Maximum Excess Reactivity	2.60% $\Delta k/k$
Maximum Regulating Rod Worth	0.48% $\Delta k/k$
Minimum Shim Safety Rod Worth	4.55% $\Delta k/k$
Maximum Shim Safety Rod Worth	7.97% $\Delta k/k$
Total Control Rod Worth	8.45% $\Delta k/k$
Maximum Movable Experiment Worth (Any Single Movable Experiment)	0.40% $\Delta k/k$
Total Movable Experiment Worth	0.60% $\Delta k/k$
Total Experiment Worth In Core	0.70% $\Delta k/k$
Minimum Startup Count Rate	2 counts/second

Table 4.1: Summary of Reactor Data (continued)

Control Systems

Shim Safety Rods:	
Number	Three
Composition	Boron-Stainless Steel
Boron Weight Percent	1.5
Cross-Sectional Shape	Grooved Oblong
Poison Section Length	26 Inches
Regulating Rod:	
Number	One
Composition	Stainless Steel
Cross-Sectional Shape	Smooth Oblong
Control Rod Stroke	24″
Nominal Drive Speed	11 cm/min
Scram Response:	
Nominal Rod Drop Time	500 Milliseconds
Nominal Magnet Release Time	30 Milliseconds

Fuel

Type	Flat-Plate MTR
Fuel Plate Characteristics:	
Loading	[redacted] ^{235}U/plate
Thickness	0.050″
Clad Material	6061 Aluminum
Clad Thickness	0.015″
Fuel Meat Thickness	0.020″
Fuel Alloy	Uranium-Silicide (U_3Si_2)
Uranium Enrichment	19.5%
Active Fuel Length	24″ (Nominal)
Fuel Elements:	
Number	21 (Standard) 5 (Control Rod) 4 (Partial)
Plates/Standard Element	16 Fueled 2 Dummy

Table 4.1: Summary of Reactor Data (continued)

Plates/Control Rod Element	10 Fueled 2 Guide Plates (Dummy)
Plates/Partial Element	4, 6, 8, 10 Fueled 14, 12, 10, 8 Dummy
Side Plate Material	6061 Aluminum
End Box Materials	6061 Aluminum
Fastening Methods	Swaging and Welding
<u>Experimental Facilities</u>	
Beam Ports:	
Number	2
Location	North Core Edge
Size	6″ Inner, 7″ Outer
Length	6′ 6″
Orientation	Perpendicular and Angle To Core
Pneumatic Tube:	
Number	1
Location	North Core Edge Above Beam Ports
Diameter	3″
Main Thermal Column:	
Extension Location	West Core Edge
Dimensions	4′4" x 4′4" x 6′
Stringer Materials	Reactor-Grade Graphite
Stringer Dimensions	4" x 4" x 57" (Removable)
Core End Shielding	Lead (3″)
Outer Face Boral Shielding	¼ Inch Boral Plate
Shield Door	2′ Barytes Concrete
BSF Pool Thermal Column:	
Extension Location	South Core Edge
Shutter Boral Shield	¼ Inch Boral Plate
Shutter Cadmium Shield	0.030" Cadmium Sheet
Column Shield	Lead (3 Inches)

5.0 Auxiliary Systems

5.1 Introduction

Several auxiliary systems aid in safe reactor operation. They provide the necessary support for a variety of operations including lighting, cooling, heating, electricity, etc. This chapter will discuss the features of these systems and their use.

5.2 Communication

Communications are provided at various places around the reactor building. The communications systems include both audio and visual indications.

5.2.1 Control Room Intercom System

A two-way intercom, located on the right-hand control console, provides the reactor operator with means to talk and listen to several stations located throughout and outside the Reactor building. These intercom stations also feature a local push-to-talk button that allows experimenters (or other personnel) to talk to the operator.

5.2.2 Building Phone System

All phones located in the building are connected to the university for both internal and external calls. The control room has its own dedicated line.

5.3 Lighting Systems

Three lighting systems are located inside and outside of the building to supply light for the personnel needs and safety requirements.

5.3.1 Normal Interior Lighting

Several high-intensity overhead lights are located in the reactor bay and machine shop area. Additionally, a combination of fluorescent and incandescent lighting is provided in all office, classroom, and storage areas.

5.3.2 Emergency Interior Lighting

In the event of power failure, four lamps, two at the front of the building and two at the rear, are activated. Each lamp has its own battery with a capacity sufficient for operation up to one hour. Additionally, several flashlight are located around the building for use in an emergency, power failure, experiments, etc.

5.3.3 Exterior Lighting

Four outdoor lamps are located on each corner of the building. These lights are automatically controlled by a light sensor affixed to the top of each lamp. There is also a light above the front entrance.

5.4 Building Services

Several auxiliary systems are provided for use by reactor personnel and experimenters. These include electricity (120V and 230V outlets), city water, 100# compressed air and natural gas.

5.4.1 Electrical, Water, Air and Gas

Electricity is provided via several 120V outlets around the building. Breaker panels located in the machine shop area and adjacent to the east service room provide remote isolation of these electric circuits. Additionally, a few 240V service outlets are also provided.

Standard city water is provided for experiments and sanitary facilities. Although demineralized water is available (from the reactor demineralizer), it is usually not provided for most experimenters, but is mainly dedicated to the reactor system.

The university supplies compressed air (100#) and natural gas for the building. Several "standard" laboratory outlets are provided throughout the building.

5.4.2 HVAC

Heating and ventilation are provided throughout the building. The rooms, located in the southeast and southwest corners of the building, house the associated blowers, motors, heating units, plenums, and associated structures. Additionally, overhead gas heaters located above the front service door and rear entrance provide heat to the reactor bay.

Air conditioning is provided by an outside unit in conjunction with the ventilation system for the east offices in the building. Several auxiliary air conditioners are provided for individual rooms, such as the control room, that have extra cooling / humidity-control needs. The control room (205) and room 104 have window-type air-conditioning units that exhaust to the outside environment. Room 109 has a window-type unit exhausting into the reactor bay area.

All building heating and air conditioning systems using air circulation fans are connected to the emergency switch located in the control for shutting off exhaust fans. This feature allows partial isolation of the building interior environment from the outside atmosphere, and may be used under certain emergency conditions.

5.5 Bridge Crane

An overhead crane, running on parallel tracks (north-south) provides capabilities for fuel handling, thermal column access, experiment movability, and other operations requiring movement of heavy objects. The crane is rated at 3 tons and has a remote control box to allow operation of the crane from ground floor. Several hooks and attachments are provided for use with various experiments and experimental facilities as well as fuel handling.

5.6 Building Alarm System

The reactor building is equipped with a security system in accordance with the NRC-approved security implementation procedures. The details of this system are discussed in the security procedures and are not subject to unrestricted disclosure. The general features of the security system include various devices and subsystems for detection of unauthorized entry, activity in certain restricted areas, and manually-actuated alarm switches.

5.7 Access Control

[redacted]

6.0 Radioactive Waste Management

6.1 Source Term Estimation

6.1.1 Liquid Effluents

Under normal conditions, no liquid is released from the reactor pool, the primary coolant loop, or the secondary coolant loop. Events that result in significant fluid releases from these systems are considered accident conditions and are discussed in Chapter 8.

Liquid-borne radioactive materials in the reactor pool and primary coolant loop eventually pass through the water processing system. The primary radionuclide detected in the resin bed ion exchange demineralizer is ^{24}Na resulting from neutron-alpha reactions with ^{28}Al in the aluminum used in and around the reactor core. Current procedures require changing of the demineralizer cartridge at the point where it cannot maintain pool water conductivity requirements. The replacement procedure allows for holding the spent demineralizer cartridge at the reactor building for several months prior to returning it to the manufacturer for regeneration. This holding period allows ^{24}Na activity to decay to negligible levels.

Thus, under normal conditions, no liquid radioactive effluent will be produced by OSURR operations.

6.1.2 Gaseous Effluents

The primary gaseous radionuclide produced by OSURR operations is ^{41}Ar. This isotope is produced whenever air is in contact with a neutron radiation field. Naturally-occurring ^{40}Ar, which comprises over 99% of all argon, undergoes a neutron capture reaction to produce ^{41}Ar, which decays by beta and daughter product (^{41}K) gamma emission, with a half-life of 1.83 hours. Argon is found in air at slightly less than 1% concentration under STP conditions.

Smaller concentrations of gaseous radioisotopes will also be produced from other activation products in air, experimental procedures, and a slight possibility of very small quantities of fission product gases released into the reactor room environment from dissolved fission product gases in the pool water. However, the quantities of these other sources are very small compared to ^{41}Ar production.

The water in the reactor pool also contains dissolved air. It is assumed that the dissolved air has an argon concentration equal to that found in atmospheric air. Some of this argon will activate and be released from the surface of the reactor pool into the building air.

Fast neutrons (with energies above about 10 MeV) can interact with the oxygen nuclei in the pool water via the neutron-proton reaction and produce the ^{16}N isotope. This nuclide has a very short half-life (7 seconds), so very little of it will reach the surface of the pool because of decay during transit from the core to the surface. Further, if the circulating pumps in the primary loop of the cooling system are on, the water rising from the core will be dispersed into the lower regions of the pool, greatly increasing the effective transit time for a given volume of water from the core to the surface. However, since ^{16}N can contribute to personnel doses as a source distributed on the surface of the pool as well as add to the radionuclide concentration in the air of the building, its production and distribution will be considered in a following section.

6.1.2.1 Argon Production in Experimental Facilities

Production of ^{41}Ar can occur in the two beam ports, the rabbit facility, the central irradiation facility (CIF) tube, dry tubes mounted near the core, and any open stringers in the main or BSF pool thermal columns. When not in use, the beam ports are normally filled with shielding plugs, which effectively reduce to zero the volume of air in the portion of the beam ports normally exposed to neutron radiation. The main and BSF pool thermal columns are also filled with graphite stringers when not in use. Movable dry tubes are stored at a location away from the reactor core when they are not in use. Thus, in many cases, the only sources of ^{41}Ar during routine operations will be the CIF tube and the rabbit tube. A puff-type release can also occur from the rabbit carrier tube after it is withdrawn from the rabbit facility following irradiation and is opened.

Isotope production can be estimated from:

$$A(t) = N\sigma\phi(1 - e^{-\lambda t_i})e^{-\lambda t}$$

where
λ = decay constant
t = time after removal of neutron flux
t_i = exposure time to neutrons
σ = microscopic cross-section for reaction of interest
ϕ = neutron flux in neutrons/cm^2/second
N = total number of target atoms available for activation.

In using this equation, it is assumed that N remains constant; that is, there is no significant "burnup" of the atoms available for the reaction forming the activation product.

In the interests of conservatism, it is assumed that activity buildup of ^{41}Ar is sufficient to achieve saturation. That is, the irradiation time (t_i) is equal to at least five half-lives of the activation product. For ^{41}Ar, with a half-life of 1.83 hours, this corresponds to an activation time of 9 hours. This is conservative in that few OSURR operations will involve irradiations of this length. Also, it is assumed that the release occurs immediately at the end of the irradiation, i.e., $t = 0$. The production equation simplifies to:

$$A = N\sigma\phi$$

where σ and ϕ are defined above. Goldman [1] reports a value of 0.61 barns (6.1×10^{-25} cm^2) for the microscopic thermal neutron capture cross-section of ^{40}Ar.

Values for neutron flux are based on thermal neutron flux measurements obtained in the various OSURR facilities at 10 kilowatt operation. It is expected that these thermal neutron fluxes will remain about the same or lower for an LEU-fueled OSURR core, and will be linear with reactor thermal power. Thus, the measured values for 10 kilowatt operation were multiplied by a factor of 50 to estimate the fluxes for 500 kilowatt operation. Table 6.1 shows data for 41 thermal neutron fluxes in the various experimental facilities. For ^{41}Ar production calculations for the main graphite thermal column and the two beam ports, the flux was taken to be half the peak flux, for reasons to be explained in a later paragraph.

The number of ^{40}Ar atoms available for activation is a function of the volume of air in the experimental facility and the concentration of argon in air under STP conditions. Etherington [2] reports a concentration of 2.5×10^{17} atoms of argon per cubic centimeter of air under STP conditions. Using the isotopic abundance of 0.996 for ^{40}Ar, a concentration of 2.49×10^{17} atoms of ^{40}Ar per cubic centimeter of air at STP is estimated.

The volume of air in each experimental facility must be estimated. Since the experimental facilities of interest are either tubes or rectangular ducts, their volume is given by their cross-sectional area times their effective length. In this case, effective length is taken to be that length over which most of the ^{41}Ar production occurs. For facilities oriented either parallel or through the core, such as the CIF tube, the rabbit facility, and movable dry tubes, the effective length is taken to be the characteristic dimension of the core over which the experimental facility tube passes. For the CIF and dry tubes, the effective length is 24 inches, which is the vertical length of the active fuel part of the core. For the rabbit facility, the effective length is the dimension of the side of the reactor core, which will be no more than 18 inches, since the grid plate is a 5 x 6 array, with each fuel element being 3 inches long. While some ^{41}Ar production occurs in the regions beyond these boundaries, it is not as great as that produced within these regions. Conservatism is added by assuming that the flux through this region is uniform and equal to the peak neutron flux along the effective length, which is the value shown in Table 6.1. In actuality, the average neutron flux in the facility will be lower than the peak value, since the spatial distribution of flux in the vertical and horizontal directions shows a reduction in thermal neutron flux as the edge of the core is approached.

Table 6.1: Measured and Estimated Thermal Neutron Fluxes in Various Experimental Facilities of the OSURR

Facility Description	Measured Thermal Neutron Flux at 10 Kilowatt Power	Estimated Thermal Neutron Flux at 500 Kilowatt Power
CIF	2.02×10^{11}	1.01×10^{13}
Beam Port 1	1.27×10^{11}	6.35×10^{12}
Beam Port 2	9.74×10^{10}	4.87×10^{12}
Rabbit	4.58×10^{10}	2.29×10^{12}
Thermal Column	1.33×10^{10}	6.65×10^{11}
Dry Tubes	4.20×10^{10}	2.10×10^{12}

Notes:

(1) The thermal neutron 2 fluxes shown above are in units of thermal neutrons/cm /second, as reported by Horning [3].

(2) Neutron energy range for thermal flux includes all neutron with energies up to 0.6 eV.

(3) The flux shown for the rabbit facility is also that assumed for calculation involving the rabbit carrier tube.

(4) The flux shown for the thermal column was that measured in the main graphite thermal column at stringer position G-7 (central location). It is assumed to be the same in both the main and BSF graphite thermal columns.

(5) Estimated thermal neutron fluxes at 500 kilowatts were obtained by multiplying the measured flux at 10 kilowatts by 50.

(6) The thermal neutron fluxes shown for the movable dry tubes are assumed to be the same for both the 2" and 4" tubes. The tubes are assumed to be mounted at the northeast corner of the core. The measured values at 10 kilowatts are those reported by Talnagi [4].

The effective length for the beam ports and graphite thermal column stringer positions, which converge at the core in a generally perpendicular direction, was estimated differently. Using a spatial neutron flux distribution measured in the open central stringer position in the main graphite thermal column, a characteristic length was determined for a neutron flux profile in air. This length was taken to be that required to achieve a 90% reduction in the initial (peak) neutron flux at the point closest to the core. For calculation of ^{41}Ar activity, the average neutron flux over this length was assumed to be one-half of the peak thermal flux occurring at the core end of the facility.

Table 6.2 shows the results for effective volumes of the various experimental facilities, and the resulting total number of ^{41}Ar atoms available for activation. Using these results in the production equation leads to the source term estimates shown in Table 6.3.

Assuming an irradiation time sufficient to achieve saturation, the rate of radioisotope production is given by:

$$R = A\lambda$$

where R = rate of isotope production in disintegrations/sec/sec
λ = decay constant, and
A = saturation activity in disintegration/sec calculated from the production equation.

The rates of ^{41}Ar production at saturation in the various facilities are listed in Table 6.3.

The rabbit system can be activated and operate continuously for a set period of time. During operation, ^{41}Ar produced in the effective volume of the rabbit is continuously purged into the reactor room atmosphere, which is exhausted to the outside atmosphere by the building vent fan. The ^{41}Ar effluent concentration can be estimated from the production rate of ^{41}Ar and the volumetric flow rate of the rabbit blower system.

The name-plate capacity of the rabbit blower system is 150 cubic feet of air per minute, or 7.0792×10^{4} cc/second. As shown in Table 6.2, the effective volume of the rabbit facility is 1737.36 cc. This implies a cycle time of 24.54 milliseconds for the effective volume. Assuming a constant neutron flux equal to that shown for the rabbit facility in Table 6.1, and the number of ^{40}Ar atoms available for activation shown in Table 6.3, an irradiation 2 time of 24.54 milliseconds gives an isotope total of 4.22×10^{-2} microcuries. This total must now be multiplied by the total number of cycles of the rabbit effective volume per second (about 41). This results in a production rate of 1.72 microcuries of ^{41}Ar per second in the exhaust of the rabbit blower.

Table 6.2: Calculated Air Volumes in Various Experimental Facilities of the OSURR

Facility Description	Cross-Sectional Area in cm^2	Effective Length in cm	Air Volume in cm^3
CIF	9.65	60.96	588.26
Beam Port 1	190.09	66.04	12553.84
Beam Port 2	190.09	66.04	12553.84
Rabbit	45.60	38.10	1737.36
Thermal Column	103.23	66.04	6817.02
Rabbit Carrier	8.43	13.97	117.77
4" Dry Tube	61.58	60.96	3753.68
2" Dry Tube	21.29	60.96	1298.11

Notes:

(1) When not in use, the two beam ports and thermal column have essentially no effective volume of air since they are filled with plugs or graphite stringers.

(2) When in use, inner volume of the main graphite thermal column is sealed with a boral-aluminum plate.

(3) The effective volume of the rabbit carrier tube may be lower than the value shown if it is stuffed with cotton, as is normally the case when it is used. The volume shown is the maximum available empty volume in the carrier tube.

(4) The effective volume of Beam Port 2 may actually be smaller than the value shown, since it intersects the face of the core at a non-perpendicular angle, which causes a sharper neutron flux gradient, reducing its effective length (see text for definition of effective length).

Table 6.3: Estimated 41 Ar Source Terms for the Various Experimental Facilities of the OSURR

Facility Description	Available ^{40}Ar Atoms [2]	Saturation ^{41}Ar Activity In Microcuries	Saturation ^{41}Ar Activity Production Rate (μcuries/sec)
CIF	1.46×10^{20}	2.43×10^{4}	2.56
Beam Port 1	3.13×10^{21}	1.64×10^{5}	17.25
Beam Port 2	3.13×10^{21}	1.26×10^{5}	13.25
Rabbit	4.33×10^{20}	1.63×10^{4}	1.71
Thermal Column	1.70×10^{21}	9.32×10^{3}	0.98
Rabbit Carrier	2.93×10^{19}	1.11×10^{3}	0.12
4" Dry Tube	9.35×10^{20}	3.24×10^{4}	3.41
2" Dry Tube	3.23×10^{20}	1.12×10^{4}	1.18

Notes:

(1) The values assumed for thermal neutron flux in the two beam ports and the thermal column is one-half that shown in Table 6.1 for 500 kilowatt operation, as discussed in the accompanying text.

(2) For isotope production rate, the half-life was assumed to be 1.83 hours, which gives a decay constant of 1.052 x 10^{-4} seconds.

(3) The estimate shown for the thermal column are assumed to be the same for the main and BSF thermal columns.

(4) All isotopic production calculation results shown above assume the fluxes given in Table 6.1 for 500 kilowatt operation, except as stated in Note 1 above.

6.1.2.2 Argon Production from Pool Water

Estimation of the ^{41}Ar production from dissolved air in the water of the reactor pool begins with a calculation of the exposure time of water passing through the core. Section 4.8 noted that the average coolant velocity through the core is 6.5 cm/second, assuming a 500 kilowatt operating power and natural convection through the core. The length of the active fuel channel is 60.96 cm (24 inches), which gives a coolant transit time of 9.4 seconds, assuming a constant average velocity through the core. This is taken to be the exposure time of the water to the average flux throughout the core.

Based on measurements of the peak thermal neutron flux in the core region at a 10 kilowatt power level, and assuming linearity of thermal flux with reactor power, the peak thermal neutron flux in the core is assumed to be 1×10^{13} neutrons/cm^2/second. Measurements of the peak-to-average thermal neutron flux at 10 kilowatts indicate that the average thermal neutron flux throughout the core will be about 60% of the peak thermal flux, or 6×10^{12} neutrons/cm^2/second.

The volume flow rate of water through the core is the product of the coolant velocity and the total flow area. Assuming a core with 18 standard fuel elements and 4 control rod fuel elements, the total flow area is the product of the flow area of an individual coolant channel and the total number of channels in the core. Section 4.8 noted that the flow area of a single coolant channel is 1.964 cm^2. The total number of flow channels is assumed to be 364 (18 standard elements with 18 flow channel each, and 4 control rod fuel elements with 10 channels each). Thus, the total core flow area is 714.896 cm^2, and the total volumetric flow rate is 4646.8 cm^3/second.

Now, the average out-of-core cycle time is given by:

$$T = V_P / \overline{V}$$

where V_P = total volume of the pool, and
$\overline{V}$ = volumetric flow rate through the core.

If the dimensions given in Section 3.1.3.1 are used for the size of the reactor pool, a volume of 2.223×10^7 cm^3 is obtained. Using the volume flow rate calculated earlier, an out-of-core cycle time of 4783.24 seconds is obtained. This can be thought of as a decay time for ^{41}Ar produced in the water of the pool.

The concentration of argon gas in the pool water can be predicted by Henry's Law. The dissolved concentration of a gas in contact with a liquid is proportional to the partial pressure of the gas and the temperature of the liquid. Dorsey [5] reports values for air at STP conditions in water that allow an estimation of 8.65×10^{15} atoms of ^{40}Ar

per milliliter of water, assuming a water temperature of 25 °C (core inlet temperature).

The saturation activity of ^{41}Ar in the pool water may be predicted from:

$$A = N\sigma\phi(1 - e^{-\lambda t}) / (1 - e^{-\lambda(t+T)})$$

where
N = concentration of ^{40}Ar atoms in the pool water,
σ = neutron capture cross section for ^{40}Ar,
λ = physical decay constant of ^{41}Ar,
ϕ = average thermal neutron flux in the core region,
t = exposure time of water in the core, and
T = average out-of-core cycle time.

Substituting appropriate constants in this equation yields an estimate of 79.02 disintegrations/second/cc. Dividing this estimate by the decay constant for ^{41}Ar gives a calculated density of $7.512x10^5$ atoms of ^{41}Ar/cc.

As water passes through the core it is heated, which reduces the solubility of air in the water. For this calculation, it is assumed that 25% of the dissolved argon is released from the water because of core heating. Some of this released argon will be redissolved as it mixes with cooler water in other regions of the pool. Measurements done at other reactors allow an estimate of 50% redissolving fraction. Thus, the argon available for release to the building air is given by:

$$S_1 = F_1(1 - F_2)N_{41}\overline{V}$$

where
N_{41} = ^{41}Ar concentration in the water at equilibrium,
F_1 = release fraction from heating (assumed to be 25%)
F_2 = redissolving fraction (assumed to be 50%), and
$\overline{V}$ = volumetric flow rate through the core.

Substituting appropriate values in this equation leads to an available release term of $4.36x10^8$ atoms of ^{41}Ar/second. This represents one component in the ^{41}Ar release from the pool water.

Another release term arises from the tendency of dissolved gas at the surface of a liquid to escape to the air across the water-air boundary. Estimating the magnitude of this release term requires calculation of an effective exchange coefficient for argon (exchange coefficient being the amount of gas in a unit volume exchanged at the surface per unit time per unit area).

Other reactor facilities have analyzed this problem and provide possible exchange coefficients that appear to cover a wide range. For example, analyzing the gas exchange at the liquid-gas boundary in terms of the diffusion coefficient of argon gas dissolved in water and the mean-square distance traversed by a molecule, an estimate of 2.35×10^{-3} cm/second is obtained. However, measurements made of the ^{41}Ar activity in the pool water of a TRIGA Mark III and subsequent analysis of these data indicate an exchange coefficient of about 2.9×10^{-4} cm/second. Further, Dorsey [5] reports approximately equal surface exchange coefficients for gases such as air, O_2, and N_2. Assuming that the exchange properties of argon are similar to those Of these gases, an exchange coefficient of about 5.7×10^{-3} cm/second is possible. Note that these estimates vary by almost a factor of 10.

In the interest of conservatism, the largest exchange coefficient (5.7×10^{-3} cm/second) is assumed in this calculation. Using this, the release rate from gas exchange at the surface of the pool is given by:

$$S_2 = 0.93 B N_{41} A_S$$

where N_{41} = concentration of ^{41}Ar atoms in the pool water,
B = exchange coefficient, and
A_S = surface area of the pool (3.646×10^{4} cm^2)

Using this equation a release rate of 1.45×10^{8} atoms/second is obtained. Now, the total source term for ^{41}Ar released from the pool water is obtained by adding this to the previous estimate for dissolved argon:

$$S_{41} = S_1 + S_2$$

$$= (4.36 \times 10^{8} + 1.45 \times 10^{8}) \text{ atoms/second}$$

$$= 6.81 \times 10^{8} \text{ atoms/second.}$$

This is the source term for ^{41}Ar released from the pool water to be used later in estimating doses and isotopic concentrations. The source term assumes 500 kilowatt operation for a time sufficient to attain saturation activity.

6.1.2.3 Nitrogen-16 Production from Pool Water

Section 6.1.2.2 above derived an exposure time of 9.4 seconds for water flowing through the core. The concentration of ^{16}N atoms per cc of water leaving the reactor core can be estimated from the following modified form of the activation product production equation:

$$N = [C \sigma_{np} \phi_f (1 - e^{-\lambda t})] / \lambda$$

where N = concentration of ^{16}N atoms leaving the core,
C = concentration of oxygen atoms in the pool water,
σ_{np} = n-p microscopic cross-section for ^{16}O,
ϕ_f = spectrum-averaged fast neutron flux (0.6-15 MeV),
t = exposure time (9.4 seconds), and

λ = decay constant for ^{16}N ($9.761x10^{-2}$ $second^{-1}$).

It remains to find appropriate values to substitute into this equation.

First, the concentration of oxygen atoms in water can be taken to be approximately $3.3x10^{22}$ atoms of oxygen per milliliter of water. This value ignores dissolved air in the water, as the number of atoms of oxygen from the water molecules far outweighs the number from dissolved air in the water.

Next, the spectrum-averaged microscopic cross-section of ^{16}O is taken to be about 0.021 millibarns, or $2.1x10^{-29}$ cm^2. This assumes an integration range of from 0.6 MeV to 15 MeV incident neutron energy.

Finally, a value must be assigned for the spectrum-averaged fast neutron flux. The cross-section threshold for the n-p reaction in ^{16}O is about 9.4 MeV, but this must be corrected for center-of-mass effects. When these are taken into account, the effective incident neutron energies are about 10.2 MeV. This relatively high threshold energy results in severe limitation of ^{16}N production, since relatively few neutrons in the OSURR in-core neutron spectrum fell above this threshold. 2 Horning [3] reports a value of $8.4x10^{10}$ neutrons/cm^2/second for neutrons above 0.5 MeV (sometimes called the "fission" component) at the central irradiation facility (assumed to be the peak flux) for the HEU-fueled OSURR. Assuming that the LEU-fueled OSURR in-core neutron spectrum is about 15% "harder" (based on experience of other core conversions), and that the power is increased by a factor of 50, the effective neutron flux above 0.5 MeV is assumed to be $4.83x10^{12}$ neutrons/cm^2/second.

Substituting values into the production equation yields an estimate of $2.01x10^6$ atoms of ^{16}N per milliliter of water leaving the reactor core. This is assumed to be an equilibrium concentration, given the very short half-life of the isotope. If the volume flow rate through the core calculated in Section 6.1.2.2 (4646.8 cc/sec) is multiplied by this concentration, a rate of $9.34x10^9$ atoms of ^{16}N per second are released from the core. Multiplying this by the decay constant for ^{16}N and converting to activity units gives a release term of 24.64 millicuries of ^{16}N per second released from the top of the core.

6.2 Liquid Effluent Waste Management

6.2.1 Pool Water Monitoring

Normally, no water is released from the reactor pool. The water level of the pool is visually checked prior to each startup of the reactor. The reactor safety system has a reactor trip function should the pool water fall below a setpoint. The water process system has a water inlet valve controlled by a second water level sensor switch to add makeup water to the reactor pool.

Concentration of gamma-emitting radionuclides in the reactor pool water is checked as part of routine maintenance and surveillance activities. Gamma dose rates above the pool are monitored continuously by an area radiation monitor (ARM). An additional ARM monitors dose rates in the vicinity of the primary heat exchanger. A third ARM monitors dose rates in the area where the reactor pool demineralizer is located. This unit traps most of the ^{24}Na activity contained in the reactor pool water. After each reactor shutdown, the on-contact dose rate of this demineralizer is measured and recorded. If necessary, the area is posted and access to it is controlled.

6.2.2 Secondary Loop Coolant Monitoring

Under normal conditions, no radionuclide concentrations should be present in the secondary coolant. However, to assure this, the dose rate at the corrosion product trapping filter near the secondary coolant pump is surveyed routinely as part of surveillance and maintenance activities.

6.2.3 Liquid Effluent Releases

If significant radionuclide concentration in either the primary or secondary coolant is suspected (above that which is routinely encountered), appropriate procedures are invoked to determine the radionuclide identity, concentration, and release pathway. Based on these tests, necessary corrective action can be taken.

Should release of all or part of the coolant inventory be deemed necessary for repair and/or maintenance activities, appropriate procedures based on the results of the radionuclide assay will be followed to assure compliance with regulations specified in 10CFR, part 20. In most cases, immediate release of the pool water to the city sewage system will be allowed. If not, the fluid will be held until sufficient decay time has elapsed to reduce radionuclide activities to permissible release levels. Otherwise, alternate storage/disposal methods and procedures will be followed.

6.2.4 Cooling System Maintenance Operations

6.2.4.1 Draining, Blowdown, and Purging

At the lowest point in the secondary loop of the cooling system, a trap and drain valve is available for drawing a small sample of secondary coolant. Additional coolant draining can be done at this point to remove larger volumes of fluid.

Maintenance of the secondary coolant chemical and fluid properties can be achieved by intermittent blowdown procedures. A small amount of fluid can be withdrawn from the drain valve and replaced with fresh fluid at the surge tank charging port. The entire volume of secondary coolant can be purged, if necessary, and refilled from the charging port.

An isotopic assay will be performed on all fluid withdrawn from the secondary coolant. If significant quantities of radionuclides are detected, they will be identified and quantified. Based on these data, appropriate procedures will be followed prior to release of any secondary coolant, and tests will be conducted to determine the primary-secondary leakage path. Appropriate repair and maintenance actions can then be taken.

6.2.4.2 Tertiary Loop Effluent Holdup

There is a very small probability that the city water supply used in the tertiary coolant loop could be contaminated by primary coolant. The probability of significant levels of contamination being present is low, since it would require a primary-secondary-tertiary leakage path. However, to assure that tertiary loop radionuclide concentration is known, part of the routine surveillance and maintenance activities will include a radionuclide isotopic assay on tertiary loop effluent during or immediately following a reactor operation involving use of the tertiary loop,

6.3 Gaseous Effluent Waste Management

6.3.1 Effluent Monitoring System

Gaseous radionuclides are detected by an effluent monitoring system. This system extracts a sidestream of the air ejected from the reactor building by the building ventilation fan. The sampled air is introduced to a shielded volume containing a double-sided pancake-type Geiger-Mueller detector. Detector output is counted on a rate meter in the control room, and recorded on a panel-mounted stripchart recorder.

System response is calibrated for ^{41}Ar activity. Detector count rate is noted in the control room. The count rate for the derived air concentration (DAC) of ^{41}Ar is posted at the recorder. Total ^{41}Ar production is tabulated on a yearly basis and compared with permissible limits.

6.3.2 Blower Effluent Monitor

An on-line monitoring system continuously samples the radionuclide content in the rabbit blower exhaust stream. This system takes the exhaust stream through a shielded volume containing a beta scintillator detector. System response for ^{41}Ar is determined and the detector output recorded on a panel-mounted stripchart recorder. Instantaneous ^{41}Ar concentration can be obtained, as well as integrated totals over various recording times.

6.3.3 Release Points

The primary gaseous effluent release point is from the building ventilation fan located at the top of the north wall of the building. This vent is about 30 feet above floor (ground) level. The ventilation fan creates a building exhaust stream that has been measured at a volume flow rate of approximately 1000 CFM, or about 4.72×10^5 cc/second. Assuming a building volume of 70,000 cubic feet, an exchange time of 70 minutes is obtained.

Other release pathways are available, such as through open building doors, windows, the vent fan in the control room, and the fume hood in room 104. However, the total capacity of these release pathways is small compared with the building vent. In addition, the pathways are normally unavailable for release during reactor operation, since building doors are closed, the fume hood is not operated continuously and is used sparingly, and windows, being located in offices and classrooms, are usually closed since these areas are serviced by the building HVAC systems. Only the vent fan in the control room is used a significant amount of the time, and the control room is normally isolated (door is closed) from the main reactor room.

Within the confines of the building, release points for gaseous effluents can be identified. For non-vented experimental facilities, gaseous effluent release is limited since the facilities are either plugged or closed when not in use. Any venting of gaseous radioisotopes will occur at the point where the facility exits from the reactor pool or shielding wall. These points are identified in Table 6.4. The exhaust point for the rabbit blower is located at the end of the exhaust pipe at the top of the building, near the north wall.

6.3.4 Estimated Releases in the Restricted Area

6.3.4.1 Types of Releases

Release of ^{41}Ar from experimental facilities can occur as either a puff or, for a vented facility such as the rabbit or the surface of the pool, a continuous stream. Section 6.1.2 discussed the estimated source terms for ^{41}Ar from experimental facilities and ^{41}Ar and ^{16}N from the surface of the pool, assuming 500 kilowatt operation. The following sections will analyze individual release scenarios and their consequences. These analyses concern releases made within the confines of the reactor building, which is defined as a restricted area.

6.3.4.2 Puff Release from the Rabbit

Saturation levels of ^{41}Ar can build up in the effective volume of the rabbit facility during a long reactor operation with the rabbit system blower turned off. Table 6.3 shows a saturation activity of 16.3 millicuries of ^{41}Ar being in the rabbit volume under these conditions. The release scenario assumes that the rabbit system blower is then

Table 6.4: Release Points of the Various Experimental Facilities of the OSURR

Facility Description	Release Point Location	Release Aperture Description
CIF	Top of Reactor Pool, About 20′ Elevation	Open Aluminum Tube, 1.38" Diameter
Beam Port 1	North Reactor Bay, About 5′ Elevation	Open Port, Flush With Shield, 7.125" Diameter
Beam Port 2	North Reactor Bay, About 4′ Elevation	Open Port, Flush With Shield, 7.125" Diameter
Rabbit	Building North Wall, About 28′ Elevation	Open Aluminum Tube, 3" Diameter
Thermal Column	West Side of Building, 1′-6′ Elevation	Open Stringer, 4" x 4" Square Opening
4" Dry Tube	Top of Reactor Pool, About 20′ Elevation	Open Aluminum Tube, 3.486" Diameter
2" Dry Tube	Top of Reactor Pool, About 20′ Elevation	Open Aluminum Tube, 2.05" Diameter

Notes:

(1) Elevations shown above are referenced to the floor of the reactor building.

(2) Aperture release points for the two beam ports assume that one or the other is open for an experiment, with no shielding plugs or other apparatus restricting access to the interior of the tube.

(3) The aperture for the thermal column assumes a single stringer position completely opened.

activated and the entire activity is instantaneously and perfectly mixed with the 70,000 cubic feet of air in the reactor building.

Diluting the 16.3 millicuries of ^{41}Ar in the building atmosphere leads to a concentration of 8.22x10^{-6} microcuries of ^{41}Ar per cc of air in the building. Table I of Appendix B, 10CFR20, shows a DAC of 3x10^{-6} microcuries/cc for ^{41}Ar, which assumes a 2000 hour working year as an available averaging time. The estimated release for this scenario exceeds the DAC by a factor of about 2.74 times. Thus, an effective exposure time of about 730 hours at the predicted concentration from the puff release would be allowed under DAC yearly restrictions, assuming that the released concentration remains constant over this time. This assumption, however, is very conservative since the concentration will diminish with time as a result of building purging and radioactive decay.

An accurate analysis of building concentration requires consideration of the reduction in concentration as a function of radioactive decay and building purging. Treatment of this problem is similar to analysis of a radioactive material passing through a biological system, where radionuclide concentration decreases after initial introduction because of physical decay and elimination from the system by purging processes. This leads to the concept of the effective half-life, defined as follows;

$$T_e = (T_d \times T_p)/(T_d + T_p)$$

where T_e = effective half-life
T_d = half-life from radioactive decay, and
T_p = half-life from building purging.

From the building exhaust rate of 1000 CFM, and a building volume 70,000 cubic feet, a purging time of 70 minutes is obtained. A relatively simple analysis of the inflow and outflow of the building, assuming an equilibrium condition, shows that the value for T_p in the above equation should be 70 minutes. Using this, and assuming a value of 1.83 hours for the radiometric half-life of ^{41}Ar, an effective half-life of 42.75 minutes is obtained for ^{41}Ar in the reactor building.

Now, if one assumes that most ^{41}Ar activity is lost after five effective half-lives have elapsed (213.75 minutes), the average concentration of ^{41}Ar in the reactor building during this time can be obtained by integrating the time-dependent concentration over this time:

$$C_{ave} = C_0 \int_{t_1}^{t_2} e^{-\lambda t} dt \,/\, (t_2 - t_1)$$

where C_{ave} = average concentration of ^{41}Ar in the Reactor building during the time interval from t_1 to t_2,

C_0 = initial ^{41}Ar concentration,
t_1 = time at the beginning of the release (0 minutes),
t_2 = time at the end of the release (213.75 minutes),
λ = decay constant ($0.693/T_e = 1.62x10^{-2}$ min^{-1}).

Performing this integration leads to the following expression for average concentration:

$$C_{ave} = (C_0/\lambda t)[1-e^{-\lambda t}].$$

Substituting t = 213.75 minutes, and appropriate values for C_0 and λ, an average concentration of $2.3x10^{-6}$ microcuries of ^{41}Ar per milliliter of building air is obtained. This average ^{41}Ar concentration is below the DAC limit of $3x10^{-6}$ µCi/ml. This calculation is conservative in that it assumes a saturation activity in the rabbit effective volume being available for a puff release.

6.3.4.3 Continuous Release from the Rabbit

Section 6.1.2.1 estimated a source term of 1.72 microcuries of ^{41}Ar per second being produced in the exhaust of the rabbit blower at 500 kilowatts. Expressing the concentration buildup of the isotope in the air of the reactor building, accounting for losses from radiological decay and building purging, leads to an equation similar in form to the production of a radioactive material by neutron irradiation, assuming a constant term for isotope production:

$$C(t) = P(1 - e^{-\lambda t})/(\lambda V)$$

where C(t) = time-dependent concentration of ^{41}Ar in the building air at time t after starting the rabbit blower,
P = production rate of ^{41}Ar in the rabbit blower exhaust stream,
t = time the blower has been running,
λ = effective half-life defined earlier, and
V = building volume.

Figure 6.1 shows the time-dependent behavior of the ^{41}Ar concentration in the building air. The concentration approaches an equilibrium value when t becomes large. For conservatism, assume that this equilibrium value has been reached. Substituting appropriate constants in the above equation leads to an estimate of $3.21x10^{-6}$ microcuries of ^{41}Ar per milliliter of air in the building at equilibrium. This about 7% above the DAC limit for a restricted area, which limits the exposure time at this concentration to about 1869 hours per 2000-hour working year. Essentially, the rabbit blower may run with the reactor at full power for most of the working year. It is difficult to postulate a reactor operation involving continuous rabbit blower operation for anything on the order of 1869 hours at a time.

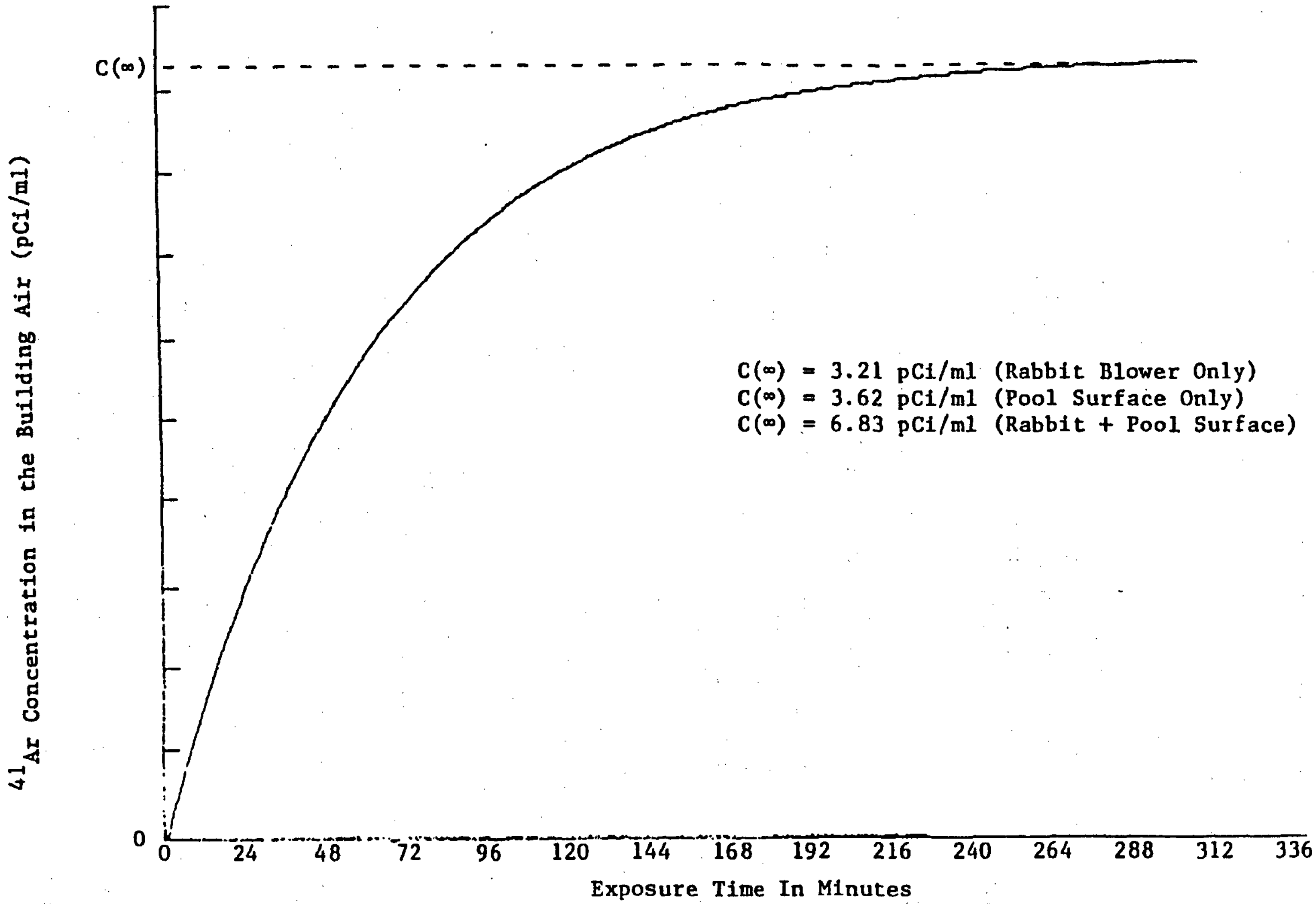


Figure 6.1: ^{41}Ar Production Curve

Another way of analyzing continuous ^{41}Ar release from the rabbit is to set C(t) in the above equation equal to the DAC limit ($3x10^{-6}$ microcuries per milliliter) and solve the equation for the rabbit blower operation time (t). Doing this, a blower operation time of about 10033 seconds is obtained. Thus, the rabbit blower may operate about 2.8 hours before a DAC level of ^{41}Ar is reached in the building air. Most rabbit operations involve blower run times on the order of 20 minutes or less.

6.3.4.4 Puff Release from the Rabbit Carrier Tube

Reactor operations involving rabbit irradiations require use of a carrier tube to insert and remove samples from the rabbit facility. Each irradiation therefore involves a puff release from the carrier tube when it is opened. Since this is a commonly-performed operation, it is analyzed separately here.

Table 6.3 lists a source term for saturation ^{41}Ar in the rabbit carrier tube of $1.11x10^{3}$ microcuries. Assuming that this source is instantaneously and perfectly mixed with the air in the reactor building, a concentration of $5.6x10^{-7}$ microcuries per milliliter results. This is less than the DAC of $3x10^{-6}$ microcuries/ml allowed for ^{41}Ar in a restricted area. Therefore, puff releases from the rabbit carrier tube are allowable even if the activity in the tube has reached saturation levels (which is unlikely in almost all conceivable rabbit operations).

6.3.4.5 Puff Releases from Other Experimental Facilities

Using the source term estimates from Table 6.3, a similar analysis for puff-type releases from the other experimental facilities was done. Table 6.5 shows the estimates for initial and average concentration of ^{41}Ar in the reactor building atmosphere. The average concentration assumes an averaging time equal to five times the effective half-life of ^{41}Ar in the building. The last column shows the relationship between the average concentration in the building air and the allowable DAC limits for a restricted area (factor = average concentration/DAC). Note that puff releases from the rabbit, the rabbit carrier tube, the thermal column, and the 2 inch dry tube result in average ^{41}Ar concentration less than restricted area DAC for releases averaged over five effective half-lives. Thus, under these conditions, no operational limits need to be established.

Table 6.6 shows operational limits on facility use for a 2000-hour work year. The last column in Table 6.6 shows the length of operation time necessary to attain a DAC level of ^{41}Ar in the building, while the other column assumes a saturation level released in the initial puff and predicts the allowed exposure duration at the resulting concentration in the building.

Assuming that a single saturation-level puff release occurs over a time equal to about five effective half-lives (213.75 minutes),

Table 6.5: Estimated ^{41}Ar Concentrations for Puff Releases of Saturation Activities of ^{41}Ar from the Various Experimental Facilities of the OSURR

Facility Description	Initial ^{41}Ar Concentration (μcuries/ml)	Average ^{41}Ar Concentration (μcuries/ml)	DAC Factor For Average Concentration
CIF	1.23×10^{-5}	3.44×10^{-6}	1.15
Beam Port 1	8.27×10^{-5}	2.31×10^{-5}	7.70
Beam Port 2	6.36×10^{-5}	1.78×10^{-5}	5.93
Rabbit	8.22×10^{-6}	2.30×10^{-6}	0.77
Thermal Column	4.71×10^{-6}	1.32×10^{-6}	0.44
Rabbit Carrier	5.60×10^{-7}	1.57×10^{-7}	0.05
4" Dry Tube	1.63×10^{-5}	4.56×10^{-6}	1.52
2" Dry Tube	5.65×10^{-6}	1.58×10^{-6}	0.53

Notes:

(1) The initial concentrations shown above assume instantaneous and perfect mixing with the building air.

(2) The average concentrations shown above assume losses of initial activity from radioactive decay and building purging. The effective half-life is taken to be that derived in the accompanying text. The release is averaged over five effective half-lives.

(3) The DAC factor shown above is calculated by dividing the average concentration by the DAC for ^{41}Ar in a restricted area (3×10^{-6} μcuries/ml).

Table 6.6: Operational Limits and Activation Time Estimates for the Various Experimental Facilities of the OSURR

Facility Description	Limit of Full-Power Hours of Operation Per Year	Activation Time Required To Attain Average ^{41}Ar Concentration Equal to DAC
CIF	1739	329 minutes
Beam Port 1	260	10 minutes
Beam Port 2	337	14 minutes
Rabbit	2608	No Limit
Thermal Column	4545	No Limit
Rabbit Carrier	45000	No Limit
4" Dry Tube	1316	170 minutes
2" Dry Tube	3774	No Limit

Notes:

(1) Where the limits on full-power operation exceed 2000 hours per calendar year, there are no limits on facility use under the assumptions of this analysis.

(2) The limits expressed above assume a puff release at saturation activity levels in a completely voided effective volume of the facility.

(3) The limits on full-power operation are in fact effective exposures times allowed at the estimated concentration. DAC limits are not exceeded if exposure times are less than or equal to this value.

dividing this release time into the calendar year hours of operation limit indicates how many releases of this type may be made during the calendar year. For example, the releases per calendar year for Beam Port 1, which has the lowest hourly limit, yields about 73 puff-type releases per year to stay within DAC limits. Similarly, for puff releases from the rabbit, about 732 releases are allowed. In all cases, based on previous operating history, it is unlikely that the total puff releases from these facilities during a calendar year will exceed these estimates.

6.3.4.6 Continuous Release of ^{41}Ar from the Pool Water

Section 6.1.2.2 estimated a release rate from the pool water for ^{41}Ar of:

$$S_{41} = 6.81\text{x}10^{8} \text{ atoms/second.}$$

Multiplying this release rate by the radiological decay constant for ^{41}Ar ($1.0519\text{x}10^{-4}$ second^{-1}) and converting activity to microcuries leads to an estimate of 1.936 microcuries/second released from the pool. Using the concept and equation developed in Section 6.3.4.3 for continuous discharge of ^{41}Ar from the rabbit blower, an equilibrium concentration of $3.62\text{x}10^{-6}$ microcuries of ^{41}Ar per milliliter of building air. This is about 21% above the DAC for ^{41}Ar in a restricted area, and thus would limit reactor operation (effective exposure time) to 1657 full-power equivalent hours each calendar year. Since this would average out to about 6.6 full-power equivalent hours of operation each day, it is unlikely that OSURR operation would result in exceeding restricted area DAC for ^{41}Ar.

6.3.4.7 Combined Continuous Release from the Pool & Blower

The release rate of 1.936 microcuries of ^{41}Ar per second calculated above for the pool water can be added to the rabbit blower source term estimated in Section 6.1.2.1 of 1.72 microcuries/second to yield a combined continuous source term of 3.656 microcuries of ^{41}Ar added to the building air per second. Using the equation developed in Section 6.3.4.3 and substituting appropriate constants gives an equilibrium concentration of $6.83\text{x}10^{-6}$ microcuries of ^{41}Ar per milliliter of building air. This is a factor of about 2.28 above the DAC allowed in a restricted area. Using an averaging time of 2000 hours, an effective exposure time of 878 hours is allowed. Thus, operation with the reactor at full power and the rabbit blower running is restricted to this number of hours each working calendar year. It allows an average of about 3.5 full-power equivalent hours each day of combined full power reactor operation with the rabbit blower running continuously.

Solving the characteristic equation for run time, assuming the production rate above, leads to a run time of about 2141 seconds, or about 35 minutes, to attain a DAC level in the building air. Again, most rabbit operations will involve run times of 20 minutes or less.

6.3.4.8 Continuous Release of ^{16}N from the Pool Water

Section 6.1.2.3 provided calculations to show that the release rate of ^{16}N from the core is about 2.01×10^6 atoms of ^{16}N per milliliter of water per second, or a total of 9.34×10^9 atoms of ^{16}N per second (24.64 millicuries/second). This source term must be diluted to account for delay in traversing the distance from the top of the core to the surface of the pool.

Using the average coolant velocity through the core of 6.5 cm/second noted in Section 4.8 and assuming a constant average coolant velocity from a point immediately above the core to the surface of the pool, the total transit time is obtained by dividing the distance from the top of the core to the surface by the velocity. Given that the minimum depth of water in the reactor pool is 15 feet (457.2 cm), a total transit time of 70.34 seconds is obtained. This estimate is conservative in that it ignores delay times resulting from pool water mixing and dispersion by the cooling system pump. Essentially, it assumes the cooling system dispersion pump is off, but the reactor is at full power. This condition is prohibited by the reactor safety system, which initiates a reactor trip if the power rises above 100 kilowatts with the primary coolant pump off.

The ^{16}N concentration at the pool top can thus be estimated from:

$$C = C_0 e^{-\lambda t}$$

where
- C = concentration of ^{16}N atoms at the pool surface,
- λ = decay constant for ^{16}N,
- C_0 = concentration of ^{16}N atoms immediately above the core, and
- t = transit time from the core to the pool surface.

Substituting appropriate constant in this equation yields a concentration of 2096.9 atoms of ^{16}N per milliliter of pool water at the surface of the pool.

As the ^{16}N-bearing water reaches the surface of the pool, it spreads across the surface in the shape of a disk, forming an area source of radiation and a release interface to the building air. For the purpose of this calculation, assume that the disk has a radius of 85 cm, which is about the width of the reactor pool. The time the water takes to spread across this area, assuming a 6.5 cm/sec constant velocity is

$$t = r/v = (85\text{ cm})/(6.5\text{ cm/sec}) = 13.07\text{ seconds.}$$

During this distribution time, the concentration of ^{16}N decays from that initially available in the rising plume from the core. The average concentration of N across the surface of the disk source given by:

$$\overline{N} = \frac{1}{t}\int_0^t N_0 e^{-\lambda t} dt = \frac{N_0}{\lambda t}(1 - e^{-\lambda t})$$

where t = spreading time across disk surface,
λ = decay constant for ^{16}N, and
N = initial ^{16}N concentration at the pool surface.

From earlier calculations, we find that N = 2096.9 atoms/ml, λ = 9.761×10^{-2} second^{-1}, and t = 13.07 seconds. Performing the calculation gives an average disk source concentration of 1184.8 atoms of ^{16}N per milliliter in the disk source at the surface of the pool.

For estimation of the release rate of gaseous radionuclides to the air of the building, the number of ^{16}N atoms diffusing from the surface of the disk source to the air must be estimated. Dorsey [5] reports an escape velocity of 9×10^{-3} cm/second for nitrogen atoms from water. Multiplying this by the average concentration of nitrogen in the disk source gives:

$$S = Cv = (1184.8 \text{ atoms/cc})(9 \times 10^{-3} \text{ cm/sec})$$

$$= 10.66 \text{ atoms/cm}^2\text{/second.}$$

The total release rate of atoms of ^{16}N to the building air would be the area of the disk (85 cm equivalent radius) times the emission rate per unit area noted above. The release rate to the building air is thus 2.4196×10^5 atoms of ^{16}N per second.

As the ^{16}N atoms enter the reactor room air, their concentration is reduced by dilution into the volume of the building, exhausting through the ventilation fan, and radioactive decay. Since the half-life of ^{16}N is very short (7.1 seconds) compared to the building purge time (70 minutes, or an effective half-life for air in the building of 70 minutes), the radioactive half-life will dominate the effects of concentration reduction resulting from decay. The rate of buildup of ^{16}N in the building air is given by:

$$\frac{d(VN)}{dt} = S - \left(\lambda + \frac{q}{V}\right)VN$$

where S = release rate of ^{16}N atoms to the air,
λ = decay constant of ^{16}N,
q = building ventilation rate,
N = concentration of ^{16}N in the building air, and
V = building volume.

Under equilibrium conditions, the time rate of change of the concentration is zero, and the above equation can be solved for N:

$$N = S/(XV + q)$$

Substituting known values in this equation gives an equilibrium concentration of 1.25×10^{-3} nuclei of ^{16}N per milliliter of building air. Converting this to activity units gives a concentration of 3.3×10^{-9} microcuries of ^{16}N per cc of building air. Table I of Appendix B, 10CFR20 does not have a listing for ^{16}N. However, it states that, "Any single nuclide not listed above with decay mode other that alpha emission or spontaneous fission and with radioactive half-life of less than 2 hours", has a DAC of 1×10^{-7} µCi/ml. The calculated value for ^{16}N concentration is well within this bound.

For this calculation, assume that the volume of the building is represented by a hemisphere with an effective radius of 980 cm (about 32 feet). The dose rate resulting from the ^{16}N concentration calculated above, dispersed in this volume, can be estimated by:

$$R = K_1NR/2K_2$$

where $K_1 = 3.7\times10^4$ photons/second/microcurie,
N = ^{16}N concentration in the building air,
R = effective building radius, and
$K_2 = 1.6\times10^5$ photons/sec-cm^2 /rad-hr.

Performing this calculation yields an estimate of 3.75×10^{-7} rads/hr in the building from dispersed ^{16}N, or about 0.4 microrads/hr.

6.3.4.9 Actual ^{41}Ar Releases

The actual releases of ^{41}Ar into the restricted area can be calculated from an effluent monitor near the intake of the building exhaust fan. It will be assumed that the concentration of ^{41}Ar measured by this monitor is representative of the concentration in the reactor building. This is a conservative assumption in that the outlet for the rabbit is very near the intake for the effluent monitor, which will result in a higher reading than that of the rest of the building. In the half-year period from 1/1/99 to 6/30/99, the effluent monitor at the building exhaust fan measured a net count of 1,923,402 counts. The calibration for this monitor is 19.3 counts/second corresponds to 3×10^{-6} µCi/ml. Assuming that ½ of the work year is 1000 hours, this gives a concentration of

$$\frac{1,923,402 \text{ counts}}{1000 \text{ hr} * 3600 \text{ sec}/\text{hr}} * \frac{3\times10^{-6}\mu\text{Ci}/\text{ml}}{19.3 \text{ counts/sec}} = 8.30\times10^{-8}\mu\text{Ci}/\text{ml}$$

This is well below the DAC limit for ^{41}Ar in restricted areas of 3×10^{-6} µCi/ml.

6.3.5 Releases from the Restricted Area

Once gaseous radionuclides are released to the building atmosphere, they begin to be discharged to the outside environment by the building ventilation fan. As noted in the earlier analyses, this fan has a measured capacity of 1000 CFM, which results in a building purge time of 70 minutes. The exhaust point is approximately 32 feet above ground level, at the roofline of the building along the north wall. The exhaust stream exits the building parallel to the ground.

The following sections will consider the dilution factors available for the building releases and analyze several release cases.

6.3.5.1 Dilution Factor

Radionuclides contained in the building exhaust stream will mix with the outside air in the lee of the building. The dilution resulting from this mixing effect can be described as:

$$A_D = Aq\psi(x)$$

where
A_D = effective exposure concentration in curies/m^3,
q = building exhaust rate in m^3/second,
$\psi(x)$ = dilution factor at distance x, in sec/m^3, and
A = activity concentration in the exhaust stream.

The dilution factor is computed for the lee of the building (x=0), and assumes that the release is made from the roofline of the building. Further, assume that the wind velocity is steady at the time of the release and is equal to 1 m/sec. The dilution factor can be written as:

$$\psi(0) = 1/[(0.5)(s)(u)]$$

where
u = wind velocity in m/second, and
s = building cross-sectional area normal to the wind direction in m^2.

Assuming that the prevailing westerly winds are blowing at the time of the release, a normal cross-sectional area of 201.6 m^2 is presented to the wind. Substituting values into the above equation indicates a dilution factor of $\psi(0) = 9.921\text{x}10^{-3}$ second/m^3.

The building exhaust rate is assumed to be 1000 CFM, or 0.47195 m^3/second. Using this for q in the above equation and the value of $\psi(0)$ calculated above, values for A can be substituted for various release cases. In the puff release cases analyzed in the following sections, the exhaust stream activity concentration is taken to be the average concentration over the release period shown in Table 6.5, column 3.

6.3.5.2 Puff Release from Various Facilities

Using appropriate data in the equations derived above, various concentrations in the air on the lee side of the reactor building were calculated. The results are shown in Table 6.7. The Effluent Concentration Limit for ^{41}Ar in unrestricted areas is specified in Appendix B, Table II, Column 1 of 10CFR20 as $1x10^{-8}$ microcuries/ml.

To maintain releases to the outside air within Effluent Concentration Limits, either the source term must be reduced or operational limits imposed so that when the releases are averaged over the permitted averaging time, the average concentration does not exceed the limit. For releases to unrestricted areas, 10CFR20 specifies an averaging time of one year. Table 6.8 shows the results of calculations based on these limits.

If one assumes that the releases occur over a time equal to 213.75 minutes (five effective half-lives of ^{41}Ar in the building), dividing the operational limits shown in Table 6.8 by this gives an approximate number of puff-type releases of this type allowed in a year. About 228 releases are allowed for Beam Port 1, and 295 releases are permitted for Beam Port 2. Considering Beam Port 1 as the most restrictive case, 228 releases each year would require an average of 4.38 releases of this type each week. It is very unlikely that OSURR operations would result in this frequent a release rate, so it is unlikely that unrestricted area limits will be exceeded.

It should be noted that the calculations shown above are conservative in that they assume a saturation activity source term in the puff releases, with the entire beam tube volume being air void. Generally, lower initial source activities will be available for release, since reactor operation times less than that necessary to achieve saturation activities of ^{41}Ar (about 9 hours) are more common than those exceeding this duration, and many, if not most, reactor operations will have much less than the full volume of the beam tube voided. Thus, lower overall concentrations in the building and in the outside air will result.

6.3.5.3 Continuous Release from the Rabbit Blower

The calculations shown in Section 6.3.4.3 indicate that an equilibrium concentration of $3.21x10^{-6}$ microcuries/ml of ^{41}Ar in the air of the reactor building will result from continuous operation of the rabbit blower. Equilibrium concentrations will be present after about five effective half-lives (213.75 minutes). At equilibrium, the equation used in Section 6.3.5.1 will predict equilibrium concentrations in the air on the lee side of the building. Substituting appropriate values leads to an estimate of $1.5x10^{-8}$ microcuries/ml of ^{41}Ar. This is above the limit for ^{41}Ar in an unrestricted area, which means that full-power reactor operation is limited to 5844 hours per year. This is well above the amount of time actually spent running at full-power in a year.

Table 6.7: Estimated ^{41}Ar Concentrations in the Lee of the Reactor Building for Puff Releases of ^{41}Ar from the Various Experimental Facilities of the OSURR

Facility Description	Average Exhaust Concentration (μcuries/ml)	Outside ^{41}Ar Concentration (μcuries/ml)	DAC Factor For Average Concentration
CIF	3.43×10^{-6}	1.61×10^{-8}	1.16
Beam Port 1	2.31×10^{-5}	1.08×10^{-7}	10.80
Beam Port 2	1.78×10^{-5}	8.33×10^{-8}	8.33
Rabbit	2.30×10^{-6}	1.08×10^{-8}	1.08
Thermal Column	1.32×10^{-6}	6.18×10^{-9}	0.62
Rabbit Carrier	1.56×10^{-7}	7.30×10^{-10}	0.07
4" Dry Tube	4.57×10^{-6}	2.14×10^{-8}	2.14
2" Dry Tube	1.58×10^{-6}	7.40×10^{-9}	0.74

Notes:

(1) The average exhaust stream concentration was taken from column 3 of Table 6.5. It assumes a puff release from the building and a purge time equal to five effective half-lives of ^{41}Ar in the building.

(2) Of the above cases, releases from the CIF, beam ports, rabbit, and 4" dry tube result in outside concentration greater than the effluent concentration limit over the averaging time.

Table 6.8: Operational Limits for the Various Experimental Facilities of the OSURR to Maintain Unrestricted Area Effluent Concentration Limits on the Lee Side of the Reactor Building

Facility Description	Limit of Full-Power Hours of Operation Per Year
CIF	7556.9
Beam Port 1	811.7
Beam Port 2	1052.3
Rabbit	8116.7
Thermal Column	No Limit
Rabbit Carrier	No Limit
4" Dry Tube	4096.3
2" Dry Tube	No Limit

Notes:

(1) The limits expressed above assume a puff release at saturation activity levels in a completely voided effective volume of the facility.

(2) The limits on full-power operation are in fact effective exposures times allowed at the estimated concentration. Effluent Concentration Limits are not exceeded if exposure times are less than or equal to this value.

6.3.5.4 Continuous Release from the Pool Water

Section 6.3.4.6 noted that a building equilibrium concentration for ^{41}Ar of 3.62×10^{-6} microcuries per milliliter would result from releases from the pool water. The equation shown in Section 6.3.5.1 predicts an outside air concentration of 1.62×10^{-8} microcuries of ^{41}Ar per milliliter of air, which is also above the limit for ^{41}Ar in an unrestricted area. This restricts operation to 5411 hours per year.

Section 6.3.4.8 gave an estimate of 3.3×10^{-9} µCi/ml for ^{16}N restricted area concentration. Using the calculation from Section 6.3.5.1, this results in a unrestricted area concentration of 1.55×10^{-11} µCi/ml. This well below the Effluent Concentration Limit for unrestricted areas of 1×10^{-9} µCi/ml. Most likely, the actual concentration will be far below that calculated since the seven second half-life of ^{16}N will result in most of it decaying away before it reaches ground level.

6.3.5.5 Combined Pool Water and Rabbit Blower Releases

If the source terms for continuous ^{41}Ar production from the rabbit blower exhaust and pool water are added, the outside air concentration at equilibrium is estimated to be about 3.20×10^{-8} microcuries of ^{41}Ar per milliliter. The combined ^{41}Ar release restricts the reactor to 2739 hours of full-power operation per year, which is still well above the number of hours actually run.

6.3.5.6 Actual ^{41}Ar Released

Using the effluent monitor data for 1/1/99 to 6/30/99 along with the calculation method shown in Section 6.3.5.1 gives an ^{41}Ar outside air concentration of 8.87×10^{-11} µCi/ml (assuming that half of a year is 4383 hours). This is well below the limit of 1×10^{-8} µCi/ml given for ^{41}Ar for unrestricted areas.

6.3.6 Steps to Limit Release Levels

Although the calculations noted in the preceding sections are conservative, it is possible to take steps to limit releases even further. This section will discuss some of these actions and their effects.

6.3.6.1 Reducing Effective Irradiated Volumes

In most cases, the volumes considered in the preceding calculations will be larger than those normally irradiated during routine operations. For example, the two beam ports are normally filled with shielding plugs, and the thermal column is filled with graphite stringers when it is not being used in an experiment, which essentially eliminate their effective volumes. Devices or samples being irradiated in the dry tubes, thermal columns, or CIF will also reduce the effective irradiated volumes. Also, the dry tubes are usually stored away from the core when not in use, so their effective volumes are not irradiated during routine operations. Air voids in experiments placed in these facilities can be limited by packing them with inert, non-activating materials (e.g., the rabbit carrier tube is packed with cotton). Only the rabbit and CIF effective volumes are irradiated when they are not in use during routine operations.

6.3.6.2 Facility Purging

The rabbit facility can be purged at any time during normal operations. The other experimental facilities can also be purged by insertion of apparatus to circulate gas through the effective volume of the facility. Purging the experimental facility limits the buildup of ^{41}Ar in the effective volume, thereby reducing the initial concentration that might be released in a puff-type expulsion of air from the facility. Limiting the purging rate can keep equilibrium concentrations of ^{41}Ar in the restricted area within DAC limits.

The experimental facilities can also be purged with nitrogen gas. When irradiated by neutrons, nitrogen undergoes very little neutron capture reactions leading to radioactive products. Replacement of air in the experimental facilities with nitrogen thereby limits the concentration of ^{41}Ar in the building air. Nitrogen gas can be introduced to the facility prior to irradiation, or in a continuous stream by gas lines inserted during the time the facility is in use.

6.3.6.3 Limiting Facility Releases

Release of ^{41}Ar from irradiated air volumes can be reduced by sealing the facilities against leakage to the atmosphere for a time sufficient to allow decay of the isotope. For example, the boral plate on the outer surface of the main graphite thermal column has a rubber gasket along its inner edge, allowing it to be hermetically sealed against the outer surface of the facility. If the experiment allows, the facility may be kept sealed until ^{41}Ar concentration has been reduced by radioactive decay. Similarly, the CIF can be left plugged for a time to reduce ^{41}Ar activity, if allowable under the conditions of the experiment. For neutron activation experiments resulting in relatively long-lived radioisotopes, overnight decay periods are generally acceptable. In this case, for example, ^{41}Ar activity in the CIF effective volume would be reduced by about 99% from that initially present at the time the irradiation ended.

6.3.6.4 Ventilation System Control

The building ventilation systems can be turned off by a switch in the control room. Such an action can thereby limit releases of gaseous radionuclides to the unrestricted area. If more rapid purging of the building is desired, ventilation rates can be increased by activating portable fans and allowing them to exhaust out of opened doors or windows. Any such releases will be recorded, and the calculated release concentrations of gaseous radioisotopes averaged into the total yearly release to the unrestricted area.

6.3.7 Estimated Doses

Release of ^{41}Ar from the experimental facilities of the OSURR will result in accumulated doses to persons exposed to the resulting isotope concentration in the air. The whole-body gamma ray dose to a person surrounded by and immersed in a semi-infinite cloud of radioactive gases can be approximated by:

$$D = 900EA_D$$

where D = dose rate in rads/hr,
E = photon energy in MeV, and
A_D = effective exposure concentration in curies/m^3

Earlier calculations and tables listed values for A_D given various types of releases. Assuming a gamma photon energy of 1.3 MeV, exposure rates can be estimated. Table 6.9 summarizes the results of these calculations.

6.4 Solid Radioactive Waste Management

Operation of the OSURR will generate very little solid low-level radioactive waste. The primary source of low-level solid waste will be the demineralizer cartridge in the reactor pool water processing system. Current procedures call for cartridges to be kept on-site for a decay period sufficient to reduce the activity of short-lived radionuclides (e.g., ^{24}Na) to negligible levels. If higher-power operation of the OSURR results in additional radioisotopes being present in the resins of the demineralizer cartridge with longer half-lives, a bulk radioassay will have to be performed to determine specific and total activity in the cartridges prior to disposal. Disposal of this low-level waste is handled by the OSU Radiation Safety Section and comprises a few cubic feet per year.

Spent fuel assemblies might also be classified as solid radioactive waste. These are stored in the fuel storage pit at the east end of the reactor pool, unless otherwise approved by the Reactor Operations Committee and the Nuclear Regulatory Commission. After suitable decay times have elapsed, spent fuel assemblies are returned to the Department of Energy for ultimate disposal. Because of their isotopic content and activity inventory, used fuel assemblies are not considered low-level radioactive waste, and are therefore handled separately from other waste forms generated by the laboratory. Spent fuel shipment is performed in accordance with approved procedures that meet appropriate federal, state, and local requirements.

Table 6.9: Estimated Dose Rates Resulting From ^{41}Ar Concentrations As A Result Of Releases From Various Experimental Facilities of the OSURR

Facility Description	Type of Release	Restricted Area Dose Rate (rads/hr)	Unrestricted Area Dose Rate (rads/hr)
CIF	Puff	4.02×10^{-3}	1.36×10^{-5}
Beam Port 1	Puff	2.70×10^{-2}	1.26×10^{-4}
Beam Port 2	Puff	2.08×10^{-2}	9.75×10^{-5}
Rabbit	Puff	2.69×10^{-3}	1.26×10^{-5}
Rabbit	Continuous	3.76×10^{-3}	1.76×10^{-5}
Pool Water	Continuous	4.24×10^{-3}	1.90×10^{-5}
Pool & Rabbit	Continuous	7.99×10^{-3}	3.74×10^{-5}
Thermal Column	Puff	1.54×10^{-3}	7.23×10^{-6}
Rabbit Carrier	Puff	1.83×10^{-4}	8.54×10^{-7}
4" Dry Tube	Puff	5.34×10^{-3}	2.50×10^{-5}
2" Dry Tube	Puff	1.85×10^{-3}	8.66×10^{-6}

Notes: (1) The puff releases calculated above assume saturation activities have been attained. The continuous discharge from the rabbit facility assumes a blower operation time sufficient to attain equilibrium concentrations in the building.

(2) The averaging time for the concentration used in the calculations is assumed to be five effective half-lives of ^{41}Ar in the building (213.75 minutes).

6.5 Liquid Radioactive Waste Management

Section 6.1.1 noted that no liquid-borne radioactive materials are discharged from the OSURR during normal operation. However, certain maintenance and repair activities may result in liquid discharges.

An alternative to demineralizer cartridge regeneration by the manufacturer or replacement of the resins by OSURR personnel would involve on-site regeneration of the cartridges by OSURR staff. In this event, some liquid-borne radionuclides would result. These eventually would have to be released from the reactor building as liquid radioactive waste.

Prior to release of liquid radioactive waste, the isotopic content of the material and specific activities of radioisotopes present in the liquid must be determined. The liquids may then be kept in a holding tank to allow decay to reduce the total activity inventory, or make it available for dilution or treatment prior to release.

6.6 Byproduct Materials Management

Operation of the OSURR will result in the production of radioactive materials as part of experimental procedures. Production of radioactive materials by neutron activation, also called byproduct materials, may be a deliberate result of the experiment (as would be the case in isotope production or neutron activation experiments), or incidental to the experiment (as would occur in materials damage studies or medical experiments). In either case, the radionuclides so produced must be handled safely.

Laboratory procedures are available for survey and assay of all materials irradiated in the OSURR. Handling and storage of activated materials is also governed by laboratory procedures. These procedures fall within the overall university guidelines for working with radioactive materials, which themselves are designed to meet or exceed the requirements specified in 10CFR20.

Typical byproduct materials would include a variety of beta and gamma-emitting radioisotopes of various half-lives, generally formed by thermal neutron-induced activation of parent (target) nuclei. A few radionuclides are formed by fast-neutron capture reactions, such as ^{58}Co (from the neutron-proton reaction with ^{58}Ni) and ^{24}Na (from the neutron-alpha reaction with ^{27}Al). A few byproduct materials emit alpha particles. Half-lives can range from seconds up to years. Most byproduct materials are in solid form, but a few are liquids, and very few are gases. Experimental procedures call for activation targets to be encapsulated, where possible. Induced activity can result in dose rates ranging from a few tenths of a millirem/hour up to a few rem or tens of rem per hour. Radiation safety procedures are followed for dealing with sources producing intense radiation fields and significant dose rates.

Radioactive materials are typically handled in the northeast corner of the building, which is designated as the radioactive materials handling and storage area. Other areas of the building may also be used for handling and storage if they are so designated and posted. Handling and storage procedures are available to assure safety in these activities. Storage of irradiated materials, if necessary, can be facilitated by using appropriate shielded containers. A variety of lead storage containers, of various shapes and sizes, are available in the laboratory.

Solid byproduct materials, if they contain essentially no transuranic radionuclides, are disposed of in designated containers. Transuranic materials are handled separately, but the quantities are generally very small. A record is kept of materials placed in the disposal container. The container is collected when required by Radiation Safety Section personnel, and added to the university total waste inventory. Liquids are released, if permitted within the framework of 10CFR20 limits, into a designated disposal sink ("hot" sink). Records are kept of liquid disposals. Liquid sources may be evaporated and disposed of as dry materials, if the radionuclides they contain are non-volatile. Gaseous radioactive materials may be vented to the air, if such procedures do not exceed averaged annual effluent release concentration limitations, or the containers holding the gases, if tight, can be disposed of in the waste container.

Miscellaneous radioactive waste, such as contaminated gloves, tools, apparatus, or absorbent pads are considered as solid byproduct waste materials and are disposed of in the waste container. The items are surveyed for dose rate prior to disposal.

The area of the building where the radioactive waste disposal container is located is surveyed for gamma dose rates are part of the area radiation surveys performed routinely at the laboratory. If necessary, the area is posted as a Radiation Area or High Radiation Area. The container is posted as a Radioactive Materials storage area.

6.7 Chapter 6 References

[1] David T. Goldman, "Chart of the Nuclides", The General Electric Company, Schenectady, N.Y., 1965.

[2] Harold Etherington, Editor, "Nuclear Engineering Handbook", First Edition, McGraw-Hill Book Company, Inc., New York, 1958.

[3] Nicholas C. Horning, "Measurement of the Neutron Spectra in the Beam Ports and Thermal Column of The Ohio State University Research Reactor", M.Sc. Thesis (unpublished), The Ohio State University, Columbus, OH, 1976.

[4] Joseph W. Talnagi, "Investigation of Perturbation Induced

by Neutron Detectors in the Spatial and Energy Dependent Neutron Flux of The Ohio State University Research Reactor", M.Sc. Thesis (unpublished), The Ohio State University, Columbus, OH, 1979.

[5] Noah E. Dorsey, "Properties of Ordinary Water-Substance In All Its Phases: Water-Vapor, Water, And All The lees", Hafner Publishing Co., New York, NY, 1940.

7.0 Radiation Protection

7.1 Radiation Sources During Operation

Operation of the OSURR will create a source of radiation in the form of particles emitted from the core. In all conceivable cases, gamma ray emission and neutron radiation will be the primary components of radiation fields observed external to the core. Beta and alpha particles will be entirely absorbed by the materials of the pool walls, water covering the core, and materials of the core itself.

The source of gamma rays includes both prompt gammas from the fission of ^{235}U, and decay gammas from the fission product inventory in the core. Neutrons will also result from ^{235}U fission and delayed neutron emitters in the fission product inventory. The intensity of the prompt neutron and gamma is proportional to reactor power, while the intensity of the delayed neutrons and decay gammas is a function of the power history of the core and, if the reactor is shutdown, the elapsed time between the time of observation of the shutdown field and the time of the last reactor shutdown.

7.1.1 Direct Exposures

Exposure to radiation directly from the core can occur in several ways. First, gammas and neutrons from the core can penetrate the walls of the pool or water covering the core and cause a dose to be accumulated by an exposed individual. A person can also be exposed to core radiation through an opened experimental facility. These cases will be considered in this section.

7.1.1.1 Gamma Dose From The Core Through Shield Walls

For this calculation, the core was modeled as a sphere with the sample volume as the rectangular solid that is the actual core geometry. This approach is considered valid since it is used in the Engineering Compendium on Radiation Shielding [1] for calculating the gamma doses from the core of the Bulk Shielding Reactor (BSR), which is similar in design to the OSURR. In the OSURR model, the effective spherical volume of the core is covered with 15 feet of light water, surrounded on all other sides by 1 foot of light water, then surrounded by 6 feet of barytes concrete. Figure 7.1 shows the representation of the core and shielding materials used in this calculation.

The energy spectrum of fission-induced gamma rays was considered to have a representative set of 8 lines, centered at energies of 1, 2, 4, 6, and 8 MeV. This spectrum was also used to represent the gamma ray energy spectrum of fission product gamma emission.

The intensity of the gamma source distributed through the core volume is estimated by considering the energy released per fission in the form of gamma radiation. For 500 kilowatt operation, the fission rate can be approximated by the following formula:

$$R = PK_1K_2K_3,$$

where P = reactor power in megawatts,
$K_1 = 10^6$ joules/MW-second (conversion factor),
$K_2 = 1$ fission/200 MeV (conversion factor), and
$K_3 = 1$ MeV/1.6×10^{-13} joules (conversion factor).

Using P = 0.5 megawatts, this calculation gives a fission rate of approximately 1.56×10^{16} fissions per second. Using a volume of 88490.2 cm^3 for the effective volume of the core, the fission rate density is estimated to be 1.77×10^{11} fissions/second/cm^3.

The intensity of the volume-distributed gamma source is calculated from:

$$S_v = DK_4$$

where D = fission rate density calculated above,
K_4 = 8.6 MeV/fission (conversion factor for gamma energy released per fission event), and
S_v = gamma source intensity.

This calculation gives a gamma source intensity of 1.52×10^{12} MeV/second/cm^3.

As the gamma rays are released from the core, they interact with the various materials making up and surrounding the core. The absorption of the gamma rays in each of these materials is a function of the energy of the gamma ray, the thickness of the material, and the absorption properties of the material. The materials to be considered include the water, aluminum, and uranium making up the core, the surrounding water, and the barytes concrete making up the walls of the pool.

The absorption properties of the core can be estimated by averaging the linear attenuation coefficients of the various materials, weighted by the percentage volume of each material. For this calculation assume that 65% of the core volume is light water, 25.5% is aluminum, and 9.5% is uranium.

At 1 MeV, K_4 = 8.6 MeV/fission, which gives a gamma source intensity of 1.52×10^{12} MeV/second/cm^3. At 2, 4, and 6 MeV, K_4 is 5.41, 1.65, and 0.392 MeV/fission, respectively. These give gamma source intensities of 9.57×10^{11}, 2.91×10^{11}, and 4.35×10^{10} for the 2, 4, and 6 MeV components, respectively.

Now, combining the gamma ray point kernel, the exponential buildup function, and the basic point kernel integral, and also assuming a constant source, spherical geometry, and buildup occurring in the concrete shield (pool walls) only (this is reasonable since water is between the core and pool walls, and buildup in the water is small compared with that in the pool walls), the following equation can be used is estimate the gamma flux:

$$\phi(z) = S_v / 2\mu) (R_s / [R_s + z]) [AE_1(\mu z(1 + \alpha) + (1 - A)E_1(\mu z(1 + B))]$$

where
S_v = gamma source intensity,
μ = attenuation factor for core or pool wall,
R_s = core (source) radius,
z = distance from the core, and
E_1 = integral function calculated from:

$$E_n(x) = x^{n-1}\int_0^{\infty}(e^{-t} / t^n)dt$$

In the above equation, A, B, and α depend on the material properties. This is the form of the shielding equation used in the Engineering Compendium on Radiation Shielding [1] to analyze the shielding requirements for the BSR at Oak Ridge.

Once the gamma ray flux is found with the above equation, the dose rate at a particular location can be estimated from:

$$\dot{D} = \phi(z)\mu_{air}(E)*(3600 \text{ sec/hr})/5.4x10^4 \text{ MeV/rem/g}$$

Using this, the following estimates are obtained for dose rate at the outside edge of the reactor pool wall at 500 KW operating power:

1 MeV : $1.2x10^{-5}$ mrem/hr,

2 MeV : $6.6x10^{-3}$ mrem/hr,

4 MeV : $1.3x10^{-2}$ mrem/hr,

6 MeV : $1.9x10^{-2}$ mrem/hr, and

8 MeV : $6.0x10^{-2}$ mrem/hr.

These results show that very little dose rate from the gamma rays emitted from the core will be present at the outside of the pool walls. The dose that does appear is a result of the higher-energy gamma rays. Almost all of the lower-energy gammas are absorbed by the materials surrounding the core.

7.1.1.2 Gamma Dose Through an Experimental Facility

Several experimental facilities are available which lead up to or into the core. These include the CIF, two beam ports, and movable dry tubes. To estimate gamma doses from opened experimental facilities, the core is modeled as a sphere with a volume equal to that of the actual core volume in the form of the rectangular solid that it actually is. This source is assumed to emit $N_0 f(\theta)$ photons per unit time per unit solid angle. The polar angle, θ, is the angle between the direction of emission and the axis of the tube which forms the experimental facility. The function $f(\theta)$ is normalized such that N_0 is the number of photons emitted into the half-space directed toward the shield per unit area per unit time. Assuming a cosine emission from the source yields the following equation to predict the gamma flux at a distance z from the core along the tube axis:

$$\phi(z) = zN_0[1-(1+R^2/z^2)]^{-1/2}$$

where N_0 is the number of gamma rays emitted from the surface of the core per unit area per unit time (after some have been absorbed by materials in the core), R is the radius of the irradiation tube, and z is the distance along the axis of the tube. This equation was used to estimate the dose rates from streaming through the CIF and beam ports. Note that this estimate does not do an exact calculation for buildup of gamma dose from secondary emissions or scattering of gamma rays from the walls of the irradiation facility. Such a calculation would involve a more complex gamma photon transport analysis using Monte Carlo techniques.

Calculation of N_0 requires consideration of the absorption properties of the materials making up the core. The total gamma absorbing effect of the core can be estimated by a weighted average of the linear attenuation coefficients of the materials composing the core. The average is weighted by the percent volume of each material in the core. For this calculation, the core volume was considered to be formed of 65% light water, 25.5% aluminum, and 9.5% uranium. The fission rate source intensity was estimated in Section 7.1.1.1, and assuming 500 KW operation and using the earlier equations, we can calculate:

$$N_0 = FK/2v_c\mu_c$$

where F = fission rate = 1.77×10^{11} fissions/cm^3/second,
K = MeV/fission released as gamma rays,
v_c = effective volume of the core, and
μ_c = core attenuation factor.

The CIF is a hollow aluminum tube 1.5 inch outer diameter, extending from the top of the pool into the geometric center of the core, about 16 feet (488 cm) long. For this calculation, it is considered to be empty of shielding plugs or experimental apparatus, and is filled with air at STP conditions.

Using equations derived earlier, the gamma flux for 1 MeV, 2 MeV, 4 MeV, 6 MeV, and 8 MeV gamma rays was estimated. The following dose rate estimates for streaming out of the open top of the CIF tube at 500 KW operation were obtained:

1 MeV : 2.1 rem/hr

2 MeV : 78.3 rem/hr

4 MeV : 23.7 rem/hr

6 MeV : 5.2 rem/hr

8 MeV : 2.2 rem/hr

TOTAL : 111.5 rem/hr

This is a high dose rate at the exit of the CIF tube. A later section will discuss protection measures to keep doses as low as possible.

Dose rates from the CIF and beam ports one hour after shutdown were also estimated. Dose rates after shutdown result from gamma rays emitted by the decay of fission products in the core. The intensity of the gamma source in the core after shutdown (sometimes called the shutdown field) is a function of the power history of the reactor and the subsequent decay time following shutdown. For this calculation, it was assumed that the recent power history would include an irradiation at 500 kilowatts for a duration of 30 hours, followed by a shutdown decay period of one hour. The fission product activity can be approximated by:

$$A = 1.4P[(t^{-0.2} - (t+T)^{-0.2}]$$

where A = fission product activity in curies,
P = operating power of the reactor in megawatts,
t = shutdown time in days, and
T = operating time in days.

Also, assume an average decay gamma ray having an energy of 0.7 MeV per disintegration.

Performing the calculation for fission product activity, a total of 3.4×10^{14} MeV/second is being generated in the source. Using the equations developed earlier, a dose rate of 13.52 rem/hr at the open top of the CIF is obtained. For the two beam ports, which are considered to be hollow aluminum tubes with a 6 inch inner diameter and 7 feet long, with no shielding plugs or experimental apparatus installed, the dose rate at the open end of the beam ports will be about 1192 rem/hour. Again, protection methodologies are discussed in a later section.

7.1.1.3 Neutron Dose Through an Experimental Facility

An opened experimental facility will allow neutrons to leak from the core to the exit of the experimental facility. A precise estimate of the neutron flux at the exit of an opened experimental facility is difficult, since it would require modeling of neutron transport phenomena along the length of the experimental facility. These transport effects would depend on the nature of the neutron source, geometry of the experimental facility relative to the core, and the materials making up and surrounding the tube through which the neutrons are passing. Effects such as neutron absorption, elastic and inelastic scattering events, and in-scattering of neutrons from the surrounding materials would have to be taken into account. However, a rough estimate of the neutron flux and dose at the exits of the experimental facilities can be obtained from considering geometry effects alone.

First, consider an experimental facility for which the neutron flux and energy distribution has been measured at the core end of the facility. The energy-dependent neutron flux is usually represented as $\sigma(E)$ neutrons/cm^2/second for a given reactor power. If the cross-sectional area of the experimental facility is known, the area-distributed neutron source can be collapsed to an equivalent point source given by $A_s\sigma(E)$, where A_s is the cross-sectional area of the experimental facility.

Now, the neutron flux passing through a unit area of a sphere surrounding this equivalent point source is simply $A_s\sigma(E)/4\pi r^2$, with r the length of the experimental facility from the core end to the exit. Table 7.1 shows the geometric parameters used in the neutron dose calculations.

Horning [2] has measured the energy-dependent neutron flux in various experimental facilities at 10 KW for the HEU-fueled OSURR. Table 7.2 summarizes the data used in the flux estimations. For these calculations, the flux measurements were normalized to 1 watt reactor power. Etherington [3] gives neutron fluence to dose conversion factors as a function of neutron energy. Table 7.3 lists the conversion factors used in these calculations.

Using the geometry equation developed earlier, and the data shown in Tables 7.1 and 7.2, estimated neutron fluxes at the exits for various experimental facilities for a 1 watt operating power were calculated. The results are shown in Table 7.4. Using the dose conversion factors shown in Table 7.3, equivalent doses were estimated. These are shown in Table 7.5.

7.1.1.4 Gamma Dose At The Pool Top Through Pool Water

Gamma radiation from the core interacts with the pool water above the core. Some of the gamma photons will travel through the 15 feet of water above the core and give rise to a dose rate at the surface of the pool. Using an approach like that taken in Section 7.1.1.1, the gamma dose was estimated for gamma rays of energy 2, 4, 6, and 8 MeV.

Table 7.1: Geometric Parameters of Various Irradiation Facilities

Facility Name	Cross-Sectional Area	Effective Length
CIF	9.65 cm^2	487.7 cm
Beam Port 1	190.09 cm^2	198.1 cm
Beam Port 2	190.09 cm^2	198.1 cm
Thermal Column G-7	103.23 cm^2	144.8 cm

Table 7.2: Measured Core-End Neutron Flux in Various Irradiation Facilities

Facility Name	Maxwellian Neutron Flux ($n/cm^2/sec$)	1/E Neutron Flux ($n/cm^2/sec$)	Fission Neutron Flux ($n/cm^2/sec$)	Total Neutron Flux ($n/cm^2/sec$)
CIF	1.94×10^{11}	1.20×10^{11}	8.40×10^{10}	3.97×10^{11}
Beam Port 1	1.22×10^{11}	5.46×10^{10}	4.08×10^{10}	2.17×10^{11}
Beam Port 2	9.47×10^{10}	3.26×10^{10}	2.67×10^{10}	1.54×10^{11}
Thermal Column Position G-7	1.29×10^{10}	5.42×10^{9}	4.96×10^{8}	1.88×10^{10}

Notes:

(1) The measurements shown above are at the core end of the experimental facility for the HEU-fueled OSURR operating at 10 KW.

(2) Neutron energy ranges are as follows:

Maxwellian: 0 - 1.85×10^{-7} MeV

1/E: 1.85×10^{-7} - 0.5 MeV

Fission: 0.5-15 MeV

(3) Measured neutron fluxes are as reported by Horning [2].

Table 7.3: Neutron Fluence to Dose Conversion Factors

Neutron Energy Range in MeV	Dose Conversion Factor in rad/neutron/cm^2
0 - 1.85×10^{-7}	3.5×10^{-10}
1.85×10^{-7} - 0.5	1.0×10^{-10}
0.5 - 15	3.5×10^{-9}

Notes:

(1) Dose equivalents are approximate values derived from visual examination of the curves shown in Etherington [3].

(2) Doses are for soft-tissue equivalent materials, based on a first-collision dose delivery model.

Table 7.4: Estimated Neutron Flux at the Exit of Various Irradiation Facilities

Facility Name	Maxwellian Neutron Flux ($n/cm^2/sec/w$)	1/E Neutron Flux ($n/cm^2/sec/w$)	Fission Neutron Flux ($n/cm^2/sec/w$)	Total Neutron Flux ($n/cm^2/sec/w$)
CIF	62.6	38.7	27.1	128.2
Beam Port 1	4702.6	2104.6	1562.7	8364.5
Beam Port 2	3650.3	1256.6	1029.2	5936.1
Thermal Column Position G-7	505.4	212.4	19.4	736.6

Notes:

(1) The estimates shown above are at the exit of the experimental facility for the HEU-fueled OSURR normalized to 1 watt operating power.

(2) Neutron energy ranges are as follows:

Maxwellian: $0 - 1.85 \times 10^{-7}$ MeV

1/E: $1.85 \times 10^{-7} - 0.5$ MeV

Fission: 0.5-15 MeV

Table 7.5: Estimated Neutron Dose at the Exit of Various Irradiation Facilities

Facility Name	Maxwellian Neutron Dose (rad/hr/W)	1/E Neutron Dose (rad/hr/W)	Fission Neutron Dose (rad/hr/W)	Total Neutron Dose (rad/hr/W)
CIF	7.89×10^{-5}	1.39×10^{-5}	3.41×10^{-4}	4.34×10^{-4}
Beam Port 1	5.93×10^{-3}	7.58×10^{-4}	1.97×10^{-2}	2.64×10^{-2}
Beam Port 2	4.60×10^{-3}	4.52×10^{-4}	1.30×10^{-2}	1.81×10^{-2}
Thermal Column Position G-7	6.39×10^{-4}	7.65×10^{-5}	2.44×10^{-4}	9.60×10^{-4}

Notes:

(1) The estimates shown above are at the exit of the experimental facility for the HEU-fueled OSURR normalized to 1 watt operating power.

(2) Neutron energy ranges are as follows:

Maxwellian: $0 - 1.85 \times 10^{-7}$ MeV

1/E: $1.85 \times 10^{-7} - 0.5$ MeV

Fission: 0.5–15 MeV

The following dose rate estimates were obtained:

2 MeV : 0.08 mrem/hr

4 MeV : 10.60 mrem/hr

6 MeV : 32.24 mrem/hr

8 MeV : 97.40 mrem/hr

TOTAL : 140.32 mrem/hr

So, at the surface of the pool, with the reactor at 500 KW, the direct gamma dose exceeds the limits for a high radiation area. This will require protection methods to be discussed in a later section.

It should be noted that this estimated dose rate is only at the surface of the pool directly above the core. During normal operation, this point is difficult to access. The most likely position for a person at the pool top closest to the core would be standing at the side of the pool, with the lower half of their body shielded by the 1-foot thick pool wall. Thus, the most likely whole-body dose rate to which an individual would be exposed is less than that estimated above.

7.1.2 Indirect Exposures

Radiation exposure can occur from sources produced by the reactor. Chapter 6 considered the doses from production of Ar and N that escapes from the reactor pool surface and/or experimental facilities and mixes with the air in the building. This section will consider other indirect sources.

7.1.2.1 Nitrogen-16 At The Surface Of The Pool

Chapter 6 discussed the production of ^{16}N isotope in the core. Section 6.1.2.3 estimated that about 24.64 millicuries of ^{16}N per second are released from the top of the core for 500 KW operation. Further, Section 6.3.4.8 considered the effects of a rising plume of ^{16}N-bearing water to the surface of the reactor pool and spreading across the surface into the shape of a disk. The average concentration of ^{16}N atoms in the disk source thus produced was estimated to be 1184.8 atoms of ^{16}N per milliliter. This estimate accounts for decay of the isotope during its rise to the surface of the pool and spreading into a disk source.

The dose rate from a disk source of ^{16}N can be estimated from the following:

$$D = [\lambda N/2\mu K][1-E_2(\mu h)],$$

where D = dose rate near the pool surface,

h = thickness of disk source (see below),
μ = linear attenuation coefficient of 6 MeV photons in water,
K = conversion factor ($1.6x10^5$ photons/cm^2/sec/rad/hr),
λ = decay constant for ^{16}N,
N = average ^{16}N concentration in the disk source, and
E_2 = second exponential integral (see Section 7.1.1.1).

The thickness of the disk source can be estimated from:

$$h = v_c t_s / A_s$$

where h = thickness of the disk source,
v_c = volume flow rate of the ^{16}N-bearing water,
t_s = time to spread into a disk source, and
A_s = area of the disk source.

Substituting appropriate values into the source thickness equation gives a value of 2.48 cm for the disk source thickness. Using this to evaluate the integral term, and substituting other constants in the above equation for D yields an estimate of 2.02 mrad/hr from the disk source of ^{16}N at the surface of the pool. This source intensity is considerably less than that from direct penetration of the water layer above the core by gamma rays from the core.

7.1.2.2 Gaseous Effluents

Operation of the reactor produces gaseous radioisotopes which diffuse into the air of the reactor building, and ultimately to the outside atmosphere. Chapter 6 considered the production of these gaseous sources, and the dose consequences both within and outside the building. Table 6.9 summarizes the dose calculation for various types of ^{41}Ar releases. Section 6.3.4.8 estimated the dose consequences of ^{16}N release from the surface of the pool.

7.1.2.3 Cooling and Process System Activation Products

Circulation of the reactor pool water through the core will induce activation of any trace elements contained in the water. Also, neutron-alpha reactions occurring in the aluminum materials in the core will produce some ^{24}Na, part of which will diffuse into the pool water. As the reactor pool water is circulated through the primary cooling loop and the water processing system (when it is activated), some of these activated elements will become trapped in the internals of these systems, giving rise to a dose rate around the components of these systems. Prediction of the amount of materials deposited in these systems and the resulting dose rates is difficult, as it depends on the power history of the reactor and the relative probability of elements being retained in the cooling or process system components. Radiation Protection methods for these systems will be discussed in a later section.

7.1.2.4 Beam Port Plug Activation

During operations not using the beam ports as experimental facilities, shielding plugs are inserted to reduce neutron and gamma exposures through the ports. These plugs have aluminum end caps and sides. The end cap of the shielding plug in Beam Port 1 is exposed to the higher neutron flux, since it rests directly adjacent to the core. Activation of the aluminum in this plug will produce radioisotopes which can produce a dose rate to personnel exposed to the shielding plug upon its removal from the beam port.

It is possible to estimate the dose rate produced from the radioisotopes in the end cap of the beam port plug. First, the basic neutron activation equation can be written as:

$$A_t^i = N^i \sigma^i \phi (1 - e^{-\lambda t'}) e^{-\lambda t}$$

where
- A_t^i = activity of isotope i at time t after irradiation in disintegrations per second,
- N^i = number of "target" atoms of the parent element for isotope i exposure to the neutron flux,
- σ^i = cross-section of the neutron reaction producing isotope i,
- ϕ = neutron flux of energy necessary to produce isotope i in the target,
- λ = decay constant of the activation product,
- t' = irradiation time, and
- t = decay time.

To estimate N^i, assume that the aluminum end cap is made of pure aluminum in the shape of a disk 6 inches in diameter and 0.25 inches thick. The volume of such a cap is approximately 15.2 cm^3. Aluminum has a density of 2.7 $grams/cm^3$, which gives an end cap mass of 41.04 grams. Now, the number of atoms of aluminum available for irradiation is:

$$N^i = m F_1^e F_2^i N_A / M$$

where
- m = total target mass,
- N_A = Avogadro's Number,
- M = molecular weight of the target element,
- F_1^e = elemental purity of the sample for the target element, and
- F_2^i = isotopic fraction of the target element that the parent isotope of the activation product represents.

Since we assume that the end cap is made of pure aluminum, F_1^e is taken to be 1. Similarly, F_2^i is also 1, since all of the naturally-occurring aluminum is ^{27}Al. Taking N_A to be 6.123×10^{23} atoms/mole, and M to be 26.98 grams/mole, the number of aluminum atoms exposed to the neutron flux is thus 9.162×10^{23} atoms.

Now, when aluminum is exposed to a polyenergetic neutron flux, several neutron reactions can occur. These are listed below:

$^{27}Al(n,\gamma)^{28}Al$:

σ = 0.235 barns
λ = 0.301 minutes^{-1}
E_n = thermal

$^{27}Al(n,p)^{27}Mg$:

σ = 0.05 barns
λ = 7.3×10^{-2} minutes^{-1}
E_n = 3 MeV

$^{27}Al(n,\alpha)^{24}Na$:

σ = 0.03 barns
λ = 7.7×10^{-4} minutes^{-1}
E_n = 7 MeV

In the above list, E_n represents the threshold neutron energy required to induce the reaction listed.

For this calculation, it is assumed that the OSURR is operated at 500 kilowatts for 30 hours, then is shut down. Neutron flux in Beam Port 1 at 10 kilowatts has been measured by Horning [2]. Assuming linearity with reactor power, the following values are assumed for neutron flux:

Thermal (0-0.185 eV) : 6.35×10^{12} nv

Fast (0.5-15 MeV) : 2.04×10^{12} nv

The value for fast flux shown above was used to estimate both ^{27}Mg and ^{24}Na activity. The calculation for isotope activities, for various decay times, is shown in Table 7.6.

Using the estimated activities, dose rates can be predicted for the various isotopes at different decay times. Dose rate can be predicted from the following approximation:

$$R = 6CE/D^2$$

where R = dose rate in R/hr,
C = isotope gamma activity in curies,
E = gamma energy in MeV, and
D = distance from the source in feet.

Table 7.6: Activity Estimates for Shielding Plug End Cap

Irradiation Time: 30 Hours
Operating Power : 500 kilowatts

Material: Pure Aluminum
Size : 0.25" thickness, 6" Diameter Disk
Mass : 41.04 grams

Decay Time (min.)	^{28}Al Activity (microcuries)	^{27}Mg Activity (microcuries)	^{24}Na Activity (microcuries)
0	3.70×10^{7}	2.53×10^{6}	1.14×10^{6}
10	1.82×10^{6}	1.22×10^{6}	1.13×10^{6}
60	5.21×10^{-1}	3.18×10^{4}	1.09×10^{6}
360	2.48×10^{-47}	9.96×10^{-6}	8.64×10^{5}
720	0	3.92×10^{-17}	6.55×10^{5}
1440	0	6.07×10^{-40}	3.76×10^{5}

This approximation is valid only for gamma emissions and if the distance from the source is such that the source subtends a small enough solid angle to be considered a point. Generally, a distance of about 5 times the maximum dimension of the source is sufficient to achieve a quasi-point source geometry. For the beam port plug end cap, this will be about three feet. Thus, at three feet, in air, the dose rates from the various activation products can be estimated. The following gamma decay properties were assumed:

^{28}Al:

$E_{\gamma 1}$ = 1.7789 MeV (100% abundance)

^{27}Mg:

$E_{\gamma 1}$ = 0.8438 MeV (72% abundance)
$E_{\gamma 2}$ = 1.0144 MeV (28% abundance)

^{24}Na:

$E_{\gamma 1}$ = 1.3685 MeV (100% abundance)
$E_{\gamma 2}$ = 2.7539 MeV (100% abundance)

The results of the calculations are shown in Table 7.7. These results indicate that the major contributor to long-term radiation exposure from shield plug activation is ^{24}Na. The shorter half-life isotopes will only contribute to doses received a short time after the end of the irradiation, and can thus be reduced or eliminated by allowing sufficient decay time before removing the shielding plug from the beam port. The remaining dose rate from ^{24}Na can be handled by shielding the plug in a cask, or increasing distance from the end cap of the plug. An overhead crane handling system is available for this type of operation.

7.1.2.5 Experimental Sample Activation

Using the equations developed in the preceding section, it is possible to estimate the doses from activation of elements in experimental samples and apparatus, provided that information of the elemental makeup of these samples is available. However, this information is not always known precisely, so it is difficult to accurately estimate the induced activity and resulting doses rates from irradiation samples in all cases. Experimental procedures require that an activity and dose rate estimate be provided based on the best available information about the material to be irradiated. Upon removal from an experimental facility, dose rate surveys are performed to assess the gamma + beta and gamma-only dose rates. If possible, an isotopic assay is also done to determine the type and quantity of induced activation products. Records of all in-pile irradiations are required to be maintained.

Table 7.7: Dose Rate Estimates for Shielding Plug End Cap

Irradiation Time: 30 Hours
Operating Power : 500 kilowatts

Material: Pure Aluminum
Size: 0.25" thickness, 6" Diameter Disk
Mass: 41.04 grams

Source Distance: 3 feet
Absorber: air

Decay Time (min)	^{28}Al Dose Rate (mr/hr)	^{27}Mg Dose Rate (mr/hr)	^{24}Na Dose Rate (mr/hr)	Total Dose Rate (mr/hr)
0	43900	1504	3133	48537
10	2157	725	3109	5991
60	0	19	2992	3011
360	0	0	2375	2375
720	0	0	1800	1800
1440	0	0	1034	1034

7.2 Protection Strategy

The approach taken to radiation protection at the NRL is to keep radiation exposures to personnel as low as reasonably achievable. This approach, sometimes designated as the ALARA concept, incorporates methods and procedures to reduce radiation exposure to individuals working within the confines of the reactor building. The ALARA approach to radiation protection includes use of methods and procedures involving shielding of radiation sources and/or personnel, increasing distance between an exposure point and a radiation source, reducing time a person might be exposed to a given dose rate, containment of sources, and careful, thoughtful, advance planning when working in an area in which might exist a radiation field.

Administrative controls maintained at the NRL are designed to compliment the ALARA approach. All experiments involving reactor use are reviewed by a qualified individual (i.e., one who holds a current, valid Senior Operator License for the OSURR). Requests for reactor use involving radioisotope production or any other potential radiation exposure of significance are examined for indications or estimates of what the level of radiation hazard might be for the experiment, and methods and/or procedures to reduce the potential hazards. If a radiation accident scenario is possible, an experimenter must provide a methodology for dealing with such a scenario and steps to be taken to mitigate its consequences. Conduct of all experiments is subject to the approval and concurrence of the Senior Operator On Duty during the reactor operation. If this individual determines that a particular reactor operation poses a hazardous or potentially unsafe condition, the run may be terminated at his/her discretion.

Production and use of radioactive materials within the reactor building fall under the guidelines issued by the university's Radiation Safety Section. These guidelines in turn fall within the regulatory framework of 10CFR20, and in most cases provide for more stringent controls than those specified in these regulations. In addition, the NRL provides in-house procedures that fall within the guidelines of both the university and federal regulations.

7.3 Protection Methodology

7.3.1 Instruction and Training

Persons working in the reactor building on a daily basis are trained in appropriate radiation protection concepts. In addition, these persons are trained to assist in responding to abnormal radiological conditions in the reactor building. Certification as to having received such instruction is provided as required by 10CFR19. Persons in this category include regular NRL operations, research, and maintenance staff, selected personnel from the University Radiation Safety Section, and student assistants working at the NRL facility.

Individuals not working in the reactor building on a daily basis, but involved in ongoing research and instructional activities at the laboratory on a routine basis, are provided instruction on self-protection against radiation exposure as per 10CFR19 requirements. Extended training in radiological emergency procedures is not given to these persons. Individuals in this protection category include faculty and students involved in research projects and instructional activities at the reactor building, NRL administrative and clerical staff, and custodial workers.

Occasional visitors to the NRL, such as commercial vendors, onetime visitors for tours and demonstrations, and non-routine experiment personnel, are given instruction in basic procedures such as wearing of dose monitors, signing in and out of the building, radiation and radioactive material storage areas. These persons are escorted by an individual more fully trained in radiation protection.

7.3.2 Access Control

The reactor building is designated as a Restricted Area. [redacted]

Certain areas of the reactor building are designated as permanent radioactive materials storage areas. [redacted]. In this area, no eating, drinking, smoking, or applying of cosmetics is allowed. These restrictions also apply to the catwalk areas along the top of the reactor pool. In addition, other areas of the reactor building may be posted as radioactive materials storage areas, in which the same precautions must be observed.

Areas may also be posted as Radiation Areas. Appropriate restrictions and precautions must be observed in these areas. Occasionally, an area may be posted as a High Radiation Area. Again, appropriate precautions must be observed, and access to this area limited.

The sign-in procedure for access to the reactor building requires certain information from an admitted person. This includes their [redacted]. Visitor's logbooks are maintained as part of the permanent records of the NRL.

7.3.3 Personnel Monitoring

Persons working full time in the reactor building are required to wear a film badge. Badge numbers may be noted in the visitor's logbook for persons not registered in the university film badge record system. Film badges used at the NRL contain gamma, beta, and neutron-sensitive materials for monitoring doses at various tissue depths.

For monitoring extremity doses, ring badges based on TLD dosimeters are available from the Radiation Safety Section. Both regular film badges and ring badges are read on a monthly basis.

Immediate dose indications are provided by wearing a self-reading pocket dosimeter. These devices, based on an air ionization and capacitor discharge effect, provide the most convenient indication of exposure in near-real time. Several pocket dosimeters are available at the NRL. These are used to monitor the exposures of individuals in classes at the reactor and are recorded in the Visitor's Logbook.

7.3.4 Area Monitoring

A variety of area radiation monitoring systems are available at the NRL. Section 3.7 discussed the Area Radiation Monitor system installed at the NRL, which provides continuous indications (local and in the control room) of the dose rate at various locations in the reactor building. Other devices are also available to provide information on area dose rates.

A beta-gamma counter is mounted at the sign-in desk at the front of the building. This unit is sometimes called a "frisker" and provides indications of gamma and beta activity. This device uses a detachable Geiger-Mueller probe which can be used to survey areas of a person's body for evidence of contamination, and it provides an indication of overall radioactivity in the area of the front door. It can be used to survey incoming and outgoing shipments of radioactive materials and devices, and can also serve as a monitor for airborne and gaseous radioactive material in the entire reactor building.

A series of four TLDs are mounted at various points in the main reactor bay. These badges, processed every three months, provide a long-term average of area dose in the vicinity of the badges. Similarly, film badges are mounted at eight different locations outside the building.

7.3.5 Survey Instruments

Several portable survey instruments are available at the NRL for routine monitoring of radiation sources. These include Geiger-Mueller and air-ionization chambers for gamma and beta + gamma detection and detector(s) for neutron detection. These survey instruments provide enough range and flexibility to monitor most expected radiation sources produced by OSURR operation. For those sources not detectable with these instruments, the Radiation Safety Section can provide additional support.

7.3.6 Shielding

Earlier sections discussed the radiation shielding features of the OSURR. Additional shielding for sources is obtainable. Normally, the beam ports are plugged with shielding materials which, when coupled with closed beam port shutters, reduce exposures at the beam port exits to negligible levels. For storage of irradiated beam port plugs, a movable shielding cask is available. This cask has a maximum lead thickness of about eight inches.

A large lead storage chest is located in the northeast corner of the building. This chest is used to store calibration sources and highly radioactive irradiated samples and devices. The lead thickness of this storage chest is about four inches. It has a motor-driven, crane-liftable lid to prevent easy access to the sources inside.

Smaller lead containers are available for storage of contained radioactive sources. These containers, commonly called "pigs", vary in wall thickness from a fraction of an inch up to several inches.

7.3.7 Administrative Controls

Safe operation of the OSURR and performance of associated experiments depends on reliable and conscientious observance of established procedures and protocols by the staff of the NRL. Experiments involving production of radioisotope sources of significant quantities and intensities, use of experimental facilities such as beam ports and dry tubes, and the use of radiation sources apart from the reactor, are subject to the approval of appropriate NRL or university administrators. Such experiments and uses must be in accordance with regulations specified in 10CRF20 or other applicable regulations. Administrative controls are established to allow NRL personnel discretion in approving and carrying out experiments and operations a safe manner. NRL personnel may themselves specify appropriate procedures to be incorporated in an experimental program to assure radiological safety. If such procedures are unavailable, the experiment may be denied on that basis. If such procedures are not observed during the experiment, the Senior Reactor Operator on duty during the operation may terminate the experiment at his or her discretion.

7.4 Chapter 7 References

[1] R. G. Jaeger et al., "Engineering Compendium on Radiation Shielding", The Internation Atomic Energy Agency, Vienna, Springer-Verlag, 1968-1975.

[2] Nicholas C. Horning, "Measurement of the Neutron Spectra in the Beam Ports and Thermal Column of The Ohio State University Research Reactor", M.Sc. Thesis (unpublished), Ohio State University, Columbus, OH, 1976.

9.0 Administrative Controls

9.1 Organization

9.1.1 Structure

The Ohio State University Research Reactor is a part of the College of Engineering administered by the Engineering Experiment Station. The organizational structure is shown in Figure 9.1.

9.1.2 Responsibility

The Director of the Engineering Experiment Station (Level 1) is the contact person for communications between the U.S. Nuclear Regulatory Commission and The Ohio State University.

The Director of the Nuclear Reactor Laboratory (Level 2) will have overall responsibility for the management of the facility.

The Associate Director (or Manager of Reactor Operations, Level 3) shall be responsible for the day-to-day operation and for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license and technical specifications. During periods when the Associate Director is absent, his/her responsibilities are delegated to a Senior Reactor Operator (Level 4).

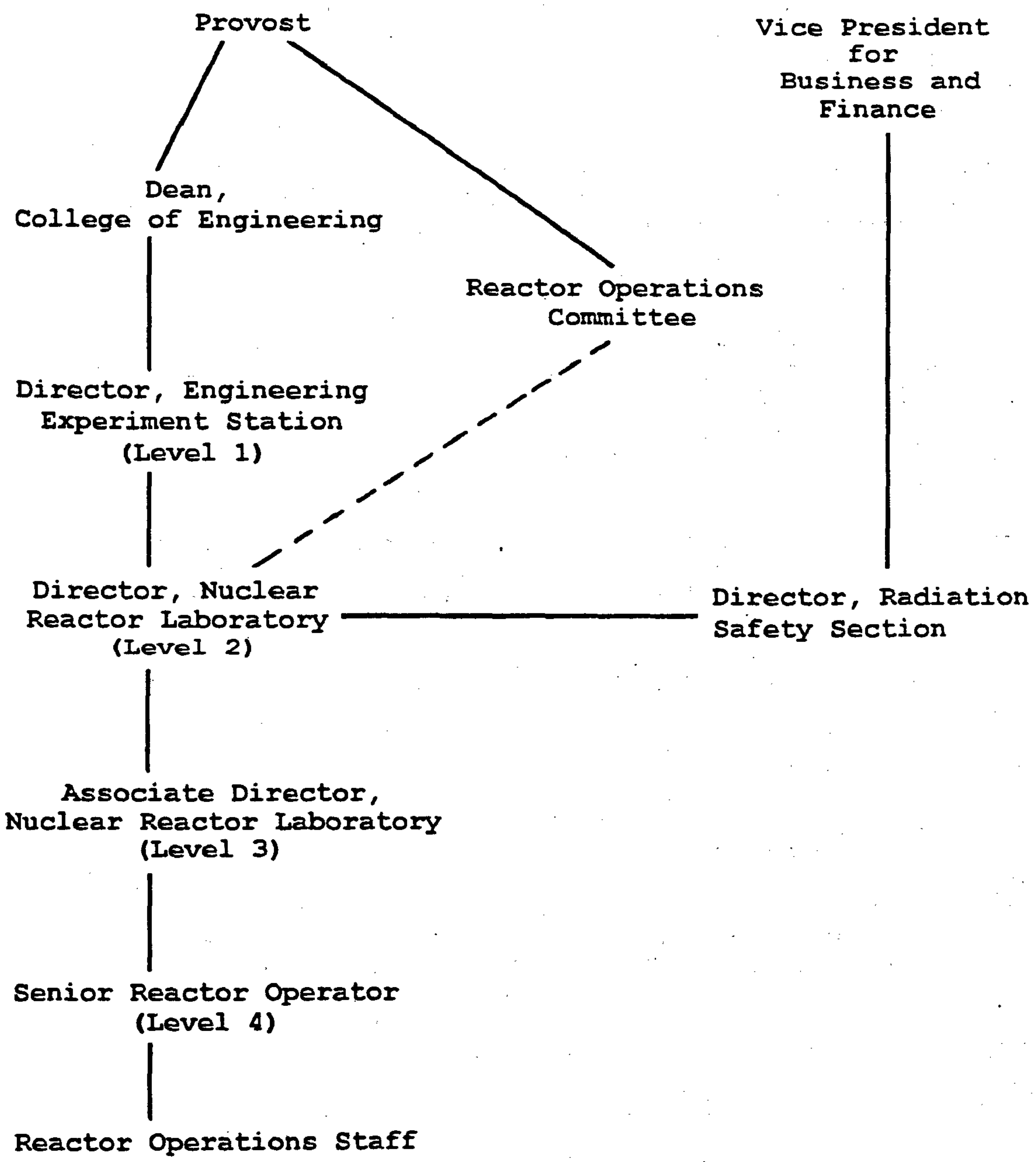
9.1.3 Support Groups

The Radiation Safety Section (RSS) of The Ohio State University provides both support and audit functions for the OSURR. They conduct and review results of area and smear wipes of the Reactor Laboratory, assure proper posting of Radiation and High Radiation Areas, assist in calibration of radiation monitoring instrumentation, inventory and wipe sealed sources, and file required SNM Reports. They are available as needed for response to unplanned emergencies at the reactor or for planned experiments that require additional monitoring. They are on the call list to respond to nuclear emergencies.

The Ohio State University Police Department (OSUPD) is the local law enforcement agency responsible for facility security. The OSUPD is equipped to provide all actions normally associated with a full-service police department, including arrest and prosecution when warranted, and special tactics and investigative units.

The Ohio State University Office of Environmental Health and Safety is available to assist in the monitoring and disposal of non-radioactive hazardous materials from the OSURR.

Fire protection and ambulance services are provided by the Clinton Township Division of Fire or the Cities of Columbus and Upper Arlington.



Solid Lines ——— Paths of Direct Responsibility
Dashed Lines - - - - Paths of Information

Figure 9.1: Administrative Organization

9.2 Training Program

The OSU-NRL shall maintain a RO/SRO requalification program reviewed and approved by the NRC. It will be designed to demonstrate Operator and Senior Operator competence, and to satisfy the requirements of 10CFR55.33(c), 10CFR55 Appendix A, and ANSI/ANS-15.4-1988 Selection and Training of Personnel for Research Reactors. The Manager of Reactor Operations (or Associate Director) shall serve as Training Coordinator and shall be responsible for the implementation, coordination and operation of the Requalification Program including the training of new operators.

9.3 Recordkeeping and Reporting Requirements

Requirements for record keeping and reporting are contained in Section 6 of the Technical Specifications for the OSURR.

9.4 Emergency Planning and Preparedness

The OSURR shall maintain an Emergency Plan reviewed and approved by the NRC. It shall be based on the requirements of Appendix E to 10CFR50 and the criteria set forth in Revision 1 to Regulatory Guide 2.6 and ANSI/ANS 15.16-1982 "Emergency Planning for Research Reactors." The plan shall include a site and facility description; normal and emergency organizational structures; an emergency classification system; an emergency response plan; and provisions to maintain emergency equipment. There shall be Emergency Plan training, drills, and periodic review.

9.5 Internal Reviews and Audits

9.5.1 OSURR Reactor Operations Committee

9.5.1.1 Responsibilities and Authority

There shall be a Reactor Operations Committee (ROC) which shall review and audit reactor operations to assure the facility is operating in a manner consistent with public safety and within the terms of the facility license. The Committee advises the Director of the NRL, and is responsible to the Provost of The Ohio State University (Figure 9.1).

9.5.1.2 Committee Membership

Committee members shall be appointed annually by the Provost of The Ohio State University. The Committee shall be composed of at least nine members including ex-officio members. The Director and Associate Director of the Nuclear Reactor Laboratory, and the Director of the Radiation Safety Section shall be ex-officio voting members of the Committee. The remaining Committee members shall be faculty, staff, and student representatives of The Ohio State University, having professional backgrounds in engineering, physical, biological, or medical sciences, as well as knowledge of and interest in applications of nuclear technology and ionizing radiation.

9.5.1.3 Committee Meetings

The Committee shall meet at least twice a year. They should meet every six months. A quorum shall consist of at least 50 percent of the members.

9.5.1.4 Subcommittees

The chairperson may appoint a subcommittee from within the committee membership to act on those matters which cannot await the regular semi-annual meeting. The full committee shall review the actions taken by the subcommittee at the next regular meeting.

A three member subcommittee shall meet annually to perform an audit of NRL operations and records or to review the results of an independent audit. At least two individuals on the Audit Subcommittee shall be ROC members. The third may be a staff member from the Reactor Laboratory or another individual appointed by the ROC chairperson. Each person should serve for three consecutive audits, at which time he or she should be replaced by a new member. In this way, each subcommittee should consist of two holdovers and one new member. The member serving for his or her second audit should be the Audit Subcommittee Chairperson.

9.5.2 Experiment Approval

All proposed experiments utilizing the reactor shall be evaluated by the experimenter and a licensed Senior Reactor Operator to assure compliance with the provisions of the utilization license, the Technical Specifications and 10CFR20. If, in the judgement of the Senior Reactor Operator, the experiment meets with the above provisions, is an approved experiment, and does not constitute a threat to the integrity of the reactor, it may be approved for performance. When pertinent, the evaluation shall include considerations of:

(1) the reactivity worth of the experiment,

(2) the integrity of the experiment, including the effects of changes in temperature, pressure, or chemical composition,

(3) any physical or chemical interaction that could occur with the reactor components, and

(4) any radiation hazard that may result from the activation of materials or from external beams.

Prior to performing an experiment not previously approved for the reactor, the experiment shall be reviewed and sanctioned by the Reactor Operations Committee. Their review shall consider at least the following information:

(1) the purpose of the experiment,

(2) the procedure for the performance of the experiment, and

(3) the safety evaluation previously approved by a licensed Senior Reactor Operator.

9.5.3 Additional Oversight

9.5.3.1 The Radiation Safety Section (RSS)

The RSS has direct lines of communication to the Director of the NRL, the Provost of The Ohio State University (through the VP for Business and Finance) and the NRC. (See Figure 9.1) This combined with their review and audit functions at the NRL provide additional control over actions that may affect the health or safety of the public.

9.5.3.2 Operations Manager

The Manager of Reactor Operations (or Associate Director) as the designee of the Director of the NRL has direct oversight responsibility for Reactor operations and experiments. He or she may terminate any operation at any time if it is deemed to be or likely to adversely affect the health or safety of the public.

9.5.3.3 SRO On-Duty

In the absence of the Operations Manager the SRO on duty has oversight responsibility and shall make decisions regarding safe operation of the OSURR.

9.6 Security

9.6.1 Security Plan

The Ohio State University Research Reactor shall implement security procedures based on Regulatory Guide 5.59 designed to meet the requirements of 10CFR50, 70, and 73. These procedures shall describe the mechanisms and organization to protect special nuclear material against sabotage and to detect theft and attempted theft.

9.6.2 Security Organization

The Director of the NRL and the Chief of Police of The Ohio State University Police Department (OSUPD) share the responsibility for implementation and operation of the security procedures at the NRL. Day to day security is the responsibility of the NRL staff during normal working hours and whenever an individual authorized access to the NRL is present in the building. At other times, the OSUPD is the local law enforcement agency (LLEA) responsible for facility security. Several other LLEA's also have jurisdiction at the reactor site and would respond if requested by the OSUPD. These include the Clinton Township Police, the Franklin County Sheriff, and in case of civil emergencies, the Ohio State Highway Patrol on order of the Governor.

9.7 Quality Assurance

9.7.1 Quality Assurance Program

The OSURR shall establish and maintain a quality assurance program based on ANS-15.8 ANSI N402-1976 to provide adequate confidence that safety-related items will perform satisfactorily in service. As a minimum, safety-related items included in the plan shall be those identified in the Limiting Conditions for Operation section of the Technical Specifications (Section 3).

9.7.2 Program Requirements

9.7.2.1 Program Requirements

The responsibility for the QA program is delegated to the Director of the NRL as shown in the "Administrative Organization" (Figure 9.1). His/her designee, normally the Associate Director, shall be responsible for the daily implementation of the program. The Reactor Operation Committee has responsibility for independent review and audit functions associated with the program. Reviews may include experimental equipment and the design of safety-related items.

9.7.2.2 Records and Documents

All activities affecting safety-related items identified in Section 3 of the Technical Specifications shall be identified and documented. Procedures shall be established to control the development, revision, and use of documents and drawings which are safety-related. Records of quality assurance activities shall include inspection and test results; QA reviews by the ROC, and analyses of modifications and design changes. Retention requirements shall be established for these records and will include duration, location, and responsibility.

10.0 Financial Qualification

10.1 Financial Ability to Operate a Non-Power Reactor

Since the issuance of the original license in 1960, The Ohio State University has provided funding to assure safe operation of the OSURR. This provides reasonable assurance that operating costs for the next five years will be made available. The current budget allocation provides funding for about 1.9 FTEs plus adequate funds for supplies and equipment. Funding for personnel includes benefits. No direct funds are required for the typical overhead costs of heat, lighting, water, electric, or space utilization.

10.2 Financial Ability to Decommission

In a letter dated July 30, 1990 the OSURR submitted its Decommissioning Funding Plan as required by 10CFR50.33(k). It provides the assurance that The Ohio State University will make funds available for eventual decommissioning.

APPENDIX A

TO

FACILITY OPERATING LICENSE NO. R-75

Technical Specifications

And Bases For

The Ohio State University

Pool-Type Nuclear Reactor

Columbus, Ohio

Docket No. 50-150

TABLE OF CONTENTS

1.0 INTRODUCTION

1.1 Scope

This document constitutes the Technical Specifications for Facility License No. R-75 and supersedes all prior Technical Specifications. Included are the "Specifications" and the "Bases" for the Technical Specifications. These bases, which provide the technical support for the individual technical specifications, are included for information purposes only. They are not part of the Technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere.

This document was, written to be in conformance with ANSI/ANS-15.1-1990. The content of the Technical Specifications includes: Definitions, Safety Limits, Limiting Safety System Settings, Limiting Conditions for Operation, Surveillance Requirements, Design Features, and Administrative Controls.

1.2 Application

1.2.1 Purpose

These Technical Specifications have been written specifically for The Ohio State University Research Reactor (OSURR).

The Technical Specifications represent the agreement between the licensee and the U.S. Nuclear Regulatory Commission on administrative controls, equipment availability, and operational parameters.

Specifications are limits and equipment requirements for safe reactor operation and for dealing with abnormal situations. They are typically derived from the Safety Analysis Report (SAR). These specifications represent a comprehensive envelope for safe operation. Only those operational parameters and equipment requirements directly related to preserving that safe envelope are listed.

1.2.2 Format

The format of this document is in general accordance with ANSI/ANS-15.1-1990.

1.3 Definitions

Administrative Controls - those organizational and procedural requirements established by the Commission and/or the facility management.

ALARA - as low as is reasonably achievable.

Channel - the combination of sensor, line, amplifier, and output devices which are connected for the purpose of measuring the value of a parameter.

Channel Calibration - an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the measured parameter. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip settings, and shall be deemed to include a channel test.

Channel Check - a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

Channel Test - the introduction of a signal into the channel for verification that it is operable.

Cold Clean Core - when the core is at ambient temperature and the reactivity worth of xenon is negligible.

Commission - the U.S. Nuclear Regulatory Commission (or NRC).

Confinement - a closure on the overall facility which controls the movement of air into it and out of it through a controlled path.

Containment - a testable enclosure which can support a defined pressure differential and which is normally closed.

Control Rod - a device fabricated from neutron absorbing material which is used to establish neutron flux changes.

Control Rod Fuel Element - a fuel element capable of holding a control rod.

Controls - mechanisms used to regulate the operation of the reactor.

Core - the general arrangement of fuel elements and control rods.

Critical - when the effective multiplication factor (k_{eff}) of the reactor is equal to unity.

Direct Supervision - in visual and audible contact.

Excess Reactivity - that amount of reactivity that would exist if all control rods were removed from the core.

Exclusion Area - that area around the reactor building in which the licensee has the authority to determine all activities as per 10CFR100.3.

Experiment - any operation, or any apparatus, device, or material installed in or near the core or which could conceivably have a reactivity effect on the core and which itself is not a core component or experimental facility, intended to investigate non-routine reactor parameters or radiation interaction parameters of materials.

Experimental Facility - any structure or device associated with the reactor that is intended to guide, orient, position, manipulate, or otherwise facilitate completion of experiments.

Explosive Material - any material that is given an Identification of Reactivity (Stability) index of 2, 3, or 4 by the National Fire Protection Association in its publication 704-M, Identification System for Fire Hazards of Materials, or is enumerated in the Handbook for Laboratory Safety published by the Chemical Rubber Company (1967).

Facility - the Reactor Building including offices and laboratories.

Fueled Experiment - any experiment that contains U-235 or U-233 or Pu-239, not including the normal reactor fuel elements.

Licensee - The Ohio State University.

Limiting Conditions for Operation (LCO) - the lowest functional capability or performance levels of equipment required fur safe operation of the facility. LCO are administratively established constraints on equipment and operational characteristics.

Limiting Safety System Settings (LSSS) - settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting shall be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.

Measured Value - the value of a parameter as it appears on the output of a channel.

Movable Experiment - one for which it is intended that all or part of the experiment may be moved in relation to the core while the reactor is operating.

Nuclear Regulatory Commission - (NRC).

Onset of Nucleate Boiling - (ONB).

Operable - a component or system which is capable of performing its intended functions in a normal manner.

Operating - a component or system which is performing its intended function.

Protective Action - the initiation of a signal or the operation of equipment within the reactor safety system in response to a variable or condition of the reactor facility having reached a specified limit.

Reactivity Limits - those limits imposed on reactor core excess reactivity based upon a reference core condition.

Reactivity Worth of an Experiment - the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter an experiment's position or configuration.

Reactor - the combination of core, permanently installed experimental facilities, control rods, and connected control instrumentation.

Reactor Operating - whenever the reactor is not secured or shutdown.

Reactor Operations Committee - (ROC).

Reactor Operator (RO) - an individual who is licensed to manipulate the controls of the reactor in accordance with 10CFR55.

Reactor Safety Systems - those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

Reactor Secured - whenever (1) all shim/safety rods are fully inserted, (2) the console key is in the OFF position and is removed from the lock, and (3) no in-core work is in progress involving fuel or experiments or maintenance of the core structure, control rods, or control rod drive mechanisms.

Reactor Shutdown - when the reactor is subcritical by at least 1% delta k/k in the cold clean core condition.

Regulating Rod - a low reactivity-worth control rod used primarily to maintain an intended power level. Its position may be varied either by manual control or by the automatic servo-controller.

Reportable Occurrence - any of the conditions described in Section 6.5.2 of these specifications.

Restricted Area - the Reactor Building to which access is controlled for purposes of protection of individuals from exposure to radiation and radioactive materials.

Safety Analysis Report - (SAR).

Safety Channel - a measuring or protective channel in the reactor safety system.

Safety Limits (SL) - limits on important process variables which are found to be necessary to reasonably protect the integrity of certain physical barriers which guard against the uncontrolled release of radioactivity.

Scram - the rapid insertion of the shim/safety rods into the reactor for the purpose of quickly shutting down the reactor.

Scram Time - the elapsed time between reaching a limiting safety system setting and the time when a control rod is fully inserted.

Secured Experiment - any experiment, experimental facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected from the normal environment of the experiment or by forces which can result from credible malfunctions.

Senior Reactor Operator (SRO) - an individual who is licensed to direct the activities of reactor operators. Such an individual may also operate the controls of the reactor pursuant to 10CFR55.

Shall, Should, and May - the word "shall" is used to denote a requirement; the word "should" to denote a recommendation; and the word "may" to denote permission, which is neither a requirement nor a recommendation.

Shim/Safety Rods - high-reactivity worth control rods used primarily to provide coarse reactor control. They are connected electromagnetically to their drive mechanisms and have scram capabilities.

Shutdown Margin - the shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems with the most reactive shim/safety rod and the regulating rod in the most reactive position (fully withdrawn) and that the reactor will remain subcritical without further operator action.

Standard Fuel Element - an element to be used or stored in the core, fuel storage pit or other approved area, but not a control rod element.

Startup Source - a spontaneous source of neutrons which is used to provide a channel check of the startup (fission chamber) channel and provide neutrons for subcritical multiplication during reactor startup.

Surveillance Time Intervals - The average over any extended period for each surveillance time interval shall be closer to the normal surveillance time, e.g. for the two year interval the average shall be closer to two years rather than 30 months.

two-year	(interval not to exceed 30 months).
annually	(interval not to exceed 15 months).
semiannually	(interval not to exceed 7-1/2 months).
quarterly	(interval not to exceed 4 months).
monthly	(interval not to exceed 6 weeks).
weekly	(interval not to exceed 10 days).
daily	(shall be done during the same working day).

Any extension of these intervals shall be occasional and for a valid reason and shall not affect the average as defined.

True Value - the actual value of a parameter.

Unscheduled Shutdowns - any unplanned shutdown of the reactor caused by actuation of the reactor safety systems, operator error, equipment malfunction, or a manual shutdown in response to conditions which could adversely affect safe operation. They do not include those shutdowns resulting from expected testing operations, or planned shutdowns, whether initiated by controlled insertion of control rods or planned manual scrams.

2.0 SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTINGS (LSSS)

2.1 Safety Limit

Applicability: This specification applies to the melting temperature of the aluminum fuel cladding.

Objective: The objective is to assure that the integrity of the fuel cladding is maintained.

Specification: The reactor fuel temperature shall be less than 550 °C.

Bases: The melting temperature of aluminum is 660 °C (1220 °F). The blister threshold temperature for U_3Si_2 dispersion fuel has been measured as approximately 550 °C. (ANL/RERTR/TM-10, October 1987, NRC NUREG 1313). Because the objective of this specification is to prevent release of fission products, any fuel whose maximum temperature reaches 550 °C. is to be treated as though the safety limit has been reached until shown otherwise.

2.2 Limiting Safety System Settings

Applicability: This specification applies to the following items associated with core thermodynamics:

(1) Reactor Thermal Power Level and

(2) Reactor Coolant Inlet Temperature.

Objective: To assure that the fuel cladding integrity is maintained.

Specification:

(1) Steady state power level shall not exceed 500 kW thermal.

(2) Reactor safety systems settings shall initiate automatic protective action so that reactor thermal power level shall not exceed 600 kW (120% of full power) during a transient.

(3) Reactor safety systems settings shall initiate automatic protective action so that core inlet water temperature shall not exceed 35 °C.

Bases: The criterion for these safety system settings is established as the fuel integrity. If the temperature of the clad is maintained below that for blister threshold then cladding integrity is maintained. This is the case for a power level of 600 kW and a core inlet temperature of 35 °C (normal inlet temperature is ≈ 20-25 °C). The maximum credible accident analysis is provided in Section 8.4.3.2 of the Safety Analysis Report. The maximum credible accident assumes steady state operation at 600 kW and a transient to 750 kW. The maximum temperature of the cladding reaches 91 °C (SAR 8.4.3.3). One may also reference SAR Sections 4.8.1, 4.8.2 for an estimate of cladding temperature during steady state operation at 500 kW (56.5 °C).

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Core Parameters

3.1.1 Reactivity

Applicability: These specifications apply to the reactivity condition of the reactor and the reactivity worths of the shim/safety rods and regulating rod under any operating conditions.

Objective: To ensure safe shutdown of the reactor and that the safety limits are not exceeded.

Specification: The reactor shall be operated only if the following conditions exist:

(1) The reactor core shall be loaded so that the excess reactivity, including the effects of installed experiments does not exceed 2.6% delta k/k under any operating condition.

(2) The minimum shutdown margin under any operating condition with the maximum worth shim/safety rod and the regulating rod full out shall be no less than 1.0% delta k/k.

(3) The total reactivity worth of the regulating rod shall be less than 0.7% delta k/k.

(4) All core grid positions internal to the active fuel boundary shall be occupied by a standard, control, regulating rod, instrumented, or blank fuel element; or by an experimental facility.

(5) The moderator temperature coefficient shall be negative and shall have a minimum absolute reactivity value of at least $2x10^{-5}/°C$ across the active core at all normal operating temperatures.

(6) The moderator void coefficient of reactivity shall be negative and shall have a minimum value of at least $2.8x10^{-3}$ /1% void across the active core.

Bases:

(1) The maximum allowed excess reactivity of 2.6% delta k/k provides sufficient reactivity to accommodate fuel burnup, xenon buildup, experiments, control requirements, and fuel and moderator temperature feedback (Section 4.2 of the SAR). Also, calculations show that this excess reactivity assures that the maximum temperature of the surface of the cladding will be well below the blister threshold of the U_3Si_2 fuel

during a design basis accident (SAR 8.4.3.2).

(2) The minimum shutdown margin ensures that the reactor can be shutdown from any operating condition and remain shutdown after cooling and xenon decay even with the highest worth rod and the regulating rod fully withdrawn.

(3) Limiting the reactivity worth of the regulating rod to a value less than the effective delayed neutron fraction assures that a failure of the automatic servo control system cannot result in a prompt critical condition.

(4) The requirement that all grid positions be filled during reactor operation assures that the volume flow rate of primary coolant which bypasses the heat producing elements will be within the range specified in Section 4.8 of the SAR. Furthermore, the possibility of accidentally dropping an object into a grid position and causing increase of reactivity is precluded.

(5) A negative moderator temperature coefficient of reactivity assures that any moderator temperature rise will cause a decrease in reactivity. The U_3Si_2 fuel also has a significant negative temperature coefficient of reactivity due to the Doppler broadening of neutron capture resonances in ^{238}U, but no credit is taken for this effect in our safety analyses.

(6) A negative void coefficient of reactivity helps provide reactor stability in the event of moderator displacement by experimental devices or other means.

3.2 Reactor Control and Safety System

3.2.1 Control Rod Drop Times

Applicability: This specification applies to the time from the receipt of a safety signal to the time it takes for a shim/safety rod to drop from fully withdrawn to fully inserted.

Objective: To ensure that the reactor can be shutdown within a specified period of time.

Specification: The reactor will not be operated unless the drop time of each of the three shim/safety rods is less than 600 msec.

Bases: Control rod drop times as specified ensure that the safety limit will not be exceeded in a short period transient.

The analysis for this is given in Section 4.3.3 of the SAR.

3.2.2 Maximum Reactivity Insertion Rate

Applicability: This applies to the maximum positive reactivity insertion rate by the most reactive shim/safety rod and the regulating rod simultaneously.

Objective: To ensure the reactor is operated safely and the safety limit is not exceeded due to a short period.

Specification: The reactor will not be operated unless the maximum reactivity insertion rate is less than 0.02% delta k/k per second.

Basis: This maximum reactivity insertion rate assures that the Safety Limit will not be exceeded during a startup accident due to a short period generated by a continuous linear reactivity insertion.

3.2.3 Minimum Number of Scram Channels

Applicability: This specification applies to the reactor safety system channels.

Objective: To stipulate the minimum number of reactor safety system channels that shall be operable to ensure the Safety Limits are not exceeded by ensuring the reactor can be shutdown at all times.

Specification: The reactor shall not be operated unless the safety system channels described in the following table are operable.

Reactor Safety System Component	**Minimum Required**	**Function**
1. Core H_2O Inlet Temp.	1	Slow scram if temp. $\geq$ 35°C
2. Reactor Thermal power level (Safety Channels)	2	Fast scram if thermal power $\geq$ 600 kW, as indicated on calibrated ionization chamber channels.
3. Reactor Period	1	Fast scram if period $\leq$ 1 sec
4. Reactor Thermal power level/coolant system pumps	1	Slow scram if coolant system pumps not on by $\geq$ 120 kW thermal power
5. Coolant Flow Rate	1	Slow scram if coolant system has no flow (primary) by $\geq$ 120 kW thermal power

Reactor Safety System Component	Minimum Required	Function
6. Pool Water Level	1	Slow scram if pool level ≤ 20 feet (15 feet above core)
7. Switches a. Magnet Power Key "On" b. Effluent Monitor counter in "Count" c. Period Generator Switch "Off" d. LOG-N Amp Calibrate Switch "Norm" e. LOG-Period Amp Calibrate Switch "Norm" f. Effluent Monitor Compressor "On"	6	Slow scram if any one switch is not properly set at the position indicated in quotes (Also prohibits startup)
8. Recorders a. LOG-N b. Linear Level c. Start-Up Channel d. Period e. Effluent Monitor	5	Slow scram if power is lost to any one of the listed recorders
9. Manual Scrams a. Control Room Console b. Pool Top Catwalk c. BSF Catwalk d. Rabbit/BP Area e. Thermal Column/BP Area	5	Slow scram upon activation of any one manual scram switch
10. Compensated Ion Chambers	2	Slow scram if voltage drops below operational specifications

Reactor Safety System Component	Minimum Required	Function
11. Safety Set Points On Recorders	4	Slow scram if associated recorder values are exceeded
a. Period		$\leq$ 5 sec
b. Linear Level		$\geq$ 120% of licensed power
c. Start-Up Channel		$\leq$ 2 cts/sec (may be bypassed if $K_{eff} < 0.9$)
12. Safety System	2	Slow scram in case of a safety amp fault or if system is discontinuous
13. Backup Shutdown Mechanisms	3	Rod drop will occur for any control rod which has excess magnet current $\geq$ 100 ma

Bases:

1. Assures safety limit is not exceeded; core inlet temperature is same as cooling system outlet

2. Assures safety limit is not exceeded

3. Assures safety limit is not exceeded

4. Assures coolant system pumps are functional before raising power > 120 kW

5. Assures there is always primary coolant flow when greater than 120 kW

6. Assures there is enough primary coolant for natural convection cooling

7. Assures nuclear instrumentation is in proper mode for operation

8. Assures information is available for observation by the reactor operator during operation, and is recorded if required as a record of reactor operations

9. Assures that the reactor can be shut down by the reactor operator in the control room or at other locations near experimental facilities if deemed necessary by other reactor

staff

10. Assures shutdown if nuclear instrumentation fails

11. Assures backup shutdown capability from short period or high power level. Assures shutdown if count rate is too low to provide meaningful startup information. The startup interlock may be bypassed if K_{eff} is ≤ 0.9

12. Assures all components of the safety system are installed and operational

13. Assures that any control rod exhibiting excess magnet current will be released and fall to the bottom due to gravity

3.3 Coolant System

3.3.1 Pump Requirements

Applicability: This specification applies to the operation of pumps for both the primary and secondary coolant loops.

Objective: To ensure that both pumps are functioning whenever the reactor is operated above 120 kW.

Specification: The reactor will not be operated above 120 kW unless both the primary and secondary coolant pumps are activated and there is flow in the primary coolant loop.

Bases: Having both pumps operating and flow in the primary loop will ensure there is adequate cooling of the primary coolant so the Safety Limit is not exceeded.

3.3.2 Coolant Level

Applicability: This specification applies to the height of the water in the Reactor Pool above the core.

Objective: To ensure there is adequate primary coolant in the Reactor Pool and sufficient biological shielding above the core.

Specification: The reactor shall not be operated unless there is 20 feet of water in the reactor pool and 15 feet of water above the core.

Bases: With the pool full of water to a level of 20 feet there is adequate primary coolant for natural convection cooling. With 15 feet of water above the core there is sufficient shielding at the licensed power level. Section 7.1.1.4 of the SAR discusses this shielding.

3.3.3 Water Chemistry Requirements

Applicability: This specification applies to the purity of the primary coolant water.

Objective: To minimize corrosion of the cladding on the fuel elements, and to reduce the probability of neutron activation of ions in the water.

Specification:

(1) The conductivity of the pool water shall not exceed the limit of 2.0 μ mho/cm.

(2) The pH of the pool water shall not exceed 8.0.

Bases: Operation in accordance with these specification ensures aluminum corrosion is within acceptable limits, and that the concentration of dissolved impurities that could be activated by neutron irradiation remains within acceptable limits.

3.3.4 Leak, or Loss of Coolant Detection

Applicability: This specification applies to the capability of detecting and preventing the loss of primary coolant.

Objective: To ensure there is adequate primary coolant in the Reactor Pool and sufficient biological shielding above the core when the reactor is operating.

Specification: The pool water level shall be at least 15 feet above the top of the fuel in the core.

Bases: The same system that functions to scram the reactor on low pool level will also be used as the detection system for this specification. Design criteria of the cooling system to prevent large losses of pool water due to siphoning are discussed in Section 3.2.2.1 of the SAR.

3.3.5 Primary and Secondary Coolant Activity Limits

Applicability: This specification applies to the buildup of radioactive materials in the secondary coolant system.

Objective: To ensure there is a level low enough so as not to exceed 10CFR20 limits if coolant is released to the sanitary sewer system.

Specification: The primary and secondary coolant system shall be monitored for the buildup of radioactivity and analyzed at least semiannually for increase in the concentration of radionuclides.

Basis: The basis for this specification is to ensure releases are legal and consistent with the ALARA principal.

3.4 Confinement Isolation

Applicability: This specification applies to the capability of isolating the reactor building from the unrestricted area outside.

Objective: To prevent the exposure of the public to airborne radioactivity exceeding the limits of 10CFR20, and the ALARA principle.

Specification: The reactor shall not be operated unless the following conditions are met:

(1) Ventilation fan operating

(2) Reactor Building bay door closed

(3) Reactor Building front and rear personnel doors closed

(4) Office windows closed

Bases: By having the capability to isolate the Reactor Building, the release of airborne radioactive material may be confined and limited to the extent analyzed in the ***revised SAR of September 1987***.

3.5 Ventilation Systems

Applicability: This specification applies to all heating, ventilating, and air conditioning systems that exhaust building air to the outside environment.

Objective: To provide for normal ventilation and the reduction of airborne radioactivity within the reactor building during normal reactor operation and to provide a way to turn off all vent systems quickly in order to isolate the building for emergencies.

Specification:

(1) An exhaust fan with a capacity of at least 1000 cfm shall be operable whenever the reactor is operating.

(2) This fan, as well as all other heating, ventilating, and air conditioning systems shall have the capability to be shut off from a single switch in the control room.

Bases: In the unlikely event of a release of fission products or other airborne radioactivity, the ventilation system will reduce radioactivity inside the reactor building or be able to be isolated. An analysis of fission product release is found in section 8.4.4 of the SAR.

3.6 Radiation Monitoring Systems and Radioactive Effluents

3.6.1 Radiation Monitoring

Applicability: This specification applies to the availability of radiation monitoring equipment which shall be operable during reactor operation.

Objective: To assure that monitoring equipment is available to evaluate radiation levels in restricted and unrestricted areas and to be consistent with ALARA.

Specification:

(1) When the reactor is operating, the building gaseous effluent monitor shall be operating and have a readout and alarm in the control room. It may be used in either the "normal" mode or "sniffer" mode.

(2) When the reactor is operating and the rabbit experimental facility is used, the rabbit monitoring system shall be operating and have a readout and alarm in the control room.

(3) When the reactor is operating, the following Area Radiation Monitors (ARMs) shall be operating and have both local and control room readouts and alarms.

a. Pool Top
b. Primary Cooling System
c. Beam Port/Rabbit Area
d. Thermal Column Area

(4) Portable survey instrumentation shall be available whenever the reactor is operating to measure beta-gamma exposure rates and neutron dose rates.

(5) Portable instruments, surveys, or analyses may be substituted for the instruments in the above sections (3.6.1.1, 3.6.1.2, or 3.6.1.3) for periods up to 48 hours. Read-out and alarms from these temporary instruments shall be reported to the reactor operator on duty at least once per hour.

Bases:

(1) The gaseous effluent monitor will detect Ar-41 levels in the reactor building. During "normal" mode operation it will sample and monitor air just before it is released from the reactor building. (SAR 6.3.1) During "sniffer" mode of operation it may be used for short periods to monitor in and around experimental facilities to determine local Ar-41 levels.

(2) The rabbit stack monitor is used with the rabbit since the rabbit system uses air as its transport mechanism and Ar-41 production takes place. This monitor will provide warning if Ar-41 levels being released in the building are too high (SAR 6.3.2 and 6.3.4.3).

(3) The ARMs provide a continuing evaluation of the radiation levels within the Reactor Building (SAR 3.7) and provide a warning if levels are higher than anticipated.

(4) The availability of survey meters enables the Reactor Staff to independently confirm radiation levels throughout the building.

(5) In the event of instrument failure short term substitutions will enable the safe continued operation of the Reactor.

3.6.2 Radioactive Effluents

Applicability: This specification applies to the monitoring of radioactive effluents from the facility.

Objectives:

(1) To ensure that liquid radioactive releases are safe and legal.

(2) To ensure that the release of Ar-41 beyond the site boundary does not result in concentrations above the Annual Average Concentration (10CFR20.1302 b(i)) for unrestricted area.

(3) To assure that the release of Ar-41 in the restricted area does not result in concentrations above the DAC.

Specifications:

(1) The release rate for radioactive liquids beyond the site boundary shall not exceed the limits as specified in 10CFR20 at the point of release.

(2) The concentration of Ar-41 at the point of release into the

unrestricted area shall not exceed the unrestricted area Annual Average Concentration (AAC) (10CFR20.1302 b(i)) when averaged over one year or 10 x AAC when averaged over one day.

(3) The concentration of Ar-41 in the restricted area shall not exceed the DAC when averaged over a 2000 hour work year.

Bases:

(1) The basis for this specification is found in Section 6.2 of the Safety Analysis Report.

(2) The basis for this specification is found in Section 6.3 of the Safety Analysis Report.

(3) The basis for this specification is found in Section 6.3 of the Safety Analysis Report and 10CFR20.1003.

3.7 Experiments

3.7.1 Reactivity Limits

Applicability: This specification applies to experiments to be installed in or near the reactor and associated experimental facilities.

Objectives: To prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specification:

(1) The absolute value of the reactivity worth of any single secured experiment shall not exceed 0.7% delta k/k.

(2) The absolute value of the reactivity worth of any single movable experiment shall not exceed 0.4% delta k/k.

(3) The absolute value of the reactivity worth of all movable experiments shall not exceed 0.6% delta k/k.

(4) The absolute value of the reactivity worth of experiments having moving parts shall be designed to have an insertion rate less than 0.05% delta k/k per second.

(5) The absolute value of the reactivity worth of any movable experiment that may be oscillated shall have a reactivity change of less than 0.05% delta k/k.

(6) The total reactivity worth of all experiments shall not be

greater than 0.7% delta k/k.

Bases:

(1) The bases for specifications 1, 2, 3, and 6 are found in Section 8.4.3.2 of the SAR which evaluates a step insertion of reactivity from an experiment.

(2) The bases for specifications 4 and 5 allows for certain reactor kinetics experiments to be performed but still limits the rate of change of reactivity insertions to levels that have been analyzed. Section 8.4.3.2 of the SAR evaluates a step insertion of reactivity from an experiment.

3.7.2 Design and Materials

Specification:

(1) No experiment shall be installed that could shadow the nuclear instrumentation, interfere with the insertion of a control rod, or credibly result in fuel element damage.

(2) All materials to be irradiated in the reactor shall be either corrosion resistant or doubly encapsulated within corrosion resistant containers.

(3) Explosive materials shall not be allowed in experiments, except for neutron radiographic exposures of items performed outside of the core and experimental facilities. The amount of explosive material contained in capsules used for radiographic exposures shall not exceed 5 grains of gunpowder.

Bases:

(1) Specification 1 assures no physical interference with the operation of the reactor detectors, control rods, or physical damage to fuel element will take place.

(2) Limiting corrosive materials in Specification 2, and explosives in Specification 3 reduces the likelihood of damage to reactor components and/or releases of radioactivity resulting from experiment failure.

(3) Limiting explosive materials to neutron radiographic exposures done outside of the core and experimental facilities reduces the likelihood of damage resulting for this experimental failure.

4.0 SURVEILLANCE REQUIREMENTS

4.1 Reactor Core Parameters

4.1.1 Excess Reactivity and Shutdown Margin

Applicability: This specification applies to surveillance requirements for determining the excess reactivity of the reactor core and its shutdown margin.

Objective: To assure that the excess reactivity and shutdown margin limits of the reactor are not exceeded.

Specifications:

(1) Whenever a net change in core configuration, for which the predicted change in reactivity is > 0.2% delta k/k, involving grid position is made, both excess reactivity and shutdown margin shall be determined.

(2) Both shutdown margin and excess reactivity shall be determined annually.

Bases: A determination of excess reactivity is needed to preclude operating without adequate shutdown margin. Moving a component out of the core and returning it to its same location is not a change in the core configuration and does not require a determination of excess reactivity.

4.1.2 Fuel Elements

Applicability: This specification applies to surveillance requirements for determining the physical condition of the reactor fuel.

Objective: To ensure that visible deterioration, corrosion, or other physical changes to the fuel elements are detected in a timely manner.

Specification: All fuel elements, both in-core and out, shall be visually inspected at least once every five years, by inspecting at least one fifth of the elements annually.

Basis: If the water purity is continuously maintained within specified limits, it is projected that chemical corrosion of the fuel clad will proceed slowly. However, faults in the basic materials or fabrication could lead to loss of cladding integrity.

4.2 Reactor Control and Safety Systems

4.2.1 Control Rods

Applicability: This specification applies to the surveillance requirements for the shim safety rods and the regulating rod.

Objective: To assure that all rods are operable.

Specifications:

(1) The reactivity worth of the shim safety rods and regulating rod shall be determined annually and prior to the routine operation of any new core configuration.

(2) Shim safety rod drop and drive times and regulating rod drive time shall be determined annually or after maintenance or modification is completed on a mechanism.

(3) The shim safety rods and regulating rod shall be visually inspected annually for indication of corrosion and indication of excessive friction with guides.

Bases: The reactivity worth of the rods is measured to assure the required shutdown margin and reactivity insertion rates are maintained. It also provides a means for determining the reactivity of experiments. Measuring annually will provide corrections for burnup and after core changes assures that altered rod worths will be known prior to continued operations.

The visual inspection of the rods and measurements of drive and drop times are made to assure the rods are capable of performing properly. Verification of operability after maintenance or modification of the control system will ensure proper reinstallation.

4.2.2 Reactor Safety System

Applicability: This specification applies to the surveillance requirements for the Reactor Safety System.

Objective: To assure the reactor safety system channels will remain operable and prevent safety limits from being exceeded.

Specification:

(1) A channel check of each measuring channel shall be performed daily when the reactor is operating.

(2) A channel test of each measuring channel shall be performed

prior to each day's operation or prior to each operation extending more than one day.

(3) A channel calibration of the reactor power level measuring channels shall be made annually. (Linear Level and LOG-N.)

(4) A channel calibration of the Level and Period Safety Channels shall be made annually. Channel tests are done on these before each day's operation.

(5) A channel calibration of the following shall be made annually

a. Core inlet temperature measuring system
b. Pool water level measuring system
c. Coolant system pumps measuring system
d. Primary coolant flow measuring system

(6) The control room manual scram shall be verified to be operable prior to each day's operation. All other manual scram switches shall be tested annually.

(7) Other scram channels shall be tested/calibrated annually.

(8) Any instrument channel replacement shall be calibrated after installation and before utilization.

(9) Any instrument repair or replacement shall have a channel test prior to reactor operation.

Bases: The daily channel tests and checks will assure that the scram channels are operable. Appropriate annual tests or calibrations will assure that long term functions not tested before daily operation are operable.

4.3 Coolant System

4.3.1 Primary Coolant Water Purity

Applicability: This specification applies to the conductivity of the primary coolant water.

Objective: To assure high quality pool water.

Specification: The conductivity and pH of the pool water shall be measured weekly.

Bases: This assures that changes that might increase the corrosion rate are detected in a timely manner and that the concentrations of impurities that might be made radioactive do not increase significantly.

4.3.2 Coolant System Radioactivity

Applicability: This specification applies to the radioactive material in the primary coolant or secondary coolant.

Objective: To identify radionuclides as potential sources of release to the sanitary sewer system.

Specification: Primary and secondary coolant shall be analyzed for radioactivity quarterly or before release.

Bases: Radionuclide analysis of the pool water or secondary coolant allows for determination of any significant buildup of fission or activation products and helps assure that radioactivity is not permitted to escape to the tertiary system in an uncontrolled manner.

4.4 Confinement

Applicability: This specification applies to the surveillance requirements for building confinement.

Objective: To assure that the building closure capability exists.

Specification: A monthly test shall be made to assure that the building exhaust fan, bay door, front and rear personnel doors, and office doors and windows are operable.

Bases: Monthly surveillance of this equipment will verify that the confinement of the reactor bay can be maintained if needed.

4.5 Ventilation System

Applicability: This specification applies to the surveillance requirements for the building ventilation system.

Objective: To assure that the ventilation system functions satisfactorily.

Specification:

(1) Ventilation fans and closures shall be checked for proper operation on a quarterly basis.

(2) The shutoff switch for all fans and air conditioning systems shall be tested an a quarterly basis.

Bases: This surveillance will assure that during normal operations the airborne radioactivity will be minimized inside the building and that the building can be isolated quickly if necessary to prevent uncontrolled escape of air-borne radioactivity to the unrestricted environment.

4.6 Radiation Monitoring Systems and Radioactive Effluents

4.6.1 Effluent Monitor

Applicability: This specification applies to the surveillance requirement of the effluent monitor.

Objective: To assure the effluent monitor is operational and providing accurate effluent readings.

Specification: The effluent monitor shall have a channel calibration annually and a channel test before each days operation.

Bases: The calibration will assure effluent release estimates are accurate and the test will assure the monitor is operable whenever the reactor is operating.

4.6.2 Rabbit Vent Monitor

Applicability: This specification applies to the surveillance requirements of the rabbit vent monitor.

Objective: To assure the monitor is operational and providing meaningful information about effluent releases from the rabbit into the reactor building.

Specification: The monitor shall have a channel calibration annually and a channel test before each day's reactor operation.

Bases: The calibration will assure effluent releases inside the building are accurately estimated and the test will assure the monitor is operable before the rabbit is used.

4.6.3 Area Radiation Monitors (ARMs)

Applicability: This specification applies to the area radiation monitoring equipment.

Objective: To assure that radiation monitoring equipment is operable whenever the reactor is operating.

Specification: A channel test of the ARMs shall be completed before each day's operation and a channel calibration shall be completed annually.

Bases: Calibration annually will insure the required reliability and a check on days when the reactor is operated will detect obvious malfunctions in the system.

4.6.4 Portable Survey Instrumentation

Applicability: This specification applies to the portable survey instrumentation available to measure beta-gamma exposure rates and neutron dose rates.

Objective: To assure that radiation survey instrumentation is operable whenever the reactor is operating.

Specification: Beta-gamma and neutron survey meters shall be tested for operability each day the reactor is to be operated and shall be calibrated annually.

Bases: Tests on days when the reactor is operated will detect obvious detector deficiencies and an annual calibration will assure reliability.

5.0 DESIGN FEATURES

5.1 Site and Facility Description

5.1.1 Facility Location

The reactor and associated equipment is housed in a building at 1298 Kinnear Road, Columbus, Ohio. The minimum free air volume of the building housing the reactor will be a 70,000 ft. There is an exhaust fan with dampers providing for control of release of airborne radioactivity. It is in the area of The Ohio State University Research Center.

5.1.2 Exclusion and Restricted Area

The fence surrounding the Research Center shall describe the exclusion area. The restricted area as defined in 10CFR20 shall consist of the Reactor Building.

5.2 Reactor Coolant System

5.2.1 Primary Coolant Loop

Natural convective cooling is the primary means of heat removal from the core. Water enters the core at the bottom and flows upward through the flow channels in the fuel elements.

5.2.2 Secondary and Tertiary Coolant Loops

The secondary coolant loop removes heat from the primary coolant. The secondary coolant (ethylene glycol and water) passes through two separate heat exchangers to remove heat if necessary. Heat is removed from the first by an outside fan-forced dry cooler. City water flow through the secondary side of an additional heat exchanger makes up the tertiary loop. It provides additional cooling for the secondary coolant.

5.3 Reactor Core and Fuel

Up to 30 positions on the core grid plate are available for use as fuel element positions. Control rod fuel elements occupy four of these positions and one is reserved for the Central Irradiation Facility flux trap. Several arrangements for the cold, clean, critical core have been investigated. Approximately [redacted] standard fuel elements in addition to the control rod fuel elements are required. Partial elements, core filler elements, and graphite elements may be utilized in various combinations to achieve the proper K excess.

The reactor fuel is The DOE Standard uranium-silicide (U_3Si_2) with a U-235 enrichment of less than 20%. It is flat plate fuel with a "meat" thickness of 0.020" and aluminum cladding of 0.015". Standard fuel elements have a total of 16 fueled plates and 2 outer pure aluminum plates. The control rod fuel elements have eight of the inner fuel plates removed to allow the control rods to enter. Pure aluminum guide plates are on the inside of this gap. The outer two plates for each control rod assembly are fueled. Partial elements are also available with 25, 40, 50, and 60 percent of the nominal loading of a standard element. These partial fuel elements are prefabricated by the vendor with fixed numbers of plates.

(1) References: NRC NUREG 1313
ANL/RERTR/TM-10
ANL/RERTR/TM-11

5.4 Fuel Storage

The fuel storage pit, located below the floor of the reactor pool and at the end opposite from the core, shall be flooded with water whenever fuel is present and shall be capable of storing a complete core loading. When fully loaded with fuel and filled with water K_{eff} shall not exceed 0.90, and natural convective cooling shall ensure that no fuel temperatures reach a point at which ONB is possible.

5.5 Fuel Handling Tools

All tools designed for or capable of removing fuel from core positions or storage rack positions shall be secured when not in use by a system controlled by the supervisor of reactor operations, or the senior reactor operator on duty.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

6.1.1 Structure

The Ohio State University Research Reactor is a part of the College of Engineering administered by the Engineering Experiment Station. The organizational structure is shown in Figure 6.1.

6.1.2 Responsibility

The Director of the Engineering Experiment Station (Level 1) is the contact person for communications between the U.S. Nuclear Regulatory Commission and The Ohio State University.

The Director of the Nuclear Reactor Laboratory (Level 2) will have overall responsibility for the management of the facility.

The Associate Director (or Manager of Reactor Operations) (Level 3) shall be responsible for the day-to-day operation and for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license and Technical Specifications. During periods when the Associate Director is absent, his responsibilities are delegated to a Senior Reactor Operator (Level 4).

6.1.3 Staffing

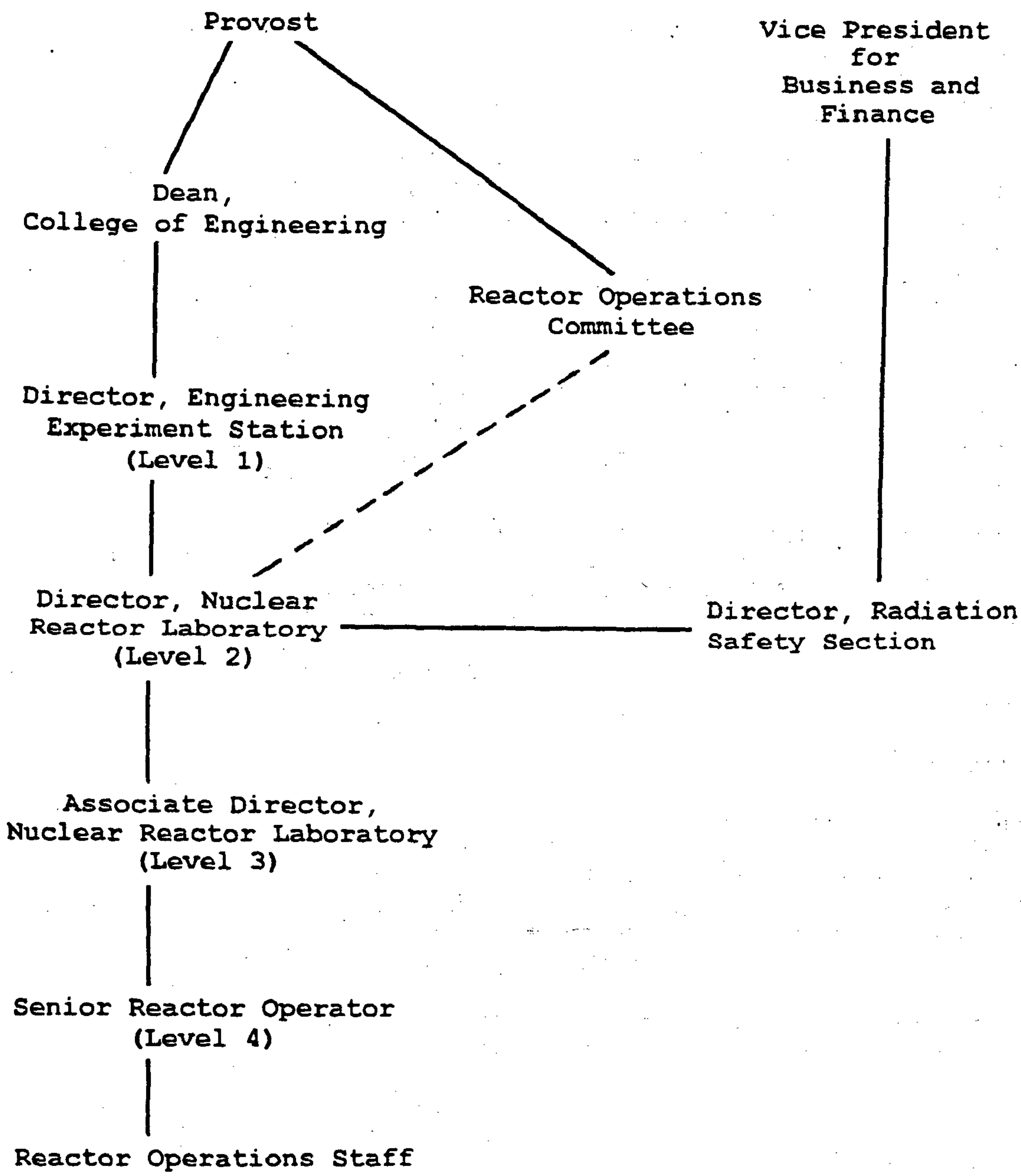
During Reactor Operations:

(1) Two or more personnel, at least one of whom is a licensed reactor operator, shall be in the building during all reactor operations. The second shall be capable of following simple written instructions for shutting down the reactor.

(2) At least two licensed operators should be in the building during any extended operations (longer than 60 minutes).

(3) Two persons, one of whom shall be a licensed senior reactor operator, shall be in the building for the first start-up of the day.

(4) Two persons, one of whom shall be a licensed senior reactor operator, shall be in the building during start-up after an unplanned shutdown.



Solid Lines ——— Paths of Direct Responsibility
Dashed Lines – – – – Paths of Information

Figure 6.1: Administrative Organization

(5) During all operations, a licensed operator shall be in the control room either as console operator or directing the activities of a student operator or trainee.

(6) A minimum of three people shall be present during fuel handling. one shall be a licensed senior reactor operator, and one shall be at least a licensed reactor operator.

6.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall meet or exceed the requirements of American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1988.

6.2 Review and Audit

There shall be a Reactor operations Committee (ROC) which shall review and audit reactor operations to assure the facility is operating in a manner consistent with public safety and within the terms of the facility license. The Committee advises the Director of the NRL, and is responsible to the Provost of The Ohio State University.

6.2.1 Composition and Qualifications of the ROC

Committee members shall be appointed annually by the Provost of The Ohio State University. The Committee shall be composed of at least nine members including ex-officio members. The Director and Associate Director of the Nuclear Reactor Laboratory, and the Director of the Office of Radiation Safety shall be ex-officio voting members of the Committee. The remaining Committee members shall be faculty, staff, and student representatives of The Ohio State University (but not part of the staff of the Reactor Lab), having professional backgrounds in engineering, physical, biological, or medical sciences, as well as knowledge of and interest in applications of nuclear technology and ionizing radiation.

6.2.2 ROC Meetings

The Committee shall meet at least twice each year. It should meet on or about six month intervals. A quorum shall consist of at least 50 percent of the members. Ex-officio members shall be counted in the quorum as follows:

(1) The Provost is an ex-officio member. Since the Provost is not appointed as a member of the ROC, the Provost is not required to act as a member, is not counted as a member when counting

a quorum, but does have the right to vote.

(2) Ex-officio members who are under the authority of the Provost serve in the same capacity as those who are appointed by the Provost, i.e., they have the right to vote and are counted as members when counting a quorum.

(3) Ex-officio members, if any, who are not under the authority of the Provost, have the right to vote, but have no obligation to participate. Accordingly, they are not counted as members when counting a quorum.

(4) All ex-officio members hold membership by virtue of their office. They cease to be members when they cease to hold office.

6.2.3 Sub-Committees

The chairperson may appoint a Subcommittee from within the Committee membership to act on behalf of the full committee on those matters which cannot await the regular semi-annual meetings. The full Committee shall review the actions taken by the Subcommittee at the next regular meeting.

6.2.4 ROC Review and Approval Function

The responsibilities of the ROC include, but are not limited to the following:

(1) Review and approval of experiments utilizing the reactor facilities

(2) Review of procedures

(3) Review and approval of all proposed changes to the license and technical specifications

(4) Determination of whether a proposed change, new test, or experiment would constitute an unreviewed safety question or require a change in the technical specifications per 10CFR50.59

(5) Review of audit reports

(6) Review of abnormal performance of plant equipment and operating abnormalities having safety significance

(7) Review of unusual occurrences and incidents which are reportable under 10CFR19, 20, 21, and 50, or Section 6.6.4 of this document, and

(8) Review of violations of technical specifications, license, or procedures having safety significance.

Relative to item (1), responsibility for review of experiments on a day-to-day basis shall lie with the Director of the Nuclear Reactor Laboratory or his designee. This day-to-day review shall determine whether a specific experiment has previously been approved in the generic sense by the ROC. A semi-annual report of performed experiments shall be provided for ROC review.

Relative to item (2), the NRL Director or his designee shall be responsible for approval of procedures or changes to procedures on a day-to-day basis. He shall provide a summary of all procedure changes to the ROC for their review.

A complete set of minutes of all Committee and Subcommittee meetings, including copies of all documentary material reviewed, and all approvals, disapprovals, and recommendations shall be kept. Minutes or reports of all Committee meetings or Subcommittee meeting should be disseminated to the Committee members prior to the next regularly scheduled meeting and should be read for approval as the first item on each agenda. A copy of the minutes, or any reports reviewed, should also be forwarded to the Director of the Engineering Experiment Station in a timely manner.

6.2.5 ROC Audit Function

A three member Subcommittee shall meet annually to perform an audit of NRL operations and records or review the results of an independent audit completed by another qualified agency. At least two individuals on the Audit Subcommittee shall be ROC members. The third may be a staff member from the Reactor Laboratory or another individual appointed by the ROC chairperson. No member shall audit a function that he is responsible for performing. Each person should serve for three consecutive audits, at which time he or she should be replaced by a new member. In this way, each Subcommittee should consist of two holdovers and one new member. The member serving for his or her second audit should be the Audit Subcommittee Chairperson. The following items shall be audited:

(1) Reactor operations for adherence to facility procedures, Technical Specifications, and license requirements

(2) The requalification program for the operating staff,

(3) The facility Emergency Plan and implementing procedures,

(4) The facility Security Plan and implementing Procedures, and

(5) The results of actions taken to correct any deficiencies that affect reactor safety, and

(6) Conformance with the ALARA Policy and the effectiveness of radiologic control.

Deficiencies found by the Audit Subcommittee that affect Reactor Safety shall be reported immediately to the Director of the Engineering Experiment Station. A written report of audit findings should be submitted to the Director of the Engineering Experiment Station and the full Reactor Operations Committee within three months of the audit's completion.

6.3 Procedures

6.3.1 Reactor Operating Procedures

Written procedures, reviewed and approved by the Director, or his/her designee, and reviewed by the ROC, shall be in effect and followed. The procedures shall be adequate to assure the safety of the reactor, but should not preclude the use of independent judgement and action should the situation require such. All new procedures and changes to existing procedures shall be documented by the NRL staff and subsequently reviewed by the ROC. At least the following items shall be covered:

(1) Startup, operation, and shutdown of the reactor,

(2) Installation, removal, or movement of fuel elements, control rods, 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 cooling system leaks, and abnormal reactivity changes,

(4) Emergency conditions involving potential or actual release of radioactivity including provisions for evacuation, re-entry, recovery, and medical support,

(5) Preventive and corrective maintenance procedures for systems which could have an effect on reactor safety,

(6) Periodic surveillance of reactor instrumentation and safety systems, area monitors, and radiation safety equipment,

(7) Implementation of Security, Emergency and Operator training and requalification plans, and

(8) Personnel radiation protection.

6.3.2 Administrative Procedures

Procedures shall also be written and maintained to assure compliance with Federal regulations, the facility license, and commitments made to the ROC or other advisory or governing bodies. As a minimum, these procedures shall include:

(1) Audits,

(2) Special Nuclear Material accounting,

(3) Operator requalification,

(4) Record keeping, and

(5) Procedure writing and approval.

6.4 Experiment Review and Approval

6.4.1 Definitions of Experiments

Approved experiments are those which have previously been reviewed and approved by the ROC. They shall be documented and may be included as part of the Procedures Manual. New experiments are those which have not previously been reviewed, approved, and performed. Routine tests and maintenance activities are not experiments.

6.4.2 Approved Experiments

All proposed experiments utilizing the reactor shall be evaluated by the experimenter and a licensed Senior Reactor Operator to assure compliance with the provisions of the utilization license, the Technical Specifications, and 10CFR Parts 20 and 50. If, in the judgement of the Senior Reactor Operator, the experiment meets with the above provisions, is an approved experiment, and does not constitute a threat to the integrity of the reactor, it may be approved for performance. When pertinent, the evaluation shall include considerations of:

(1) The reactivity worth of the experiment

(2) The integrity of the experiment, including the effects of changes in temperature, pressure, or chemical composition

(3) Any physical or chemical interaction that could occur with the reactor components, and

(4) Any radiation hazard that may result from the activation of materials or from external beams

6.4.3 New Experiments

Prior to performing an experiment not previously approved for the reactor, the experiment shall be reviewed and approved by the Reactor Operations Committee. Committee review shall consider the following information:

(1) The purpose of the experiment,

(2) The procedure for the performance of the experiment, and

(3) The safety evaluation previously reviewed by a licensed Senior Reactor Operator.

6.5 Required Actions

6.5.1 Action To Be Taken In the Event A Safety Limit Is Exceeded

(1) The reactor shall be shut down, and reactor operations shall not be resumed until authorized by the NRC.

(2) The safety limit violation shall be promptly reported to the Director of the Reactor Laboratory.

(3) The safety limit violation shall be reported by telephone to the NRC within 24 hours.

(4) A safety limit violation report shall be prepared. The report shall describe the following:

a. Applicable circumstances leading to the violation including, when known, the cause and contributing factors,
b. Effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public, and
c. Corrective action to be taken to prevent recurrence.

(5) The report shall be reviewed by the Reactor Operations Committee and shall be submitted to the NRC within 14 working days when authorization is sought to resume operation of the reactor.

6.5.2 Action To Be Taken In The Event Of A Reportable Occurrence

A reportable occurrence is any of the following conditions:

(1) Operating with any safety system setting less conservative than stated in these specifications,

(2) Operating in violation of a Limiting Condition for Operation established in Section 3 of these specifications.

(3) Safety system component malfunctions or other component or system malfunctions during reactor operation that could, or threaten to, render the safety system incapable of performing its intended function.

(4) An uncontrolled or unanticipated increase in reactivity in excess of 0.4% delta k/k,

(5) An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of an unsafe condition in connection with the operation of the reactor, and

(6) Abnormal and significant degradation in reactor fuel and/or cladding, coolant boundary, or confinement boundary (excluding minor leaks) where applicable that could result in exceeding prescribed radiation exposure limits of personnel and/or the environment.

(7) Any uncontrolled or unauthorized release of radioactivity to the unrestricted environment.

In the event of a reportable occurrence, the following action shall be taken:

(1) The reactor conditions shall be returned to normal, or the reactor shall be shutdown, to correct the occurrence.

(2) The Director of the Reactor Laboratory shall be notified as soon as possible and corrective action shall be taken before resuming the operation involved.

(3) A written report of the occurrence shall be made which shall include an analysis of the cause of the occurrence, the corrective action taken, and the recommendations for measures to preclude or reduce the probability of recurrence. This report shall be submitted to the Director and the Reactor Operations Committee for review and approval.

(4) A report shall be submitted to the Nuclear Regulatory Commission in accordance with Section 6.6.2 of these specifications.

6.6 Reports

Reports shall be made to the Nuclear Regulatory Commission as follows:

6.6.1 Operating Reports

An annual report shall be made by September 30 of each year to the Director, Office of Nuclear Reactor Regulation, NRC, Washington, DC 20555, with a copy to the NRC, Region III, in accordance with 10CFR 50.4, providing the following information:

(1) A narrative summary of operating experience (including experiments performed) and of changes in facility design, performance characteristics, and operating procedures related to reactor safety occurring during the reporting period.

(2) A tabulation showing the energy generated by the reactor (in kilowatt hours) and the number of hours the reactor was in use.

(3) The results of safety-related maintenance and inspections. The reasons for corrective maintenance of safety-related items shall be included.

(4) A table of unscheduled shutdowns and inadvertent scrams, including their reasons and the corrective actions taken.

(5) A summary of the Safety Analyses performed in connection with changes to the facility or procedures, which affect reactor safety, and performance of tests or experiments carried out under the conditions of Section 50.59 of 10CRF50.

(6) A summary of the nature and amount of radioactive gaseous, liquid, and solid effluents released or discharged to the environs beyond the effective control of the licensee as measured or calculated at or prior to the point of such release or discharge.

(7) A summary of radiation exposures received by facility personnel and visitors, including the dates and times of significant exposures.

6.6.2 Special Reports

(1) A telephone or telegraph report of the following shall be submitted as soon as possible, but no later than the next working day, to the NRC Region III Office:

(a) Any accidental offsite release of radioactivity above authorized limits, whether or not the release resulted in property damage, personal injury, or known exposure.

(b) Any exceeding of the safety limit as defined in Section 2.1 of these specifications.

(c) Any reportable occurrences as defined in Section 6.5.2 of these specifications.

(2) A written report shall be submitted within 14 days to the Director, Office of Nuclear Reactor Regulation, US NRC, Washington, DC 20555 with a copy to the NRC Region III, in accordance with 10CFR 50.4, of the following:

(a) Any accidental offsite release of radioactivity above permissible limits, whether or not the release resulted in property damage, personal injury, or known exposure.

(b) Any exceeding of the safety limit as defined in Section 2.1.

(c) Any reportable occurrence as defined in Section 6.5.2 of these specifications.

(3) A written report shall be submitted within 30 days to the Director, Office of Nuclear Reactor Regulation, US NRC, Washington, DC 20555, with a copy to the NRC, Region III Office in accordance with 10CFR 50.4, of the following:

(a) Any substantial variance from performance specifications contained in these specifications or in the SAR,

(b) Any significant change in the transient or accident analyses as described in the SAR, and

(c) Changes in personnel serving as Director, Engineering Experiment Station, Reactor Director, or Reactor Associate Director.

(4) A report shall be submitted within nine months after initial criticality of the reactor or within 90 days of completion of the startup test program, whichever is earlier, to the Director, Office of Nuclear Reactor Regulation, U.S. NRC, Washington, DC 20555, with a copy to the NRC, Region III upon receipt of a new facility license, an amendment to license authorizing an increase in power level or the installation of a new core of a different fuel element type or design than previously used.

The report shall include the measured values of the operating conditions or characteristics of the reactor under the new conditions, and comparisons with predicted values, including the following:

(a) Total control rod reactivity worth,

(b) Reactivity worth of the single control rod of highest reactivity worth, and

(c) Minimum shutdown margin both at ambient and operating temperatures.

(d) Excess reactivity

(e) Calibration of operating power levels

(f) Radiation leakage outside the biological shielding

(g) Release of radioactive effluents to the unrestricted environment.

6.7 Records

Records or logs of the items listed below shall be kept in a manner convenient for review, and shall be retained for as long as indicated.

6.7.1 Records to be Retained for a Period of at Least Five Years

(1) normal plant operation,

(2) principal maintenance activities,

(3) experiments performed with the reactor,

(4) reportable occurrences,

(5) equipment and component surveillance activity,

(6) facility radiation and contamination surveys,

(7) transfer of radioactive material,

(8) changes to operating procedures, and

(9) minutes of Reactor Operations Committee meetings.

6.7.2 Records to be Retained for at Least One Requalification Cycle

Regarding retraining and requalification of licensed operations personnel, the records of the most recent complete requalification cycle shall be maintained at all times the individual is employed.

6.7.3 Records to be Retained for the Life of the Facility

(1) gaseous and liquid radioactive effluents released to the environment,

(2) fuel inventories and transfers

(3) radiation exposures for all personnel,

(4) changes to reactor systems, components, or equipment that may affect reactor safety,

(5) updated, corrected, and as-built drawings of the facility.

(6) records of significant spills of radioactivity, and status,

(7) annual operating reports provided to the NRC,

(8) copies of NRC inspection reports, and related correspondence