UNIVERSITY OF MISSOURI, COLUMBIA
MISSOURI UNIVERSITY RESEARCH REACTOR
LICENSE NO. R-103
DOCKET NO. 50-186

PRELIMINARY HAZARDS SUMMARY REPORT IN SUPPORT OF AN APPLICATION TO CONSTRUCT A RESEARCH REACTOR

MARCH 1961

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUR OR DENOTED BY BRACKETS

University of Missouri
Research Reactor Facility

Preliminary Hazards Summary Report

and

Hazards Summary Report

PRELIMINARY HAZARDS REPORT

University of Missouri

Research Reactor

Submitted to the

United States Atomic Energy Commission

March, 1961

University of Missouri

Columbia, Missouri

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1.0. INTRODUCTION

This report is being submitted by the University of Missouri to the Hazards Evaluation Branch of the Atomic Energy Commission's Division of Licensing and Regulation. It is submitted to support an application being made by the University of Missouri for a license to operate a nuclear reactor research facility on the Campus of the University of Missouri. This report is a preliminary one, intended to supply sufficient data for the issuance of a construction permit. It is the intention of the University of Missouri to submit more detailed information regarding operation and research programs after completion of construction. This will be submitted in the form of a Final Hazards Report.

The University of Missouri would propose to build a research reactor of the heterogeneous type using highly enriched uranium fuel, cooled and moderated by light water. This reactor consists of a pressurized tank, similar to the Oak Ridge Research Reactor, but with the addition of a flux trap position in the center of the core and control elements located as a shroud surrounding the pressurized tank. In turn, the pressurized tank would be surrounded by a beryllium and graphite reflector. This reactor is designed for eventual operation at ten-megawatts thermal power. It is proposed that the initial installation will permit operation at a maximum of five-megawatts. Sections of this report having to do with hazards analysis are based upon an operating thermal power of ten-megawatts.

The selection of the annular core, flux-trap-type reactor for the University research facility was based upon a number of advantages offered by this design. First, there is a position in the center of the core which offers a very high peak thermal flux; second, a study of the available beam-tube currents, relative to those attainable from an equivalent open-pool reactor, indicates a factor of two increase; and third, the thin-fuel annulus (which one has in a flux-trap device) permits external rod control and eliminates rod penetrations and seals into a pressurized tank resulting in a considerable savings in cost. These factors, together with the desire of the University of Missouri to enter into a "second phase" of research reactor development, has led this group to propose the construction of a flux-trap reactor on this Campus.

The reactor, control room, three offices, and two small laboratories will be housed in a square containment-type building having the dimensions of approximately 60 by 60 feet. The process equipment, including the demineralizers, the heat exchangers and pumps, as well as the delay tank will be located in an adjacent below-grade room. The piping connections between the process equipment room and the containment vessel will be led through a u-type water seal beneath the edge of the containment vessel. Adjacent to this gas-tight structure, and connecting it through a door which may be sealed, will be a 26,000-square-foot laboratory and office building housing a variety of research facilities, including a hot cell, a Cobalt-60 gamma irradiation facility, machine and electronic shops, radioactive waste handling equipment, various counting rooms, animal quarters, and research laboratories. The containment and laboratory buildings are being designed by the firm of C. L. T. Gabler and Associates of Detroit, Michigan.

The design of the reactor described in this report is not complete. Certain design alternatives exist which can be resolved only in the course of detailed design. The possible alternatives are mentioned wherever these have bearing on the hazards evaluation. In each case the presently preferred solution is indicated and this is intended to serve as the basis for preliminary hazard evaluation.

Particularly significant to the hazards evaluation is the intended use of the facility:

> This reactor will be used exclusively for research programs conducted by faculty and graduate personnel. It will not be used for reactor experiments nor for student training.
>
> No particularly hazardous types of irradiation or experiment are intended, such as fuel element testing or circulating fuel experiments. Initial usage of the center test hole or flux-trap facility will be limited to experiments which can be moved or altered only upon reactor shutdown. Means for alternate usage of this facility would be installed only upon the approval of an addendum to the Final Hazards Report.

It is the intent of the University of Missouri to proceed with detailed design of the facility described in a preliminary manner in this report. The intention is that the detailed design will closely follow the specifications described. The University of Missouri will submit to the Atomic Energy Commission supplements, as well as a Final Hazards Report, detailing the exact design characteristics of the device as built on this Campus. It is the intent of this report to secure for the University of Missouri a construction permit to build the device outlined.

1.1. The University of Missouri

The University of Missouri was created in 1839 by the General Assembly of the State of Missouri. It was the first state university established in the Louisiana Purchase territory, which was acquired by the United States during the administration of President Thomas Jefferson. The University graduated its first class in November of 1843. The University serves a three-fold function: teaching, research, and service to the inhabitants of the State of Missouri.

The University's corporate name is the Curators of the University of Missouri. It is located on two campuses, at Columbia and Rolla, Missouri. On the Columbia Campus, the University is comprised of the: College of Arts and Science, Division of Agricultural Sciences, School of Business and Public Administration, College of Engineering, School of Journalism, School of Law, School of Medicine, and the Graduate School. There are eight engineering and related science departments located on the Rolla, Missouri Campus.

1.2. The Board of Curators

The government of the State University is vested in a Board of Curators consisting of nine members appointed by the Governor, by and with the advice and consent of the Senate.

The term of service of the Curators is six years, the terms of three expiring every two years.

The functions of the Board are to appropriate monies annually to the University for probable expenses they will incur. The faculty and other members of the University are appointed and removed by the Board, and the Board defines and assigns the duties of the faculty and fixes their compensation accordingly. When a person is judged worthy of a degree, the Board has the power to confer such honor. The Board provides the protection and improvement, and adapts the improvements to the uses of the University. The Curators cause to be made annually a careful and complete inventory and appraisement of all property belonging to the University. They also prescribe rules and regulations in order to secure a careful inspection of the property and comparison of the same with prior inventories.

At the close of each University year, the Board makes a detailed report to the Governor of Missouri on the progress, conditions, and wants of the institution, along with a list showing the courses of study, number, and names of the officers and students.

1.3. The President

The President of the University, appointed by the Board of Curators, is the chief executive officer.of the University. He has general supervision and direction over all administrative officers, teachers, and employees, and has the authority to nominate the aforesaid for appointment by the Board. He has the right to preside and vote at all faculty meetings and to appoint all committees of the University faculty. Also, he is authorized to make academic and non-academic appointments.

He presents an annual report to the Board on the progress, conditions, and wants of the institution, and at such other times as the Board deems necessary.

He attends all meetings of the Board so far as his duties permit.

Conferring of degrees, authorized by the Board, is in the President's power. He may refuse admittance to any student who does not qualify under the regulations of the University.

He may cause to be made, for his information, an examination or inquiry regarding the work of the University.

Suspension of exercises of the University is under the President's power, and also he may authorize any member of the University faculty to exercise temporarily his authority as President during his absence.

1.4. Enrollment

The University of Missouri, like all state colleges across the country, has experienced a phenomenal growth over the past ten years. There is every reason to believe that as the college-age population increases in the next ten and more years, this growth will continue at its present astounding rate. To illustrate the growth which the University of Missouri has had since 1953 through 1959, Table 1.1 is presented. The total enrollment in 1953 was 7,379 students. Six years later the total enrollment was 10,196 students.

Not only is the number of students increasing at a rapid rate, but also the distribution of the students is changing. An analysis of the enrollment records indicates that upper-class and graduate enrollment has increased from a percentage of 49% in 1953 to 58.5% in 1959. The number of graduate students enrolled at the University of Missouri has increased over this same period of time from 11% to 13.5% of total enrollment.

On the basis of a state-wide survey, which includes an estimate of the Missouri college population of ages 18 through 22 and a projection of the expected enrollment at the University of Missouri, a table has been prepared which, it is felt, is a realistic projection of the expected enrollment through the year 1975. This is presented as Table 1.2.

1.5. University Staff

The University of Missouri staff has grown through the past ten years in keeping with the increase in student enrollment. A growth of approximately 100% has taken place over these ten years.

TABLE 1.1

UNIVERSITY OF MISSOURI AT COLUMBIA ENROLLMENT

Fall Semester	Total Enrollment
1953	7,379
1954	8,070
1955	8,983
1956	9,538
1957	9,904
1958	10,261
1959	10,196
1960	11,089

TABLE 1.2

PROJECTION UNIVERSITY OF MISSOURI ENROLLMENTS
FALL SEMESTERS
Adapted from the Wadsworth Study
November, 1959

Fall Semester	Projected Enrollment University of Missouri Columbia
1958	10,261
1959	10,140
1960	11,046
1961	11,524
1962	12,012
1963	12,460
1964	13,294
1965	14,053
1966	14,880
1967	15,880
1968	16,500
1969	17,238
1970	18,040
1971	18,868
1972	19,710
1973	20,570
1974	21,392
1975	22,230

Table 1.3 is included to illustrate the present staffing of the University and the rate of growth for various years from 1951 with an estimate through 1963. This is a tabulation of full-time employees of the University of Missouri. It does not include various student assistants and part-time employees.

1.6. Physical Plant

The University of Missouri owns more than 38,000 acres of land. A summary of the land ownership and its distribution is presented in Table 1.4. The main Campus of the University of Missouri at Columbia consists of 733 acres of land. The majority of this acreage is within the city limits of Columbia. There are, on this land, more than 100 major buildings owned, operated, and maintained by the University of Missouri.

The University owns approximately 3,500 acres of cultivated farm land in the immediate surroundings of Columbia. This acreage is used for agricultural teaching and research within the College of Agriculture of the University.

TABLE 1.3

FULL-TIME EMPLOYEES OF THE UNIVERSITY OF MISSOURI

Fiscal Year	Academic	Non Academic	Total
1951-52	573	1,439	2,012
1952-53	573	1,554	2,127
1953-54	572	1,575	2,147
1954-55	591	1,653	2,244
1955-56	634	1,741	2,375
1956-57	681	2,126	2,807
1957-58	732	2,481	3,213
1958-59	782	2,654	3,436
1959-60	810	2,683	3,493
1960-61 Estimated	836	2,760	3,596
* 1961-62 Estimated	944	2,867	3,811
* 1962-63 Estimated	981	2,918	3,899

* Based on the assumption that all amounts requested from the State of Missouri for additional staff for the biennium 1961-63 will be allowed.

TABLE 1.4

LAND OWNED BY THE UNIVERSITY OF MISSOURI

Main Campus - Columbia, Missouri	733 Acres
Campus at Rolla, Missouri	538 Acres
Other Land in Boone County, Missouri	6,089 Acres
Land in State, But Outside Boone County	30,535 Acres
Land Outside of the State of Missouri	154 Acres
TOTAL	38,049 Acres

2.0. SITE CHARACTERISTICS

2.1. Geography*

Boone County, Missouri, consists of an area of about 683 square miles located in the central part of the State (Figure 2.1). The County lies between 38° 40 minutes and 39° 15 minutes north latitude, and between 92° 5 minutes and 92° 35 minutes west longitude. The County is approximately 41 miles in its greatest north-to-south length, and about 22 miles in its greatest east-to-west width. The southwestern boundary is formed by the Missouri River, and the southeastern boundary is formed almost entirely by Cedar Creek. These streams follow rather sinuous courses, which produces a great deal of irregularity in the County's outline (Figure 2.2).

Columbia, the county seat and the largest city (1960 census - 36,510), is located in the center of Boone County. The second largest city is Centralia, which is located in the northeastern corner of the County and has a population of approximately 3,188. There are a number of other smaller towns and villages, having populations of less than 1,000 people, which serve as trading centers for their surrounding areas. The total population of Boone County (1960 census) is 54,969.

There are two major highways which cross the County and intersect at Columbia. U. S. Highway 40 extends east and west across the middle width of the County and provides connections between St. Louis and Kansas City. U. S. Highway 63 extends north and south across the mid-length of the County, connecting with Jefferson City in the south and with U. S. Highway 24 at Moberly in the north (Figure 2.2). In addition to these main highways, the County is served by a network of secondary roads making most parts of the County easily accessible.

The Gulf, Mobile and Ohio Railroad and the Wabash Railway cross the northeastern corner of the County through Centralia, from where the Wabash Railway extends a branch-line to Columbia. The Missouri-Kansas-Texas Railroad passes along the southeastern edge of the County, essentially parallel with the Missouri River. A branch of this line extends from McBaine to Columbia.

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* From: Geology of Boone County, Missouri, by A. G. Unklesbay, 1952, State of Missouri, Geological Survey and Water Resources.

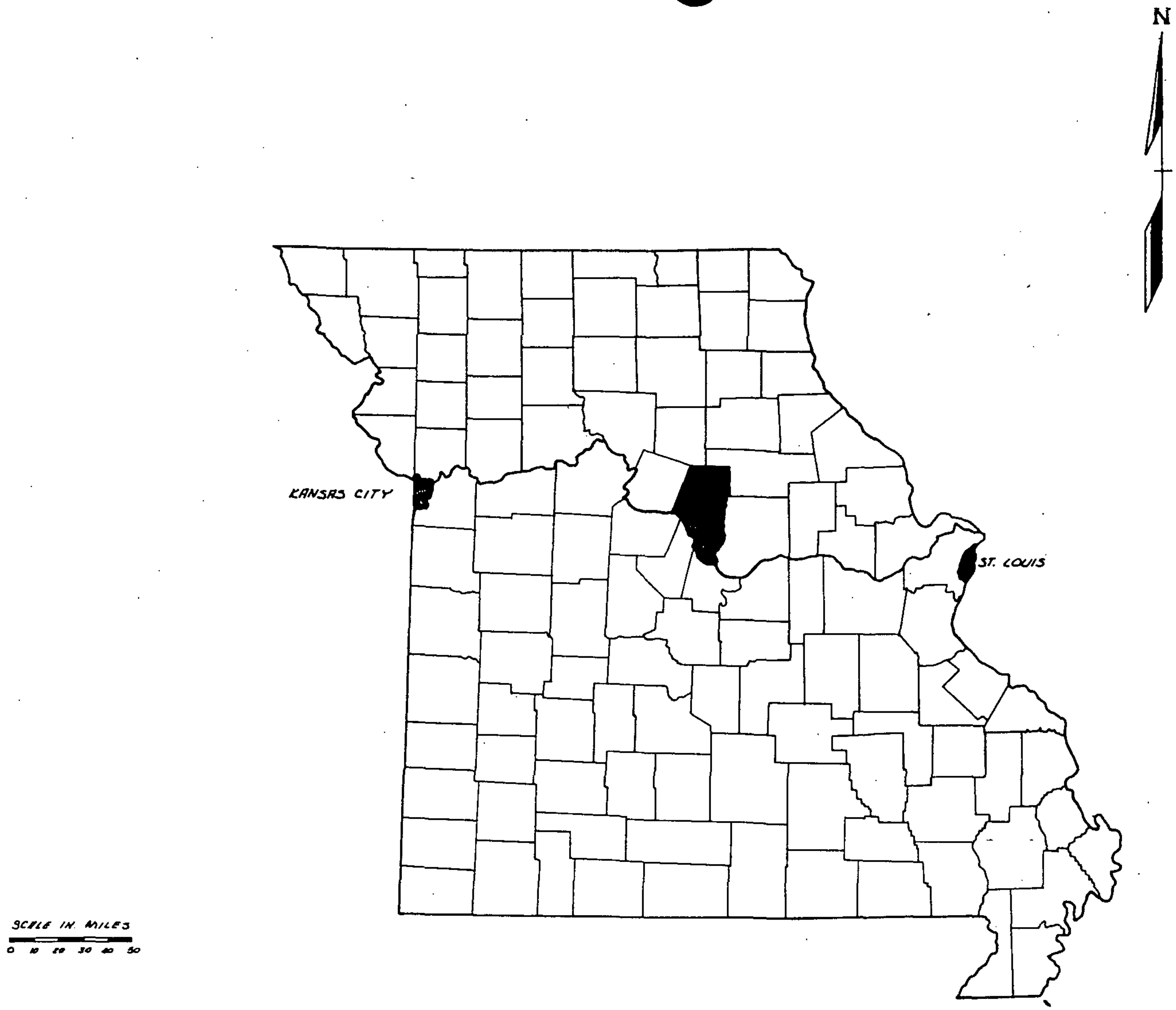

FIGURE 2.1 KEY MAP OF MISSOURI, SHOWING LOCATION OF BOONE COUNTY

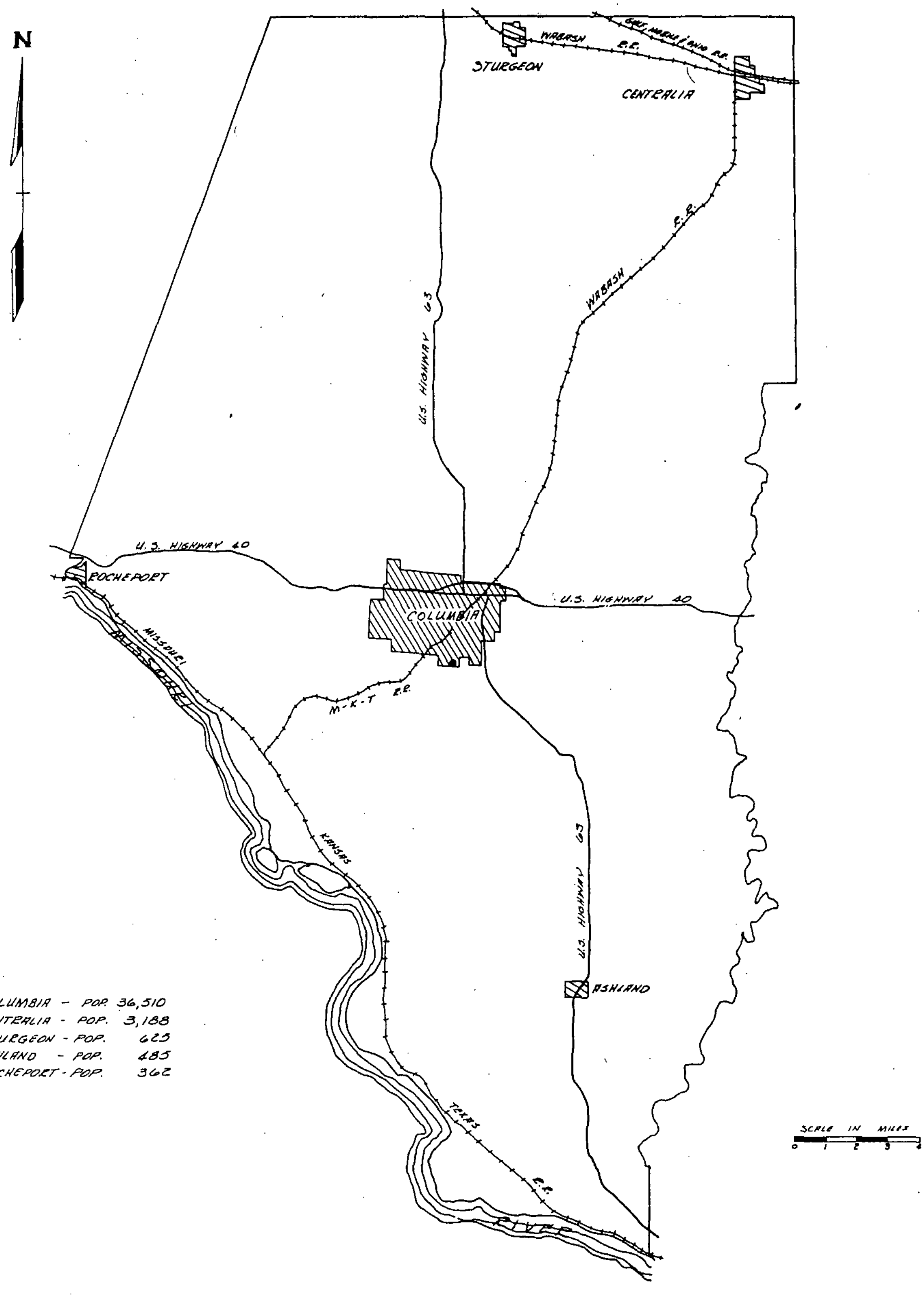

FIGURE 2.2 BOONE COUNTY, LOCATION OF MAJOR CITIES, RAILROADS, AND HIGHWAYS

There are a number of small industrial activities in the area, but for the County as a whole, agriculture is the leading activity. Agriculture is especially important in the upland area where there are good soils derived largely from glacial materials and where the area is rather flat. The fertile soil of the Missouri River flood plain is used for truck gardening in the vicinity of Hartsburg. The limestone deposits in the State form the basis for a number of quarrying operations for the production of road aggregates, agricultural limestone, and building stone. The limestone deposit in the vicinity of Easley is used in the manufacture of rock wool. There are Pennsylvanian coal deposits in the area which form the basis for a limited coal mining industry. The clays and shales in the County supply material for a small amount of brick and tile manufacture. One of these brick manufacturing concerns is located at Columbia, Missouri.

Columbia is the home of the University of Missouri, Stephens College, and Christian College.

2.2. Physiography

Boone County contains a variety of topographic features ranging from rugged maturely dissected hills to flat flood plains and flat uplands. The northeastern portion of the County, between Sturgeon and Centralia, is a flat upland plain which averages about 850 to 900 feet above sea level. This upland extends southward from Centralia toward Hallsville and Murray, but is somewhat dissected by the upper portion of Silver Fork and its tributaries. In this same general area, a few portions of the plain grade gradually upward to the highest point in the County, which is slightly above 940 feet.

Southwest from this upland area, the land surface is more completely dissected in a wide zone extending to within a few miles of the Missouri River. Over most of this region the interstream areas stand at about the same elevation so that the skyline is remarkably uniform.

The most rugged part of the County lies between this dissected upland and the flood plain of the Missouri. This portion is 3 to 5 miles wide. It parallels the Missouri River and extends some distance up the tributary valleys. In this rugged area the local relief ranges up to about 250 feet, and there is very little flat upland or lowland.

Many of the slopes are very steep, and there are many vertical bluffs along the Missouri River and some of the major tributaries.

The river flood plain varies in width, but along much of the southwestern part of the County, it is about 2 miles wide. The lowest point in the County is 540 feet above sea level on the flood plain at the extreme southern tip of the County.

Drainage from most of the County is into the Missouri River. Perche Creek* is the largest stream in the County. It enters the County near the northwestern corner and flows southward across the western half. On the Missouri River flood plain it turns southeastward and parallels the river for about 5 miles before entering it at a place where the river approaches a steep bluff. The chief tributaries of the Perche enter it from the east or northeast. These include the Hinkson which rises in the area northeast of Columbia, and is fed by Grindstone and Hominy Creeks. All surface drainage from property identified as the University of Missouri is to one of two branches of the Hinkson Creek, which, in turn, feeds into Perche Creek. The two branches of the Hinkson which receive surface runoff from the University of Missouri are known as Flat Branch and Grindstone Creek. Figure 2.3 illustrates the area of Boone County which drains via Perche Creek into the Missouri River.

2.3. Geology, Hydrology, and Water Supplies**

The area about Columbia is underlaid by a considerable thickness of sedimentary rocks of which the carbonates, limestone, and dolomite, make up the major portion. These rock formations are essentially horizontal but have a slight dip to the north and in the immediate vicinity of Columbia are relatively free of faults, folds, or other major structural features.

The generalized column for this area is presented in Figure 2.4. Figure 2.5 shows the outcrop pattern of the formations in the Columbia vicinity. The general features of the separate units and their water-bearing characteristics are described in the following paragraphs.

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* In older records of the State of Missouri, that creek today known as Perche Creek was identified as Roche Perche Creek.

** Prepared by: Athel G. Unklesbay, Professor of Geology, University of Missouri.

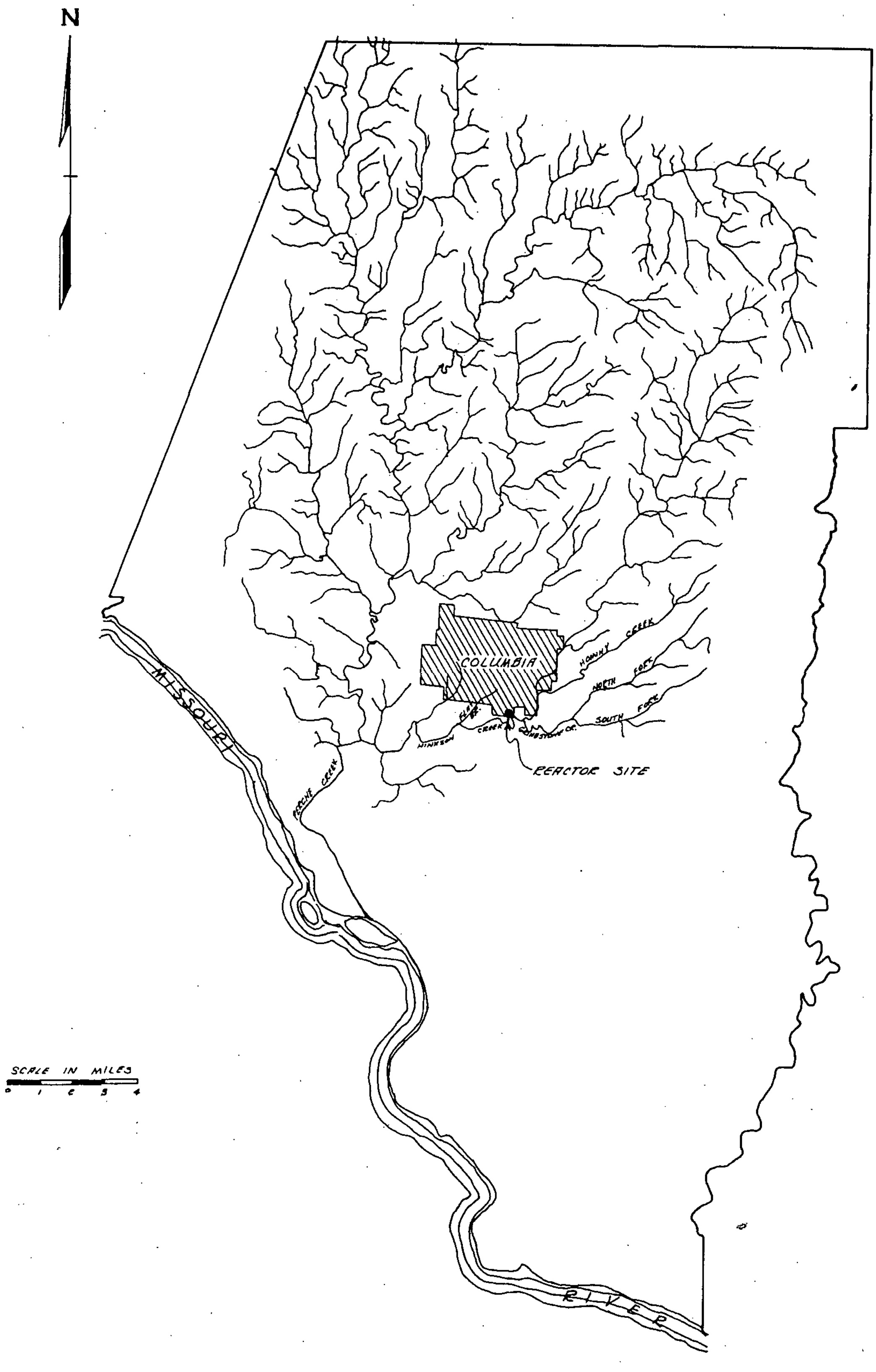
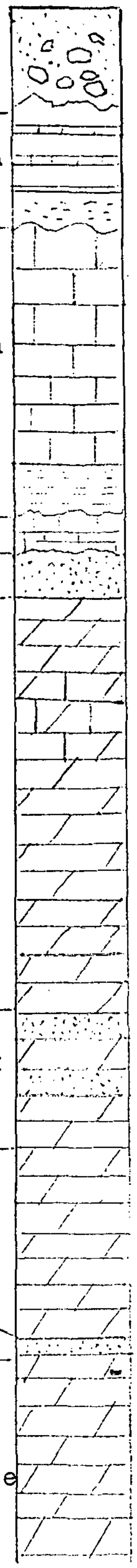


FIGURE 2.3 MAP OF PERCHE CREEK DRAINAGE AREA IN BOONE COUNTY, MISSOURI

FIGURE 2.4

GEOLOGIC COLUMN AT COLUMBIA MISSOURI



	Glacial	0-50'	Glacial deposit. Clay, sand, gravel. Locally produces limited quantities of water.
	Pennsylvanian	0-100'	Shale, limestone, coal. Small amounts of highly mineralized water.
	Mississippian	50-100'	Limestone, mostly coarse grained. Crevices and solution channels produce moderate supplies. Lower part finer grained.
	Devonian	0-50'	Limestone. Produces no appreciable water.
ORDOVICIAN	St. Peter	0-30'	Sandstone. Locally absent and not reliable.
ORDOVICIAN	Jefferson City	400'	Dolomite and magnesian limestone. Fine grained. Dependable source for small to medium amounts for domestic and farm supplies. Much used in the Columbia area.
ORDOVICIAN	Roubidoux	100'	Interbedded dolomite and sandstone. Produces large supplies, and is important to municipal supplies.
ORDOVICIAN	Gasconade	50'	Cavernous dolomite. Dependable for municipal supplies. Water moderately hard.
ORDOVICIAN	Gunter	15'	Sandstone. Good source. Sometimes over-rated.
CAMBRIAN	Potosi Eminence	100'	Cavernous dolomite. Important contributor to newer municipal wells in this area.

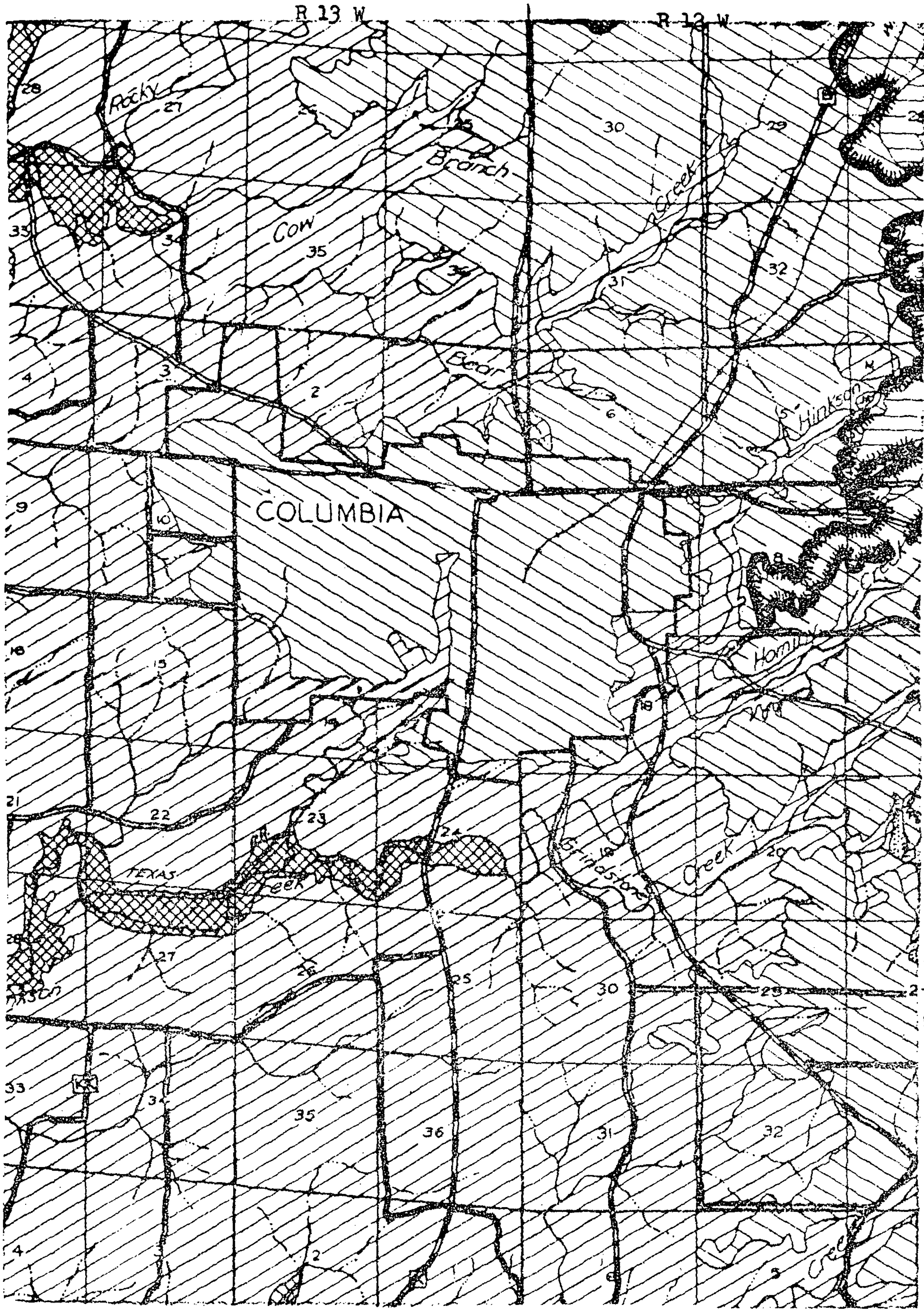


FIGURE 2.5

GEOLOGIC MAP OF COLUMBIA AREA
Scale 1" to 1 mile

Alluvium Pennsylvanian Burlington

Glacial Deposits

These deposits mantle the upland areas and consist of a heterogeneous mixture of clay, sand, and pebbles of diverse rock types. They vary greatly in thickness and are as much as 140 feet thick in the northern part of Boone County. This material is relatively impermeable and supplies very little water to wells, but may be used locally for small quantities.

Pennsylvanian

This part of the column consists largely of clay and shale with minor amounts of coal and thin impure limestone. The total thickness may be as much as 100 feet. These beds produce only small quantities of water and are not used in this area as a source of supply. The water in them is usually high in iron and sulphur.

Burlington Limestone

This formation is a coarse-grained clastic limestone with many beds and nodules of chert. It is the limestone exposed in quarries, creek banks, and road cuts around Columbia. It may also contain minor amounts of pyrite and limonite. It is about 160 feet thick in the Columbia area, but the thickness is variable. The Burlington contains many relatively shallow drilled wells and yields sufficient quantities for rural domestic supplies. The water is relatively hard.

The limestone is readily soluble and the Burlington contains many caverns and solution passages and where it is the surface formation, there are many caves and sinkholes.

Chouteau Limestone

This carbonate unit is very fine-grained and is, for the most part, evenly-bedded bluish gray limestone. The upper part is somewhat massive and is rather high in magnesium. The Chouteau, because of its fine texture, is relatively impermeable and the movement of water in it is restricted to joints and small fissures. It offers poor prospects as a source of water but yields very small quantities to a few wells.

Devonian Limestone

These limestones are variable in lithology and range from very fine-grained to coarsely-textured beds. Some are slightly sandy. In the Columbia area these beds are only about 30 feet thick and in some of the Columbia wells they seem to be completely absent. These Devonian beds are not valuable as water producers.

St. Peter Sandstone

This formation, which is a very important aquifer in eastern and northern Missouri, has no importance whatever in the Columbia area. It is present only as localized masses in depressions in the older rocks.

Jefferson City Formation

This predominantly dolomite formation averages about 400 feet in thickness in the Columbia area and wells which bottom in it produce moderate quantities of relatively hard water. It is probably the "most-used" aquifer in this vicinity because of its relatively shallow depth and ability to produce quantities of water. It probably has more rural domestic wells terminating in it than any other formation in the area. Drillers report "large" solution openings in wells.

Roubidoux Formation

This formation consists of alternating sandstone and dolomite beds and averages about 100 feet in thickness. It is a very dependable water producer and is penetrated by all of the Columbia municipal and University wells. Its production has never been tested alone because most of the wells in this area are uncased below the Jefferson City formation and all beds contribute to the supply.

Gasconade Formation

This unit consists mostly of light gray dolomite with a sandstone (Gunter) at the base. The thickness is about 280 feet. This dolomite unit is very cavernous and contains many interconnected solution passages. The sandstone, about 15 feet thick, is very permeable and has a wide areal extent. It usually gets the credit for producing most of the water to wells in this area, but probably is no more valuable than the cavernous dolomites associated with it.

Cambrian Dolomites

The lower 400-450 feet of the section penetrated by wells near Columbia consists largely of dolomite which is somewhat cavernous and which contains a few thin beds of sandstone. This thickness contributes considerable quantities of water to the deeper wells, but like the Roubidoux has never been tested separately.

Water Levels and Recharge Areas

The position of the water levels in the various wells is controlled by the topographic position of the well, the hydrostatic pressure in the aquifer, and pumping conditions.

The information given in Table 2.1 is typical of the water levels in a few of the wells in the city limits of Columbia.

No systematic survey of the position of ground-water surfaces in Central Missouri has ever been made, but is presumed that the water in the Ordovician and Cambrian aquifers is moving northward and that it enters the aquifers in their outcrop area some 90 miles southwest of Columbia. The water in the Burlington, however, is probably from local recharge through near surface out-crops and through sinkholes and solution passages.

Major Producing Wells in Area

The only large supplies of water are produced from wells owned by the City of Columbia and the University. The locations of these major producing wells are shown on the topographic map (Figure 2.6). These wells penetrate and draw their water from the Jefferson City-Cambrian portion of the column. The city wells are rated at a capacity of 1 to 1.3 million gallons per day. The average production of the city wells in 1957 was 3.2 million gallons per day. The peak production on record by the municipal supply was 6.04 million gallons per day recorded in 1954.

TABLE 2.1

DATA ON REPRESENTATIVE WATER WELLS

(Location Shown on Fig. 2.6)

	Well	Surface Elev.	Depth	Casing	Water Level Below Surface	Date	Producing Formation
1.	City Well #5	805'	1200'	605'	-352'	1954	Jefferson City
					-385'	1957	Gunter
2.	City Well #6	707'	1160'	538'	-300'	1954	Jefferson City
					-295'	1957	Cambrian
3.	City Well #4	782'	1505'	600'	-392'	1954	Jefferson City
					-385'	1957	Cambrian
4.	City Well #1	777.7'	1200'	500'	-313'	1954	Jefferson City
					-332'	1957	Cambrian
5.	City Well #3	777.4'	1200'	400'	-283'	1928	Jefferson City
					-375'	1954	Cambrian
					-390'	1957	Cambrian
6.	Univ. Well	700.5'	1060'	450'	-202'	1938	Jefferson City
					-266'	1954	Cambrian
					-314'	1957	Cambrian
7.	Univ. Well	733.8'	1203'	500'	-187'	1938	Jefferson City
					-262'	1957	Cambrian
8.	Garrett	744'	120'	-	- 50'	1937	Burlington

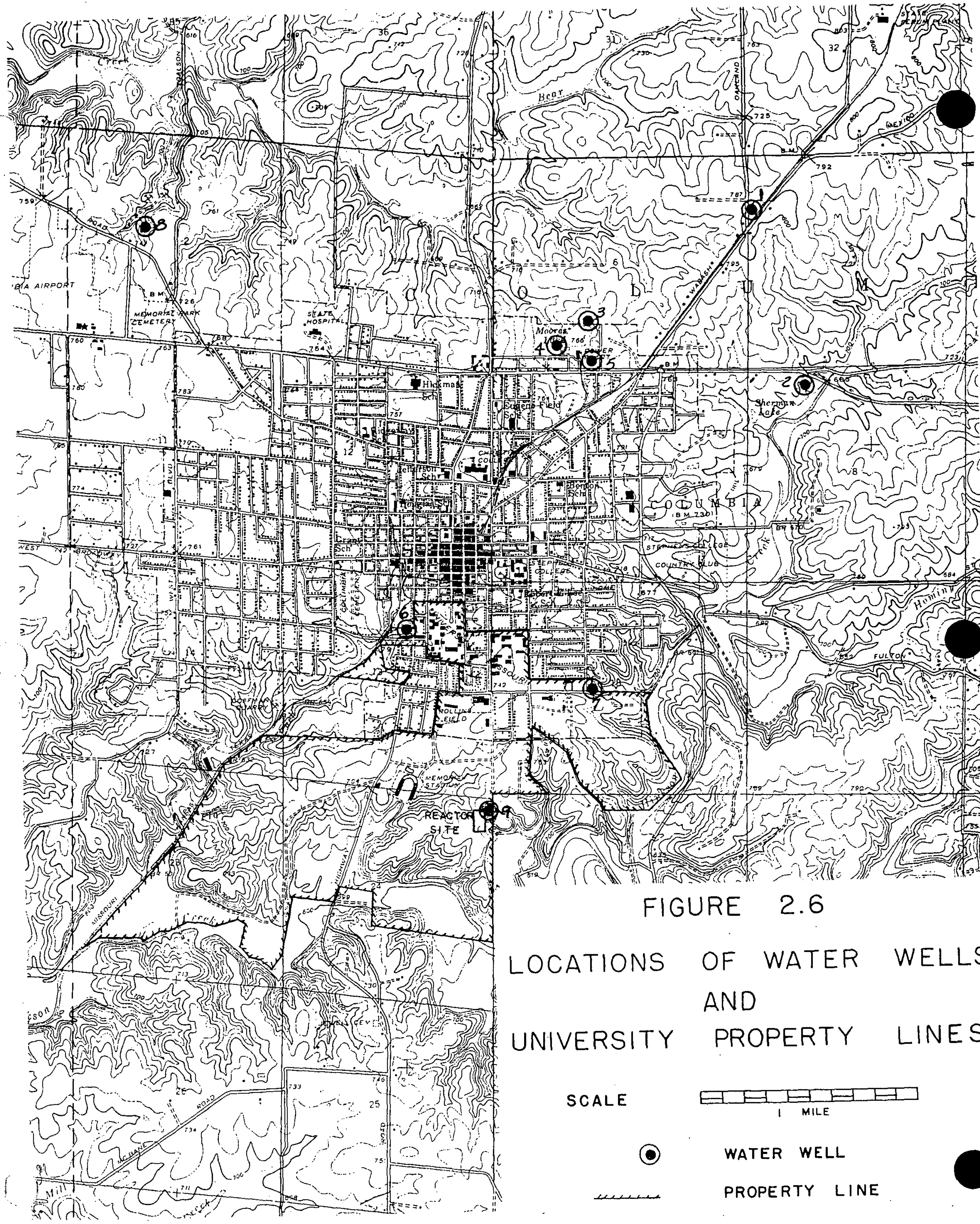


FIGURE 2.6

LOCATIONS OF WATER WELLS AND UNIVERSITY PROPERTY LINES

SCALE

1 MILE

WATER WELL

PROPERTY LINE

2.4. Meteorology and Climatology*

2.4.1. General Meteorological Features

The climate of central Missouri is continental having the usual features attributed to such a climate. This includes cold winter and warm summer temperatures, and a warm season maximum in precipitation amounts.

The general climatic conditions for Columbia have been summarized by the U. S. Weather Bureau, and this summary is reproduced in Table 2.2. From the data in this table the seasonal distribution of temperature, precipitation, and other climatic characteristics may be obtained.

Missouri's climate is determined by the frequency and properties of air masses invading the state. Throughout the year, but particularly in winter, there are invasions of cold continental polar air from the north. The properties of this air mass are greatly modified by the time it arrives over Missouri. The layer of air near the surface has been warmed permitting considerable mixing.

Occasionally the movement of maritime tropical air over cold winter surfaces produces prolonged periods of foggy weather. This air mass has moved northward from the Gulf of Mexico and Caribbean Sea and contains much moisture. During the warmer seasons of the year, the maritime tropical air masses are not associated with foggy conditions. In fact, considerable vertical cloud development is possible in these air masses during spring, summer and fall.

Air masses sometimes move over Missouri from the Rocky Mountains. This flow of air is normally associated with clear weather and is quite frequent in winter, though not unknown during the remaining seasons.

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* Prepared by Wayne L. Decker, Assoc. Prof. of Climatology, University of Missouri.

TABLE 2.2

CLIMATIC CONDITIONS FOR COLUMBIA, MISSOURI

Columbia, Missouri

Normals, Means and Extremes[1]

Temperature and normal degree days

Month	Temperature – Normal – Daily maximum	Temperature – Normal – Daily minimum	Temperature – Normal – Monthly	Temperature – Extremes – Record highest	Year	Temperature – Extremes – Record lowest	Year	Normal degree days
(a)	(b)	(b)	(b)	19		19		(b)
J	38.8	20.7	29.8	77	1950	-18	1940	1091
F	43.1	24.3	33.7	76	1954	- 9	1951	876
M	53.0	32.6	42.8	85	1956+	- 5	1948	698
A	65.0	43.9	54.5	90	1956	20	1940	326
M	74.0	53.2	63.6	93	1956+	33	1954+	135
J	82.9	62.9	72.9	102	1954+	41	1945	14
J	88.8	66.8	77.8	113	1954	50	1944	0
A	87.0	65.4	76.2	103	1947	46	1950	6
S	79.5	57.9	68.7	102	1954+	29	1942	62
O	69.1	47.0	58.1	92	1953	21	1952	262
N	52.9	33.4	43.2	82	1949	4	1940+	654
D	41.9	24.3	33.1	75	1948	-12	1943	989
YEAR	64.7	44.4	54.6	113	JULY 1954	-18	JAN 1940	5113

Precipitation

Month	Normal total	Maximum monthly	Year	Minimum monthly	Year	Maximum in 24 hrs.	Year	Snow, Sleet – Mean total	Snow, Sleet – Maximum monthly	Year	Snow, Sleet – Maximum in 24 hrs.	Year
(a)	(b)	19		19		10		10	10		10	
J	1.80	5.96	1949	.42	1956	1.90	1951	4.0	14.1	1958	10.3	1958
F	1.57	4.15	1951	.22	1947	1.73	1955	2.8	9.5	1951	7.4	1951
M	3.01	5.76	1945	.51	1956	1.60	1952	5.2	8.9	1958	7.8	1958
A	3.80	9.53	1944	.61	1948	2.23	1954	.6	4.0	1957	2.3	1957
M	4.84	13.30	1943	2.61	1944	2.40	1956	T	T	1953	T	1953
J	4.90	8.93	1945	1.06	1953	3.47	1955	.0	.0		.0	
J	3.01	11.45	1958	.33	1954	3.86	1957	.0	.0		.0	
A	3.89	7.23	1944	.85	1957	2.71	1953	.0	.0		.0	
S	4.62	11.80	1945	.09	1940	2.45	1953	.0	.0		.0	
O	3.21	12.67	1941	.16	1952	4.05	1955	.1	.4	1954	.4	1954
N	2.61	4.94	1946	.21	1955	2.14	1958	1.4	7.5	1951	7.5	1951
D	2.07	4.42	1942	.19	1955	2.07	1949	3.5	7.1	1954	6.9	1954
YEAR	39.33	13.30	MAY 1943	.09	SEPT 1940	4.05	OCT 1955	17.6	14.1	JAN 1958	10.3	JAN 1958

Relative humidity, wind, sunshine and sky cover

Month	Relative humidity – Midnight CST	6:00 A.M. CST	Noon CST	6:00 P.M. CST	Wind – Mean hourly speed	Wind – Prevailing direction	Fastest mile – Speed	Fastest mile – Direction	Fastest mile – Year	Pct. of possible sunshine	Mean sky cover sunrise to sunset
(a)	19	19	19	19	10	9	10			8	19
J	76	79	65	69	10.9	S	56	NW	1951	47	6.4
F	74	79	63	65	11.2	NW	45	NW	1952	53	6.0
M	72	79	57	59	12.5	WNW	51	NW	1951	55	6.1
A	70	78	53	54	11.9	WNW	57	NW	1953	58	6.0
M	77	82	56	57	9.8	SSE	58	SW	1950	67	6.0
J	82	85	60	61	8.5	SSE	58	NW	1951	71	5.4
J	80	85	55	56	7.8	SSE	61	NW	1958	76	4.6
A	81	87	56	60	7.7	SSE	56	NW	1954	75	4.7
S	76	85	52	59	8.5	SSE	63	NW	1952	75	4.1
O	72	82	51	58	9.3	SSE	43	W	1954	67	4.5
N	72	79	56	62	11.4	S	49	NW	1955	57	5.7
D	76	80	64	68	11.1	S	56	NW	1957	48	6.4
YEAR	76	82	57	61	10.1	SSE	63	NW	SEPT 1952	64	5.5

Mean number of days

Month	Sunrise to sunset – Clear	Sunrise to sunset – Partly cloudy	Sunrise to sunset – Cloudy	Precipitation .01 inch or more	Snow, Sleet 1.0 inch or more	Thunderstorms	Heavy fog	Temperatures – Max. – 90° and above	Temperatures – Max. – 32° and below	Temperatures – Min. – 32° and below	Temperatures – Min. – 0° and below
(a)	10	10	10	19	10	19	10	19	19	19	19
J	7	6	18	8	1	*	3	0	10	27	2
F	8	6	14	8	1	1	3	0	5	23	*
M	7	10	14	11	2	3	1	0	2	17	*
A	7	8	15	11	*	5	1	*	0	3	0
M	7	10	14	12	0	8	1	1	0	0	0
J	9	11	10	11	0	11	*	8	0	0	0
J	10	12	9	8	0	8	1	14	0	0	0
A	13	10	8	9	0	8	1	13	0	0	0
S	15	7	8	7	0	5	1	6	0	*	0
O	14	8	9	7	0	3	1	*	0	2	0
N	12	7	11	7	*	1	1	0	1	15	0
D	8	7	16	7	1	1	2	0	7	24	1
YEAR	117	102	146	106	5	54	16	42	25	111	3

(a) Length of record, years. (b) Normal values are based on the period 1921-1950, and are means adjusted to represent observations taken at the present standard location.

Means and extremes in the above table are from the existing location (or last comparable location). Annual extremes have been exceeded at prior locations as follows: Lowest temperature -26 in February 1899; maximum monthly precipitation 14.86 in June 1928; minimum monthly precipitation 0.06 in August 1909; maximum precipitation in 24 hours 6.61 in September 1918; maximum monthly snowfall 23.4 in March 1912; maximum snowfall in 24 hours 12.8 in March 1937.

1. Table from Local Climatological Data with Comparative Data, Annual Summary. U. S. Weather Bureau, Department of Commerce

In summer, conditions sometimes develop which cause the air to subside from high levels in the atmosphere to levels nearer the surface. The air mass, often called superior air, associated with this condition occurs during droughty periods. The cloud-free atmosphere permits much warming at the surface and there is considerable mixing within this surface layer of air.

It has been noted by Decker (1) that Missouri's climate results from air masses which occasionally have properties which permit the accumulation of contaminants. But he notes that this condition is almost invariably of short duration and often occurs only during the hours of darkness and in valleys.

2.4.2. Meteorological Factors and Pollution

The meteorological factors affecting the collection of contaminants at the surface are clearly understood. These factors have been described by Thornthwaite et.al. (2) and Hewson (3). The general features of the meteorological picture are (a) contaminants drift with the wind from the source of origin, (b) the presence or absence of mixing action in the atmosphere either encourages or prevents the dilution of the pollutants in the atmosphere and (c) the occurrence of precipitation washes the contaminants from the atmosphere yielding fallout downwind from the source.

2.4.3. Variations in Wind at Columbia, Missouri

As indicated elsewhere in this report, the variations in topography around Columbia are very small. There should not be any consistent variations in wind direction and speed observed at one point when compared with that of another location. This is assuming that the comparisions are made at the same height above the surface and at locations where free air movement is permitted. For this reason the wind observations taken at the Municipal Airport in Columbia should be essentially the same as those reported elsewhere in the general vicinity.

In Figure 2.7 are shown the seasonal wind roses for Columbia, Missouri. This presentation shows the percentage of the time that the wind has blown from each direction by wind speed groups. A review of these charts reveals that southerly winds are the most frequent in all seasons. In summer, half the time the wind blows from between southwest and southeast. In spring and winter the frequency of winds from these directions is over a third of the time. The wind blows from east of south with a little higher frequency than from west of south in all seasons of the year.

There is a secondary maximum in wind direction corresponding to winds from the west-northwest and northwest. This is particularly noticeable in the winter when 21 percent of the time there are winds from these directions.

High wind speeds at the surface are more to be expected in winter and spring than in summer and fall. This not only increases dispersion of atmospheric contaminants by increased drift but increases the vertical mixing in the atmosphere.

Pollutants entering the atmosphere do not normally drift near the surface, but are carried aloft as lateral dispersion occurs. Shown in Figure 2.8 are the seasonal wind roses at 5,000 feet over central Missouri. These wind roses are quite different from those for surface wind observations. Not only are the wind speeds greater in all seasons, but winds from the west prevail. In winter, over 75 percent of the time, the wind blows from southwest to northwest at 5,000 feet. In summer southwesterly winds are more prevalent with air flow from the south-southwest to west-southwest occurring 36 percent of the time.

2.4.4. Stability of the Atmosphere at Columbia, Missouri

By definition a stable atmosphere retards mixing and the dilution of atmospheric contaminants. On the other hand, an unstable atmosphere is associated with mixing and the dilution of pollutants throughout the atmosphere.

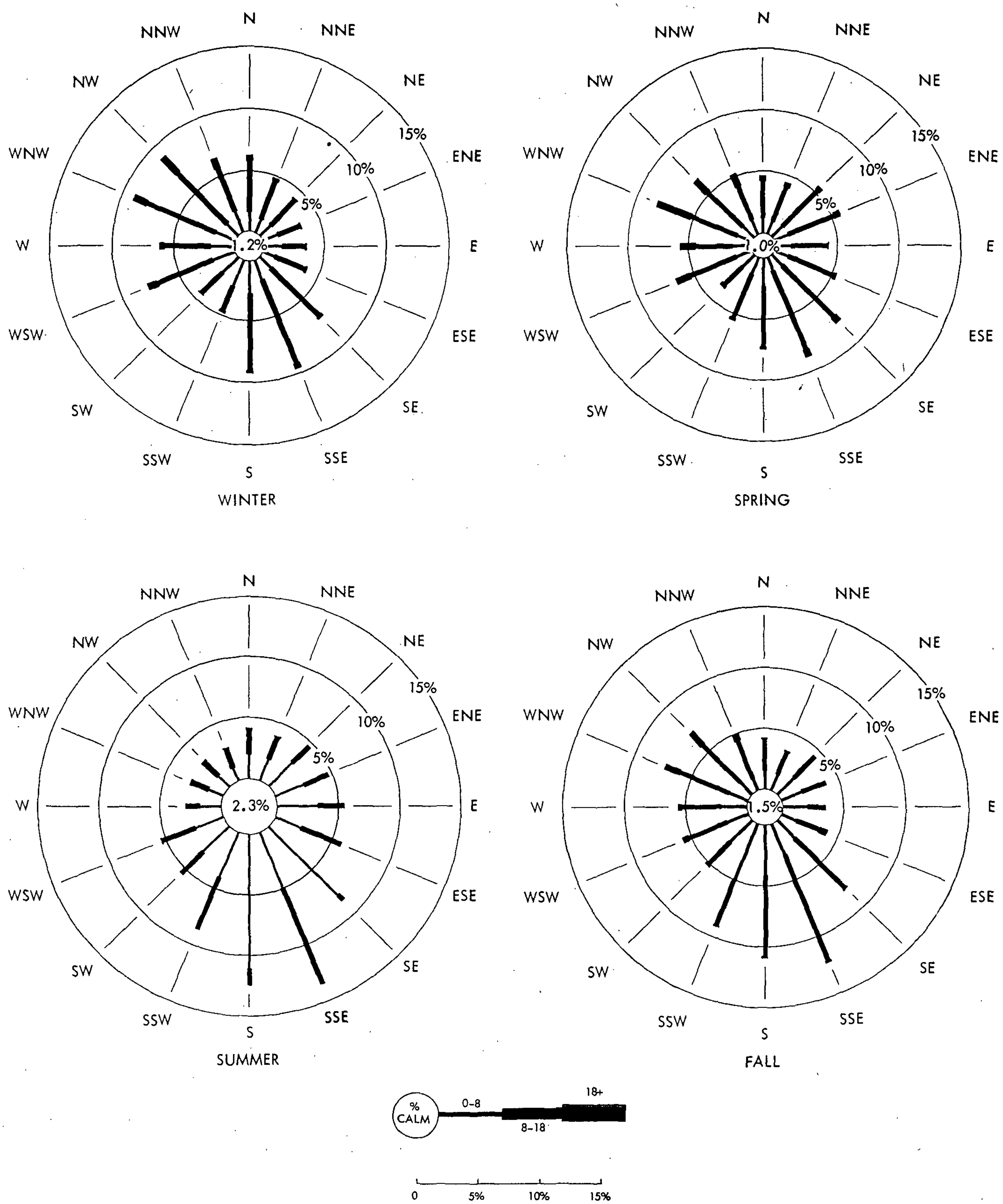

FIGURE 2.7 Surface Wind Roses for Columbia, Missouri for the period 1951 - 1959

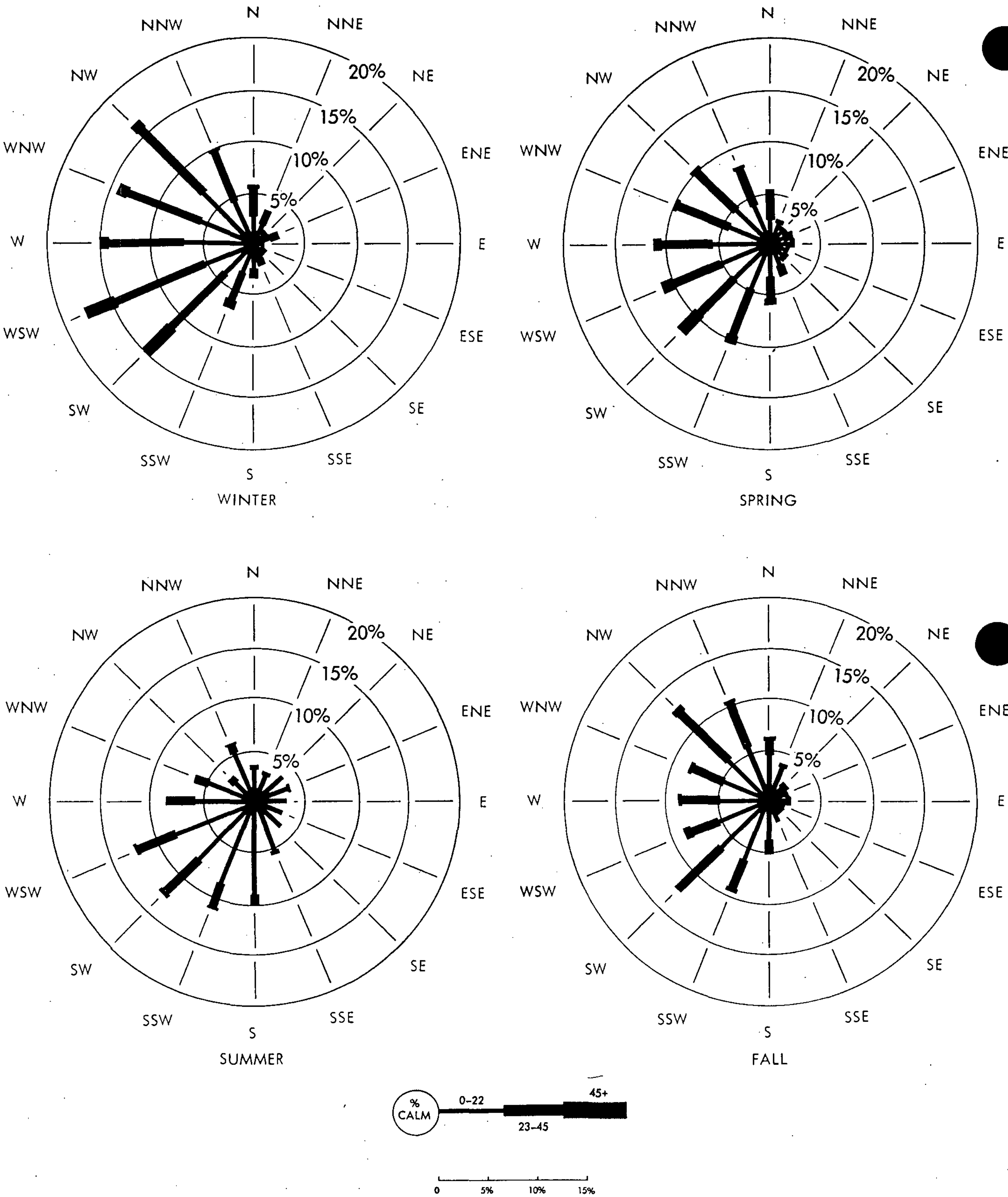

FIGURE 2.8 Wind Roses for 5000 feet over Columbia, Missouri (850 millibar Level) for the Period July 1949 – June 1956

The physics determining the atmosphere's stability has been described by Sutton (4) and others. In general, the atmosphere is more unstable during the periods of great wind shear and/or a relatively large decrease of temperature with height. Conversely, a stable atmosphere is associated with a small wind shear and/or a small or negative decrease of temperature with height (a negative decrease would actually be an increase of temperature with height).

To climatically describe the atmospheric stability one needs to obtain records of the wind shear and temperature gradients at Columbia. Unfortunately, the former is entirely lacking while the latter has been collected as a macro-observation and is virtually useless. If wind and temperature profiles are needed for the operation of a reactor facility, instruments must be installed at the site to measure the wind shear and temperature gradients.

Since measurements are made at a single height at the Weather Bureau installation, wind shear measurements are non-existent. For some years radiosonde flights have been made at Columbia and from these data temperature profiles are available. Since the data are intended to indicate the gross stability of the atmosphere, little effort is made to derive with precision the stability in the layer of air near the ground. This raises doubt as to whether the stability of the air near the surface may be obtained from such measurements. A greater objection to the use of radiosonde data for characterizing atmospheric stability is that the observation times do not sample all periods of the day. Prior to July 1957, the radiosonde and rawinsonde instruments were sent up into the atmosphere at about 9 a.m. and 9 p.m. Since this date the release times are 6 a.m. and 6 p.m. Since stable conditions in the layers of air near the surface prevail at night and unstable conditions at midday and afternoon, one would expect statistics derived from these soundings to be biased toward the stable condition in the lowest layer of the atmosphere. As a result, fallacious conclusions might be drawn from the use of such data. It was decided that little would be gained by obtaining summaries of these data for the studies.

A better indicator of the stability in the atmosphere is the occurrence of restrictions to visibility since these can only occur during periods with stable atmosphere. These restrictions follow a diurnal and annual trend, being more frequent at night and during the winter. The percent of time with restricted visibility is less than one percent in summer and fall, but is 20 percent in winter and 12 percent in spring.

During the periods with restricted visibility it is of interest to discover how wind directions vary. The seasonal wind roses for periods with restricted visibilities are shown in Figure 2.9. During summer and fall the prevailing directions of wind with restricted visibility are from the southeast and south-southeast. But during the winter and spring the prevailing directions are north and north-northeast. These values indicate that in the winter, when stable conditions are more frequent, contaminants will be carried toward the south and west.

2.4.5. Precipitation at Columbia, Missouri

Contaminants entering the atmosphere will be returned to the surface by precipitation. In Table 2.2 is shown the general seasonal distribution of precipitation at Columbia, Missouri. Although the greatest precipitation totals are received in the warm season of the year, the highest percentage of hours with precipitation are in winter. In winter and spring about 15 percent of the hourly observations report precipitation while in fall and summer the percentage was less than 10.

The seasonal variation in wind direction during periods with precipitation are shown in Figure 2.10. In summer and fall precipitation is most often associated with southerly and southeasterly winds with a second maximum during summer associated with north to northeast winds. The latter maximum is probably due to shift in winds associated with squall-line passages.

In winter, precipitation occurs during periods when north and northeasterly winds prevail. It is

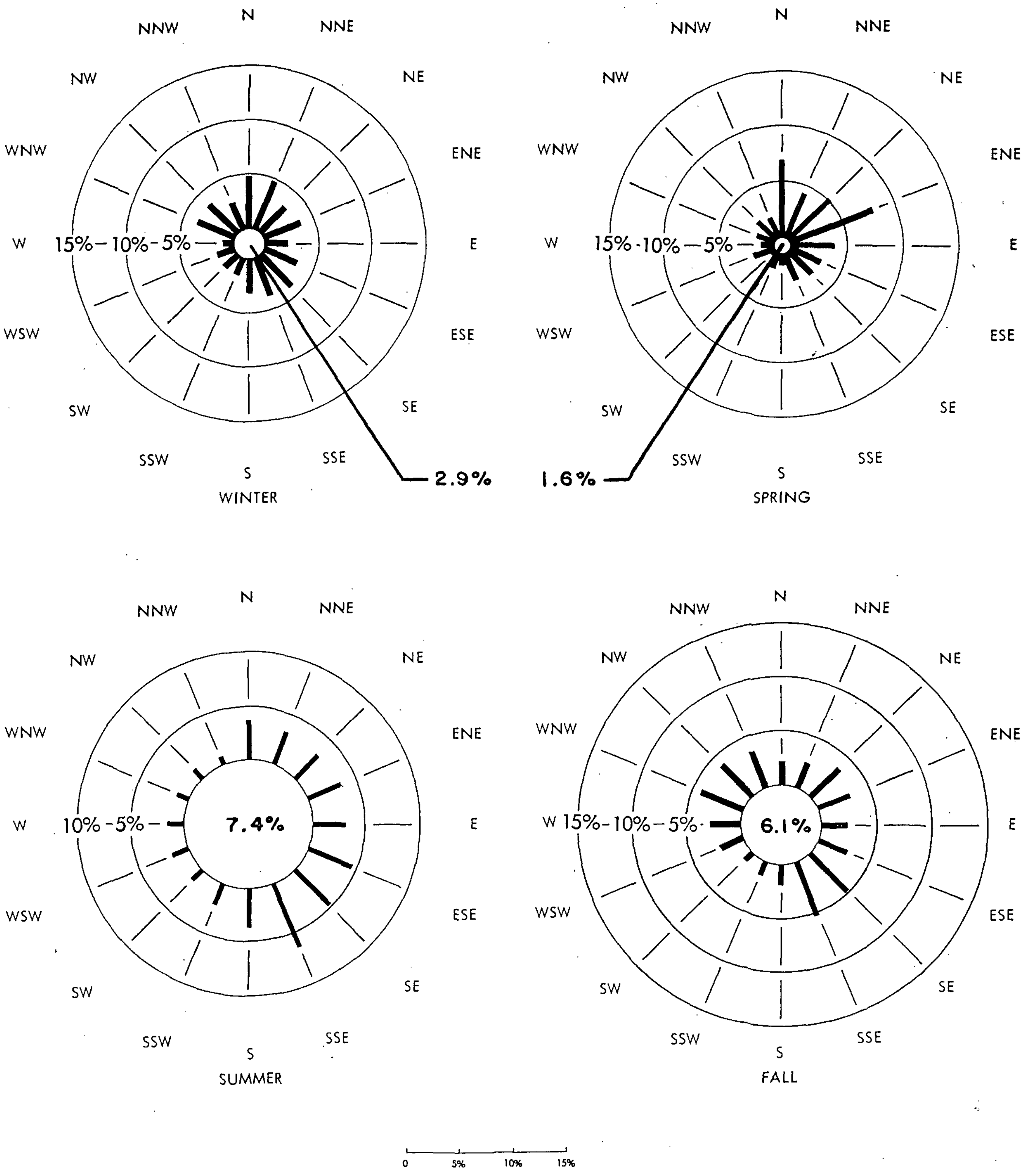

FIGURE 2.9 Wind Roses for Observation Times with Restricted Visibility at Columbia, Missouri for the period 1956 - 1959

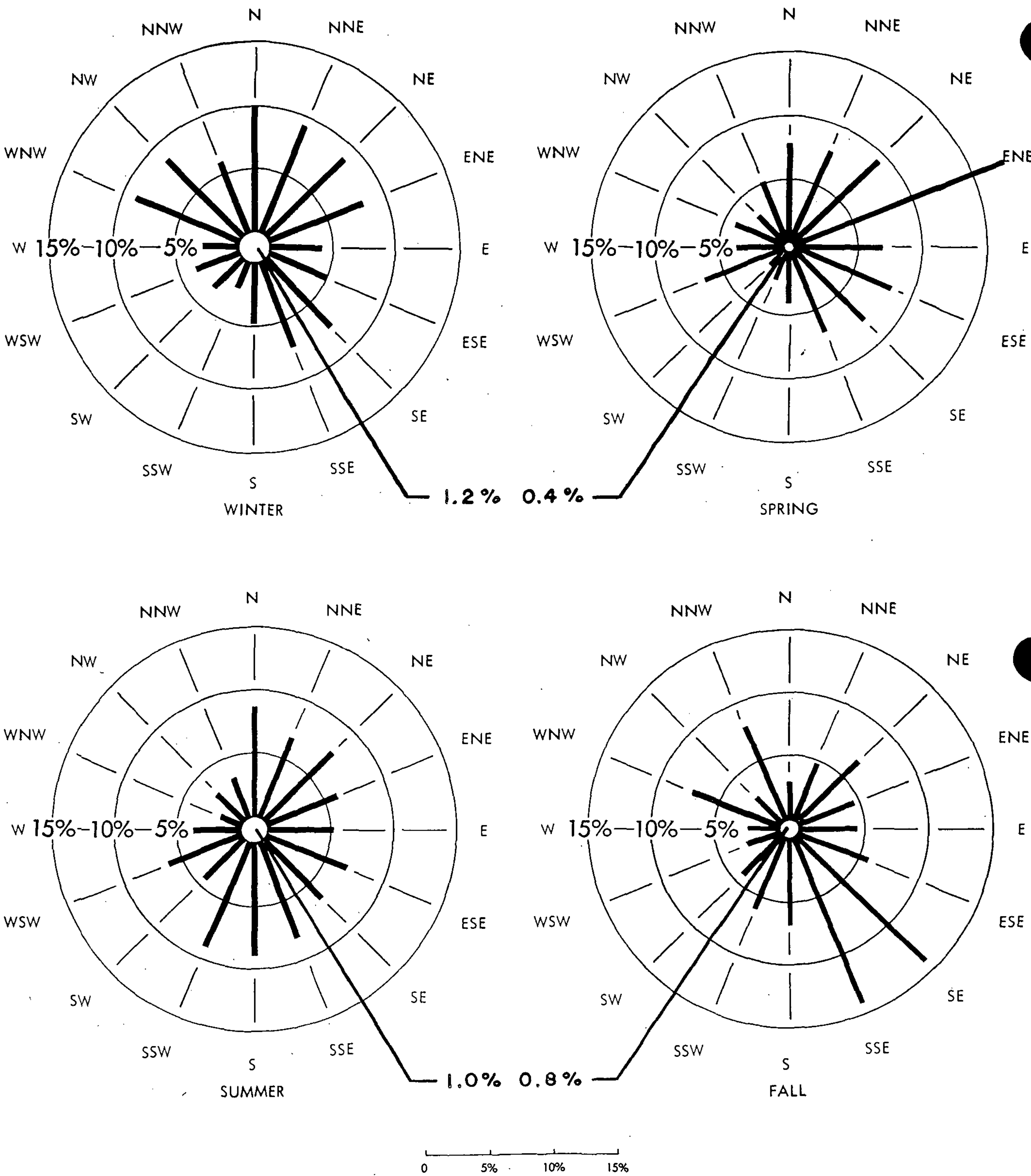

FIGURE 2.10 Wind Roses for Observation Times with Precipitation at Columbia, Missouri for the Period 1956 – 1959

significant that the same condition occurred during periods with restricted visibility. This indicates that in winter precipitation and restricted visibilities often occur simultaneously.

It is of some interest to note whether there are times during the day when precipitation is more likely to occur. The diurnal variation in precipitation is shown in Figure 2.11. These data indicate that there is not a great deal of difference through the day. In winter the highest percentage occurrence is in midmorning. For the other seasons there is a tendency for two maxima in rainfall occurrence. One of these is in the early morning before daylight while the second occurs in the late afternoon.

2.4.6. Frequency of Damaging Winds at Columbia, Missouri

Wind speeds up to 50 miles per hour are not uncommon at Columbia, Missouri. In Table 2.3 is a summary of the wind speeds for varying recurrence intervals. At Columbia, wind speeds of 50 miles per hour will occur on the average every other year while speeds above 100 miles per hour rarely occur.

TABLE 2.3

AVERAGE RECURRENCE INTERVAL FOR A GIVEN WIND SPEED

Average recurrence Interval (Years)	Wind Speed (Miles Per Hour)
2	50
5	59
10	65
20	72
50	85
100	94
200	105

Tornadoes represent a separate hazard to structures. Not only are extremely high winds encountered but falling atmospheric pressures can cause unvented

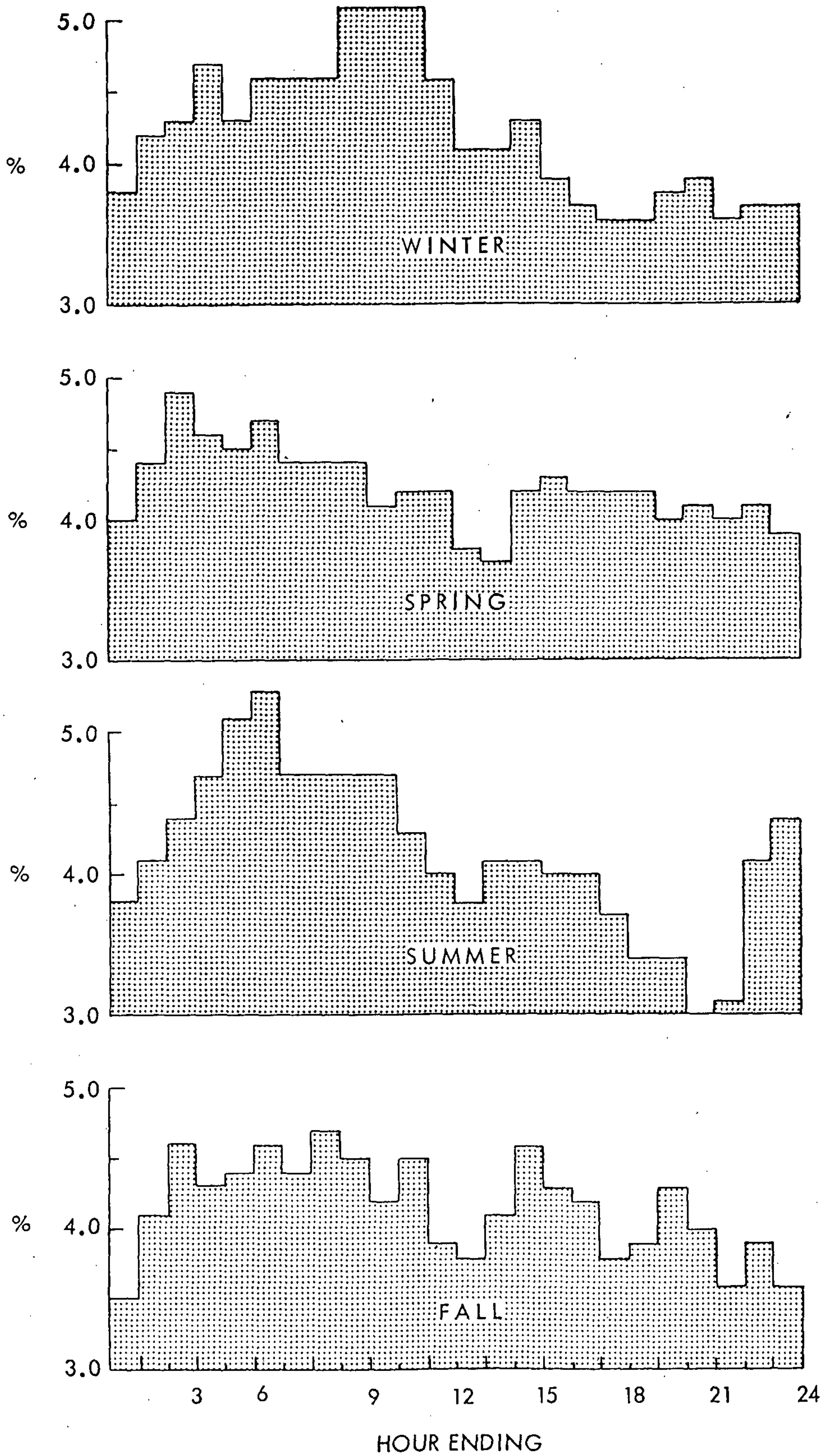

FIGURE 2.11 Diurnal Variation in the Frequency of Observations with Precipitation at Columbia, Missouri for the Period 1951 - 1959

buildings to "explode." Statistics relative to the intensity of wind velocities and pressure falls in tornadoes are not complete. This lack of data is due to inability of measuring devices to withstand the storms and to the fact that most storms occur where no measurements are being recorded.

Estimates of wind speeds within tornadoes vary from 200 to 500 miles per hour. Pressure falls associated with the passage of a storm, range from one to five inches of mercury (33 mb to 150 mb).

Fortunately the frequency of tornadoes is so low that it is difficult to estimate the probability of the event. In most of the midwest, there are on the average about 2.5 tornadoes per year in a 10,000 square-mile area. Since the average area damaged by a tornado is about 3 square miles, only about seven-hundreths of one per cent of the area in a region will be damaged during an average year.

2.4.7. Summary of the Effect of the Climate of Columbia upon Operation and Location of the Reactor Facility

1. Data presented indicate that in most cases contaminants entering the atmosphere will drift toward the north and northwest near the surface, but as the pollutant rises, it will likely drift in an easterly direction.

2. There are no reliable direct means of evaluating the stability of the atmosphere so that the seasonal variation may be obtained.

3. The occurrence of precipitation and stable conditions (as indicated by restriction to visibility) are somewhat similar. Both precipitation and restricted visibility occur during a greater number of hours in the winter and spring seasons than in summer and fall. For both of these occurrences the prevailing wind is from the north and northeasterly direction in winter. This indicates that when the hazard of contamination is greatest the drift will be toward the south and southwest.

2.4.8. References Cited

(1) Atmospheric Pollution and Meteorology, Proc. 5th Annual Air and Water Pollution Conf. Univ. of Missouri, Engr. Expt. Sta. Bull. 47, (1959).

(2) Thornthwaite, C. W., Mather, R. Jr., and Davis, F. K., Site Evaluations for Nuclear Reactors, Climatology 13: 1-91 (1960).

(3) Hewson, E. W., Meteorological Aspects of Atmospheric Pollution, Proc. U. S. Conf. on Air Pollution, McGraw-Hill, pp. 779-782.

(4) Sutton, O. G., Micrometeorology, McGraw-Hill, New York (1953).

2.5. Seismology*

2.5.1. Introduction

The most reliable evaluations of possible earthquake damage to an area are based on an examination of the area's seismic history. Fortunately, the U. S. Weather Bureau and the U. S. Coast and Geodetic Survey have compiled records of earthquakes occurring in Missouri for many years. These records and others have been ably summarized by Dr. Ross R. Heinrich (1), Director of the Seismological Observatory at St. Louis University. The writer is indebted to Dr. Heinrich for many helpful suggestions and criticisms.

2.5.2. Earthquake Districts in Missouri

In 1941 Heinrich, as is shown in Figure 2.12, mapped the zones of seismicity within and affecting the State of Missouri. According to him, no seismic activity has occurred since that date which would change the pattern of the map (Heinrich, 1960, personal communication). Six more or less definite

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* Prepared by: George W. Viele, Asst. Prof. of Geology, Univ. of Missouri.

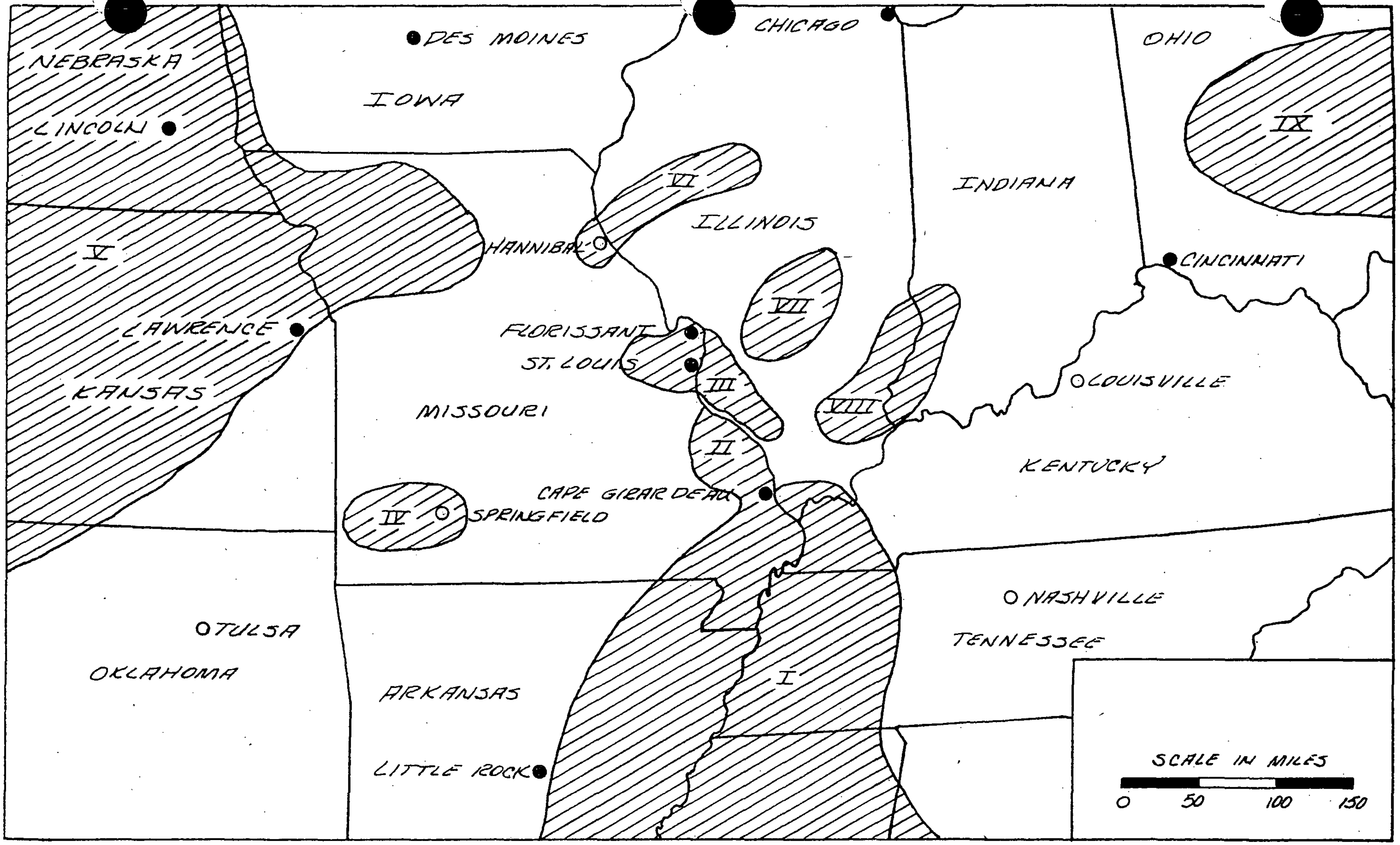

Fig. 2.12. Distribution of seismic areas in and about Missouri, and location of regional seismograph stations. Filled circles represent seismograph stations. Crosshatched areas represent seismic zones. These zones are called: I. New Madrid area. II. Saint Marys fault region. III. Saint Louis district. IV. Springfield region. V. Kansas-Nebraska zone, with the Northwest Missouri district represented as an extension. VI. Hannibal district. VII. Illinois Basin area. VIII. Harrisburg, Illinois, district. IX. Western Ohio area near Anna.

seismic districts are suggested for Missouri, and they are called because of their locations, the New Madrid, St. Marys, St. Louis, Hannibal, Springfield, and Northwestern Missouri seismic districts. Of these, the New Madrid district is by far the most active, followed by the St. Marys district. Approximately 84 per cent of the total seismic activity in Missouri between 1920 and 1939 came from these two areas; the remaining 16 per cent was divided among the other four districts (see Table 2.4 from Heinrich).

Heinrich shows that 85.4 per cent of all earthquakes in Missouri were of slight to moderate intensity (Table 2.5). Only about 10 per cent of these were destructive to property and dangerous to human life and 2 1/2 per cent of this value was due to the New Madrid earthquake of 1811-1813 alone.

The New Madrid District

The New Madrid district, the epicenter of the well-known earthquake of 1811-1813, is confined to the north end of the Mississippi embayment. The borders of the district extend southwestward from Cape Girardeau to the vicinity of Poplar Bluff, Missouri, and from Cape Girardeau south-southeast along the eastern margin of the Mississippi River flood plain. Little is known about the basement configuration of the faults, which cause the earthquakes, except that most of the epicenters lie below the towns of New Madrid and Caruthersville, Missouri.

The term, "New Madrid earthquake" is applied to a series of shocks beginning late in 1811 and continuing into the early part of 1813. The most violent shocks occurred on December 16, 1811, and caused several topographic changes in the area, the most notable being the formation of Reelfoot Lake, Tennessee. The earthquake was felt over an area of some 1,000,000 square miles. No other earthquake in the United States has been felt over a wider area.

Since 1813 numerous shocks have been felt in the New Madrid district. Earthquakes in 1843, 1865, 1883, 1895, 1903, 1905, 1920, and 1934, have caused structural damage, consisting mostly of cracked walls and fallen chimneys, in the area. None has ever approached the intensity of the earthquake of 1811.

TABLE 2.4

EARTHQUAKES OF MISSOURI ORIGIN AS CLASSIFIED BY SEISMIC AREAS, 1920-1939

Seismic Area	Number of Earthquakes	Per Cent of Total
New Madrid Area	30	58.8
St. Marys Fault Region	12	23.5
St. Louis District	4	7.9
Hannibal District	2	3.9
Northwestern Missouri District	2	3.9
Springfield District	1	2.0
TOTAL	51	100.0

TABLE 2.5

INTENSITIES OF EARTHQUAKES FELT IN MISSOURI*

Intensities	Number of Earthquakes	Per Cent of Total
I	6	85.4
II	7	
III	30	
IV	44	
V	19	
VI	11	
VII	8	5.8
VIII	1	0.7
IX	2	1.5
IX-XII	3	2.2
Undetermined	6	4.4
TOTAL	137	100.0

* The earthquakes which have been felt in Missouri are listed according to the number of each degree of intensity on the Wood-Neumann Scale. The intensity gradations range from the minimum (I) to the maximum intensity recorded for any earthquake. In the tabulation only the three main shocks of the New Madrid series were considered.

The St. Marys District

Earthquakes in the St. Marys District are probably caused by a belt of faulting, which crosses the northeastern flank of the Ozark uplift and extends northwesterly from Wittenberg in Perry County through St. Genevieve County. The St. Marys District is an active seismic area for Missouri, but the intensities of the earthquakes, compared to those of the New Madrid district, are generally low.

Other Districts

Seismic activity, both in terms of frequency and intensity, is low in the remaining 4 districts surrounding Missouri, and the districts do not, as a whole, show as significant relationships to the regional structures.

The exception is district 5, which is related very clearly to the Nemaha Ridge, a north-south high formed in Nebraska, Kansas, and Oklahoma during Pennsylvanian time. On April 9, 1952, a mild earthquake was felt along the entire length of this ridge. Lee (2) indicates that it was felt in Kansas City, but the area affected did not extend eastward as far as Columbia, Missouri.

2.5.3. Boone County, Missouri

Although a few local faults exist, no earthquakes are known to have originated in Boone County. The largest fault is exposed in Fox Hollow in the SE 1/4 sec. 12, T. 46 N., R. 13W, where it has a throw of about 120 feet. Unklesbay (3) mentions that there are other small faults associated with the Browns Station anticline in the northern part of the county. These structures are early Paleozoic in age; they are not active; and they do not constitute an earthquake hazard.

2.5.4. History of Earthquakes Felt in Columbia, Missouri

1811, December 16, New Madrid, Missouri...Although no records from Columbia are available which describe the effect of this quake, it was undoubtedly felt in this area. The intensity of the shock was such that it might have caused structural damage had the area been inhabited.

1843, January 4, New Madrid, Missouri...A strong earthquake occurring in the New Madrid district was felt from the seacoast of Georgia to beyond the western frontier posts (Heinrich (1), p. 193). It was probably felt in Columbia, Missouri, but the intensity of the shock is not known.

1867, April 24, Eastern Kansas...Scattered reports indicate that an earthquake occurring in eastern Kansas was felt as far eastward as Chicago. It may have been noticeable in Columbia.

1886, August 31, Charleston, South Carolina...The well-known earthquake occurring on this date had an intensity of II on the Wood-Newmann Scale at St. Louis. According to Capt. Clarence R. Dutton (4) it was felt as far westward as Columbia, Missouri; there was no report of structural damage.

1895, October 31, Charleston, Missouri...An earthquake that damaged every building in the commercial block of Charleston was felt slightly in Columbia, Missouri; no damage was reported. In the New Madrid district this shock probably ranks second in intensity to the New Madrid series of 1811-1813.

1917, April 9, St. Genevieve, Missouri (5)...A sharp disturbance at St. Genevieve and St. Marys, Missouri, had a Rossi-Forel intensity at Columbia of IV and a Mercalli intensity of IV. At the epicenter it had a Rossi-Forel intensity of VI.

According to the Daily Missourian (6) for April 9th the earthquake was not felt in Columbia, but on the following day several people reported feeling the shock. They thought that it was an explosion. No damage accompanied the earthquake.

1920, May 1, St. Louis, Missouri (7)...An earthquake on this date shook buildings in the entire St. Louis area. Two shocks were felt in Mt. Vernon, Illinois, and three were felt in Centralia, Illinois. An erroneous entry in the Earthquake History of the United States (8), as compiled by the U. S. Coast and Geodetic Survey, gives Columbia, Missouri, as the epicenter. The actual epicenter of this earthquake is unknown. The writer suggests that it was well to the east of Columbia, probably in Illinois.

In the Evening Missourian (9) of May 1, 1920, George Reeder of the U. S. Weather Bureau said that the shock was not felt in Columbia. In a later investigation, however, a few people reported feeling a slight tremor.

2.5.5. Summary and Conclusions

Undoubtedly, seismic activity will continue in the New Madrid and St. Mary's districts, and occasional strong shocks in these areas will be perceptible in Columbia, but they should not have damaging effects. The possibility of an earthquake as severe as the 1811-1813 New Madrid quake, however, cannot be eliminated. The records which are available do suggest that it was a rare and unusual occurrence.

In conclusion, Columbia's position within the central stable area of Missouri, and the seismic history of the area indicate that the probability of seismic damage to the area is extremely low.

2.5.6. References Cited

(1) Heinrich, Ross R., 1941, A contribution to the seismic history of Missouri: Seismological Soc. America Bull., V. 31, pp. 187-224.

(2) Lee, Wallace, 1954, Earthquake and Nemaha anticline: Am. Assoc. Petroleum Geologists Bull., V. 38, p. 338-340.

(3) Unklesbay, A. G., Geology of Boone County Missouri, State of Missouri, (1952), p. 105-109.

(4) Dutton, Clarence R., The Charleston earthquake of August 31, 1886; U. S. Geol. Survey, Ann. Rept. 4, pp. 461-463, 1888-89.

(5) U. S. Weather Bureau, 1917 Monthly Weather Review: v. 45, pp. 187-188.

(6) The Daily Missourian, Number 187-188, April 9-10, 1917.

(7) U. S. Weather Bureau, 1920 Monthly Weather Review: v. 48, pp. 307 & 310.

(8) U. S. Coast and Geodetic Survey, 1956, Earthquake History of the United States: Part I, No. 41-1.

(9) The Evening Missourian, Number 207, May 1, 1920.

2.6. Sewerage System*

2.6.1. Storm Sewers

The City of Columbia has relatively few storm sewers because of the natural drainage system. A map showing the location of the few storm sewers is not available. The City may, however, be divided into three general drainage areas. The section of town south of Broadway and south and east of the Wabash Railroad drains south to Hinkson Creek (Figure 2.13). Runoff from the western section of this area discharges into Hulen's Lake, the overflow of which drains to Hinkson Creek. The northwest section of town drains west and eventually into Hinkson Creek. The northeast section of town that lies north of the Wabash Railroad drains north to Bear Creek.

The University storm sewer system is shown in Figure 2.14. One system of drainage is directed generally from the northeast corner of the campus to the southwest corner. The runoff from the northeast

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* Prepared by: Edgar A. Jeffrey, Asst. Prof. of Sanitary Engineering, University of Missouri.

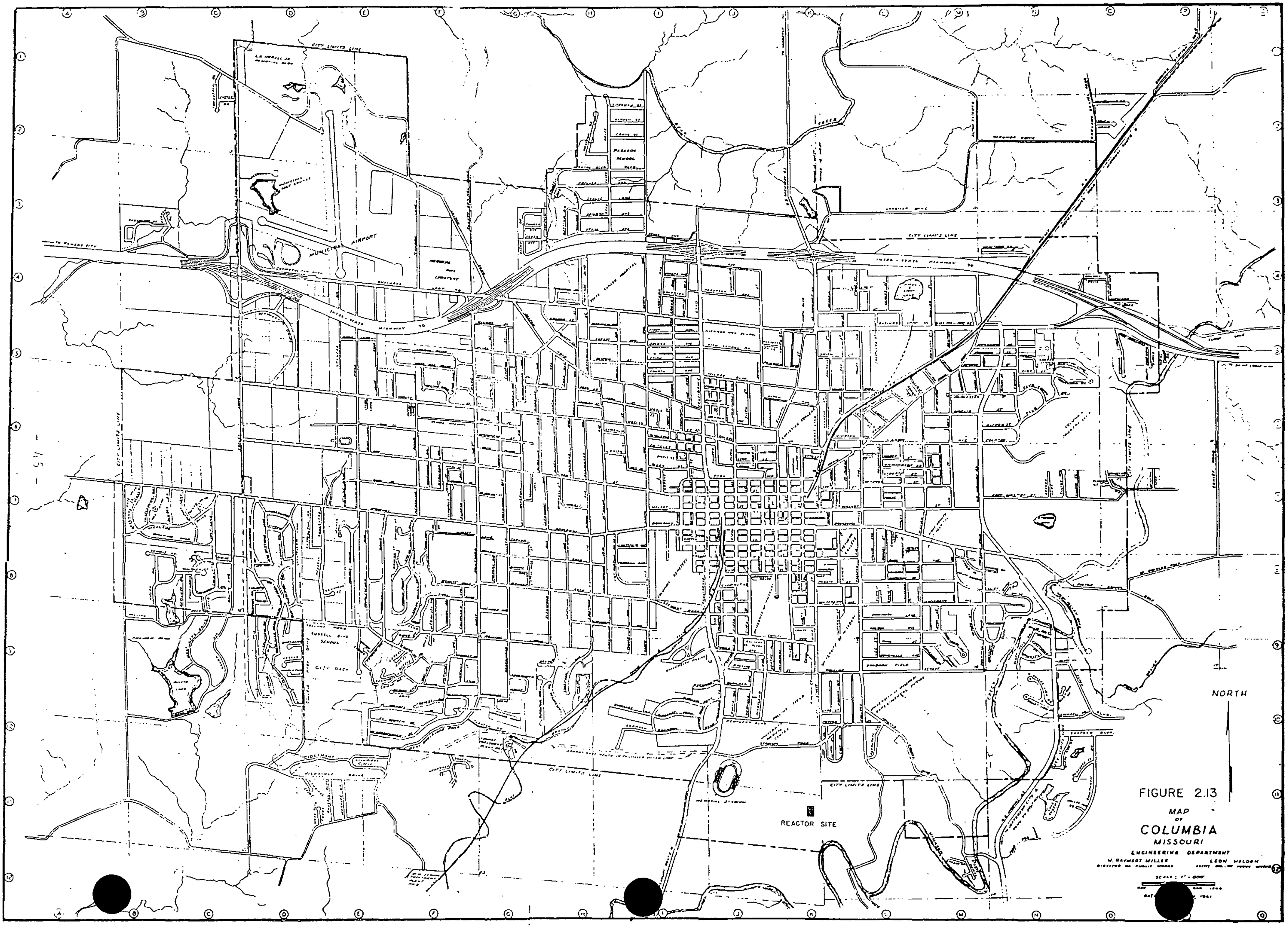


FIGURE 2.13

MAP OF COLUMBIA MISSOURI

section is picked up by a box culvert that crosses the men's athletic field, drains south under the stadium, and is discharged to Hinkson Creek. Run-off from the hospital area drains south and then west to combine with the flow from the box culvert mentioned above. The southwest section of the campus drains overland south and east to Hinkson Creek. The northwestern section of the campus drains west to Flat Branch Creek, which flows into Hinkson Creek.

2.6.2. Sanitary Sewers

University System

The University sanitary sewer system is shown in Figure 2.14. The northeast section of the campus is serviced by a sewer system that follows the same general direction as the storm sewer described above, that is, from northeast to southwest. It discharges into a city sewer that crosses the men's athletic field and drains eventually to a main sewer on Providence Road. Another sanitary sewer servicing the extreme east section of the campus flows west on Rollins and combines with this sewer at Rollins and Hitt Streets.

The formation of buildings just northeast of the stadium (Medical Center, faculty and student housing, the new women's dormitories, and the proposed dormitories on Providence Road and Kentucky Avenue, as well as the proposed reactor facilities) drain into a 12 inch line that generally follows Stadium Road west to the stadium and then northwest to a city sewer on Providence Road.

These two systems cross Providence Road, combine, and flow west to an outfall sewer that discharges directly to the sewage treatment plant.

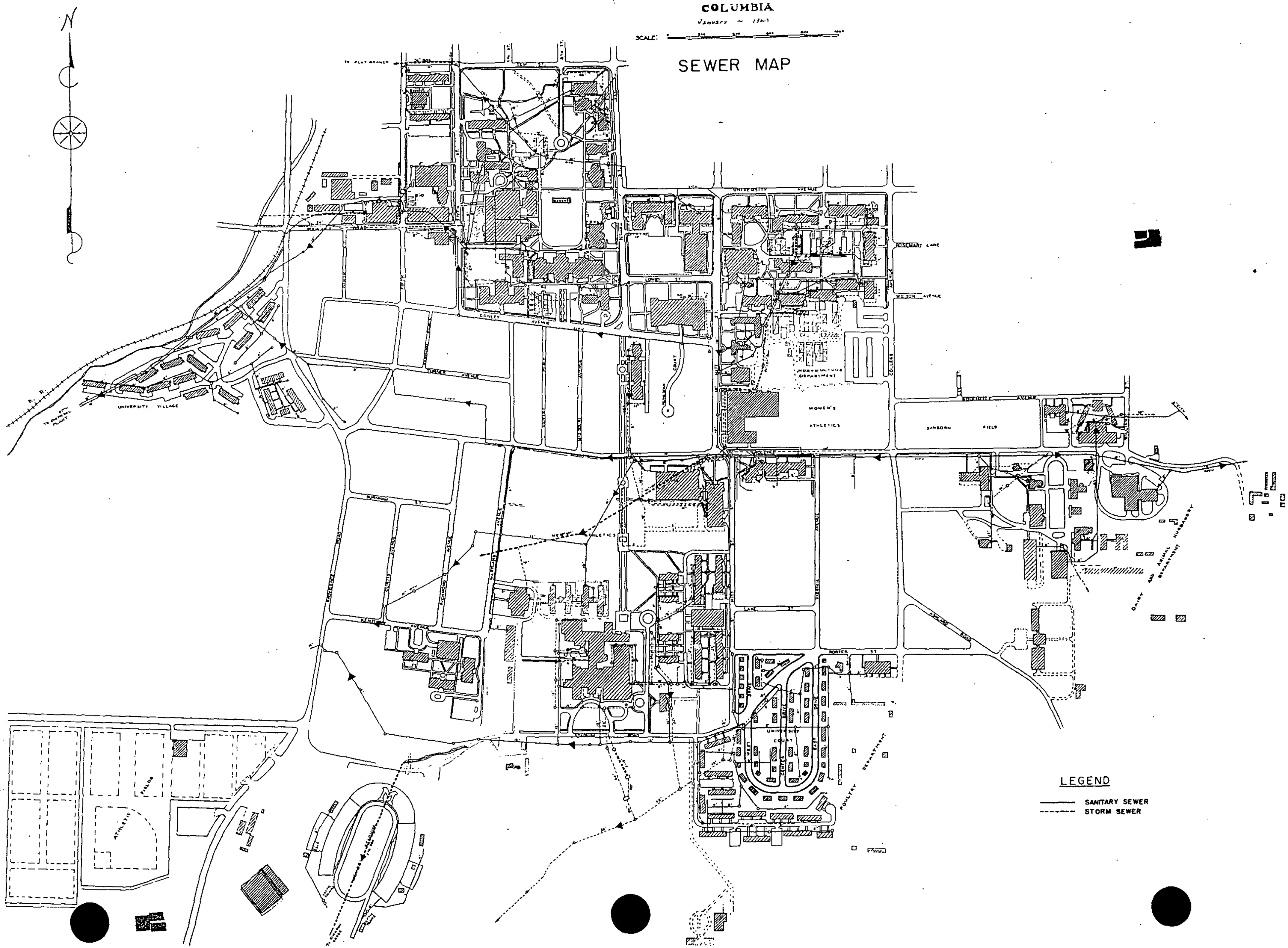
The northwest section of the campus is serviced by a line that flows west, approximately along Stewart Road, and discharges into a city sewer that follows Flat Branch Creek to the treatment plant.

Columbia System

The Columbia sanitary sewer system is shown in Figure 2.15. The drainage areas are practically

FIGURE 2.14

CAMPUS
UNIVERSITY OF MISSOURI
COLUMBIA



the same as those defined in the section on storm sewers. The eastern and southern sections of the city are serviced by an outfall sewer that follows Hinkson Creek to the treatment plant (trickling filter plant). The central section of town is serviced by a system that follows Flat Branch Creek to the activated sludge plant. Two lift stations are utilized in the north and west sections to accommodate some of the flow from this area, but most of the system flows by gravity to the plant.

The City of Columbia operates two sewage treatment plants. One is an activated sludge plant that was constructed in 1939 for a design population of 25,000 and a hydraulic loading of 2.5 million gallons per day. This plant is located on the Flat Branch Creek in the southwest section of the city.

A trickling filter plant was built in 1956. The plant was originally designed for 2.67 million gallons per day, but because of a lack of funds, the design flow was reduced to 1.7 million gallons per day. This corresponds to a design population of 17,000. The plant is located just south of the activated sludge plant at the junction of the Flat Branch and Hinkson Creeks. The new plant is not equipped with a flow measuring device, but preliminary investigations by the sanitary engineer with the City of Columbia indicate that the flow to both plants is approximately 3.1 million gallons per day, with a maximum of approximately 4.3 million gallons. In light of the recent federal census, the contributing population is estimated to be 33,000. Both plants are 85 to 90 percent efficient in BOD removal.

A raw sewage lagoon at the western edge of the city services approximately 2,000 persons. This lagoon was constructed to serve as a temporary treatment facility, and the sanitary sewage will eventually be directed to a treatment plant.

2.7. Site Location, Topography, and Surface Drainage

The campus of the University of Missouri is situated south of the main business district of the City of Columbia.

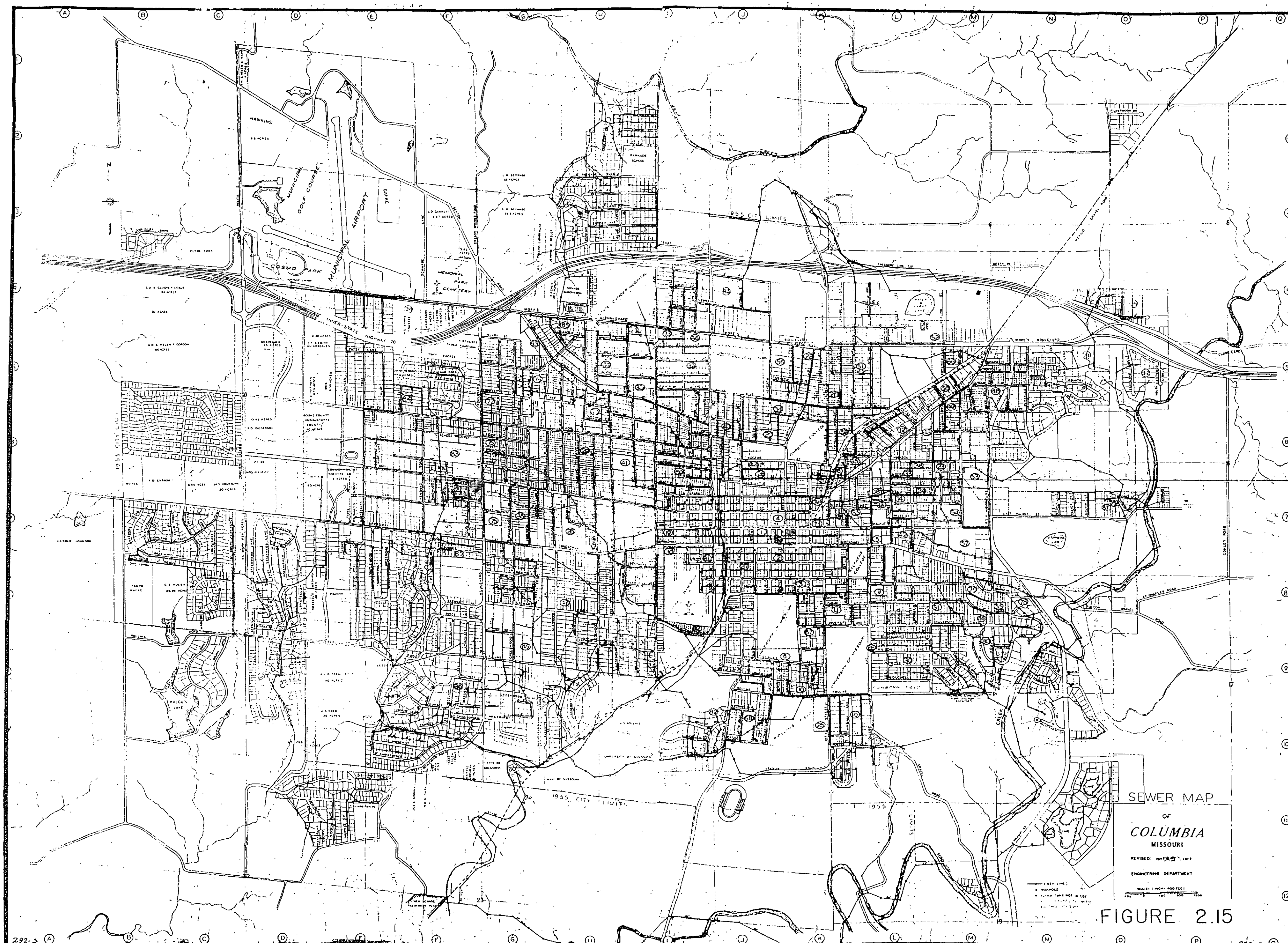

SEWER MAP
OF
COLUMBIA
MISSOURI
ENGINEERING DEPARTMENT
SCALE: 1 INCH = 400 FEET
MUNICIPAL AIRPORT
MUNICIPAL GOLF COURSE
COSMO PARK
MEMORIAL PARK CEMETERY
INTER-STATE HIGHWAY 70
1955 CITY LIMITS
CREEK

FIGURE 2.15

Approximately 733 acres of land comprise the main campus area. Of this acreage, approximately 500 acres are within the city limits. The property lines of the University are delineated on Figure 2.6. The reactor will be located at the south edge of the campus area.

Figure 2.16 shows the location of the reactor site with respect to the main buildings on the campus. It will be noted that the nearest occupied housing unit is 900 feet north-northeast, the Medical Center and University Hospital are 1750 feet north, and the football stadium is 1700 feet west-northwest from the site. There is an old cattle barn located immediately adjacent to the reactor site on the east side. This barn will be converted into a temporary warehouse facility and will ultimately be removed. There is an abandoned quarry located 1200 feet due east of the site.

The relationships of these features, one to the other, and with respect to the site, may be easier to visualize when reference is made to Figure 2.17. This is an aerial photograph taken from a point approximately one mile south of the site and looking north.

A map showing the topography of a 48 square-mile area around Columbia, Missouri, and the reactor site is presented as Figure 2.18. As may be seen, the area south of Columbia is hilly, with steep bluffs adjacent to the various creeks.

The topography in the immediate vicinity of the research reactor site is shown in Figure 2.19. There has been a certain amount of fill completed since this topographical map was prepared. This fill very nearly brings the depression, depicted on Figure 2.19 to the northwest of the site, up to the same grade as the site. None of the new construction for the research reactor will be made on fill. The reactor site has been selected to be at the top of the ridge pictured in the topographical map. Falling away from this ridge, toward the south, is a depression which contains Hinkson Creek. Hinkson Creek drains to the Perche Creek and the Perche Creek drains to the Missouri River. Surface drainage from the site will move south to enter the Hinkson Creek, which has a drainage area of about 43 square miles.

Any radioactive contaminants carried by surface run-off would find their way into Hinkson Creek. The volume of surface run-off from the area draining into the Hinkson Creek will be a large portion of a rainfall since Hinkson Creek is

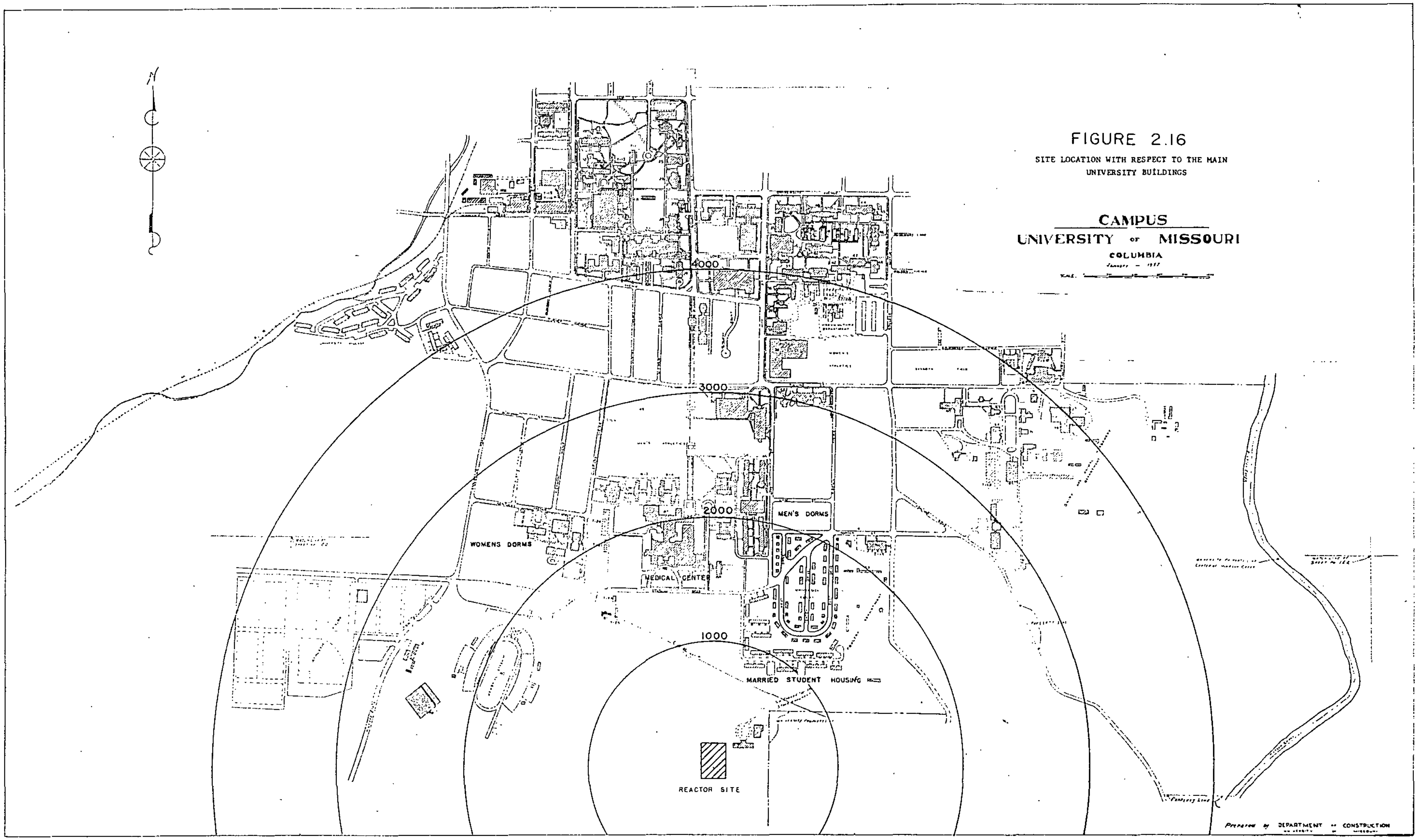


FIGURE 2.16

SITE LOCATION WITH RESPECT TO THE MAIN UNIVERSITY BUILDINGS

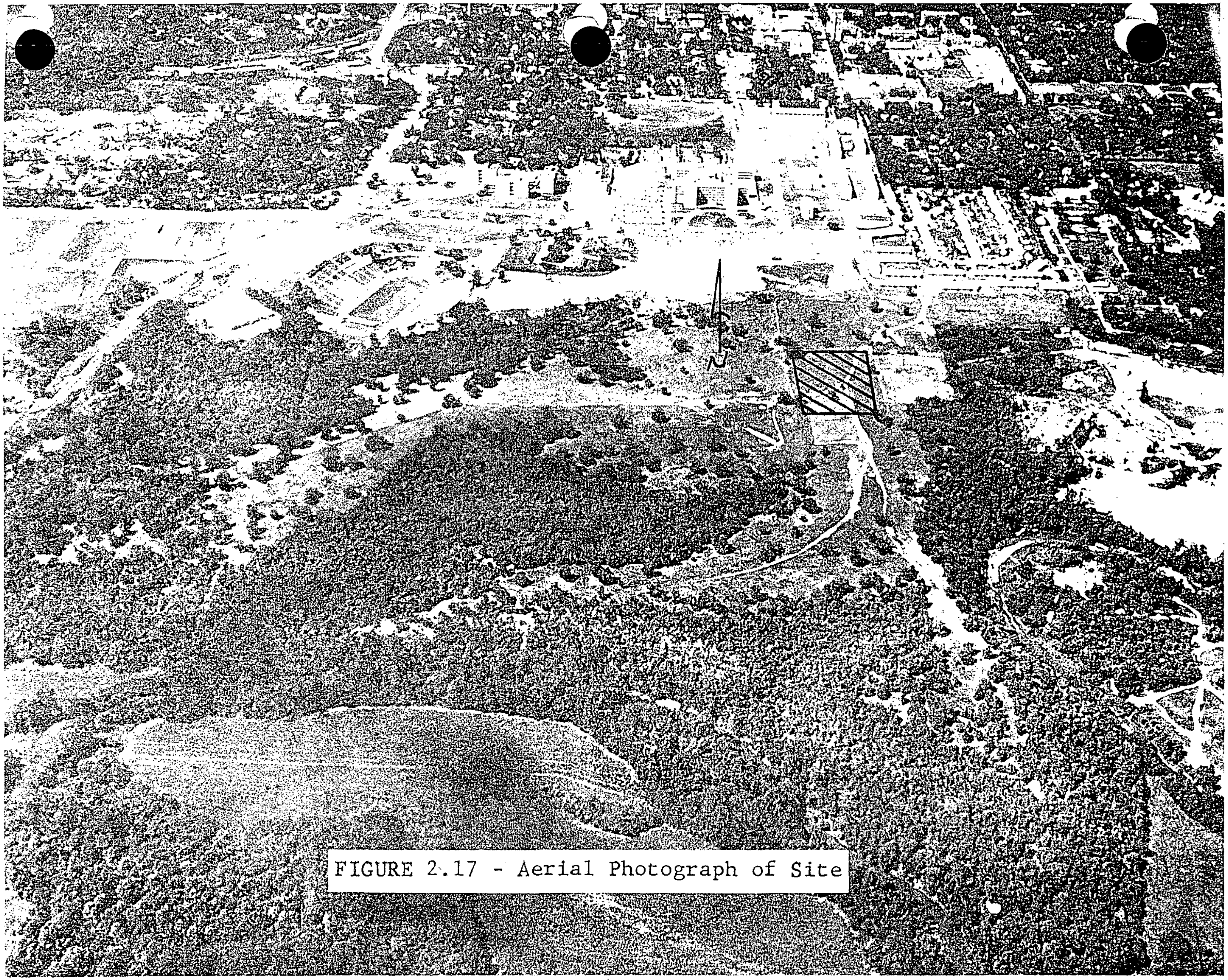
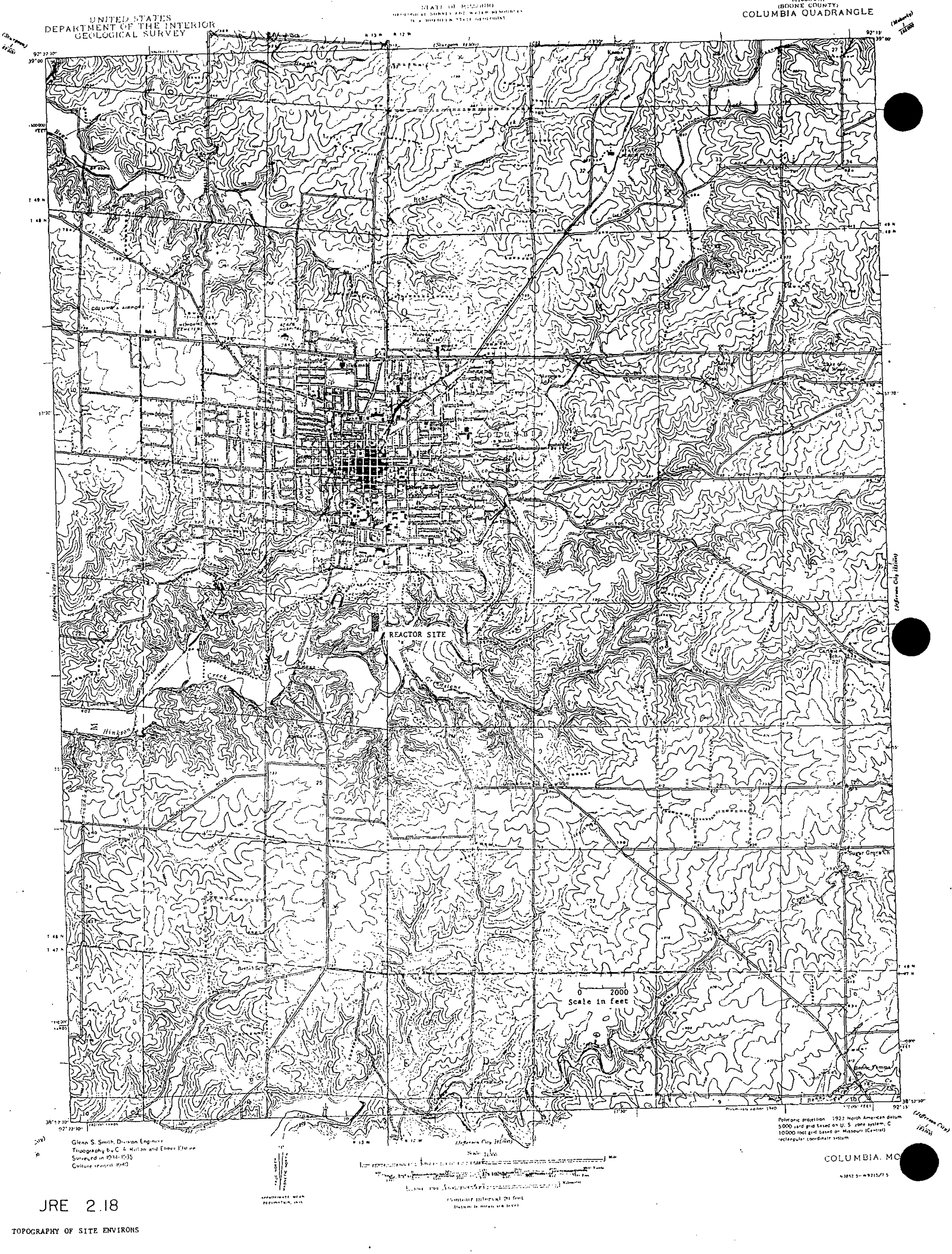


FIGURE 2.17 - Aerial Photograph of Site

FIGURE 2.19 SITE TOPOGRAPHY



IGURE 2.19 SITE TOPOGRAPHY

JRE 2.18

TOPOGRAPHY OF SITE ENVIRONS

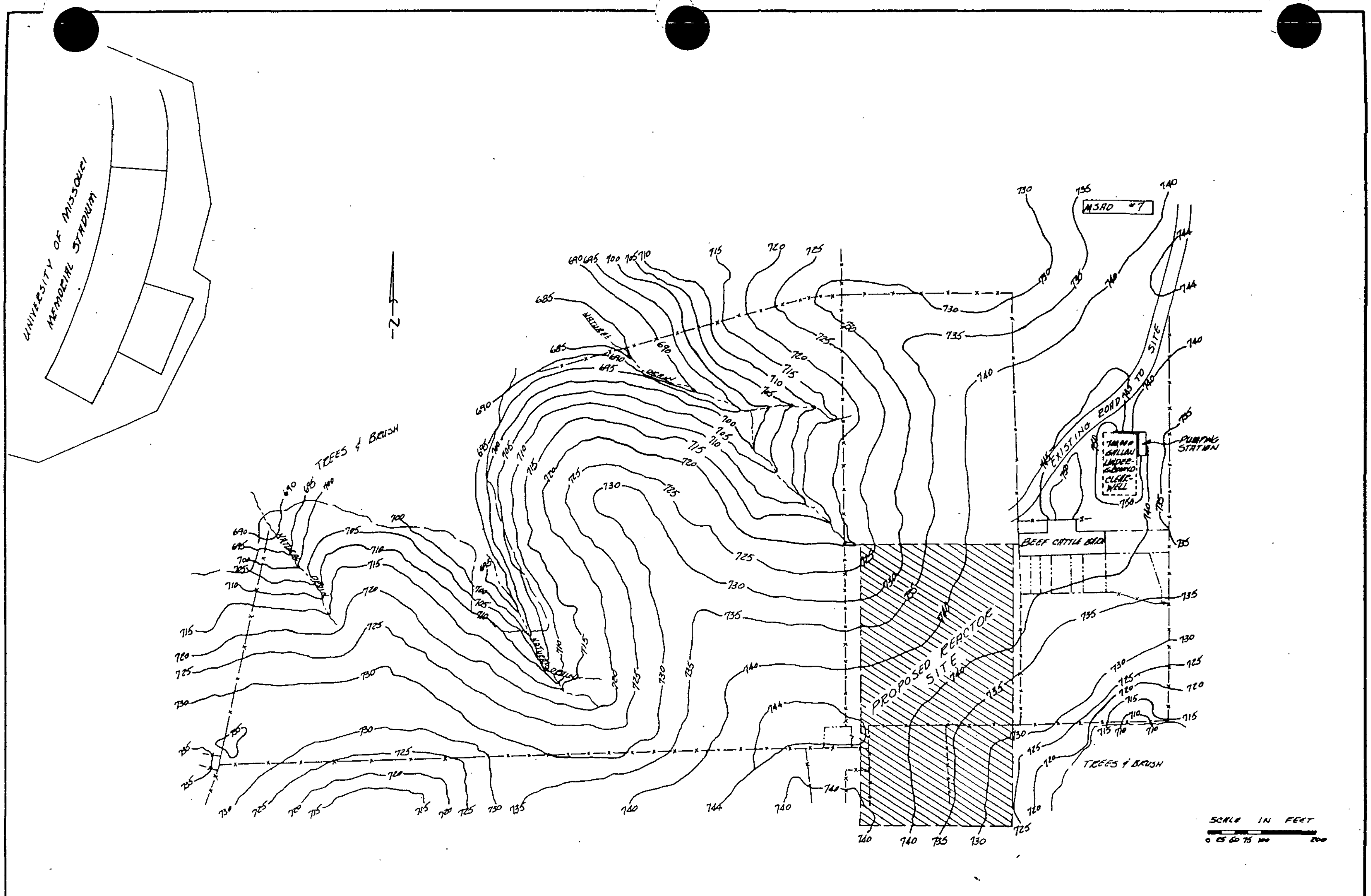


FIGURE SITE TOPOGRAPHY

bordered by relatively steep slopes and the soil structure is such as to minimize the infiltration of water. All of the land lying between the reactor site and the entrance point of runoff water from this site is owned and controlled by the University of Missouri. The nearest point of merging of runoff water from the site with Hinkson Creek is approximately 2,200 feet south of the site.

2.8. Disposal and Dilution of Surface Run-off Water

In the event of an incident leading to the release of radioactive contaminants which were precipitated to the ground and carried off by surface water runoff to the Hinkson Creek, there would be dilution achieved by the runoff water from the total area drained to the Hinkson Creek (Table 2.6). There are a limited amount of data available on the average flow of the Hinkson Creek. These data are reproduced in Table 2.7. It is to be noted that the flow rate of Hinkson Creek varies enormously from one season of the year to another.

The Hinkson Creek drains into Perche Creek. Perche Creek receives the surface runoff water from 405 square miles of land in Boone County (Figure 2.3 and Table 2.6). The flow of this body of water is quite seasonally dependent.

Perche Creek drains into the Missouri river at a point approximately 8 1/2 miles south-southeast from the site. The actual river miles from the point of entry of site runoff water to the mouth of Perche Creek is 19 miles. It is estimated (on the basis of drainage areas involved) that the minimum dilution factor achieved prior to entry of surface runoff water into the Missouri River would be a factor of 100.

Prior to the completion of up-stream reservoirs on the Missouri River, which were constructed under the Pick-Sloan Plan of the U. S. Army Corps of Engineers, this river was subject to high-flood flows from spring into the fall and very low flows during the winter months. With the upper and main stem reservoir system coming into operation in 1957, flood waters can now be impounded and released during drought periods of low flow. (The combined capacities of these five main stem reservoirs is 73,500,000* acre-feet which is

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* Effects of Missouri River Basin Development on River Flow, Wendell E. Johnson, Chief Engineer, Missouri River Division, U. S. Army Corps of Engineers.

TABLE 2.6

DRAINAGE AREAS FOR STREAMS AND RIVERS IN
THE VICINITY OF COLUMBIA, MISSOURI*

Stream	Drainage Area in Square Miles
Hinkson Creek	42.9
Rocky Ford Creek	20
Grindstone Creek	14.7
Perche Creek	405
Missouri River (above Jefferson City, Mo.)	509,000

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* Beckman, H. C.

"Water Resources of Missouri 1857-1926" Vol. XX, Second Series, Missouri Bureau of Geology and Mines, 1927.

TABLE 2.7

HINKSON CREEK DISCHARGE MEASUREMENTS*

Date	Flow in cfs
April 27, 1942	8.27
October 13, 1942	0.46
August 30, 1943	0.05
October 30, 1945	2.76
May 14, 1946	33.5
June 22, 1946	2.67
September 17, 1952	1.97
September 10, 1953	0.07
October 22, 1959	6.02
October 23, 1959	6.55
October 25, 1959	5.62
October 27, 1959	6.39

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* Bolon, Harry C. "Missouri Geological Survey and Water Resources," Vol. XXXIV, Second Series, (Surface Waters of Missouri, 1940-1949) 1952.

Also:

Personal letter from Elmer H. Sandhaus, Hydraulic Engineer, Water Resources Division, U. S. Department of the Interior, April 13, 1960.

approximately three times the average annual flow of the Missouri River at Sioux City, Iowa.)

Each summer the Reservoir Control Center in Omaha, Nebraska, with the assistance of the U. S. Weather Bureau and interested state groups, makes a forecast of the probable water to be available for an eighteen-month period in advance. They then correlate all water flow requirements for water supply, pollution control, irrigation, navigation, and power production from the various involved state agencies. A tentative operating plan is prepared, discussed, and revised in an effort to develop a final plan acceptable to all agencies for eighteen months in advance. These combined groups have established minimum flows at Kansas City as follows:

June through September	14,700 cfs
January, February, May and December	9,620 cfs
March, April, October and November	7,715 cfs

The first downstream municipality which uses Missouri River water as a drinking water supply is Jefferson City, located approximately 23 river miles below the confluence of the Perche Creek with the Missouri River. A summary of the minimum flow volumes for the Missouri River was given in the preceding paragraph.

The maximum flow volume of Perche Creek is never greater than 1/1000 the flow of the Missouri River. (Most of the year it is less than 1/10,000 of the flow.) Then a dilution factor of 100,000 would be a conservative estimate of the dilution achieved by site-runoff water prior to its receipt at the first municipality using this water as a drinking water supply.

This discussion completely ignores the adsorption-precipitation phenomena which takes place in a turbid body of water. Perche Creek flows through intensively farmed bottom land, heavily pastured hills, and unmanaged timber lands, and is characterized by high flood flows. All of these factors contribute to the turbidity of the stream and would indicate a very rapid silting rate. The Missouri River, frequently called the "Big Muddy," has a very high turbidity in all seasons of the year. This high turbidity and rapid silting rate would contribute toward the removal of any radioactive particulate contamination which entered these bodies of water.

All of the storm sewers in the City of Columbia, including the storm sewers to be installed on the reactor site, drain into Hinkson Creek. The sewage treatment plants of the City of Columbia also drain into Hinkson Creek. Then those comments made in preceding paragraphs relative to dilution faotors attainable from the Hinkson Creek to the Perche Creek and ultimately to the Missouri River, apply equally to storm sewer drain-off as well as sewage treatment plant effluents.

2.9. Population Density Study

It is not intended that a controlled exclusion area will be provided around the reactor, however no new construction is envisioned within 700 feet, the distance to an east-west highway which will be built between the reactor and the main campus. The nearest permanent resident is approximately 1000 feet from the proposed reactor location. A maximum of 8040 people would reside within a one-mile radius, 7124 of them occupying a quadrant centered on north, and 7912 located in the area north of the reactor (the northeast quadrant plus the northwest quadrant).

A medical center and 441-bed hospital are located 1850 feet north of the reactor. The University stadium has a capacity of 35,000 people and lies WNW at 1800 feet. It is occupied 5 days a year for 4 hours a day.

The technique employed for the determination of the population density in the immediate area surrounding the nuclear reactor site has been as follows: The area surrounding the reactor has been divided into a series of concentric circles of various radii, and the total compass of 360° has been divided into eight sectors. These sectors have been numbered from 1 through 8, beginning at the North Pole of the compass. The North Pole of the compass conforms to the direction of true North on the site locality. the Number of private residences in each segment of a given sector is tabulated in Table 2.8.

Another study was made to evaluate the maximum number of people in each sector of the circular areas. Because there are included in these areas various University buildings, the peak population of these buildings is included in this study. The results are presented in Table 2.9. There is an estimated maximum of 8040 people within a radius of 5000 feet.

TABLE 2.8

NUMBER OF PRIVATE RESIDENCES WITHIN
5000 FEET OF THE REACTOR

Distance in Feet	Sector Number - Clockwise from North 1	2	3	4	5	6	7	8	ToTal	Cumulative Total
0 - 500	0	0	0	0	0	0	0	0	0	0
500 - 1000	0	0	0	0	0	0	0	0	0	0
1000 - 1500	0	0	0	0	0	0	0	0	0	0
1500 - 2000	2	0	0	0	0	0	0	0	2	2
2000 - 2500	26	8	0	0	0	0	4	24	62	64
2500 - 3000	31	9	2	0	0	0	9	58	109	173
3000 - 3500	32	3	1	0	0	0	17	63	116	289
3500 - 4000	18	0	6	3	1	0	10	37	75	364
4000 - 4500	63	2	6	3	4	0	8	17	103	467
4500 - 5000	141	16	3	2	1	0	4	37	204	671
TOTALS	313	38	18	8	6	0	52	236	671	

TABLE 2.9

ESTIMATED* NUMBER OF PEOPLE WITHIN
5000 FEET OF THE REACTOR

Distance in Feet	Sector Number - Clockwise from North 1	2	3	4	5	6	7	8	Total	Cumulative Total
0 - 500	0	0	0	0	0	0	0	0	0	0
500 - 1000	17	0	0	0	0	0	0	0	17	17
1000 - 1500	181	358	0	0	0	0	0	0	539	556
1500 - 2000	879	40	0	0	0	0	0**	571	1490	2046**
2000 - 2500	921	62	0	0	0	0	16	798	1797	3843
2500 - 3000	443	36	8	0	0	0	36	232	755	4598
3000 - 3500	128	12	4	0	0	0	68	330	542	5140
3500 - 4000	228	0	24	12	4	0	40	524	832	5972
4000 - 4500	252	8	24	12	16	0	32	404	748	6720
4500 - 5000	564	64	12	8	4	0	16	652	1320	8040
TOTALS	3613	580	72	32	24	0	208	3511	8040	

* Based on these assumptions:

(1) Four (4) people per private residence,

(2) All students in their dormitories, and

(3) All hospital beds filled.

** Neglects University Stadium of capacity 35,000 people, 5 afternoons per year for 4 hours.

A table of projected population growth for Columbia, Missouri is given in Table 2.10. The technique used to make this estimate consisted of computing the average increase in resident population per decade as well as the incremental increase (or decrease) from decade to decade. Then to the last firm base year (census year 1960 in this instance) is added the average increase per decade and the incremental increase. This gives a forecast of the resident population of this City for the future (decade) years. The estimates of student enrollment made by the various schools have been added to this to arrive at a total for the City.

TABLE 2.10

PROJECTED POPULATION GROWTH FOR
COLUMBIA, MISSOURI, 1960-1980

Year	Resident Population	Total Students*	Grand Total
1950	25,335	10,909	36,244
1960	26,510	10,000	36,510
1970	32,330	20,000	52,330
1980	36,860	26,000	62,860

* Includes enrollment of University of Missouri, Stephens College, and Christian College.

3.0. REACTOR FACILITY

3.1. Introduction

3.1.1. Proposed Design

The proposed facility is a modified tank-type reactor, consisting of a closed reactor immersed in an open pool of water.

The reactor core consists of a cylindrical annulus occupied by curved-plate U-Al alloy fuel elements which are cooled by light water. A light water island and a test hole within the fuel annulus provide a flux trap facility for neutron activation purposes.

The reactor core is contained in a 12 in. diameter aluminum pressure vessel, and the vessel is pressurized to about 50 psi over the hydrostatic pressure of the pool, with a maximum pressure of 80 psia in the reactor coolant loop.

Both the reflector and the control rods are external to the pressure vessel, with the rods forming a shroud between the reflector and the vessel. The reflector, the rod gaps, and the center test hole are all cooled by unpressurized pool water which is drawn through these regions and circulated through a pool water loop having heat exchangers, a bypass demineralizer, and a holdup tank.

The core and the island are cooled by the pressurized reactor loop which, at 10 Mw, circulates 3600 gpm through an external heat exchanger loop and returns it at 140^{o}F.

The more significant design parameters may be summarized briefly as follows:

Initial Power Capability	5 Mw
Eventual Power Level with Additional Equipment	10 Mw
Coolant	40 to 80 psia H_2O
Fuel	6 kg enriched U in Al-U plates, clad with Al
Dimensions of Fuel Annulus	24 in. height 5.54 in. I.D. 11.62 in. O.D.
Reflector	BeO + graphite
Control	External rod shroud between vessel and reflector
Shielding	29 ft pool with barytes concrete walls

Experimental Facilities:

1	1.5 in. D. center test hole
4	Radial beamports
2	Tangential beamports
1	Shield Plug facility
4	Pneumatic facilities
11	Reflector irradiation baskets
1	Gamma exposure facility

3.1.2. Advantages of the Annular Core Design for this Application

The pressurized annular core concept has been developed as a transition between the open-pool type reactor, which is best suited for power levels lower than 5 Mw, and the more expensive tank-type reactor, which finds use at higher power levels.

Preliminary studies indicated that the open-pool type reactor is limited in power level by several effects which are significant at 10 Mw:

(1) Heat Exchanger Requirements

The core water in an open-pool reactor must enter at pool temperature; hence the mean temperature difference in the heat exchanger is small, and the heat exchanger surface must be large.

(2) Pool Activity

The use of aluminum-clad fuel leads to sodium activity from the recoil atoms produced at the clad surface by the reaction Al^{27} (n, α)Na^{24}. Maintenance of tolerable dose levels at the surface of a 10 Mw open-pool reactor would require either very high demineralizer flow rates or design features which would seriously compromise the advantage of accessibility normally offered by this type of reactor.

Relative to the tank-type reactor, as typified by the Oak Ridge Research Reactor, the proposed design presents the following advantages:

(1) Higher Beam-Tube Currents

A study of the available beam-tube currents, relative to those attainable from an equivalent solid-core reactor operated at the same power, indicates that the annular core provides twice as much beam-tube current. This results from the higher power density in the smaller core - the beam tubes "see" a higher fission density and a higher scattering density. In this sense the annular core appears desirable from the standpoint of external utilization of neutrons, and the flux trap itself might be regarded as a bonus.

(2) Elimination of Rod Penetrations and Seals

A thin fuel annulus may be controlled by an external shroud of rods. By locating these between the pressurized core and a good reflector, the pressurized reactor can be controlled adequately by "unpressurized" rods. The relative savings, which apply equally to the open-pool reactor, may amount to $70,000, plus the possible elimination of a sub-pile room if inconvenient rod penetrations would otherwise compel the use of bottom-driven rods.

(3) Inexpensive Reactor Vessel

Only the small diameter core need be pressurized, and the only penetrations consist of the inlet and outlet piping. The resulting vessel is far smaller and simpler than those required for typical tank-type reactors.

3.1.3. Design Requirements

Realization of these advantages requires that certain design conditions be fulfilled and that certain problems be solved which are unique to this annular core concept.

(1) Sufficient control must be available from the partial shroud of rods permitted by engineering design considerations. Also, since the rods form a curtain in front of the beam tubes, it is desirable to terminate beam tubes below the core centerline and to insure that the rods will be half-way withdrawn during operation.

Study of the reactivity balance indicates that sufficient rod worth is available and that the half-withdrawal conditinn can be fulfilled if a reasonable fraction of experimental absorber remains in place during operation. The amount of withdrawal required for the shutdown margin, the temperature effect, minimum concentrations of xenon and samarium, and minimum experimental absorber then corresponds to half-withdrawal.

(2) It is desired that certain of the beam ports terminate at the inner face of the reflector to provide maximum fast neutron current. This requirement is particularly severe for a thin annular core.

The proposed design minimizes the effect of beam ports upon reactivity and power distributions by terminating all beam tubes at the outside of the inner reflector, and extending only two tubes across the reflector by means of small empty cans having the same diameter as the anticipated collimator.

(3) The control rod requirements favor a thin fuel annulus, while the inner fuel radius is essentially fixed by the optimum island dimension ($\sim$ 4 cm total water thickness). Consequently a high power density and relatively low heat transfer surface results.

The reactor is pressurized 50 psi over pool pressure to raise the saturation temperature and permit non-boiling operation with reasonable flow, even at this high power density. The small amount of overpressure requires essentially no increase in the thicknesses of the vessel or the coolant equipment and should permit use of a simple closure device.

The overpressure also permits a higher core inlet temperature and results in consequent savings in heat exchanger surface because of the larger mean temperature difference in the heat exchangers. This is offset somewhat by the increased pool heat exchanger surface required because of conduction from the reactor loop to the pool.

3.1.4. Development of Flux-Trap Reactors

The flux-trap reactor is characterized by thin fuel regions adjacent to a good moderator which slows down the neutrons and causes the thermal flux to peak in a region accessible for experiments. The simplest geometry and the most effective for small test samples is an annular core which surrounds a light water island containing a test hole, as proposed in the present design.

Although operating experience is lacking for this type of reactor, the concept has been well investigated by several agencies. Considerable background information has been accumulated in the following developments.

Early interest in flux-trap reactors led to the conceptual design of the Advanced Engineering Test Reactor (1). From this concept has evolved the design of a very high flux testing reactor scheduled to be constructed at Idaho Falls by the Atomic Energy Commission (2). Oak Ridge National Laboratory is also constructing a flux trap - the High Flux Isotope Reactor (3). University application of a flux trap reactor was considered in Internuc 22 (4).

3.2. General Description

3.2.1. Overall Plant

3.2.1.1. Reactor Building

The reactor pool, the control room, and certain office and laboratory rooms are enclosed in a 60 ft x 60 ft concrete reactor building. The reactor building extends three stories above grade and is surrounded at grade level by the rooms of the laboratory building. Effectively, it extends two stories above the roof of the laboratory building.

The beam-port floor or lower floor of the building is 16 feet below grade. The upper floors extend only to the edge of the pool structure so that the pool and experimental facilities are accessible to an overhead crane (see Figures 4.1 through 4.4).

Personnel entry is controlled through the second floor corridor to the laboratory building. Equipment and fuel casks may be delivered through a truck entrance below grade at the beam-hole floor. Coolant and utility lines entering the building are run through water seals to preserve integrity of the containment and all lines can be isolated within the building by check and stop valves.

The structure and penetrations of the building are described in greater detail and the containment function is discussed in Sections 4.0 and 6.0.

3.2.1.2. Pool Structure

The reactor is located eccentrically within a 10-ft -diameter pool, 29 ft deep (see Figures 3.1 and 3.2). Twenty-three feet nine inches of demineralized water covers the fuel during operation, reducing the direct gamma dose rate at the pool surface to about 1 mr/hr at 10 Mw.

The radial shield is comprised of the pool water plus about 5 to 6 1/2 ft of barytes concrete

[redacted]

[redacted]

(density: 3.5 gm/cc), sufficient to reduce the 10 Mw dose rate to 0.75 mrem/hour at the shield surface (see Figure 3.3).

During refueling operations, the pool level is lowered 6 1/2 feet to permit easier access to the fuel from either of two catwalks which are pinned to a ledge on the pool wall. At two days after shutdown from 8 hour daily operation at 10 Mw, one element delivers about 7.5 mr/hr to the observer on the catwalk as it is lifted over the lip of the vessel to be deposited in temporary storage racks on the pool wall.

Permanent storage facilities are provided behind a thick conrete weir which retains sufficient shielding water to shield fuel elements whenever the pool is emptied. The vessel lid and alternate reflector pieces can be stored on a shelf in this area. The barytes concrete thickness is sufficient to reduce the dose rate to 0.75 mrem/hour at the shield surface, with a layer of spent elements stored against the wall.

Experimental facilities which penetrate the shield wall include 6 beam ports, one shield-plug facility which may be used as a thermal column, four pneumatic-tube facilities and a gamma-exposure facility.

The entire pool is lined with 3/8 in. aluminum. The pool liner plus all aluminum penetrations in the concrete are coated to prevent corrosion of the aluminum.

The rod drives are supported on a traveling bridge structure which can be retracted to permit access to the reactor (Figures 3.1 and 3.4). Normally the rods are disconnected from their extensions just above the reactor vessel and the bridge, together with the immersed rod extensions, are backed to a stop immediately above the thermal column extension.

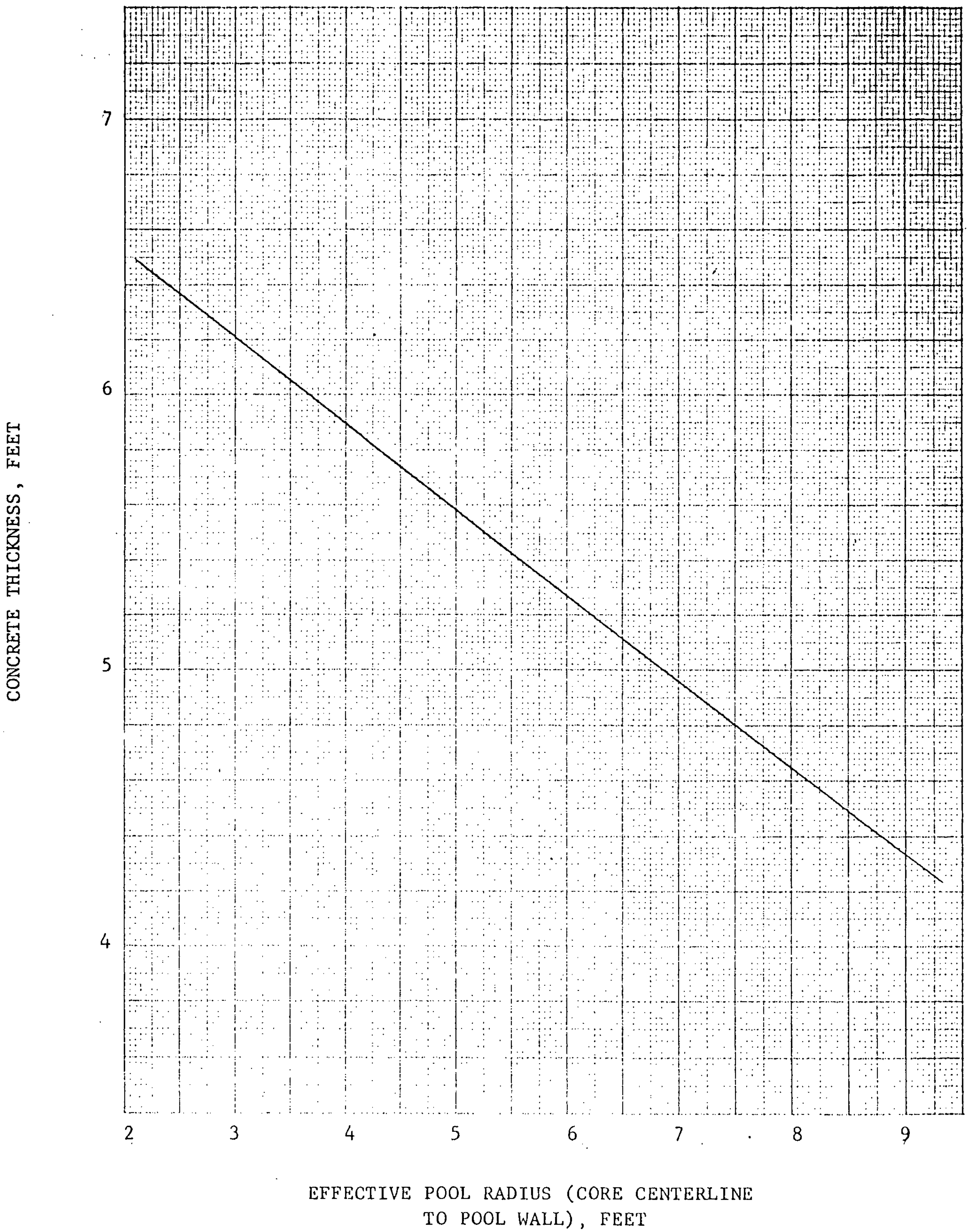
3.2.1.3. Coolant-Equipment Area

Coolant Loops

The reactor is cooled by a pressurized loop; and the pool, reflector, and test hole are cooled by a separate loop. Both loops enter and return through

FIGURE 3.3

THICKNESS OF BARYTES CONCRETE (ρ = 3.5)
REQUIRED FOR 0.75 MREM/HR AT 10 MW



[redacted]

the pool floor. All lines have check or stop valves at the point where they pass beneath the shield face and enter a trench leading to the coolant equipment room. The trench in the beam-port floor is shielded with removeable five-foot thick ordinary concrete blocks to reduce the N^{16} dose rate to 0.75 mr/hr.

The lines pass through a water seal and enter the coolant equipment room, which is separated from the reactor building by at least 4 ft of ordinary concrete or 6 ft of dirt.

Coolant Equipment Rooms

The coolant equipment is separated into the three areas as indicated in Figure 4.1:

(1) An area shielded on all exposed sides and on the ceiling with 4 ft of ordinary concrete or 6 ft of dirt. This area contains coolant equipment for the two loops, both of which are intensely radioactive because of the N^{16} activity.

(2) An accessible area containing secondary pumps and motors, pressurizing control equipment, the water softener and make-up demineralizer, control valves on non-radioactive lines, and gages and recorders.

(3) Shielded cells for the demineralizers, with reach rods penetrating [redacted] ft of concrete into region 2 for switching flow to the spare demineralizer. The ceiling is equivalent to at least [redacted] ft of concrete.

The accessible area is provided primarily because of the possibility that the reactor will be operated continuously. If it is to be operated always on an 8 hour-per-day schedule, limited access to the equipment in region 1 should be feasible each morning previous to start-up, and the advantage of an accessible area is reduced. It may be noted that the pool coolant equipment, other than the hold-up tank, may be located in the accessible area if the holdup tank is upstream from the other components.

Demineralizers

Mixed-bed demineralizers are proposed, with equipment provided for their regeneration. A spare 50 gpm unit is included which can be valved into either the pool or reactor loop when a demineralizer in that loop is to be valved off and allowed to decay. After a week or two, the sodium and other short-lived activities will have vanished and the only activity remaining should be long-lived nuclides resulting primarily from impurities in the aluminum. It is anticipated that the dose levels at a shutdown unit will be reduced sufficiently after this cooling time to permit direct access for bed separation; however, experience from similar installations will be reviewed during the detailed design.

The demineralizers require $\sim$ 4 ft of ordinary concrete shielding during operation because of N^{16}, and $\sim$ 1.5 ft after shutdown because of Na^{24}. All the cells are located behind a common 4 ft wall. Within the wall the units are shielded from each other by 1.5 ft walls and on top by 1.5 ft removable concrete blocks. After shutdown, entry can be gained to the corridor above the cells, and blocks are removed from above a decayed unit by means of a 1-to 2-ton traveling crane.

Tanks for preparing and storing regenerant solutions are located outside the building at grade level. All waste regenerant solutions, as well as any leakage water, flow to a shielded sump beneath the floor of the coolant equipment room from which they may be pumped to the liquid-waste treatment area in the hot laboratory.

3.2.1.4. Grade Level Facilities

Cooling towers and a demineralized water storage tank are provided at grade level. The required storage volume is determined primarily by the need to store nearly 6,000 gallons of water when the pool level is lowered during refueling.

3.2.1.5. Data Summary

Initial operation would be at 5 Mw or less, with 8-hour daily operation. The equipment provided initially would be adequate for 5 mw, with both provision for 8-hour cyclic operation with overnight cooling, and sufficient demineralizer flow for continuous operation.

At 10 Mw, additional equipment and some piping would be added as indicated in the text and in the drawings. All shielding and pipe sizes are designed for 10 Mw.

Reactor data for both 5 and 10 Mw operation are summarized in Table 3.1.

TABLE 3.1

REACTOR DATA SUMMARY

Power	(5 Mw)	(Either)	(10 Mw)
Initial Power Capability	5 Mw		
Eventual Power with Modifications			10 Mw
Average Power Density in Annular Core	155 Kw/liter		310 Kw/liter
Region-Averaged Thermal Neutron Flux			
In Test Hole with H_2O	2.26×10^{14}		4.51×10^{14}
In Test Hole with Na*	1.89×10^{14}		3.78×10^{14}
In Core	3.03×10^{13}		5.56×10^{13}
Reactor Core			
Geometry		Annular	
Inner Fuel Radius		2.77 in.	
Outer Fuel Radius		5.81 in.	
Active Fuel Height		24 in.	

* Na is chosen only to represent a typical absorbing material.

Reactor Core (Cont.)	(5 Mw)	(Either)	(10 Mw)
Volume		32.227 liters	
Fuel Loading	~ 5.5 Kg U		34 w/o
Cladding		Al	
Number of Elements		8	
Fuel plates/element		24	
Gap Thickness		.080 in.	
Fuel Plate Thickness		.050 in.	
Meat Thickness		.020 in.	
Clad Thickness		.015 in.	
Reflector			
Inner Reflector		BeO canned in Al	
Overall Thickness		2.96 in.	
Outer Reflector		Graphite canned in Al	
Overall Thickness		9 in.	
Height		36 in.	
Coolant		Pool Water	
Control Rods			
Location		Outside Vessel	
Type		Curved Plate	
Material		Boral	
Clad		Al	
Overall Thickness		3/16"	
Number:			
Shim-Safeties		4	
Regulating		1	
Test Hole			
Inner Diameter		1.5 in.	
Coolant		Pool Water	
H_2O Island			
Thickness		1.575 in.	
Coolant		Reactor Loop Water	

Core Coolant	(5 Mw)	(Either)	(10 Mw)
Total Flow Rate	1800 gpm		3600 gpm
Coolant Pressure:			
at Pressurizer	80.3 psia		80.3 psia
at Core Inlet	78.9 psia		75.0 psia
at Core Exit	73.6 psia		55.7 psia
Inlet Temperature	140°F		140°F
Outlet Temperature	157.4°F		157.9°F
Demineralizer Flow	50 gpm		100 gpm
Loop Volume with Pressurizer	2000 gal		2300 gal
Pool Coolant			
Heat Removal Load	10%		7.5%
Flow Rate	350 gpm		500 gpm
Max. Mixed Pool Temp.	100°F		100°F
Pool Inlet Temperature	97.5°F		97.7°F
Pool Exit Temperature	107.2°F		108.0°F
Demineralizer Flow	50 gpm		100 gpm
Pool Volume	20,700 gal		20,700 gal
Holdup Tank Volume	1350 gal		1350 gal
Heat Transfer - Reactor Loop			
Heat Transfer Area		190 ft^2	
Heat Flux, $Btu/hr\text{-}ft^2$			
Average	0.84×10^5		1.68×10^5
Maximum	2.54×10^5		5.08×10^5
Radial Max/ave.		1.84	
Axial Max/ave. (assumed)		1.64	
Hot Channel Factors			
Bulk Temperature		1.275	
Film Drop		1.350	
Max. Wall Temp.	267°F		270°F

	(5 Mw)	(Either)	(10 Mw)
Pressure Drop:			
Fuel Element	5.3 psi		19.3 psi
Reactor Loop	~ 11		~ 40
Control			
Reactivity Requirements, $\Delta k/k$			
400 MWD Fuel Burnup + Fission Products			0.021
Xenon and Samarium			0.054
Temperature Effect			0.004
Minimum Reactivity Requirement			0.079
Available for Experiments			0.038
Cold Clean Reactivity	0.110*		0.117
Rod Worth with Complete Shroud of Thermal Poison, Reactor Hot Clean, $\Delta k/k$			0.1748
Core Lifetime	400 MwD		400 MwD
Average Fuel Burnup	10.34%		9.47%
Experimental Facilities:			

1 1.5 in. D. center test hole
4 Radial beam ports
2 Tangential beam ports
1 Shield plug facility
4 1.5 in. I.D. pneumatic facilities
11 Reflector irradiation baskets
1 Gamma exposure facility

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* See Section 3.3.5.

3.2.2. Reactor

3.2.2.1. Reactor Vessel

The reactor vessel is 9 feet in overall length and is fabricated of 6061 aluminum. For most of its length it is 12.34 in. O.D. with 1/4-inch wall thickness (see Figures 3.5 and 3.6).

The center-test-hole wall consists of a 1 1/2 in. D. extra strong aluminum pipe which extends through nearly the full length of the vessel and conveys pool water for cooling of samples within the test hole. The pipe makes a 90° bend near the bottom of the vessel and is welded to the vessel wall where it penetrates the vessel and enters the lower pool plenum. It penetrates the vessel lid through a seal held in place by an insert which threads into the vessel lid.

The design pressure of only 50 psi over pool pressure permits use of a simple clamped or threaded closure for the vessel. A threaded aluminum lid similar to a pipe plug is indicated in Figure 3.5. The lid screws into a stainless steel transition piece, which in turn is threaded and pinned to the aluminum vessel wall. Deterioration of the threads should require replacement of only the lid.

The vessel wall is penetrated by two 6 in. aluminum inlet lines at the top and the vessel necks down to match a 10 in. flange on the exit line at the bottom.

The vessel is supported by this flange, by the reflector grid plate which is anchored to the pool floor by the walls of the lower plenum (see Figure 3.1), and by additional supports on the pool floor if these are found necessary.

The region between the test-hole wall and the vessel wall is separated into two regions, the core and the island, by the island wall. A perforated annular plate welded between the lower end of the island wall and the test-hole wall serves as a flow restriction for the island. Thin spacing ribs are welded between the two walls at the top of the island.

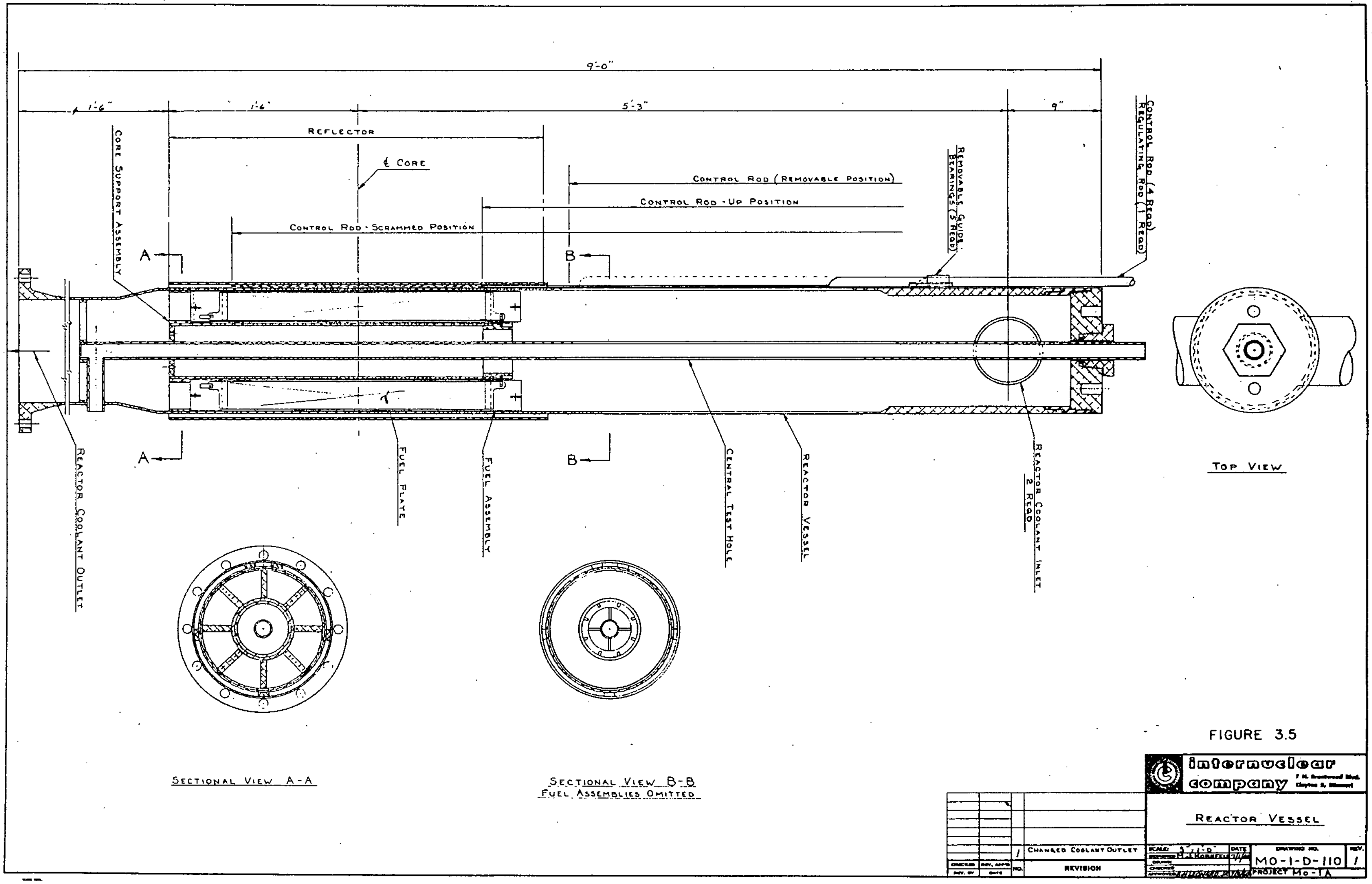
9'-0"
1'-6"
1'-6"
5'-3"
9"
REFLECTOR
℄ CORE
CONTROL ROD (REMOVABLE POSITION)
CONTROL ROD - UP POSITION
CONTROL ROD - SCRAMMED POSITION
CORE SUPPORT ASSEMBLY
REMOVABLE GUIDE BEARINGS (3 REQD)
CONTROL ROD (4 REQD)
REGULATING ROD (1 REQD)
A
A
B
B
REACTOR COOLANT OUTLET
FUEL PLATE
FUEL ASSEMBLY
CENTRAL TEST HOLE
REACTOR VESSEL
REACTOR COOLANT INLET 2 REQD
TOP VIEW
SECTIONAL VIEW A-A
SECTIONAL VIEW B-B
FUEL ASSEMBLIES OMITTED
internuclear company
REACTOR VESSEL
CHANGED COOLANT OUTLET
REVISION
MO-1-D-110
PROJECT MO-1A

FIGURE 3.5

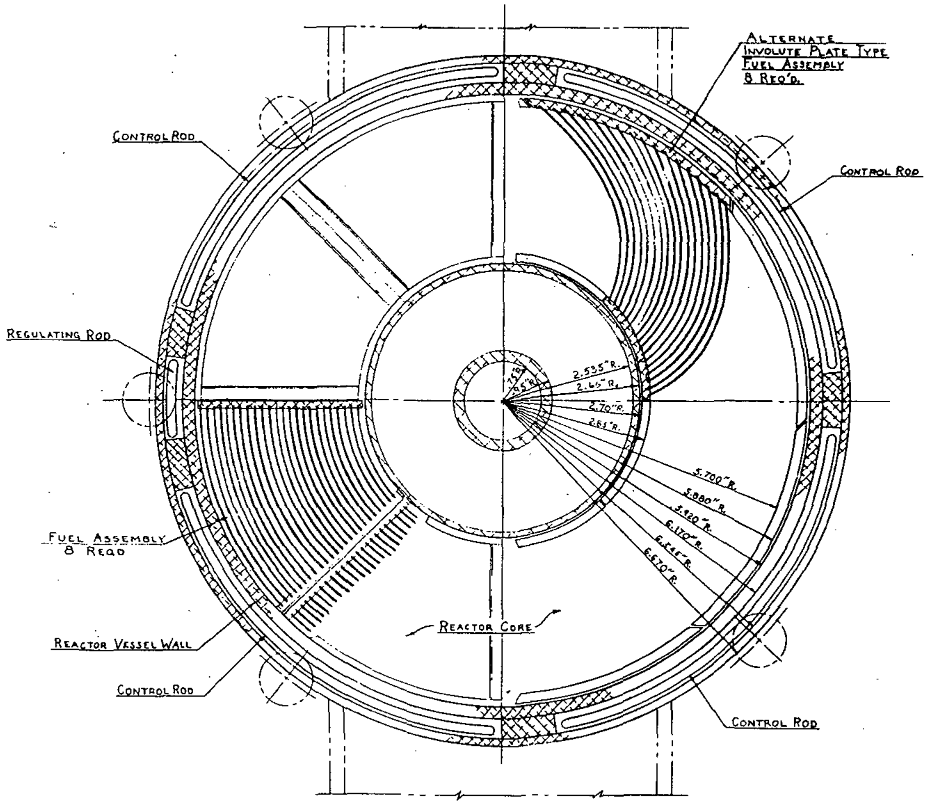

FIGURE 3.6

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REACTOR CORE

MID-PLANE CROSS SECTION

NO.	REVISION
1	CHANGED FUEL ASSEMBLY

SCALE: FULL SIZE

DESIGNED: H. J. KORNFELD

DRAWING NO. MO-1-D-111 REV. 1

PROJECT MO-1A

Fuel elements are supported in the annular core region by a bottom spider support having grooved channels. The elements are aligned by these channels and by positioning slots in the top of the island wall which align with pins protruding from the tops of the elements.

A three foot long aluminum cylinder of 1/8 in. thickness surrounds the vessel to provide control rod slots 3/8 in. thick which are separated by aluminum spacers (see Figure 3.6).

Mounted to the vessel above this shroud are five tapered slots which accept linear bearings mounted on the control rod extensions.

Design and fabrication of the vessel will conform to Section VIII of the ASME Boiler and Pressure Vessel Code and appropriate nuclear case interpretations.

Radiation heating at the inner face of the vessel wall is estimated at 15.6 watts/cc based on a 1.64 axial max/ave. power density. This would cause a maximum temperature rise of only $1^{o}F$ within the wall if the faces were at equal temperature.

A larger temperature difference exists because the water within the vessel is about 40^{o} warmer. With the velocities anticipated along the wall at 10 Mw, the aluminum resistance accounts for only 5.7^{o} of the 40^{o} drop. A thermal tensile stress of about 634 psi is indicated at the inner face, or a total tensile stress of $\sim$ 1900 psi, including hoop tension. At reduced velocities, the film drops will account for a greater fraction of the total temperature drop and the thermal stress will be less.

3.2.2.2. Fuel Elments

Eight fuel elements comprise the annular core (see Figures 3.6 and 3.7). Each element has 24 curved fuel plates of varying circumferential length which span the distance between the radial side plates of the element.

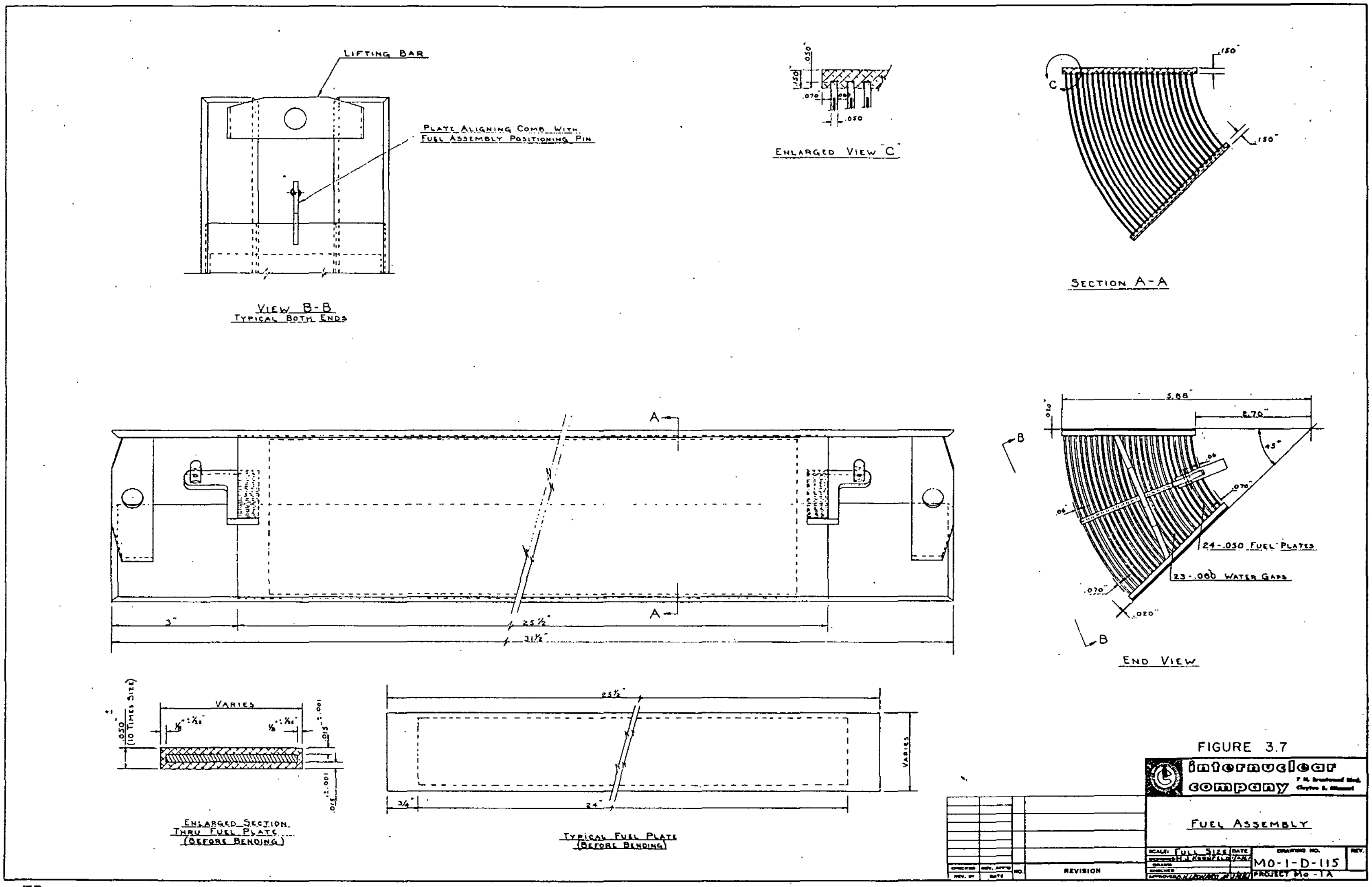

LIFTING BAR
PLATE ALIGNING COMB WITH FUEL ASSEMBLY POSITIONING PIN
VIEW B-B
TYPICAL BOTH ENDS
ENLARGED VIEW "C"
SECTION A-A
A
A
3"
25 1/2"
31 1/2"
5.88"
2.70"
45°
24-.050 FUEL PLATES
25-.060 WATER GAPS
END VIEW
VARIES
.050" (10 TIMES SIZE)
ENLARGED SECTION THRU FUEL PLATE (BEFORE BENDING)
25 1/2"
24"
3/4"
VARIES
TYPICAL FUEL PLATE (BEFORE BENDING)
internuclear company
FUEL ASSEMBLY
SCALE: FULL SIZE
REVISION
MO-1-D-115
PROJECT MO-1A

FIGURE 3.7

The maximum curvilinear span of the fuel plates is 4.22 in., slightly greater than that of the elements for the Advanced Test Reactor (ATR), which are intended for much higher fluid velocities. The similarity in all dimensions other than length suggests the possibility of using shortened ATR elements directly by making the necessary dimensional changes. This possibility will be explored previous to final design calculations.

The fuel plates consist of a 20 mil meat section of U-Al alloy (34 w/o U with 6 kg), clad on both sides with 15 mils of aluminum. The water gaps are 80 mils thick and a 110-mil water gap is left beyond the inner and outer plates. The active fuel length is 24 inches; the overall element length is 31 inches.

The fuel elements would be supported at the bottom by a spider having radial ribs which span the fuel annulus, and which are grooved to form channels for the side plates of the elements. The elements would be held down only by the flow. They would be aligned at the top by pins which rest in slots in the island wall. The pins would be toggled to the comb of the fuel element so as to point in either of two directions, downward when the pin is at the bottom of the element or inward when it is at the top (see Figure 3.5). By this means, fuel elements could be inverted individually.

All nuclear and thermal calculations have been based on an alternate fuel element which has essentially the same thermal and nuclear characteristics. The alternate fuel element is indicated in Figures 3.6 and 3.8. It has plate curvature conforming to the involute of the circle describing the inner boundary of the fuel annulus.

The involute element would be supported and aligned differently because the side-plates of the element are circumferential and do not span the annulus. They would be supported at the bottom by shoulders on the vessel wall and on the island wall. The shoulder on the vessel wall would be notched in 16 places to accept the 8 positioning vees on either end of the elements. To invert the elements (all 8 at once

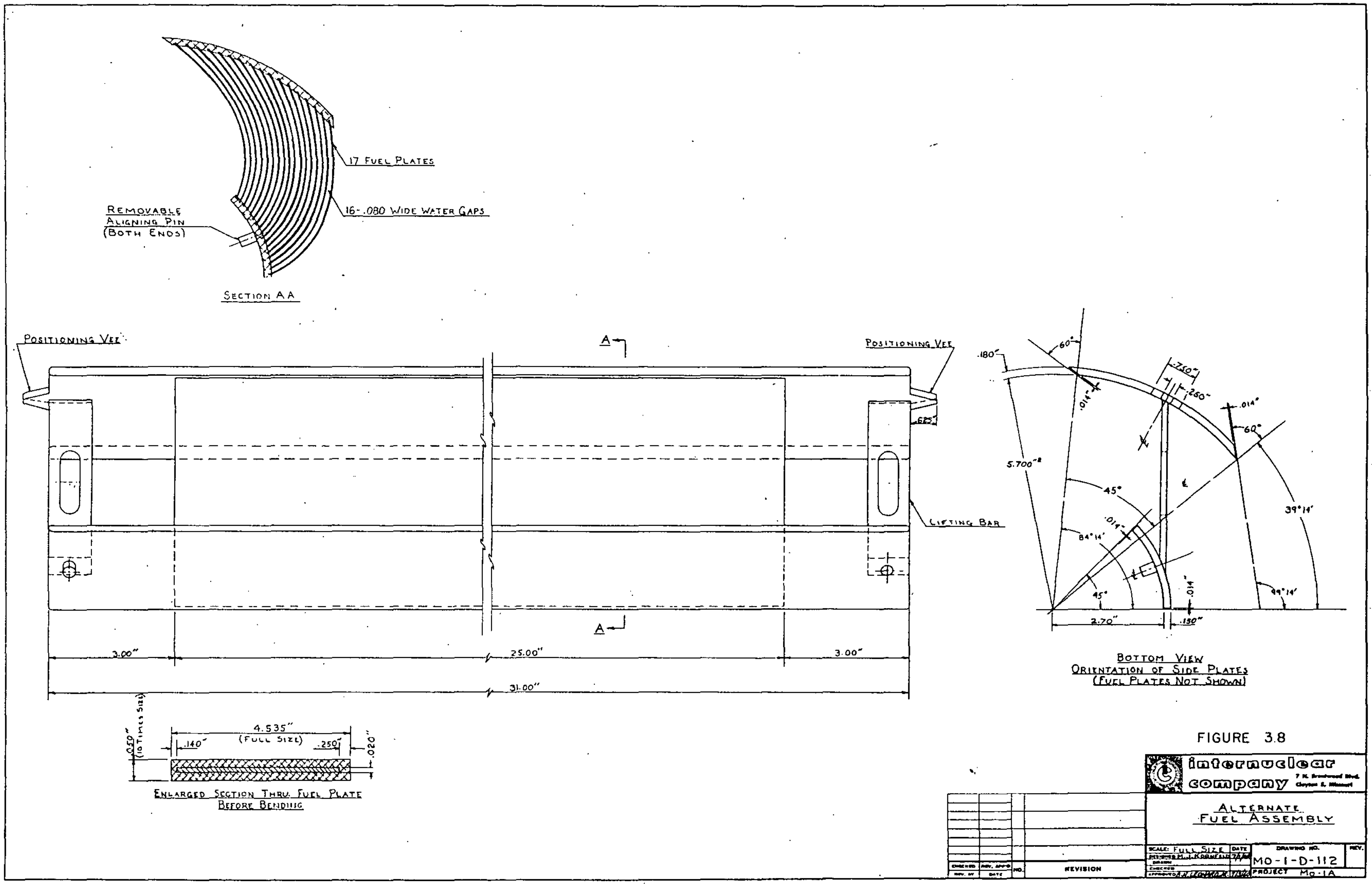

17 FUEL PLATES
16-.080 WIDE WATER GAPS
REMOVABLE ALIGNING PIN (BOTH ENDS)
SECTION A A
POSITIONING VEE
POSITIONING VEE
LIFTING BAR
3.00"
25.00"
3.00"
31.00"
4.535" (FULL SIZE)
.140"
.250"
.020"
.050" (10 TIMES SIZE)
ENLARGED SECTION THRU FUEL PLATE BEFORE BENDING
BOTTOM VIEW
ORIENTATION OF SIDE PLATES
(FUEL PLATES NOT SHOWN)
5.700"
2.70"
.150"
45°
60°
39°14'
internuclear company
ALTERNATE FUEL ASSEMBLY
SCALE: FULL SIZE
DRAWING NO. MO-1-D-112
PROJECT MO-1A
REVISION

FIGURE 3.8

because the plates then curve the opposite way), a removable alignment pin at the top of the element would be transferred to the other end, and the vees would then match the second set of notches on the vessel-wall shoulder.

The circumferential plate assembly is presently preferred, primarily because of its greater rigidity. It should be regarded as the reference design for purposes of preliminary hazards evaluation.

3.2.2.3. Reflector and Thermal Column Extension

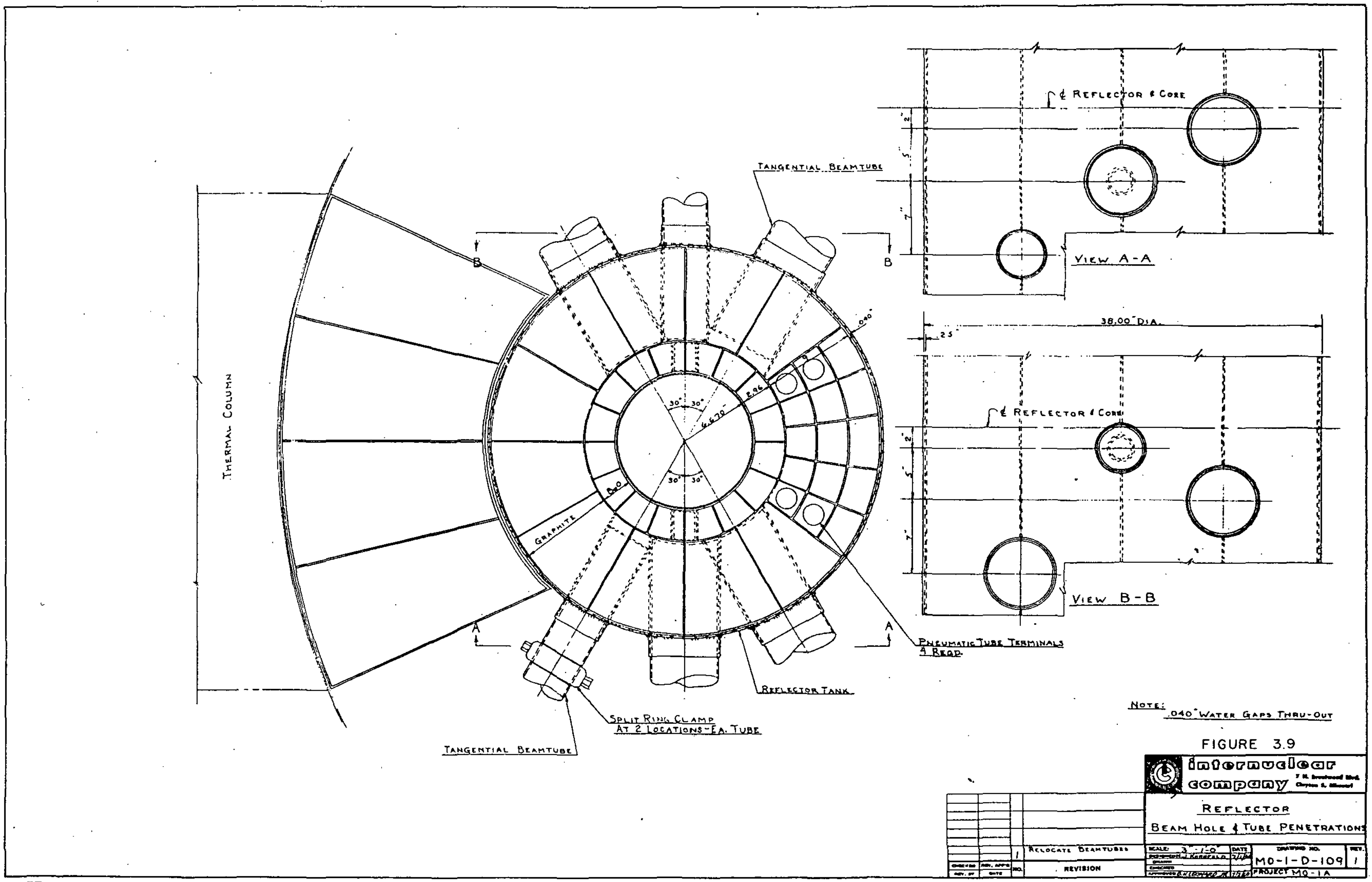
The reference reflector consists of a layer of beryllia approximately 3 in. thick, followed by a 9-in. layer of graphite (see Figure 3.9). Either BeO or Be metal is desirable for the inner layer because of the better reflecting properties of Be and also because the fast neutron dose at this position would probably preclude the use of graphite. All reflector pieces are 3 ft long and are canned with aluminum.

The reflector elements are supported by a grid which is welded between the vessel and the walls of the lower plenum. The plenum walls are extended to form a reflector retaining wall which prevents pool water from entering the reflector by crossflow. Design of the grid and the lower ends of the reflector elements would be performed in a subsequent design phase.

Beryllium Oxide

Nuclear calculations have been based on 3 in. of BeO. The heat production in the BeO at 10 Mw operation and peak axial power is estimated to be $12.6\ e^{-0.2r}$ watts/cc, where r is the radial distance outward from the inner face in centimeters.

Calculation of the temperature distribution in the BeO and the tensile stress at its inner face now indicates that the tensile strength of BeO will be exceeded if its thermal conductivity is reduced appreciably with irradiation. Further attention will be given to this problem, to determine the need for thinner elements or the desirability of using Be metal.



THERMAL COLUMN
TANGENTIAL BEAMTUBE
GRAPHITE
SPLIT RING CLAMP AT 2 LOCATIONS-EA. TUBE
TANGENTIAL BEAMTUBE
REFLECTOR TANK
PNEUMATIC TUBE TERMINALS 4 REQD.
.040
6.70
8.96
B
B
A
A
30°
℄ REFLECTOR & CORE
VIEW A-A
38.00" DIA.
VIEW B-B
NOTE: .040" WATER GAPS THRU-OUT
internuclear company
REFLECTOR
BEAM HOLE & TUBE PENETRATIONS
SCALE 3"=1'-0"
DRAWING NO. M0-1-D-109
REV. 1
PROJECT M0-1A
RELOCATE BEAMTUBES
REVISION

FIGURE 3.9

Four of the BeO elements are shaped to envelop two beam-tube extensions (see Figure 3.10). The extensions are empty Al cans which are secured by two elements strapped together. The curved reflector faces adjacent to the can walls are cooled by pool water flowing between the elements.

Graphite Reflector

Heat production in the graphite is less than that in the BeO by a factor of 4.6, because of the attenuation in BeO. The graphite elements indicated in Figure 3.9 are cooled adequately. Radiation damage effects in the graphite and the possible need for its replacement during plant life will be examined previous to final design.

Several of these elements must be shaped to accept beam tube extensions. The extensions are inserted through the reflector retaining wall and between the reflector elements afterthe graphite elements are in place (see Figure 3.11).

The 15 smaller graphite elements in the irradiation area are removable and replaceable with special elements containing samples or experiments. Four of the special elements would contain terminals for the pneumatic tubes.

Thermal Column Extension

The thermal column extension consists of graphite elements, clad with aluminum and cooled by natural convection of pool water.

3.2.2.4. Control Rods and Drives

The control rods form nearly a complete shroud around the reactor core, They travel in slots between the pressure vessel and the reflector and are cooled by pool water. Thus, the pressurized reactor is rod-controlled without the need to provide pressure seals on the rod drives or extensions.

The rods are curved blades of Boral, clad with Al. The four wide blades are shim-safeties

FIGURE. 3.10

<u>Typical BeO Reflector Elements with 2" Dry Beamtube Extension</u>

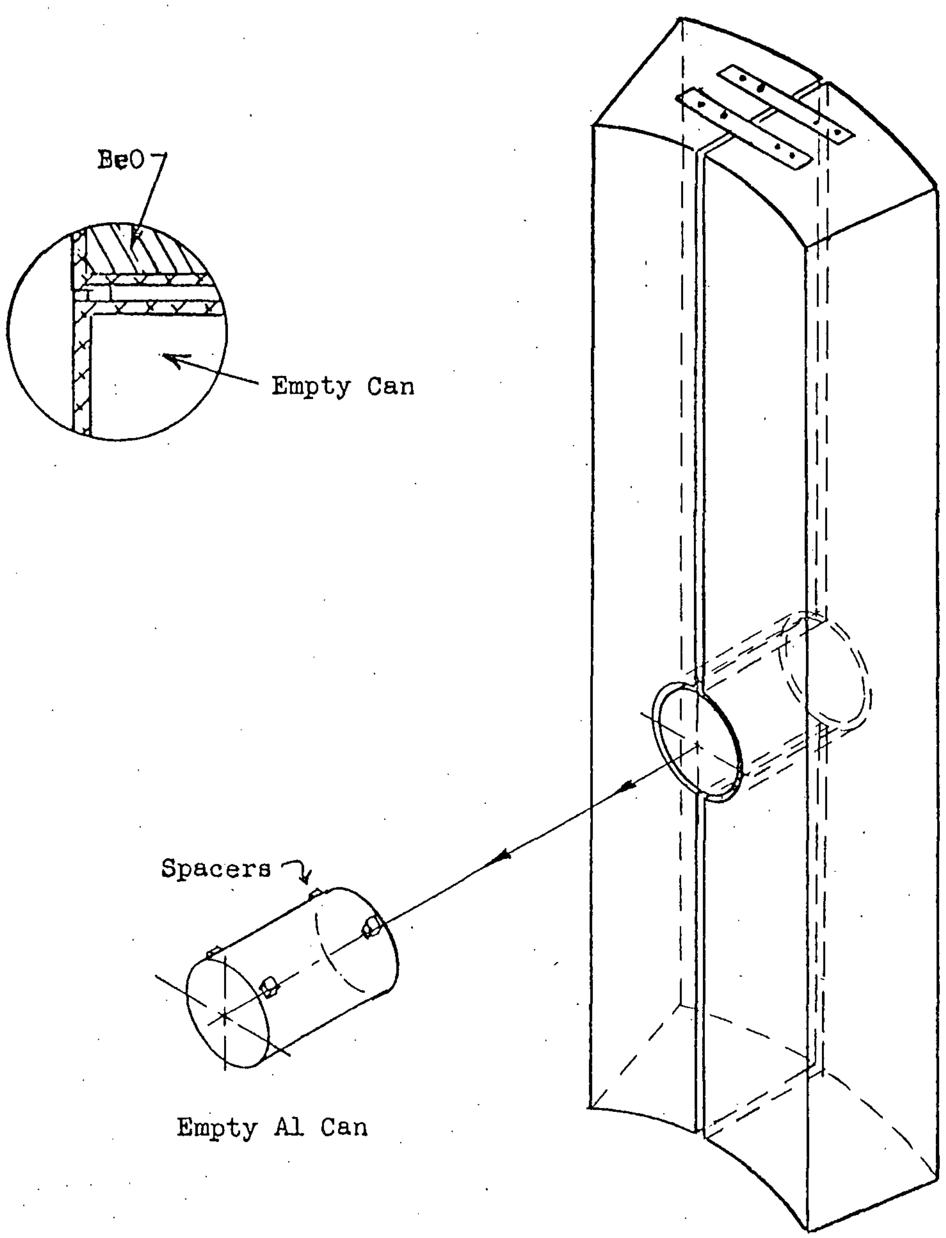
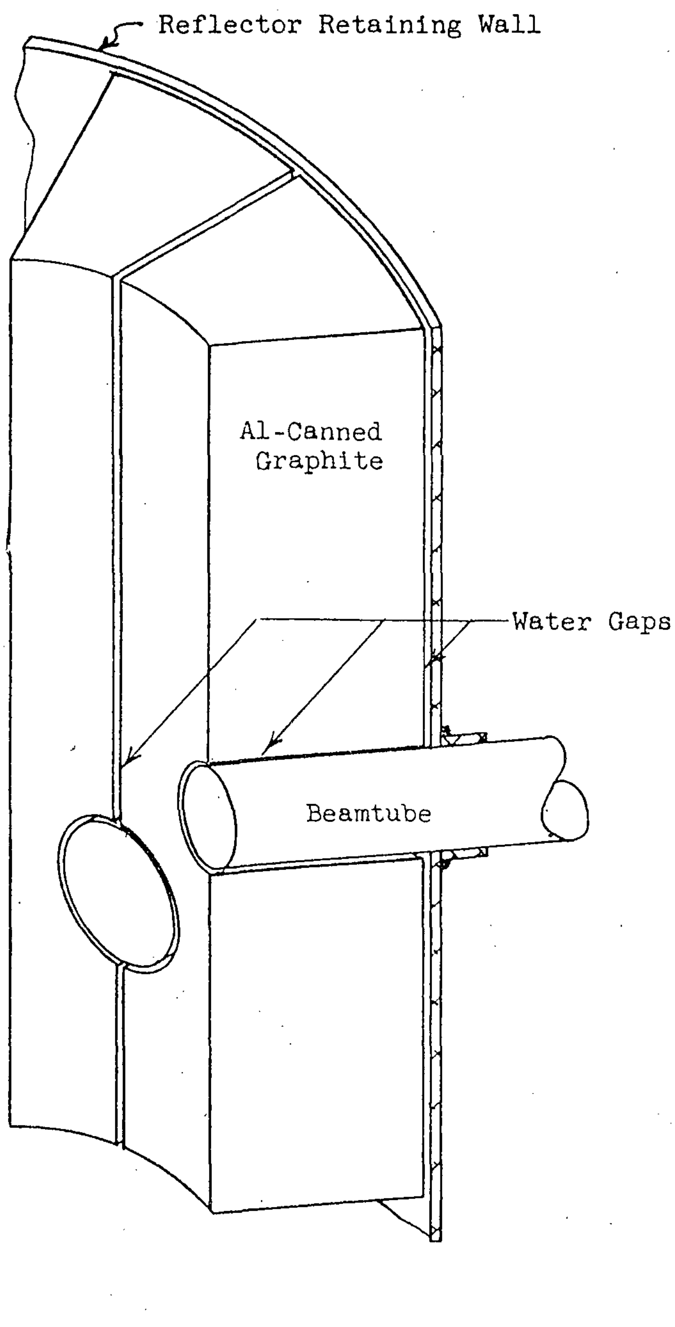


FIGURE 3.11

Typical Graphite Reflector Elements
with 4" and 6" Floodable Beamtube Extensions



and the smaller blade is the regulating rod which is controlled either manually or by the flux servo.

The control rods are guided at two points during their normal travel. The blades are guided by the slots between the pressure vessel and the reflector, and the lower rod extensions are guided by linear ball bearings. The bearings are mounted in a wedge-shaped assembly and the whole unit encircles the lower rod extension and is free to slide on it above a lower limit stop on the rod. The bearing assembly seats in an open tapered receptacle mounted to the pressure vessel.

At full rod withdrawal, the blade is still partly inserted in its slot and the bearing remains in its receptacle. Removal of the rod requires that it be withdrawn past the travel limit of the actuator until the blade clears the slot and the bearing is pushed clear of its receptacle by the lower limit stop on the extension rod. The rod can then be moved outward through the opening of the receptacle and the entire control rod assembly can be maneuvered from beneath the inlet coolant lines.

The bearings will be designed so that they will remain seated in their receptacles during all normal rod movement and they will be pinned or latched if this is found necessary.

Adequate stops will be provided at the bottom of the control-rod slots to prevent loss of the rods when they are disconnected or in the event that they are somehow driven past their lower travel limit.

Assurance of reliability will require that particular attention be directed during final design to the ways in which a rod may possibly hang up. The possibility of a rod freezing by straightening within its curved slot is of particular concern, and consideration will be given to rods which are made flexible in cross-section, by means of vertical hinges, for example.

A mechanical disconnect is required in the extension rod at a point which is near the upper end of the vessel when the rods are fully inserted. After disconnecting the rods, the drives and extension rods are moved and parked over the thermal column extension by means of the traveling bridge. The reactor is then accessible so that the lid can be removed and refueling operations can be performed.

The design anticipates that the release magnets and the shock absorbers on the shim-safety rods will be located on the traveling bridge. At least one commercially-available actuator provides both functions within the actuator, and this unit is indicated on the drawings. Alternate actuators, which do not provide shock absorbers or, in some cases, neither shocks nor magnets, might be used by coupling them to guide tubes suspended beneath the actuators on the traveling bridge, and providing the necessary magnets or shock absorbers within the guide tubes. The regulating rod requires no magnet or shock absorber, and has only a pinned coupling to the actuator, so alternate regulating-rod drives would be acceptable for this purpose.

Electrical leads to the actuators would be conveyed from the pool wall to the bridge by flexible cables, and from the pool wall to the control room by a covered trench recessed in the top of the pool shield.

The traveling bridge will be securely locked in place on fixed alignment equipment during operation, and interlocked by a circuit on the locking device such that magnet current will be interrupted and the rods will drop if the lock is released.

Retracting the rod extensions to facilitate moving the bridge and rod extensions requires that the magnets first be raised individually, and this can be done only if the magnet currents are off. Each extension rod is then released from its fully inserted control rod at the submerged disconnect mechanism and raised several inches and pinned clear of the vessel and control rod. Visual observation that each extension rod is disconnected before withdrawal is thought to be adequate because the extension rods will be raised one at a time for only a short distance, and the rod worth will be such that one rod can be completely withdrawn without causing criticality.

The possible need for additional safeguards will be reviewed during the design and such features will be incorporated wherever they contribute significantly to the safety of operation.

3.2.3. Coolant System

All structural materials in the pool, in the reactor-coolant loop and in the pool-coolant loop will be of aluminum or stainless steel. Standard components will be used throughout the cooling systems to ensure reliability of operation.

3.2.3.1. Reactor Coolant System

The reactor coolant flows downward through the core and the island. A flow-restricting plate with small apertures is located at the base of the island and causes about 4% of the total flow to be bypassed through the island.

The 10-in. inlet line has a check valve at the outer face of the pool wall. The inlet flow is split between two 8-in. lines within the pool which are reduced to 6 in. at the vessel inlets to facilitate control rod clearances on the small diameter vessel.

The single 10-in. exit line has an invert loop buried in the concrete pool wall. The invert loop extends over the shield-plug facility or thermal column, as indicated in Figure 3.1. A minimum thickness of 3 ft of barytes concrete envelopes the loop to reduce the N^{16} dose rate at the shield face to tenth-tolerance. Up to 2 in. of lead may be required on one side of the pipe near core elevation to compensate for the concrete displaced by the pipe.

A vacuum-breaker valve connected to the top of the invert provides a siphon break and ensures retention of water above the core in the event of an external loop rupture. A manual stop valve is provided on the exit line at the outer pool wall to isolate the reactor loop during unattended periods.

A convective loop and fin-tube heat exchanger are provided within the pool and in parallel with the core. A spring-loaded check valve on this loop is held closed against the spring by the design pressure drop across the core. Upon loss of flow, the valve springs open to permit natural circulation through the convective loop.

A bleeder line at the high point of the convective loop allows gases to rise and be collected at the top of the invert loop so they will not bind the convective loop. This bleeder line is always open, causing a small amount of the exit flow to be bypassed through the convective loop during operation.

The initial coolant room equipment includes one pump, two heat exchangers in parallel, a 50 gpm mixed-bed demineralizer, and a pressurizer. Previous to 10-Mw operation, a parallel loop would be added within the coolant equipment room with duplicate heat exchangers and demineralizer, and the pump capacity would be increased.

The flow arrangement is shown in Figure 3.12, where the added equipment necessary for 10 Mw operation is indicated in phantom.

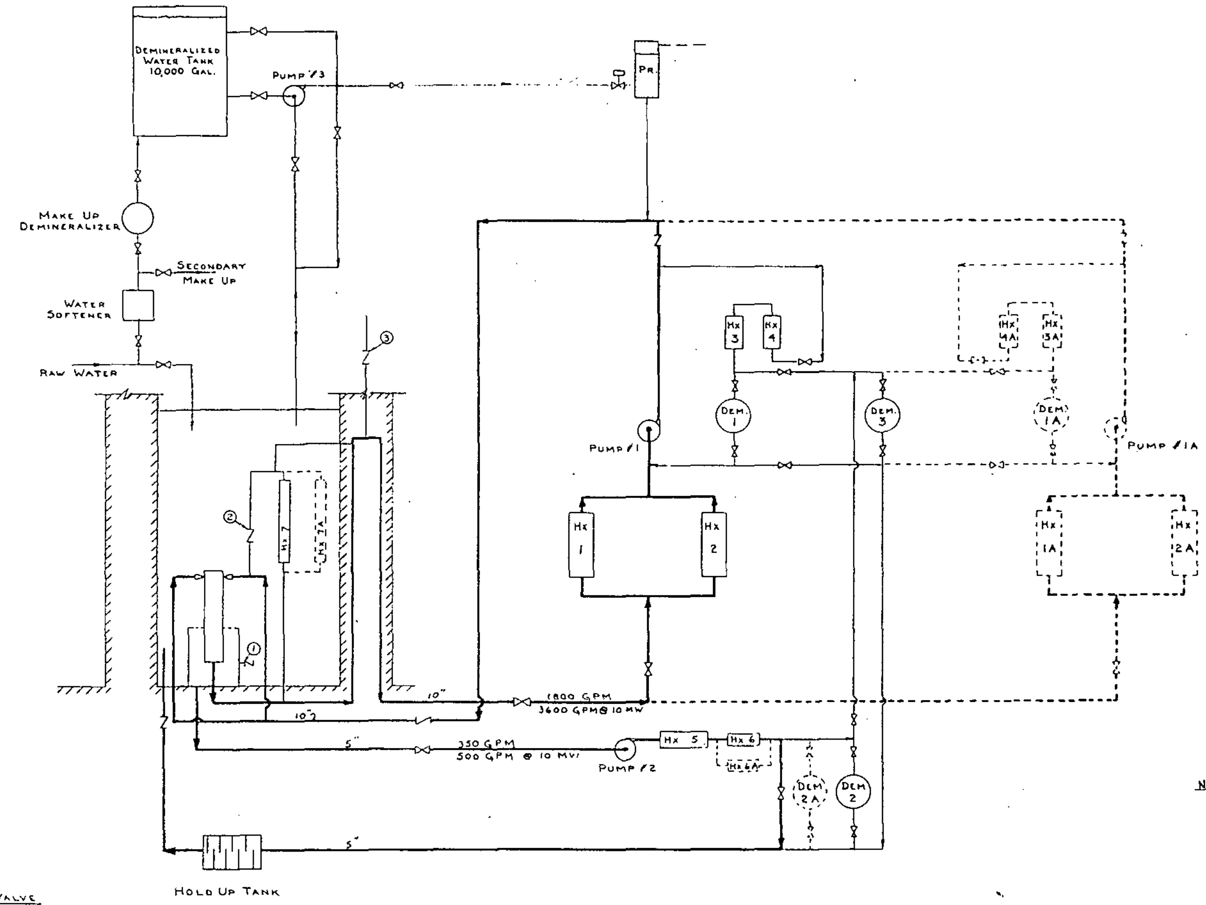
Heat Exchangers

The reactor-loop heat exchangers are all aluminum. The following conditions apply approximately at 5 and at 10 Mw, based upon a total fouling factor of 0.002 and design data for specific heat exchangers.

TABLE 3.2

REACTOR-LOOP HEAT EXCHANGERS

Tubeside inlet	158°F
Tubeside outlet	140°F
Primary Flow	1800 gpm
Shellside Inlet	85°F
Shellside outlet	105°F
Shellside flow	1600 gpm
Required surface per unit	481 ft^2
Number of units	2 at 5 Mw 4 at 10 Mw



Note: Piping and Equipment Shown in Phantom To Be Added For 10 MW Operation.

FIGURE 3.12

Valve Legend.
① Flapper Valve
② Spring Loaded Check Valve
③ Vacuum Breaker Valve

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Flow Sheet
Reactor Coolant System

No.	Revision
1	Redrawn For Single Loop Coolant System

Drawn: H. J. Kornfeld
Approved: E. Leonard
Drawing No. MO-1-D-113 Rev. 1
Project MO-1A

Pumps

Centrifugal constant-speed pumps are proposed, with mechanical seals and controlled leakage. At 10 Mw and 3600 gpm, the total pumping requirements are about 100 hp. The pump arrangement which is most economical with regard to modification for 10-Mw operation will be determined during the detailed design.

Pressurizer

Loop pressure is maintained by compressed air admitted to the pressurizer at 80 psia by a pressure-regulating valve. It is intended that radiolytic gases plus air will be discharged continuously from the pressurizer through a fixed orifice. Liquid level would be controlled automatically between high- and low-level limits.

Subsequent design phases will determine the volumetric requirements and the required disposal facilities for off-gases.

Demineralizers

One 50-gpm demineralizer at 5 Mw and two at 10 Mw should maintain the sodium activity in the loop at a level such that complete mixing with the pool, following shutdown from continuous operation, would increase the surface dose rate by $\sim$ 7 mr/hr.

With stop and check valves closed previous to opening the vessel, the increase in dose rate will be correspondingly less because of the smaller amount of loop water to be mixed. Following 8-hour daily operation the maximum increase in dose rate would be less than half as large as for continuous operation. These considerations, when combined with more detailed analysis of the effective flux causing recoils, may lead to reduction in the loop demineralizer capacity.

Figure 3.12 includes two small heat exchangers in series before each demineralizer to reduce the coolant temperature from 140°F to 100°F, in order to extend the lifetime of the resins.

Piping and Valves

All piping is 6061 aluminum or equivalent, schedule 40, with all welded joints except within the coolant equipment room. All piping and valves shall conform to the ASA B31 Piping Code.

Convective Loop

The convective loop would include initially an aluminum fin-tube heat exchanger mounted on the pool wall. This heat exchanger would be capable of transferring the shutdown heat to the pool following 5-Mw continuous operation, without causing excessive temperatures in the core. For 10-Mw operation, a second unit would be installed in the pool.

The top of the convective loop would be below the siphon break and the heat exchangers would be mounted as high as practical in the loop.

3.2.3.2. Pool-Coolant System

Pool water is drawn through the reflector, the control rod gaps, and the center test hole and mixed in a plenum beneath the reactor. From there it passes under the shield and through a shielded trench to the coolant equipment room. The pool loop includes a pump, heat exchangers, a bypass demineralizer, and a hold-up tank having baffled internal flow paths. The pool discharge line has a manually-operated stop valve, and the pool inlet line has a check valve. A flapper valve is provided on the lower plenum which opens automatically to admit pool water when the pump on the pool loop stops, and this causes the reflector region to be cooled by natural convection.

Heat Exchangers

Calculations for specific heat exchangers indicate the desirability of multiple units in series, to maintain reasonable velocities and heat transfer coefficients with the low available mean temperature difference. Two units in series are proposed initially, with a third added in series-parallel when the system is modified for 10-Mw capability (see Figure 3.12).

The data of Table 3.3 apply to the heat exchangers calculated for the pool loop. All-aluminum units are proposed and these are sized on the basis of 100°F mixed pool water temperature occurring during the hottest days when the design wet-bulb temperature of 78°F is applicable.

TABLE 3.3

POOL-LOOP HEAT EXCHANGERS

	5 Mw	10 Mw
Arrangement	2 in Series	3 in Series-Parallel
Tubeside inlet	107.2°F	108.0°F
Tubeside outlet	97.5°F	97.7°F
Primary flow	350 gpm	500-250 gpm
Shellside inlet	85°F	85°F
Shellside outlet	95°F	95°F
Secondary flow	350 gpm	500-250 gpm
Total fouling factor	0.0015	0.0015
Required surface	435 ft^2	676 ft^2

Demineralizers

One 50 gpm demineralizer at 5 Mw and two at 10 Mw are calculated to reduce the Na^{24} dose rate at the pool surface to about 4 mr/hr during continuous operation. The sodium dose rate will be less than half this value with 8-hour daily operation.

Hold-up Tank

The 29.4 sec O^{19} activity in the mixed flow entering the lower plenum is estimated to be 32,000 dis/cc-sec at 10 Mw and 500-gpm flow. A 1,350-gallon hold-up tank added to the pool loop volume of $\sim$ 350 gallons provides a 3.4-minute total hold-up before discharge back into the pool. This reduces the pool surface dose rate to 5 mr/hr if the return is directly to the surface, and to 0.1 mr/hr if complete mixing occurs within the pool. At 5 Mw and 350-gpm flow, 500 gallons less hold-up would yield nearly the same dose rates.

The 7.4-sec N^{16} gammas and the 4-sec N^{17} neutrons will contribute negligible activity to the pool because of the long decay time necessitated by O^{19}.

Valve on Plenum

The flapper valve would be similar to those which have been installed on open-pool reactors. An adjustable weight would be set so that gravity will open the valve when pump suction is lost. Ideally, this valve would also be closed automatically by pump suction when the pump starts; however, the design may require that the valve be closed by a pull-chain or other means after the pump is started and previous to rod withdrawal.

3.2.3.3. Secondary Coolant System

Two-cell cooling tower units are proposed which will cool 2,400 gpm from 105°F to 85°F with a design wet-bulb temperature of 78°F. One two-cell unit would be installed for initial 5-Mw operation and another would be added at 10 Mw.

All secondary piping will be of aluminum. Make-up water to the secondary system will pass through a water softener. Make-up to the demineralized storage tank will pass through this softener and then through a 10-gpm make-up demineralizer.

3.2.4. Control and Instrumentation

3.2.4.1. Control-Rod Movement

The size and travel of the regulating rod will be such that it controls no more than 0.5% $\Delta k/k$. Its movement will be controlled either manually or by the flux servo. Its rate of movement will be adequate to compensate for Xe burnout at restart and will be limited to a safe value based on the results of the accident analysis. Continuous position indication and upper-and lower-limit switches will be provided, with both limit switches connected to audible and visible alarms.

The four shim-safety rods and their extension rods will be suspended individually from electromagnets at the top of the pool. Manual or automatic scram signals cause interruption of the current to the common bus supplying the magnets and the rods then drop individually into their guide slots until the rod extensions seat in their shock absorbers at the top of the pool. Position indication will be provided for the full range of travel of the magnets. Switches will be provided to indicate whether or not the magnet and the extension rod are in contact and also to indicate whether or not the extension rod has dropped to its lower travel limit.

Except for certain conditions causing powered run-in of the shim rods, they are inserted or withdrawn only manually by means of a single switch. With the magnet current on, the magnet positions are interlocked electrically so that the rods move simultaneously and are always inserted to uniform depth.

The drive motors will operate on alternating current and will be of such a type that the maximum speed is limited by the frequency of the a.c. supplied. The motors will be instantly reversible, and suitable gear trains will be provided so that the maximum rate of combined rod withdrawal does not exceed the safe value determined in the accident analysis.

An alarm will be given and automatic run-in of the magnets will be initiated upon selected signals and also whenever the regulating rod reaches its lower limit. The latter function is important because reactivity is added continuously by Xe burnout during weekday mornings with cyclic operation. The run-in cannot be overridden by manual or servo control; in the case of the regulating rod reaching its limit, the run-in is stopped only by raising the regulating rod from its lower limit. The regulating rod may require such repositioning six times following 10 Mw-startup on weekday mornings.

Under no circumstances will the shim rods be withdrawn automatically. When the regulating rod reaches its upper limit of travel, only an alarm will be given.

Any of the normal modes of operation, manual-shim rods, manual-reg rod, or servo-reg rod can be initiated at any time without disturbing the reactor operation. Initiation of any of these modes will cancel the signals from a previous mode except during scram or rod run-in.

An additional mode will be provided which will permit, for maintenance purposes, raising of individual magnets whenever the magnet currents are off.

3.2.4.2. Safety and Shutdown Signals

A tentative list of conditions which would cause scram or rod run-in is given in Table 3.4.

TABLE 3.4

SAFETY AND SHUTDOWN SIGNALS

		Scram	Prevents Startup	Rod Run-in	By-pass Possible
1.	High neutron level	X			
2.	Short period			X	
3.	Shorter period	X			
4.	Low reactor-loop pressure	X	X		X
5.	Loss of reactor-loop pump	X	X		X
6.	Loss of pool-loop pump	X	X		X
7.	High loop water temp.		X	X	X
8.	Power failure	X	X		

TABLE 3.4 (Cont.)

	Scram	Prevents Startup	Rod Run-in	By-pass Possible
9. Low pool level	X	X		
10. Selected fault conditions in experiments	X	X		X
11. Reg. rod at lower limit			X	
12. Individual magnet raised	X	X		
13. Truck entry open	X	X		
14. Bridge lock released	X	X		
15. Ion-chamber low voltage	X	X		
16. Magnet actuator amplifier output low	X	X		
17. Calibration switches off "operate" (log N-period and power level amplifiers)	X	X		
18. Instrument range switches off lowest scale		X		
19. Shim rods not full in		X		
20. Low count-rate from start-up chamber		X		
21. Start-up counter not full in		X		
22. Manual scram button	X			

A multipoint annunciator system will be provided to indicate fault or unusual conditions in the reactor or coolant system, including all conditions in Table 3.4. Certain conditions such as high temperature in the lower-pool plenum, secondary-loop pump failure, or high radioactivity levels in either loop, in the building, or at the stack cause alarm signals.

Closure of the containment will be initiated automatically by any reactor scram or as a result of sounding either the building or area evacuation alarms, as described in Section 5.9.

3.2.4.3. Nuclear Instrumentation

Seven nuclear instrumentation channels would be provided; two for start-up, two for the period range, and three for level safety and for the servo signal. The channels are shown schematically in Figure 3.13. To promote minimum response time and maximum reliability, substantially solid-state systems are preferred. All circuits will be designed to fail safe.

The neutron detectors will be located on two sets of inclined racks in the pool water at core level (see Figure 3.2). The ion-chamber leads will be conveyed to the pool surface by aluminum tubes clamped to the pool wall, and they will have quick disconnect junctions at the pool surface to facilitate removal of the tubes and chambers. The fission counters for start-up will be removable by means of a drive mechanism and they will require flexible water proof leads.

Low-Level Start-up Channels

The start-up channels measure neutron flux and reactor period from source level to approximately 10^{-5} of full power. No period scram is generated by these channels. Two separate channels are provided, primarily to ensure reliability consistent with the need for daily start-up.

Each channel consists of a fission counter, linear preamplifier, linear amplifier, pulse-height discriminator, logarithmic count rate meter, low-level period circuit, and high-voltage power supply. There will be a scaler included with appropriate

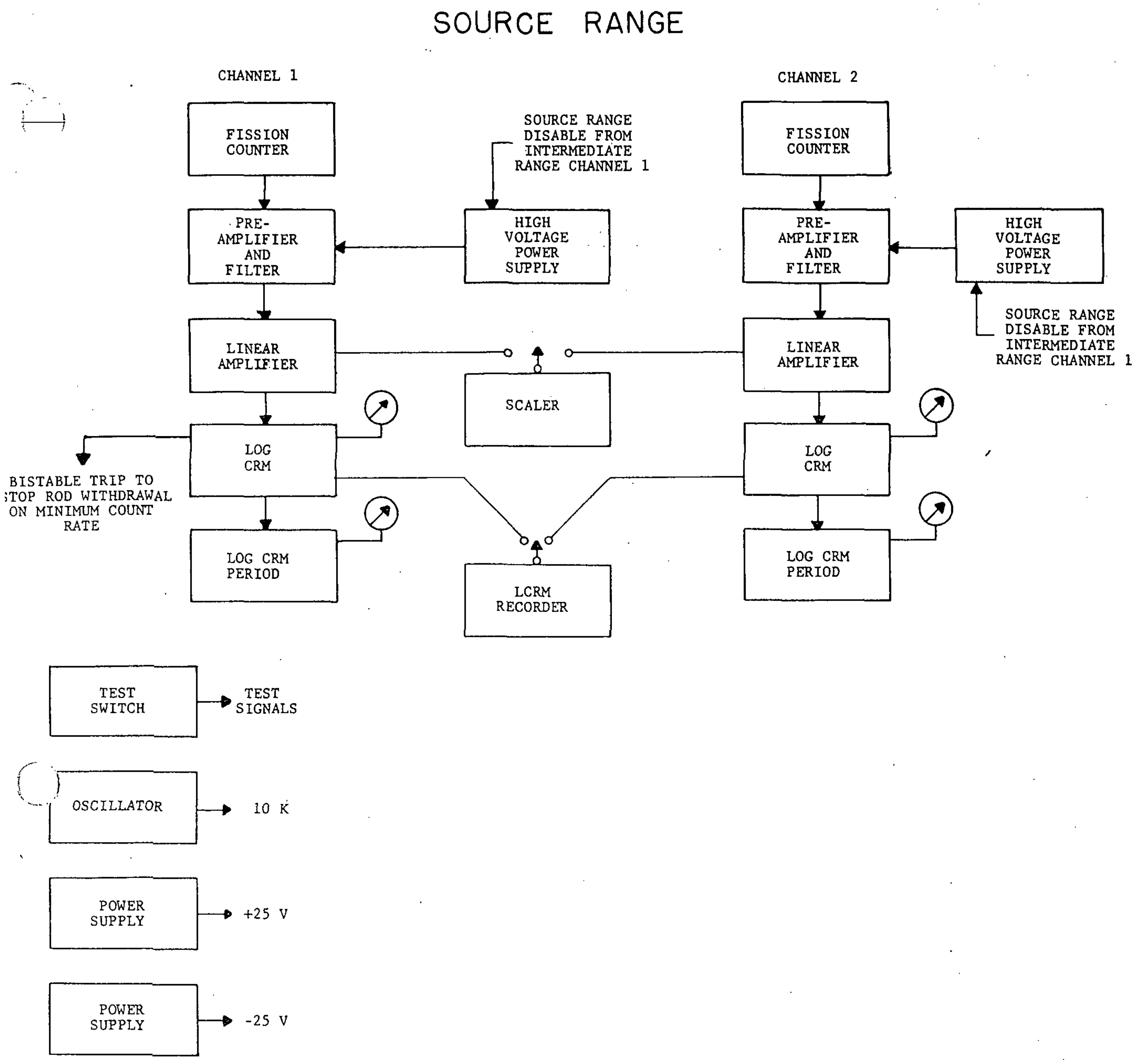

FIGURE 3.13 - NUCLEAR INSTRUMENTATION

INTERMEDIATE RANGE

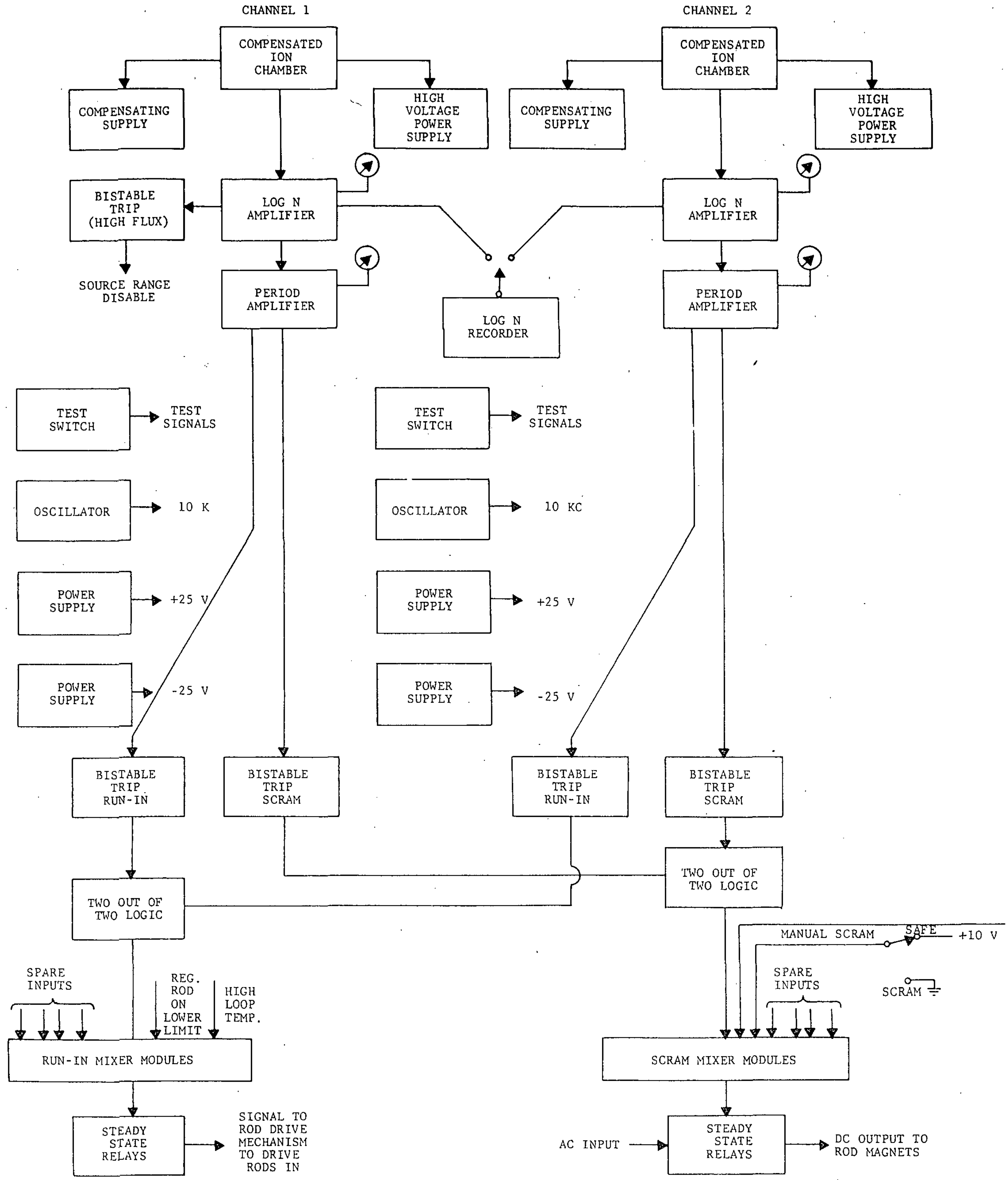

FIGURE 3.13 - NUCLEAR INSTRUMENTATION (Cont.)

POWER RANGE

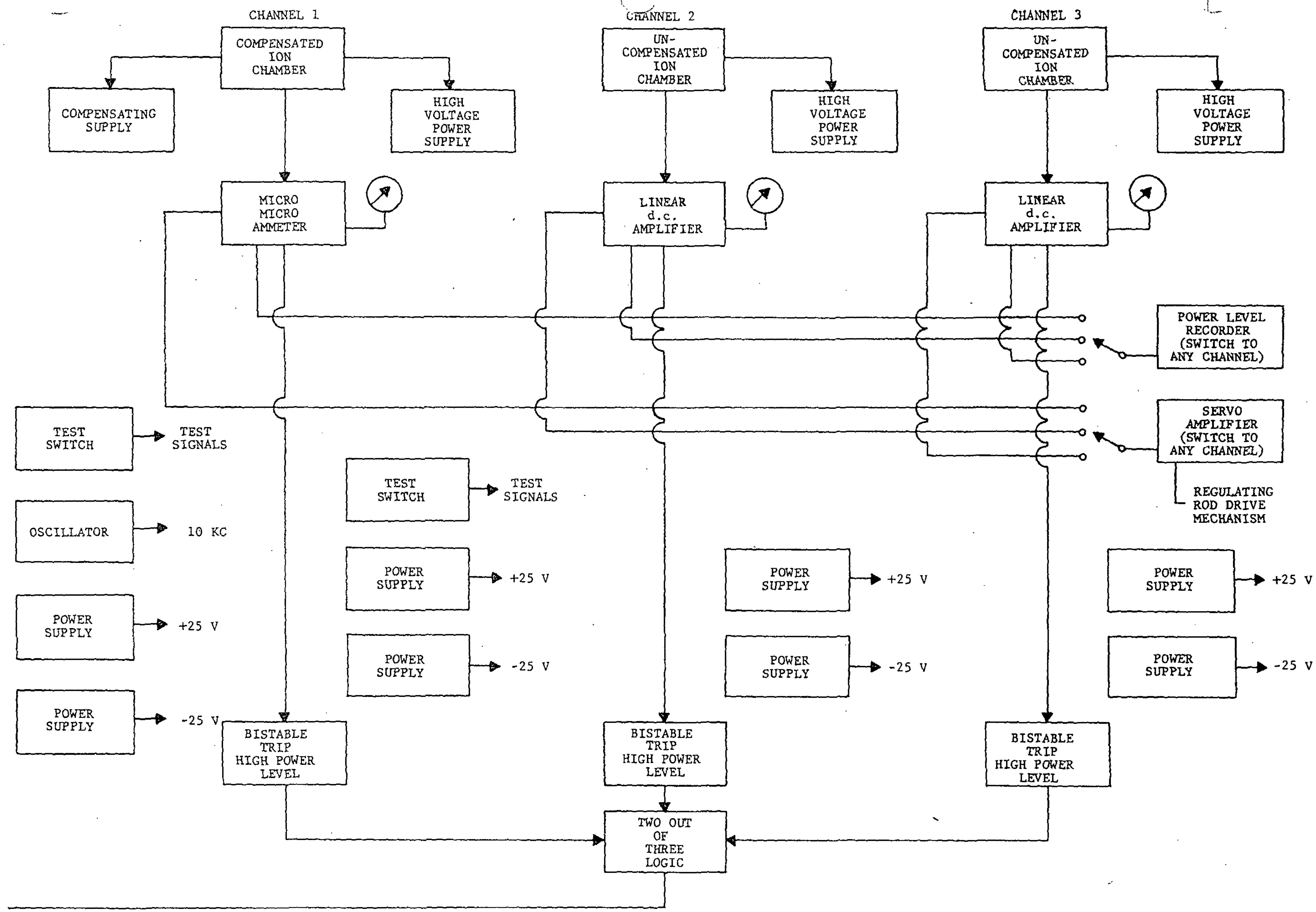

FIGURE 3.13 - NUCLEAR INSTRUMENTATION (Cont.)

switching circuits to permit use with either channel. The scaler will have an adjustable timer with preset time and preset count features. There will be one log count rate recorder which can record the signal from either of the start-up channels.

During full power operation, the counters will be withdrawn from their start-up positions to prevent damage to the counters. The counter positioning devices will have limit switches and lights to indicate positions and interrupt motor currents at three positions: IN, CENTER, AND OUT. Interlock circuits will be provided to prevent rod withdrawal if the count rate is less than one count per second as indicated by the log-count-rate meter.

Log N and Period Channels

The range of the two log N-period channels overlaps the start-up channel range by at least one decade and extends up to full power. These channels provide rod run-in or scram when the reactor period is less than the safe values which will be determined in the accident analysis. The rod run-in and scram signals will require coincident signals from both period circuits before rod movement is initiated. If one system is removed for service, the remaining system will be able to scram the system since the removal of one channel provides one of the two necessary signals.

Each channel consists of a compensated ion chamber, log-N-amplifier, period amplifier, high-voltage power supply, a single log-N recorder, and log N and period remote indicating meters. The recorder and indicators have switching circuits which enable them to be connected to either channel.

Linear Power Level and Safety Channels

Three channels of instrumentation will be provided to measure the neutron flux in the power range and provide scram signals when the neutron flux exceeds a safe level which will be determined in the accident analysis. The high-flux scram signals are connected in 2-out-of-3 coincidence. When one channel is down for service, a signal from either remaining channel causes scram.

One channel consists of a gamma-compensated ion chamber feeding a zero-stabilized micromicro-ammeter. Two channels will consist of un-compensated ion chambers feeding linear d.c. amplifiers. Each channel will have an indicator meter at the console, and one recorder with switching circuits will be used to record the power level from any one of the channels.

Safety Circuits

The safety circuits will consist of bi-stable transistor-controlled units providing logic functions to initiate scram signals upon receipt of proper coincident signals and sharp magnet-holding current cutoff upon receipt of scram signals. The magnet-holding current will be constant at all power levels and high enough to minimize nuisance scrams.

Servo Channel

This channel will consist of a power-schedule unit, a servo amplifier and the servo motor and regulating-rod drive mechanism. It will operate on a signal proportional to power level derived from one of the safety-channel amplifiers.

3.2.4.4. Process Instrumentation

The reactor will have complete coolant-system monitoring and control equipment for the primary loop, the demineralizer loops and the pool-cooling loop. Critical parameters of pressure, temperature, flow, conductivity, etc. will be metered and recorded in the reactor control room. Failures of the cooling system will be interlocked with the reactor control and safety system to automatically shut down the reactor.

3.2.4.5. Activity Monitoring

Activity of the reactor loop, the pool loop, and the secondary coolant loop will be monitored continuously. The reactor-loop monitor will be capable of indicating serious fuel defects. The assembly proposed for this is shown in Figure 5.3.

Radiation levels will be monitored continuously at the stacks. Liquid wastes will be monitored before discharge to a sanitary sewer.

An area-monitoring system will be provided for the reactor building, the coolant equipment room, the laboratory building, and the environs.

All radiation monitoring systems are described in greater detail in Section 5.8.

3.2.4.6. Control Panels and Console

A desk-type control console will be provided, containing all meters, switches and indicating lights necessary for operation of the reactor. Controls and meters will be functionally grouped for ease of operation; e.g., position indicators for the drive mechanisms will be grouped with the rod-control switches and nuclear channel meters will be arranged conveniently with appropriate range-switching equipment. The annunciator and operator scram button will be prominently located in the center of the console.

Nuclear amplifiers, scalers, and recorders will be functionally arranged in a vertical amplifier cabinet located in a position convenient to the operating console. Cooling system equipment, controls, meters and recorders will also be functionally grouped in a complementary cabinet.

3.2.5. Experimental Facilities

3.2.5.1. Center Test Hole

The center test hole is described as part of the vessel. Its use requires the insertion of a special irradiation canister which can be inserted or removed only during reactor shutdown. The canister is described in Section 5.7.2.

3.2.5.2. Beam-Port Facilities

The beam-port arrangement is indicated in Figures 3.2 and 3.9. Four radial and two "tangential" beam ports are provided; four with 6 in.-diameter tubes and two with 4 in.-diameter tubes. The tangential beam ports also approach the vessel radially but they terminate within the graphite reflector below the core and do not view the reactor core directly.

All radial tubes terminate at the inner face of the graphite, except for one 4-in. and one 6-in. tube, which have extensions in the form of empty aluminum cans penetrating the BeO layer. The can diameters need be only as large as the anticipated collimator, and it is intended that they be no greater than 2 in. in diameter.

The presence of these cans has not yet been included in reactor analysis, but it is expected that, with this size restriction, the effect on reactivity and power distributions will not be of great importance.

The beam tubes are welded to the Al pool liner at the point where they enter the pool. At the reactor, they are loosely fitted first through a sleeve which penetrates the reflector retainer wall, and then through the shaped holes in the graphite, so that they are free to expand into the coolant gap surrounding the vessel (see Figure 3.9). Two flanges are provided in the pool water, the purpose of the second flange being to permit withdrawal of the last tube section to unkey the graphite. A split-ring clamp is indicated on the drawing, the design of which is yet undetermined. Ideally, the clamp should be easily removable and replaceable under water.

It is expected that a standard design of a beam-port facility will be available from the selected manufacturer which will satisfy the design requirements. A general description of the proposed facility follows.

In the shield wall the tube diameter is stepped to reduce radiation streaming around collimators and plugs. The tubes terminate at the shield face in a lined recess provided with an air-exhaust vent. The recess is backed with lead to compensate for the reduction in concrete thickness. Movable lead gamma shutters are not indicated on the drawings but they would be provided on all beam ports. The need for shutters on the tangential beam ports will be determined later.

When the beam tube is not in use, a barytes-concrete plug is inserted in the tube and the tube is sealed with a cover plate to permit flooding of the empty portion of the tube to complete the shield. The tube would be flooded with demineralized water through a helical coil in the plug, and drained through a line within the shield. Drain and fill valves would be recessed in the shield wall between beam ports.

Beam-port utilities include demineralized water, cold tap water, compressed air, electricity, an off-gas exhaust duct, and leads to the control room. All utilities would be manifolded along the ledge in the shield wall above the beam ports, and outlets would be supplied between the beam ports.

At 10 Mw, it appears necessary to cool and circulate the water within the beam ports. The drain and fill circuit would then be left open to flooded ports and it would include a heat exchanger and pump.

3.2.5.3. Shield-Plug Facility

The "shield-plug facility" is intended for multiple usage: as a thermal column, as a medical irradiation facility, and as a location for shielded packages of experimental apparatus.

The present concept is a split facility with graphite segments within the pool and an Al-lined cavity in the pool wall which terminates at the Al pool liner.

Used as a thermal column, the cavity would be filled with a 4-ft cube of graphite, backed with a boral curtain and concrete shielding blocks. Certain alternate usages would involve replacement of two graphite segments within the pool and two graphite reflector pieces with similar units which envelop and hold empty Al cans, the cans serving essentially as beam port extensions. Shielded experimental packages or equipment for medical exposure would be inserted in the cavity according to experimental requirements.

Subsequent design work will include an evaluation of the advantage of flexibility presented by the split facility vs. the neutron losses resulting from the water gaps in the extension pieces. Depending on the nature of the anticipated experiments, it may be found preferable to design the facility as an integral Al-cased unit extending outward into the pool. A lead block cooled by pool water would shadow the hot end of the facility, eliminating need for internal cooling.

Pending the results of this evaluation, the split design is proposed, as indicated on the drawings. Initially, solid graphite segments would be used in the pool, and the cavity would be filled with barytes-concrete blocks. In this condition, the maximum 10-Mw heat production at the inner concrete block face is calculated to be 2.5×10^{-4} watts/cc, causing only a $1.5^{o}F$ temperature peak to occur at about one foot depth in the concrete. With the graphite segments removed and replaced with pool water, these values increase to a maximum heat production of .011 watts/cc and a temperature rise of $65^{o}F$.

Since the graphite segments will remain in place, the only need for cooling results from radiation down the beam port. Appropriate cooling equipment would be incorporated in experimental packages which require the use of the beam port.

3.2.5.4. Pneumatic Facilities

Four pneumatic facilities enter the pool below the water surface but above the refueling level and arc down through the pool to the irradiation area of the graphite reflector. They terminate in modified reflector elements. Within the pool the lines consist of concentric tubes. The inner (rabbit) tube is 1.5 in. I.D. and the outer diameter of the tube assembly is 2.5 in.

Two tubes enter on each side of the pool through a flange which is located above the refueling level. The tube bundle, from the flange to the reflector, is removable for replacement or maintenance.

3.2.5.5. Reflector Irradiation Baskets

Eleven graphite reflector elements are removable and replaceable with modified reflector elements (irradiation baskets) containing samples. The modified element must appropriately restrict the flow to maintain uniform flow through the reflector, and all reflector spaces must be occupied.

3.2.5.6. Gamma-Exposure Facility

A lined and stepped cavity is provided in the shield face adjacent to the deep storage racks in the storage canal. The back face of the cavity consists of the 3/8-in. Al pool lining stiffened with aluminum angles.

Barytes blocks would initially fill the cavity. A shield-block package would later be designed around equipment which would permit insertion and withdrawal of samples. The initial blocks and the eventual shield package would be equivalent in shielding effectiveness to the pool wall.

3.3. Reactor Analysis

3.3.1. Scope of Calculations

The nuclear design of the reactor is based on the results of one-dimensional multigroup calculations which include the effects of fuel burnup and the effects of xenon variation during cyclic operation.

The analysis was directed toward selection of design parameters which would satisfy the following basic nuclear requirements:

(1) The core should be capable of at least 300-Mw-days operation at 10 Mw with sufficient reactivity retained at the end of this period to compensate for experiments.

(2) Throughout this period the reactor must be capable of over-riding xenon after the 16-hour shutdown associated with 8-hour daily operation.

(3) The external shroud of control rods must provide an adequate shutdown margin when fully inserted.

(4) The reactivity balance must be such that during operation the shroud of control rods is at least half-withdrawn, so that the beam-port terminals at the lower half of the core are not subjected to flux depression by the rods.

The following areas were also studied in the analysis:

(1) the required fuel loading

(2) radial power distributions

(3) flux distributions in the core and in the experimental regions

(4) void coefficients in the various regions of the reactor

(5) a comparison of beam-tube currents available from the reference reactor relative to those available from a typical solid-core reactor for the same application.

3.3.2. Reactor Configuration and Composition

The reactor dimensions were selected on the basis of results of extensive parameter studies performed previously for similar flux-trap reactors (4). The reference configuration is intended to maximize the thermal neutron flux available in a 1.5-inch test hole, consistent with the requirements of heat removal in the core.

The reactor is 2.0 ft in active length and is assumed to be reflected on either end by 1 cm of aluminum followed by an effectively infinite length of light water.

The composition in the radial direction described in Table 3.5. Calculations were based on an oxide fuel element rather than a U-Al alloy because a greater fuel mass was anticipated. The results are thought to describe adequately the case of the alloyed element.

TABLE 3.5

RADIAL COMPOSITION OF THE REFERENCE REACTOR

Region Number	Material	Outer Radius (cm)	Thickness (cm)
1	H_2O or Na	1.905	1.905
*2	Al	2.2225	.3175
*3	H_2O	6.1214	3.8989
*4	Al	6.7564	.635
5	H_2O	6.858	.1016
6	Al	7.239	.3810
**7	Core (UO_2-Al-H_2O)	14.478	7.239
8	Al	14.9352	.4572
9	H_2O	15.0368	.1016
10	Al	15.6718	.635
11	H_2O	15.9098	.238
***12	H_2O	16.3863	.4765
13	H_2O	16.6243	.238
14	Al	16.9418	.3175
15	H_2O	17.0434	.1016
16	BeO-Al-H_2O	24.5618	7.5184
17	C-Al-H_2O-.0813 void	35.9918	11.43
18	C-Al-H_2O-.0412 void	47.4218	11.43
19	H_2O	53.4218	6.0

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* Small changes in the aluminum thickness have been made subsequently to include the island in the pressurized loop (see Figure 3.7).

** UO_2 assumed rather than U-Al alloy; dimensions based on involute fuel elements.

*** Region 12 is fully occupied with the control rod when it is inserted.

The voids shown in the graphite reflector regions represent the homogenized equivalent of the beam tube voids over a two-foot length of reflector.

All reactor regions except the core (which is considered to be uniformly 150°F in temperature) are assumed to be at a temperature of 110°F. The material compositions are listed in Table 3.6.

TABLE 3.6

MATERIAL COMPOSITIONS

Region Composition	Element	*Density (gms/cc)	Volume Fraction
Na	Na	.9682	1
H_2O	H_2O	.99072	1
Al	Al	2.68	1
BeO-Al-H_2O	BeO	2.8	.91
	Al	2.68	.072
	H_2O	.99072	.018
C-Al-H_2O-.0813 void	C	1.6	.8751
	Al	2.68	.0347
	H_2O	.99072	.0089
	void	0	.0813
C-Al-H_2O-.0412 void	C	1.6	.9133
	Al	2.68	.0362
	H_2O	.99072	.0093
	void	0	.0412
6 kg. Core	UO_2	10.2	.022195
** (Uranium is .9317	Al	2.68	.362420
U^{235})	H_2O	.9804	.615385

* Effective density equals this density times the volume fraction.

** UO_2 fuel assumed rather than U-Al alloy; involute fuel elements.

3.3.3. Analysis of Hot Clean Core

All the neutron calculations were performed using the nuclear codes WANDA-4 or INA, both of which employ few group one-dimensional theory. Four groups of neutrons were used, with the separation points of the groups at 821 kev, 5.53 kev, and .625 kev.

The epi-thermal neutron group constants were obtained by the MUFT-IV code according to the P_1 self-consistent age approximation with a fission-spectrum neutron source. The thermal group neutron constants for the core were obtained from the SOFOCATE code solution with the Wigner-Wilkins approximation, and the thermal constants for all other reactor regions were computed with the assumption that the thermal spectrum is Maxwell-Boltzmann.

The reactivity changes resulting from various perturbations on the operating reference reactor are summarized in Table 3.7.

3.3.3.1. Flux and Power Distributions

The radial flux distributions are presented in Figure 3.14 for the reference design, with sodium occupying the test hole and the rods fully withdrawn. Sodium is a convenient material to simulate a typical test sample since it is a weak absorber with essentially no moderating properties. The greater thermal flux in a water-filled test hole is shown in Figure 3.15. In Figure 3.16 is shown the effect of fully inserting the rods.

It should be noted that the thermal flux in regions far from the fuel is underestimated if the reactor is supercritical in the condition assumed for the calculation. Previous studies show that the under-estimate is about equal to the multiplication factor k. Therefore, the test hole fluxes are expected to be about 12% greater than shown in Figures 3.14 and 3.15. With this correction, the following thermal flux values are indicated for 10-Mw operation.

FIGURE 3.14

Radial Flux Distributions - Na in Test Hole
Rods Out; Case W-6; 10 Mw

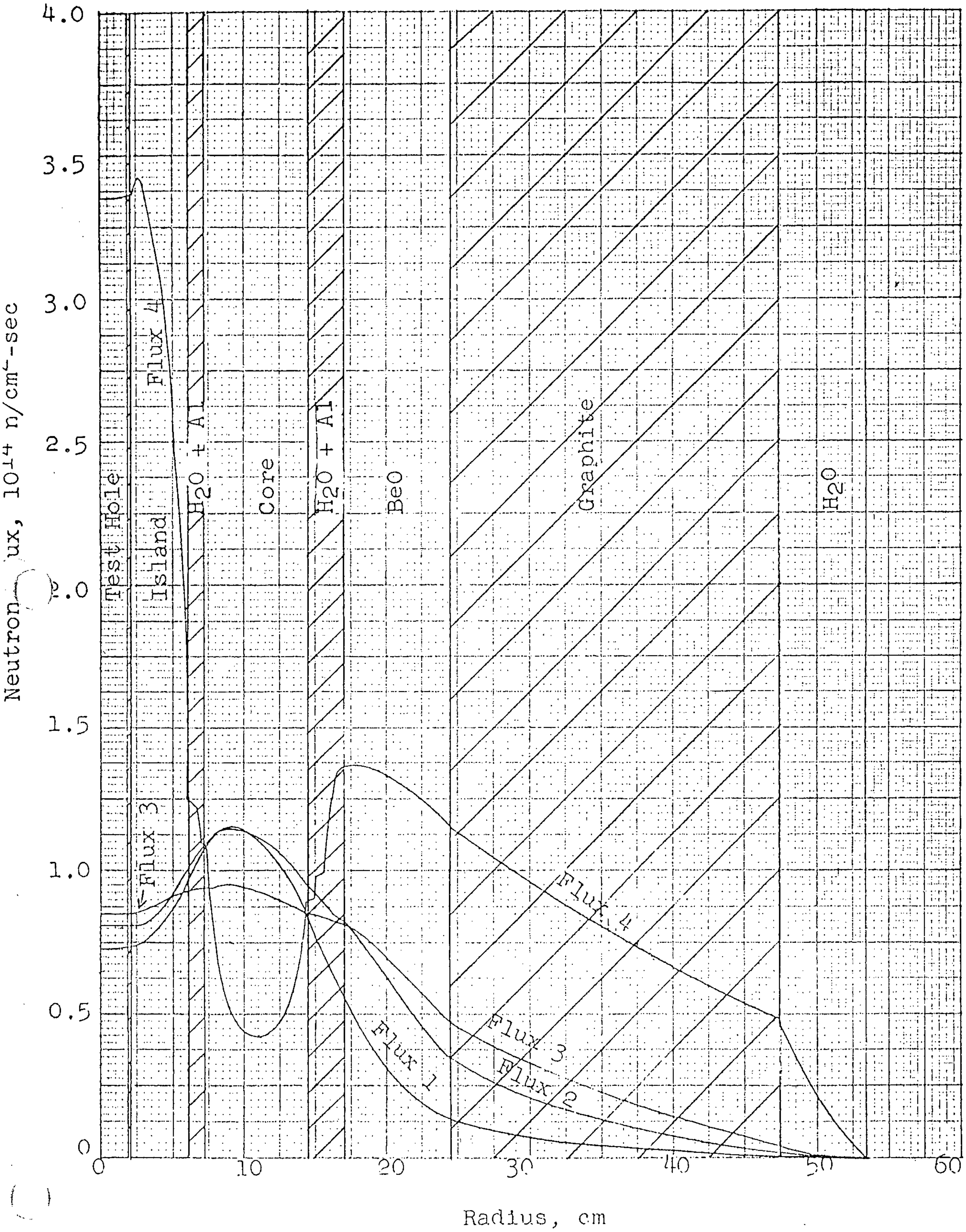


FIGURE 3.15

Thermal Fluxes with H_2O or Na in Test Hole
Cases W-6 and W-9; 10 Mw

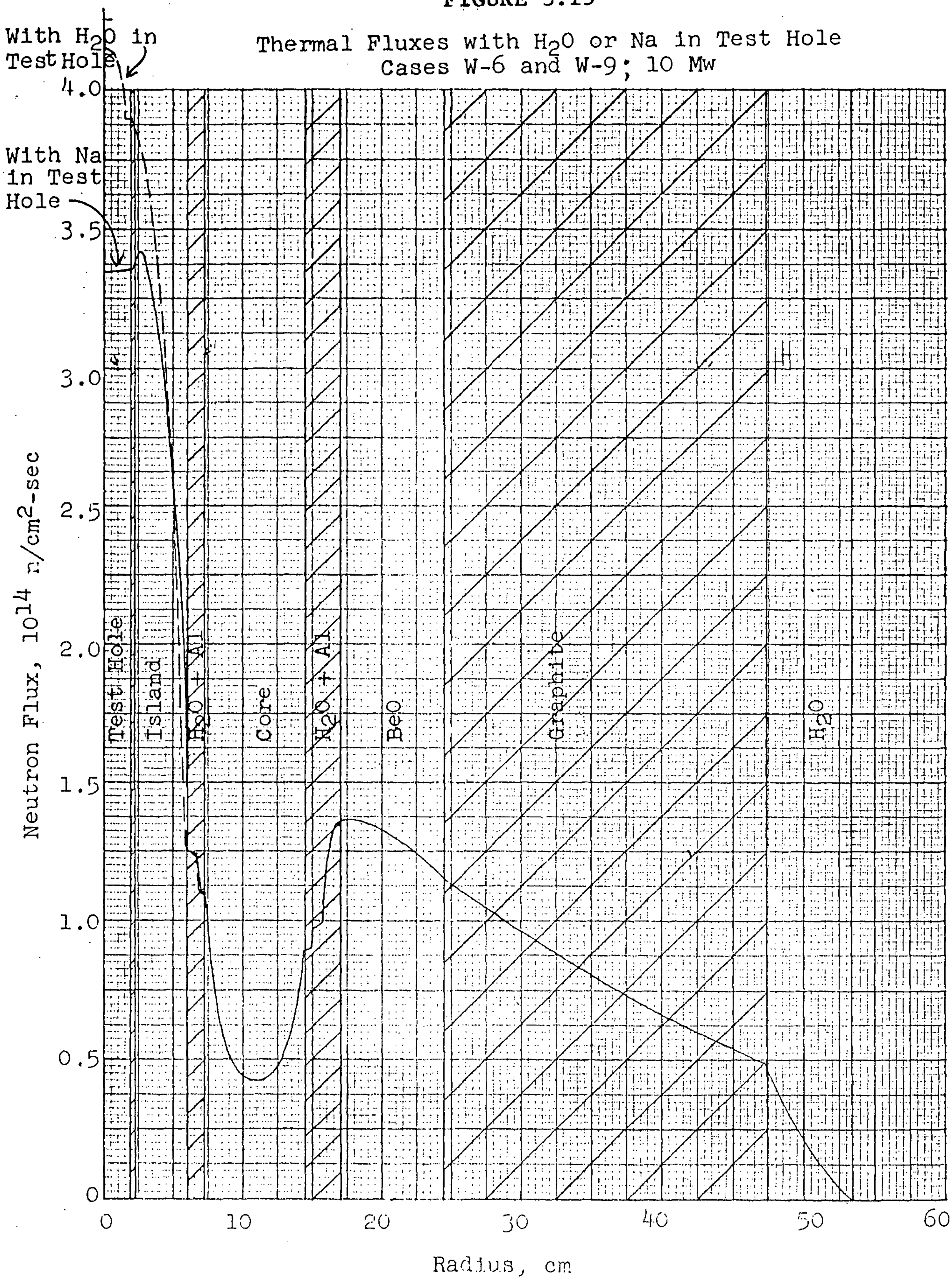
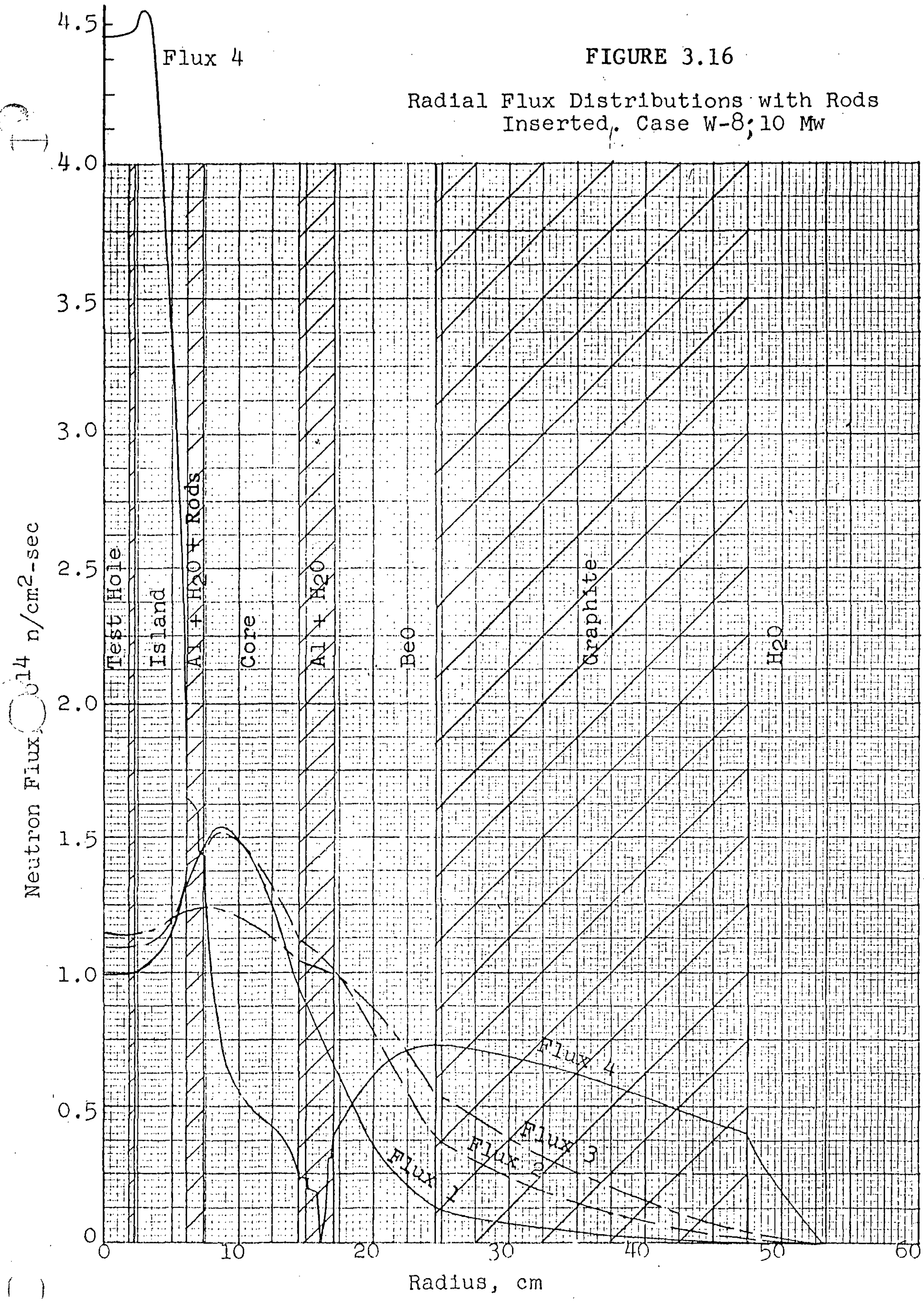


FIGURE 3.16

Radial Flux Distributions with Rods Inserted, Case W-8, 10 Mw



	Thermal Flux Average in Core	Average in Test Hole
W-6: Na in Test Hole; Rods Out	5.56×10^{13}	3.78×10^{14}
W-9: H_2O in Test Hole; Rods Out	5.56×10^{13}	4.51×10^{14}
W-8: Na in Test Hole; Rods In	5.33×10^{13}	4.20×10^{14}

These values do not include an axial peaking factor (nor a depression factor in case W-8). The axial distribution depends on the rod position and the presence of the beam ports in the lower half of the reflector, so it could be determined only by two-dimensional calculations. With the rods fully withdrawn, a one-dimensional solution gives the axial distribution of Figure 3.17. With the 1.64 max/ave. axial flux distribution assumed for heat transfer calculations, the peak thermal fluxes in the test hole would be approximately 6.2 and 7.6×10^{14} in cases W-6 and W-9, respectively.

The radial power distribution peaks at the inner fuel surface, as shown in Figure 3.18. A maximum-to-average ratio of 1.84 occurs with the rods out (case W-9) and a ratio of 2.41 occurs with the rods in (case W-8).

3.3.3.2. Variation in Materials

Figure 3.19 presents the variation in Δk with fuel loading as given by cases W-5, 6 and 7.

Cases W-13, W-12, and I-8 of Table 3.7 indicate the reactivity value of the BeO layer relative to that of the graphite which surrounds it. Replacing the graphite with water causes a loss in Δk of 0.033; replacing it with void loses 0.091. Replacing both graphite and BeO with H_2O causes a loss of 0.182.

3.3.3.3. Void Effects

Reduction of water density in the core causes a decrease in reactivity. An equivalent change in water density in any other region, without coincident density change in the core, causes a reactivity change of opposite sign but of lower magnitude than occurs in the core (see Table 3.7).

FIGURE 3.17

Axial Flux and Power Distributions with Rods Out
10 Mw

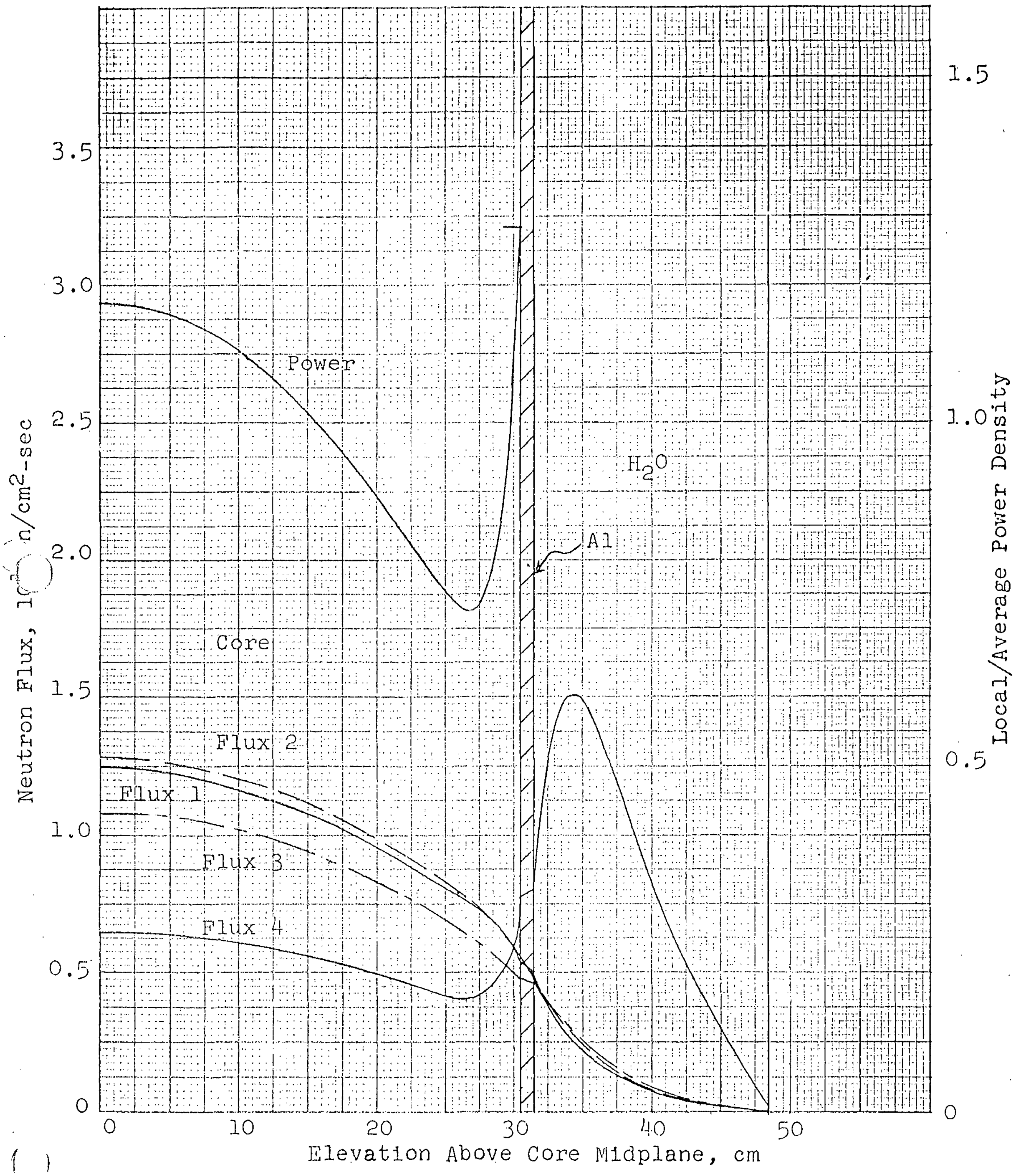


FIGURE 3.18

Radial Power Distributions

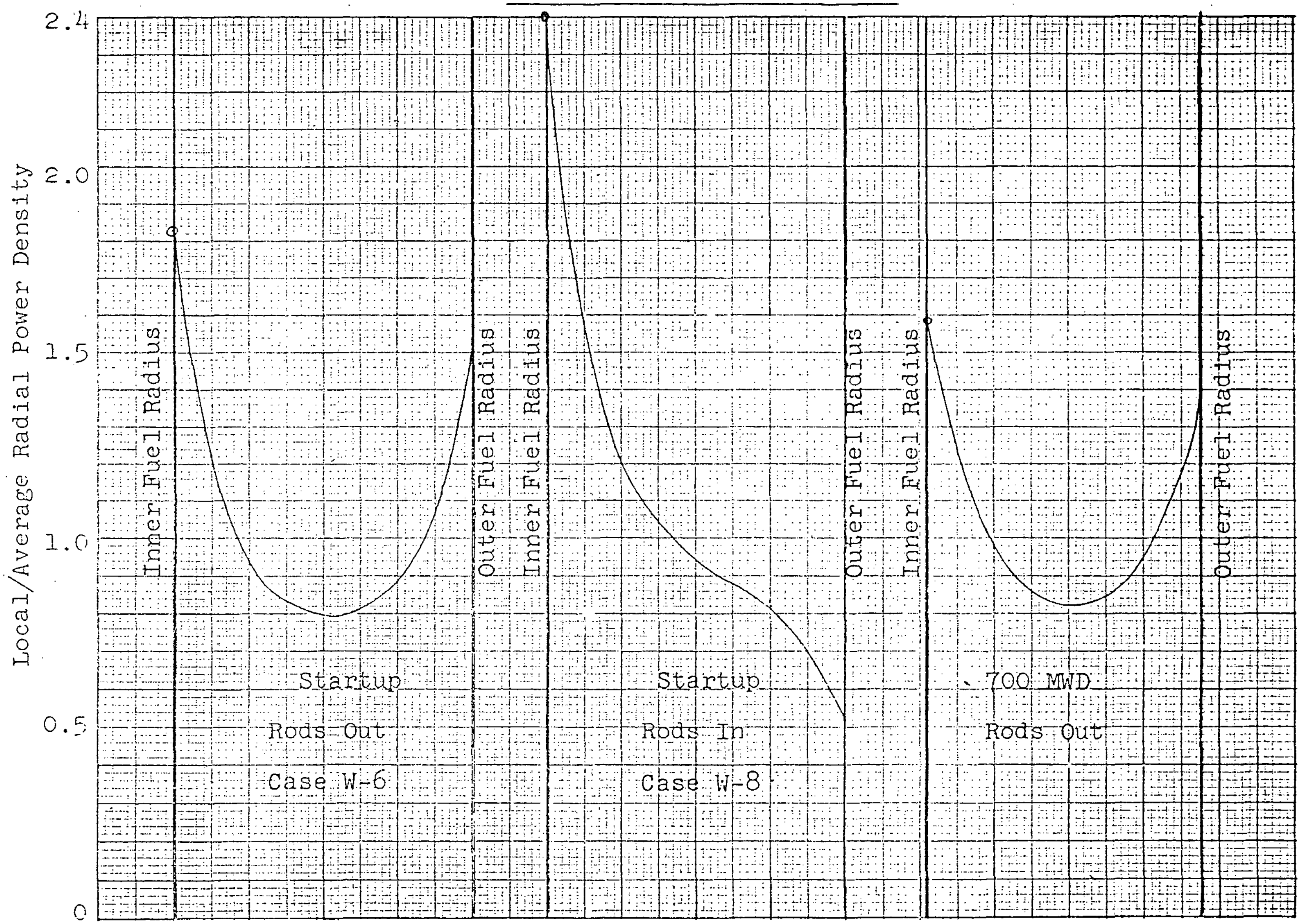


FIGURE 3.19

Δk versus Fuel Loading

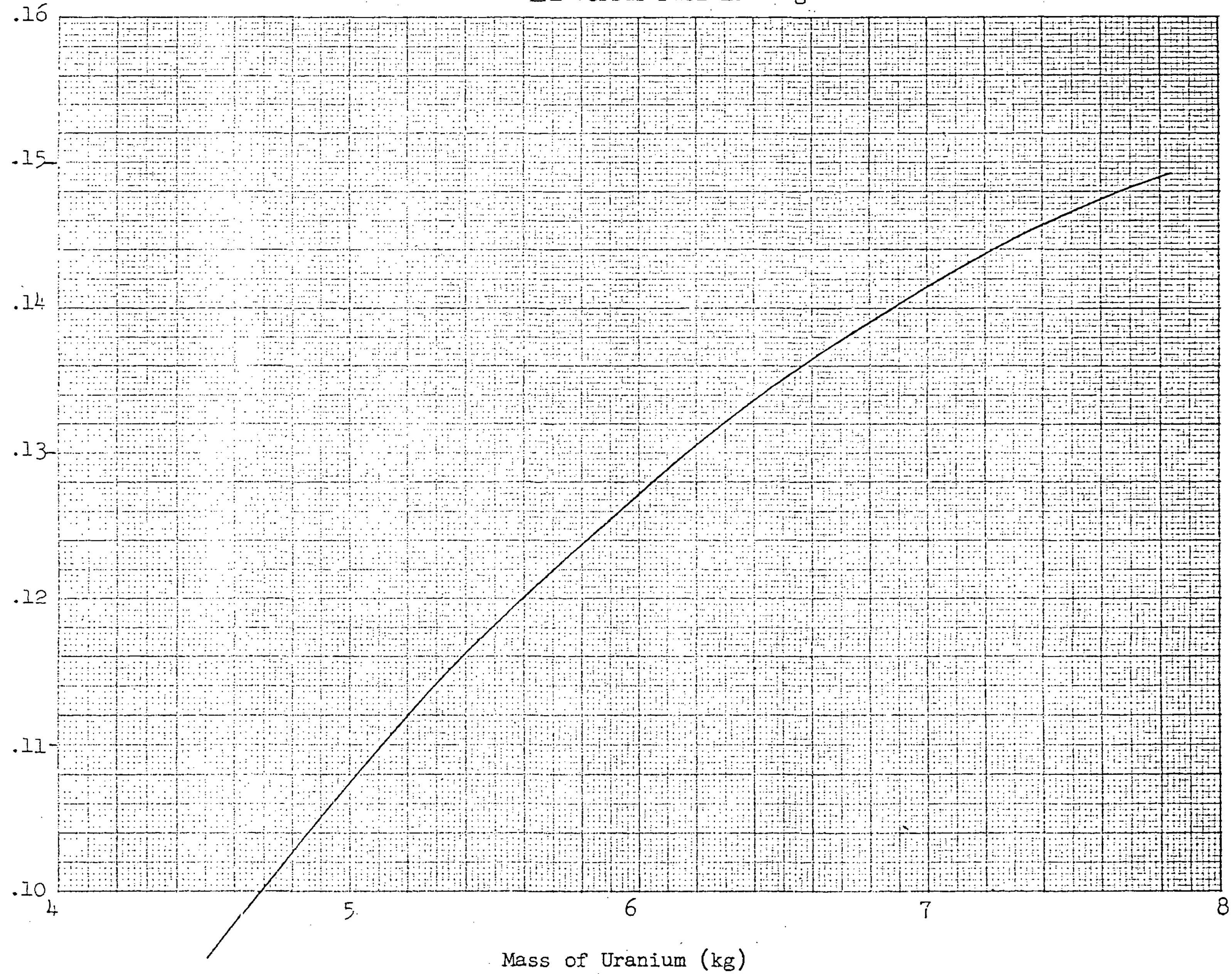


TABLE 3.7

REACTIVITY SUMMARY FOR HOT CLEAN CORE CALCULATIONS

Case	Perturbation	Test Hole Material	k	Δk, Relative to Case ( )
	6 kg Reference Case			
W-6	Na in Test Hole	Na	1.12718	
W-9	H_2O in Test Hole	H_2O	1.12099	-.00619 (W-6)
W-8	Rod Shroud Fully In	Na	.94161	-.18557 (W-6)
	Varied Fuel Loading			
W-5	5 kg U	Na	1.10727	-.01991 (W-6)
W-7	7 kg U	Na	1.14162	+.01444 (W-6)
	Reflector Variation			
W-17	Outer 1.5 in. of BeO Replaced by Graphite + Beam-Tube Voids	Na	1.10818	-.01900 (W-6)
W-13	H_2O Replaces Graphite and Voids in Outer Reflector	Na	1.09386	-.03332 (W-6)
W-12	H_2O Replaces Both BeO & Graphite Reflectors	Na	.94547	-.18171 (W-6)
W-18	Void Outside BeO	Na	1.03579	-.09139 (W-6)
	Void Effects			
W-20	H_2O Density Reduced 10% in Core	Na	1.10059	-.02659 (W-6)
W-19	H_2O Density Reduced 10% in Test Hole	H_2O	1.12203	+.00104 (W-9)
W-10	H_2O Density Reduced 10% in Island	Na	1.13428	+.00710 (W-6)
W-11	H_2O Density Reduced 10% in Test Hole & Island	H_2O	1.12948	+.00848 (W-9)
W-14	All H_2O Coolant Removed from BeO Region	Na	1.13452	+.00734 (W-6)
	Reactivity Effect-Beam Tubes			
W-15	All Beam-Tube Void Replaced by H_2O	Na	1.12659	-.00059 (W-6)
W-16	All Beam-Tube Void Replaced by Graphite	Na	1.13204	+.00486 (W-6)

The positive void coefficient in regions other than the core is of concern with regard to boiling in these regions. The significance of this effect in hazards considerations is discussed in Section 3.6.1.

3.3.3.4. Flooding of Beam Tubes

Case W-15 of Table 3.7 indicates a small reactivity loss upon flooding all beam tubes. The assumption of homogenized beam-tube void space may be poor, and it remains possible that flooding causes a small reactivity increase.

Case W-16 shows that only a small positive change in reactivity results from the introduction of a good moderator into an empty beam tube.

3.3.4. The Effects of Cyclic Xenon and Fuel Burnup

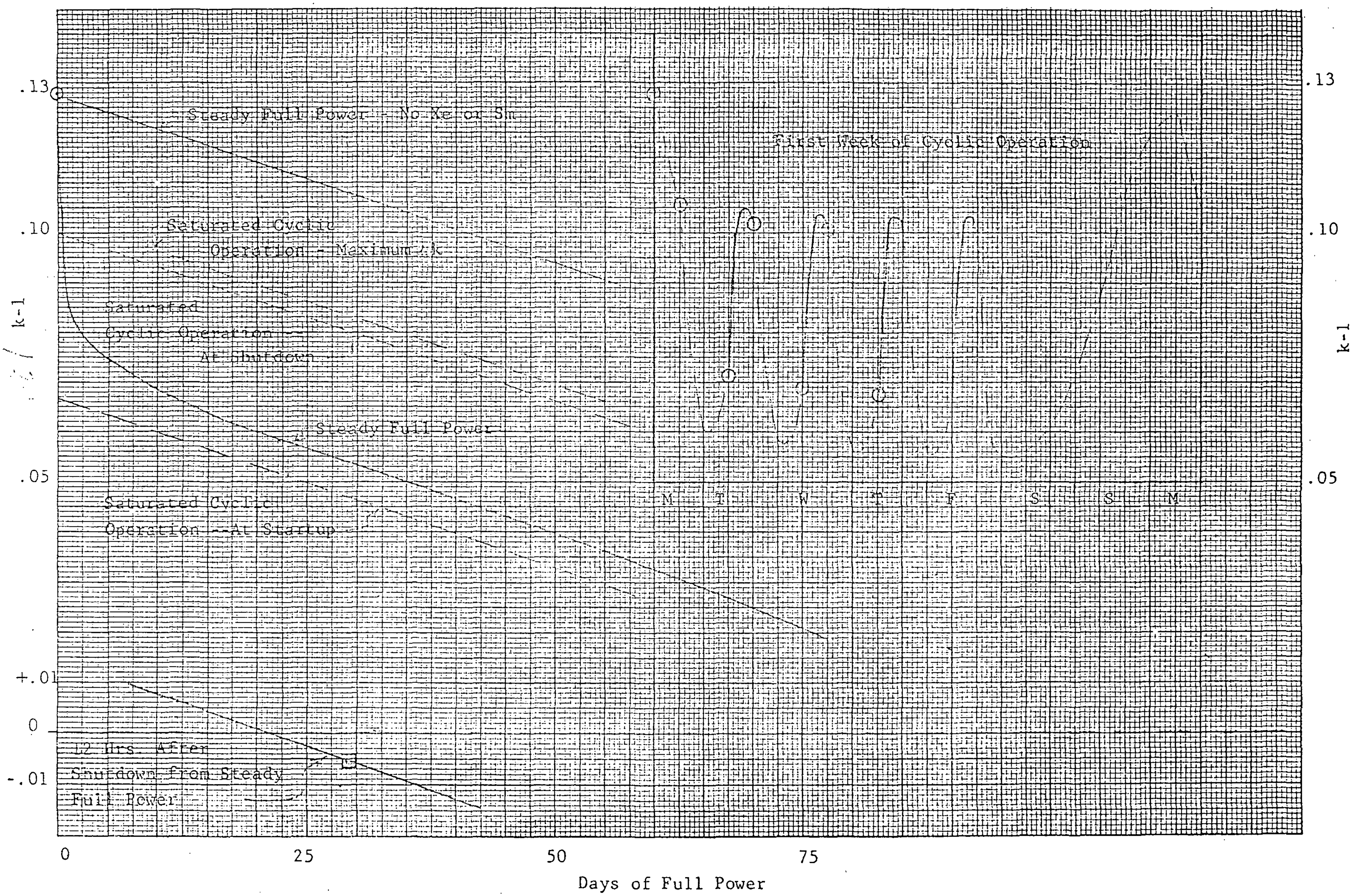
The reactivity that must be compensated by the control rods varies continuously during operation on an eight-hour-per-day basis because the xenon does not have time to reach an equilibrium value. The reactivity requirement is then somewhat different from that for continuous operation, and the relative requirements depend on the power level and the fuel loading.

The effects of fuel burnup and fission product poisons were determined for both cyclic and continuous operation at 10 Mw by a single burn-up calculation using the INA code. The calculation included the radial and time variation of low cross-section fission products, U^{236}, Xe^{135}, and Sm^{149}; and the depletion of U^{235}.

Figure 3.20 presents the results in terms of Δk for the hot reactor vs. time of operation at 10 Mw. The expanded time scale at upper right indicates the variation in k during the first week of cyclic operation. The circled points represent the values at start-up and shutdown during the first three days, as given by the burn-up calculation. The curve through these points has been determined approximately by deriving the equations which describe the cyclic I and Xe concentrations with fuel burn-up and spatial variations neglected.

FIGURE 20

Δk Loss with Fuel Burnup
10 Mw Power Level--6 kg U



After three days, the oscillations during the eight-hour-on, sixteen-hour-off schedule are of nearly constant amplitude. If this schedule is interrupted by a 64-hour weekend shutdown, the Xe essentially vanishes, as indicated by the curve.

After three simulated days of cyclic operation, the burn-up calculation proceeded at continuous 10-Mw operation for 70 days, except for a short shutdown at 30 days to observe the maximum loss in Δk following shutdown from 10-Mw continuous operation. This maximum loss occurs at $\sim$ 12 hours after shutdown from continuous operation and it is indicated by the point on the lower curve of Figure 3.20. This lower curve represents the greatest loss in Δk due to xenon which can occur at 10 Mw.

The reactivity loss during 10-Mw continuous operation is indicated by the curve labeled "steady full power." This curve was extrapolated back to zero days of operation and a parallel line was then drawn through the initial hot clean Δk value to obtain a theoretical burn-up curve with Xe and Sm neglected.

From the observed variation in Δk during the first three days of cyclic operation, lines can also be drawn to represent approximately the reactivity at start-up and at shutdown during saturated cyclic operation. These are shown as dashed lines in Figure 3.20, together with the dashed line corresponding to the minimum value of Xe occurring during each 8-hour period of operation. This minimum value occurs at about 4.6 hours after start-up.

It is noted from the calculations that the Xe content which must be overridden at start-up during cyclic 8-hour, 10-Mw operation exceeds the equilibrium value of Xe which would occur with continuous 10-Mw operation, by a value corresponding to about .007 $\Delta k/k$. The cyclic equations for Xe and I indicate that the reverse situation occurs at 5 Mw; the cyclic start-up concentration is then somewhat less than the steady-state value.

Radial power distributions are also available at each time step of the burn-up calculation. The distribution after 700 MWD of operation is shown in Figure 3.18. The maximum-to-average power ratio for the burned-out core is 14% lower than at start-up.

3.3.5. Reactivity Balance

3.3.5.1. Fuel Lifetime

Figure 3.20 indicates that a 6-kg core operated for 400 MWD (40 days continuously at 10 Mw or 168 days on an 8-hours-per-day, 5-days-per-week basis) would still provide .0394 Δk for experiments. This operating time corresponds to 9.47% in average U^{235} burnup, after which further burnup should be attainable by using a mixed loading of fresh and used fuel elements.

The reactivity endurance is somewhat overestimated in this calculation because the spatial variation of burn-up was considered only in the radial direction.

At 5 Mw, the reduced xenon poisoning would permit some reduction in initial fuel loading to obtain the same MWD endurance with the same amount of experimental absorber present at the end of fuel lifetime. Burn-up calculations have not been performed at 5 Mw; however the equivalent 5-Mw loading is estimated to be about 5.5 kg uranium, based on equilibrium Xe and approximate corrections to the 10-Mw burn-up data. The reduced loading at 5 Mw is of less interest from the hazards standpoint because the installed reactivity will then be less.

Converting the Δk values at 400 MWD to reactivity, $\Delta k/k$, the reactivity breakdown for 10-Mw operation is as shown in Table 3.8. The approximate reactivity values are indicated also for the case of 5-Mw operation with reduced fuel loading.

3.3.5.2. Reactivity in Rods

According to calculations W-6 and W-8 of Table 3.7, a complete shroud of rods provides a change in $\Delta k/k$ of -0.1748. Calculation W-8 assumed that the rod material was identical to Al except in its thermal properties, which were essentially for a mixture of Cd and Al. It is intended instead to use boral which, from previous studies, is expected to be worth about one-fourth more because of its epithermal absorption.

TABLE 3.8

REACTIVITY BREAKDOWN ($\Delta k/k$)

	At 10 Mw With 6 kg U	At 5 Mw With 5.5 kg U
400 MWD Burn-up + Fission Products	0.021	~ 0.023
Sm + Xe at 8-hr Shutdown	0.024	-
Added Xe at Start-up, 16 hrs. later	0.030	-
Sm + Equilibrium Xe	-	~ 0.045
Temperature Effect*	0.004	0.004
Minimum Reactivity Requirement	0.079	~ 0.072
Available for Experiments with 6-kg loading	0.038	~ 0.038
Cold Clean Reactivity	0.117	0.110

* Computed from density effects in Table 3.7:

Core and Island: 70°F to 150°F

Other Regions: 70°F to 110°F

Assuming that the added boron absorption will compensate for the incomplete shroud required by design considerations, and neglecting temperature effects on the rod worth, about 6% in shutdown reactivity may be expected.

This is more than ample to ensure that the reactor can be shut down with one rod stuck in its withdrawn position. If more detailed analysis indicates that the attainable rod worth is too low, consideration will be given to the use of Be metal as a reflector rather than BeO.

3.3.5.3. Rod Position During Operation

The four shim-safety rods travel together so that their depth of insertion is always uniform. During operation they must be at least half-withdrawn in order not to depress the flux near the beam-port terminals.

The fraction of the rod worth released by the rods when they are in their lowest operating position may be estimated as follows:

$$\text{Fractional Rod Worth} = \frac{(\text{Temp. effect}) + (\text{Min. Xe} + \text{Sm}) + (\text{Experiments}) + (\text{Shutdown Margin})}{(\text{Cold Clean}) + (\text{Shutdown Margin})}$$

With uninterrupted cyclic operation, the minimum worth of Xe is appreciable, and the half-withdrawal requirement is easily satisfied. However, weekend shutdowns cause the Xe + Sm term nearly to vanish, and full use of the beam ports on Monday mornings requires that the rods be designed for half-withdrawal under this more restrictive condition.

Assuming the following values:

Temp. Effect = .0042

Min Xe + Sm = 0

Shutdown Margin = .06

Cold Clean = .117 at 10 Mw & 6 kg U

and assuming that the rods are half-withdrawn when their reactivity is half-released, the half-withdrawal point requires .0243 $\Delta k/k$ to be held in experiments.

The requirement on shutdown margin and quantity of experimental absorber necessary to ensure half-withdrawal is reduced as burn-up proceeds, and it is also reduced for fuel loadings having less installed reactivity. Consequently, the most restrictive design requirement is for Monday mornings with a fresh 10-Mw loading.

If detailed design studies indicate difficulties in satisfying the half-withdrawal requirement, consideration will be given to the use of partially loaded fuel elements, burnable poisons, or removable poisons. In any event, the requirement that the beam ports be exposed during operation does not appear to present a serious design problem.

3.3.5.4. Reactivity in Experiments

The reactivity in Table 3.8 assumes a test hole completely filled with sodium, a typical weak absorber. Replacement with water will cause a $\Delta k/k$ loss of 0.0049. From studies of similar reactors, a test hole loaded with half water and half stainless steel for its full length might cause a further loss of 0.009 relative to the all-water case. The useable experimental volume within the test hole will be restricted as indicated in Section 5.7.2.

The pneumatic facilities and reflector experiments are located in the irradiation sector of the graphite. Replacing all the graphite in this 70 degree sector with H_2O may cause a loss of $0.0053 \Delta k/k$, assuming that the reactivity loss varies linearly with the fraction of circumference replaced; replacing it with void would lose 0.0152 on the same basis.

Beam tube experiments will have little reactivity effect and the BeO will be permanent and will have no experiments located within it.

Although insufficient data are available to estimate the total reactivity expected to be held in experiments, the above considerations suggest that it will be less than 4% and that adequate allowance is made in Table 3.8.

3.3.6. Beam-Tube Current

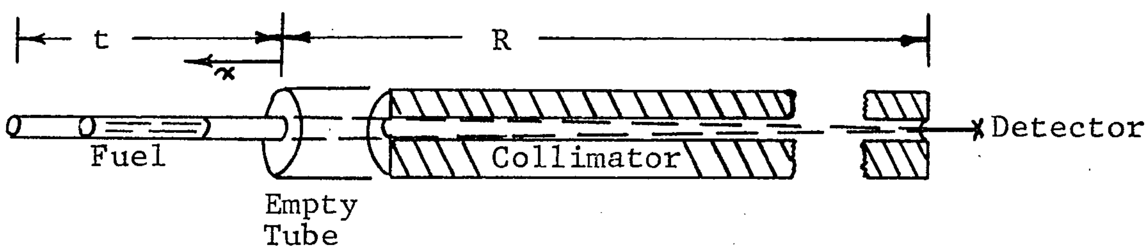
An analysis of available beam-tube currents was performed in an attempt to determine the merit of a flux-trap reactor for beam-tube use, relative to that of a typical solid-core research reactor.

Some uncertainty in the method of calculation compels caution in use of the absolute currents predicted; however, the results permit more confidence than do estimates based on flux levels at the inner end of the beam tube.

More useful information results from a comparison of results for the cases examined. From such comparison, it is concluded that the beam-tube current varies nearly as the power density in the fuel. Consequently, beam currents from this flux-trap reactor should be about twice those available from the larger-volume solid core which might be selected for the same application.

3.3.6.1. Method of Analysis

Only those neutrons which travel directly outward along the path of the collimator are of interest. The neutrons traveling in this direction will be attenuated exponentially from their point of birth or from their point of last scattering; hence, if the distributions of both the fission density and the scattering density are known throughout the thin conical segment subtended by the collimator angle, the current at the detector can be determined.



Assuming isotropic scattering, the flux or current received at the detector from a single energy group is given approximately by:

$$J = \frac{\pi r^2}{4\pi R^2} \int_0^t \left[S_o(x) e^{-\Sigma_t x} + \Sigma_s(x) \; \phi(x) e^{-\Sigma_t x} \right] dx$$

where $S_o(x)$ = source density at x, which exists only in the fueled region

$\Sigma_s(x)$ = elastic scattering cross-section

$\Sigma_t(x)$ = total cross-section for absorption and scattering

$\phi(x)$ = flux at x

r = radius of collimator

R = distance to detector

t = thickness of scattering region.

Radial flux printouts were available for the source plus four neutron flux groups in the various reactor calculations; hence, the integration could be performed numerically, to estimate the current from each of the four scattered groups plus the uncollided fission neutrons.

The cross-sections assumed for each region and for each energy group were obtained from the input of the radial flux calculations or were calculated from the same basic data used to obtain slowing-down cross-sections for those calculations. All the source neutrons were attenuated identically to the first group, which has a lower energy cutoff to 0.82 Mev. Three-fourths of the fission neutrons are born in this group and the beam-tube end is near the fuel in all cases of interest, hence this procedure seems valid. If greater attenuations were involved, it would be necessary to count only the very high energy neutrons and to attenuate these with the lower cross-sections at high energies, as in shielding problems.

3.3.6.2. Flux-trap vs. Solid-Core

To determine the advantage of the flux-trap reactor relative to an equivalent solid core, another WANDA-4 problem was run to determine radial flux distributions for a typical solid-core research reactor operated at the same power level and having essentially the same amount of reactivity. The solid core was assumed to be a cylinder of the same two-foot height and of radius 9.260 inches, equivalent in cross-sectional area to 28 MTR fuel elements. The core composition was based on the same metal-to-water ratio as in MTR and an assumed loading of 3.5 kg U. The same thicknesses of BeO and graphite reflector were assumed to be present, but these regions were located immediately adjacent to the core, as might be the case in an unpressurized reactor. The reflector of the flux-trap reactor is separated from the fuel by an inch of Al and H_2O because of the pressure vessel and slots for the external control rods.

Currents from the two reactors are compared assuming the beam tube to terminate at the inner face of the BeO, a point reached by the extensions of two of the beam tubes in the reference reactor. These extensions are only about 2 in. in diameter, the anticipated size of the collimator. The calculation is based on the fluxes which would exist with a complete reflector; i.e., depression by the beam tube is neglected.

With a 2-in. collimator and with the detector located 9 ft -4 in. from the inner face of the BeO, the currents are:

		Currents		Ratio of Currents
		Flux Trap 10 Mw	Solid Core 10 Mw	Flux Trap/Solid Core
Source	> 821 Kev	0.911×10^9	0.472×10^9	1.93
1	> 821 Kev	1.391×10^9	0.769×10^9	1.81
2	5.53-821 Kev	1.724×10^9	0.845×10^9	2.04
3	0.625-5530ev	1.579×10^9	0.816×10^9	1.94
	Total Fast	5.605×10^9	2.902×10^9	1.93
4	Thermal	2.398×10^9	1.219×10^9	1.97

The flux-trap reactor apparently gains a factor of about 2 in currents. The ratio of power densities is 2.842 and the relative advantage would be more nearly equal to this value if the annular core were not handicapped by the attenuation of the vessel and rod gaps.

Currents were also calculated at the outer face of the BeO, based on a radial flux calculation with only void beyond the BeO, to approximate the depression caused by the large beam tubes. The total fast current with the same collimator and tube length was 1.67×10^9, and the thermal current was 0.54×10^9. This calculation was performed only for the annular core, so comparison with the solid core was not made. However, approximately the same advantage would be expected at this position, as may be inferred from the ratio of fluxes at the BeO interfaces when graphite rather than void is present beyond the interface.

	ϕ, Flux Trap/ϕ, Solid Core	
	BeO, Inner Face	BeO, Outer Face
Total Fast	1.81	1.72
Thermal	1.97	1.83

3.4. Heat Removal

3.4.1. Reactor Loop

3.4.1.1. Design Considerations

A high power density is desirable from the standpoint of beam-tube currents, and is also necessitated by geometrical requirements in that the inner fuel radius is fixed by the optimum flux-trap dimension and the fuel thickness is limited to that which permits effective control by an external shroud of control rods.

The reactor is designed to avoid surface boiling at the hottest fuel plate. Although boiling in the core does not in itself present any hazard, abundant steam production would probably have undesirable effects because of the high specific volume of low-pressure vapor.

Pressurizing the reactor raises the saturation temperature and permits reasonable design velocities, without boiling, at these high power densities. The core design pressure of 50 psi over pool pressure requires essentially no increase in material thicknesses relative to an unpressurized reactor.

The choice of design inlet temperature must compromise the reduction in heat exchanger area gained by high inlet temperature with the increased heat conducted to the pool and the consequently greater size of the pool heat exchanger. An inlet temperature of 140°F is selected on this basis.

Consideration was given to initial 5-Mw operation as an open pool, drawing pool water through the open lid of the vessel and circulating it through the heat exchangers and back to the pool. However the inlet temperature would then have to equal the pool temperature. The consequently low temperature difference between primary and secondary water would require much more heat exchanger surface at 5 Mw than the total area proposed for 10-Mw pressurized operation. Therefore pressurized operation is proposed at both power levels.

3.4.1.2. Power Distribution

The radial power distribution across the fuel annulus has been determined for all the conditions described in Table 3.7, and also at various time points during burn-up and during cyclic operation with transient xenon effects. With rods removed, the worst radial peaking occurs at startup, with a maximum-to-average ratio of 1.84 at the inner fuel radius. With the rods fully inserted, a value of 2.41 results at this point.

Determination of the axial power distribution would require two-dimensional calculations which consider the presence of control rods and beam tubes. In the absence of such calculations, a peak value of 1.64 was extracted from data reported for the Oak Ridge Research Reactor with rods partially inserted.

The peak radial value of 1.84 is used for design purposes, and this is assumed to occur coincidentally with the axial power peak of 1.64. The greater radial peak of 2.41 would occur at elevations where the rods are present, but the axial power density at those elevations would probably be less than average, hence the maximum power density should occur below the rods.

The axial power peak is assumed to occur at the lower end of the fuel where the pressure and saturation temperature are least. The closest approach to boiling consequently is assumed to be at the core exit.

3.4.1.3. Heat Transfer and Fluid Flow Data

Hot Channel Factors

The hot channel factors are those determined previously for a similar reactor:

F_b(bulk temperature rise) = 1.275
F_f(film drop) = 1.350

Film Coefficients

The film coefficient is determined at the average film temperature by the following correlation:

$$h = 0.02 \frac{k}{D} \quad (Re)^{0.8}(Pr)^{0.4}$$

Core Pressure Drop

An analytic expression is used for the friction factor, which best fits the Crane Handbook data over the range of Reynolds numbers and at the hydraulic diameter of interest. This expression, applied to the Fanning equation, gives a pressure drop in a 2-ft length:

$$\Delta p = 0.0519\ v^{1.865}$$

where the drop is in psi and the velocity v is in ft/sec.

The calculated core pressure drops also include entrance and exit losses at the ends of the fuel plates.

The pressure assumed at the core inlet for the 10-Mw calculation corresponds to 80.3 psia at the pressurizer, when the pressure drop between the core and pressurizer is considered. All calculations assume this pressure at a pressurizer the same elevation as the fuel.

Core and Island Data - Involute Fuel Elements

Gap thickness	0.080	in.
Flow area between plates	0.3271	ft^2
Total core flow area	0.3421	ft^2
Heat transfer area	190.0	ft^2
Heat fraction released in core	0.94	
Island area	0.1205	ft^2

Results

At 10 Mw, surface boiling occurs at the hot spot with a core velocity of 18.7 ft/sec or a total loop flow of 2918 gpm. A flow of 3200 gpm would still cause boiling to occur at only 10.6 Mw and this may be too close for nominal 10-Mw operation. Therefore, 3600 gpm is specified (4-900 gpm or 2-1800 gpm pumps). Boiling is then predicted to occur at 11.4 Mw; and at 10 Mw the maximum fuel plate temperature is 18° below boiling.

At 5-Mw operation, 7.55-ft/sec core velocity or 1178 gpm would cause boiling. With 1800 gpm, the hottest fuel surface is 40°F below boiling, and 6.6 Mw would be required to boil.

The coolant conditions at design flow are as follows:

TABLE 3.9

DESIGN DATA - Reactor Loop

	5 MW	10 MW
Core Flow	1732 gpm	3465 gpm
Island Flow	68 gpm	135 gpm
Total Flow	1800 gpm	3600 gpm
Core Velocity	11.5 ft/sec	23.1 ft/sec
Island Velocity	1.25 ft/sec	2.5 ft/sec
<u>Heat Flux</u>		
Average	0.84×10^5 Btu/ft^2-hr	1.68×10^5 Btu/ft^2-hr
Maximum	2.54×10^5 Btu/ft^2-hr	5.08×10^5 Btu/ft^2-hr
<u>Temperatures</u>		
Core Inlet	140°F	140°F
Core Exit	157.4°F	157.9°F
Maximum Wall	267°F	270°F
Saturation at Exit	307°F	288°F
Heat Added to Island Flow	40 kw	84 kw
Island Δt	4.1°F	4.3°F
<u>Pressure Drop</u>		
Across Core	5.3 psi	19.3 psi
Entire Loop	~ 11 psi	~ 40 psi

At 1800 gpm the 5 Mw core is relatively over-designed. A design point equivalent to the 3600-gpm, 10-Mw case might result by reducing the flow to 1450 gpm or by reducing the pressure at the pressurizer to 60 psia.

Boiling Burnout

At 10-Mw design conditions the hottest bulk liquid is sub-cooled 128°F relative to the design pressure, or 52°F with respect to atmospheric pressure. The maximum heat flux at 10 Mw is a factor of 4 to 6 less than would be required for burn-out with this amount of sub-cooling.

Lower factors would apply at the coolant conditions which would have to exist in order for burn-out to be approached. The occurrence of these conditions is hard to visualize; e.g., full power with a high steam fraction in the channels at the lower end of the core.

From the standpoint of design and operation, boiling burn-out appears to be of no concern. Only under severe runaway conditions does it seem to be a credible occurrence, and it then represents only a mechanism for the partial meltdown postulated in the hazards analysis.

3.4.2. Regions Cooled by Pool Water

The gamma heat production in regions outside the core has been estimated by scaling the results of machine calculations of gamma heating for a nearly identical geometry, and the gamma absorption in the island and test hole has been calculated approximately. With the added contribution of neutron slowing-down and absorption of water-capture gammas, the space-integrated heat production at 10 Mw is estimated as follows:

TABLE 3.10

HEAT PRODUCTION IN REGIONS OTHER THAN CORE AT 10 MW

Test Hole and Liner (test hole nearly 1/2 filled with steel)	20 kw
Island	68 kw
Inner Core Wall	42 kw
Outer Core Wall	43 kw
Rod Gap	43 kw
BeO	255 kw
Graphite	83 kw
Pool	70 kw
	624 kw = 6.24%

At 5 Mw half this amount, or the same percentage, would be generated external to the core.

Pool water is drawn through the reflector, the rod gaps, and the test hole; loop water flows through the island. The pool water must absorb not only the heat generated in the regions which it cools, but also the heat conducted from the loop water through the piping and vessel walls and much of the heat produced in the walls. Assuming typical velocities in each region and computing heat flow across each surface where the temperature differences of 40 to 60°F exist, the heat input to the pool water is estimated as follows:

TABLE 3.11

HEAT INPUT TO POOL WATER

Region	V, ft/sec	gpm	Heat Input, kw 10 Mw	Heat Input, kw 5 Mw
In Pool		353	165	130
Reflector	4.25	205	338	169
Rod Gaps	2 & 4	93	176	143
Test Hole	10	55	46	36
In Plenum		353	20	20
	TOTAL		745	498

With 140°F inlet to the core and 100°F pool temperature, the pool water apparently absorbs about 7.5% of the heat at 10 Mw, and about 10% at 5 Mw.

The most critical velocity in these regions is in the reflector, where a velocity of about 3.5 ft/sec is calculated as necessary to prevent boiling at the inner face of the BeO at 10 Mw.

To ensure that adequate velocities are obtained despite leakage around reflector experiments and beam-port penetrations, a flow rate of 350 gpm is proposed at 5 Mw, and 500 gpm at 10 Mw. With 100°F mixed mean temperature in the pool, this results in pool inlet and exit temperatures of 97.5 and 107.2°F at 5 Mw and 97.7 and 108.0°F at 10 Mw.

3.4.3. Overnight and Weekend Cooling

The convective loop and the heat exchanger within the pool will be designed to transfer the decay heat to the pool at a rate sufficient to prevent the occurrence of boiling or excessive temperatures in the reactor core.

During overnight or weekend shutdown, the reactor will be unattended, with stop valves closed on the pool loop and the reactor loop. During this time the heat capacity of the pool is sufficient to absorb the decay heat.

In the worst case anticipated at 5 Mw, shutdown following 30 days of continuous operation, the time-integrated heat release and the temperature rise in the pool are as follows:

TABLE 3.12

INTEGRATED HEAT RELEASE FOLLOWING 30-DAY CONTINUOUS OPERATION AT 5 MW

Time After Shutdown	Integrated Heat Release, Mw-sec	Pool Temp. Rise, °F
1/2 Hour	207	1.4
16 Hours (Weekday Morning)	2000	13.5
64 Hours (Monday Morning)	3445	24.5

Evaporative losses, without air circulation, are not significant and conduction to the pool walls has been neglected.

During the first two years, the operating level will not exceed 5 Mw and 8-hour operation is intended, hence the heating will be less. It is intended to run the pool loop for a short time following shutdown to lower the pool temperature before leaving the facility, and to run the pool loop again in the morning during the pre-start-up procedure, whenever this is found necessary.

Following 10-Mw continuous operation for 30 days, the values in the table would be doubled. The same operating procedure might then be adequate, but cooling times of up to 1.3 hr. on weekday mornings and 1.9 hr. on Monday mornings are indicated with the cooling equipment presently proposed.

Subsequent design work may indicate the desirability of cooling the pool overnight following operation at 10 Mw. A flow of about 50 gpm with heat rejection to one cooling tower or to a radiator is estimated to be adequate. In this case, the pool-loop modifications for 10-Mw operation would include an invert loop within the pool and a siphon break, as are presently provided on the reactor loop, to prevent loss of water overnight through a ruptured pool loop. Blind-flanged pipe stubs would be provided initially for this purpose and the eventual usage of this equipment would be discussed in the Final Hazards Report.

The heat released overnight from the hot pool will be removed from the building by the air conditioning units which will operate continuously. Loss of pool water will cause the containment to be closed automatically by a signal from the pool-level detector.

3.5. Pool-Surface Dose Rates

The pool water is exposed to high neutron levels in transit through the test hole, the rod gaps, and the reflector. The dose rate at the pool surface consists primarily of contributions from sodium recoils, activated sodium impurity, and the activity induced in water itself.

Specific activities were calculated for the pool water on the basis of the flow and residence time in each region previous to mixing in the lower plenum, using appropriate equations which include the cascade effect of an infinite number of exposures to the flux, and to the demineralizer bypass in the case of sodium.

Calculations of surface dose rates assume the pool to be a half-space of uniform source strength, and they include the effects of buildup due to scattering.

In the case of thermal neutron activations, average fluxes are available for each region and these are assumed to apply to 150% of core length, in order to approximate the effect of flux peaking in the end reflectors. One ppm sodium impurity is assumed to persist in the pool loop.

Estimates of sodium recoil activity are scaled from LITR and ORR data according to power density, flow, and recoil surface. The estimates for the pool loop are approximate because the effective activation flux for this threshhold reaction is uncertain in the external regions. However, the thermal activation of the assumed sodium impurity exceeds the calculated recoil activity in the pool loop.

The total surface dose rate from the 15 hr sodium activity is estimated to be ~ 4 mr/hr for the proposed conditions of operation at 5 Mw and at 10 Mw.

A holdup tank is provided to decay the 29 second O^{19} before returning it to the pool. The shorter-lived N^{16} and N^{17} activities are negligible with this amount of holdup. With 1700 gallon total holdup in the pool loop, and with 500-gpm flow at 10 Mw, the surface dose rate from O^{19} is estimated to be 5 mr/hr if the return is directly to the surface, and 0.1 mr/hr with complete mixing.

The total surface dose rate with continuous operation is estimated as follows:

	5 Mw, 350 gpm 50 gpm demin.	10 Mw, 500 gpm 100 gpm demin.
Na^{24} with 1 ppm Na	4 mr/hr	4 mr/hr
O^{19}	< 1/4 mr/hr	< 1 mr/hr
Core Gammas	1/2 mr/hr	1 mr/hr
	~ 5 mr /hr	~ 6 mr/hr

With repeated 8-hour daily operation the maximum Na^{24} activity would be 0.31 to 0.46 times as great, depending on whether or not the flow through the demineralizer is continued during shutdown.

When the reactor loop is opened for refueling, mixing of the reactor loop with the pool loop will increase the pool-surface dose rate. The added contribution from reactor-loop sodium would be 7 mr/hr following continuous operation at 5 Mw with 50-gpm or at 10 Mw with 100-gpm demineralizer flow in the reactor loop. The incremental dose rate will be less if the valving prevents complete mixing of the reactor loop with the pool, or if the reactor has been operated only for eight hours each day.

3.6. Reactor Safeguards Considerations

3.6.1. Void Coefficients

Reduction of water density in the core causes reactivity to decrease, as occurs in other small water-moderated reactors. However, reduction of water density in any region other than the core causes a relatively smaller but positive reactivity change, unless it is accompanied by similar density reduction in the core.

This behavior appears to be an unavoidable characteristic of flux-trap reactors of this type, which are partially reflector moderated and which contain light water in regions other than the core.

The proposed design essentially eliminates concern with boiling in the island by diverting core coolant through this region, and it relies on protective circuitry to minimize the hazard associated with boiling in the other regions, in which boiling would cause less reactivity to be released.

The calculated reactivity changes accompanying density changes ($\Delta\rho/\rho$) of the water in each region are presented as density or void coefficients in the following table, where the void coefficient is defined as the density coefficient with opposite sign.

TABLE 3.13

VOID COEFFICIENTS

Region	Void Coefficient $-\left[\frac{\Delta k}{k} / \frac{\Delta \rho}{\rho}\right]$
Test Hole	+ .0082
Island	+ .0556
Core	- .2143
Rod Gap	Not Computed
BeO Coolant Gaps	+ .0057

The void coefficient in the rod gap should also be positive but much less than in the island. The region of greatest concern with respect to a positive void coefficient is the island.

3.6.1.1. Boiling Within the Island

The island flow is in parallel with the core; thus independent flow reduction in one region cannot occur and, during forced convection, the ratio of flow rates in these regions should be essentially independent of the total flow rate.

Boiling will not occur in either region during operation because the pump speed is constant and the design flow is adequate.

Loss of flow or pressure does not cause boiling because either condition scrams the reactor, and loss of flow activates the convective loop within the pool, which will be sized to prevent the occurrence of boiling in either region. Boiling within the island or any other region, of course, is safe after shutdown because the rods provide an adequate shutdown margin.

Assuming that the reactor were somehow run at reduced flow through malfunction or deliberate bypassing of the safety circuits, boiling should occur first in the core, causing a reactivity reduction.

The flow reduction which causes surface boiling at the hottest fuel plate is predicted to be 20% at 10 Mw, although it may be somewhat greater depending on the power distribution and the validity of the hot-channel factors. The water in the island, however, will boil only through volumetric boiling induced by radiation absorption in the water. This heating causes a temperature rise in the island of only 4°F at design conditions, while a rise of at least 150°F would be required for bulk boiling at design pressure. To a first approximation, then, a velocity reduction of 37.5 can occur before boiling takes place in the island.

During shutdown operation with the pool convective loop in service, conditions are different in that the velocity ratio changes, pressure may be lost, and the ratio of heat production in the two regions is altered. (Immediately after shutdown, the core may be generating only 5% of the previous power, while the energy absorption in the island may be about half its previous value.) With the relative resistances corresponding to the design velocities proposed for the core and the island, it appears probable that boiling under natural convection conditions, if somehow it were to happen, would still occur first in the core.

3.6.1.2. Boiling Within Other Regions

The reflector, test hole, and rod gaps are cooled by pool water, hence loss of flow in the pool-cooling circuit could conceivably cause boiling without a compensating density change in the core.

Surface boiling in these regions is of greatest concern in the reflector region. At the hottest spot, on the inner face of the BeO reflector, a velocity of about 3.5 ft/sec appears necessary to avoid boiling at 10 Mw. The heat production is greatly attenuated through the rest of the reflector so this flow requirement exists only at the inner face. However, an average velocity of 6 ft/sec is proposed for the entire reflector.

Inadequate flow would cause surface boiling to occur first at this inner face. With surface boiling alone, the bulk liquid would still be sub-cooled and the average density reduction should be small. Reduction of flow by a factor of ~ 15 is required to produce bulk boiling at the exit of the inner face. Essentially complete flow stoppage is thus required to approximate the condition of total water loss. Expulsion of all the water around the BeO is calculated to release less than a prompt critical amount of reactivity.

Heat fluxes from the control rods and from non-fuel samples in the test hole will be low, hence flow reductions nearly sufficient to cause bulk boiling must occur before surface boiling results. With the design temperature rises of $\sim 10^{o}F$ in these regions, boiling would result only with large flow reductions. Flow reduction in either region would cause the bulk temperature to rise and dump considerable heat into adjacent regions; however, with flow stoppage, the heat transfer coefficient on one side of the wall would be greatly reduced, effectively insulating the region, and boiling would occur.

The occurrence of boiling in all three regions cooled by the pool flow is prevented during critical operation by:

1. designing for much greater flow than is required to prevent boiling,

2. scramming the reactor upon loss of pressure drop across the pool pump,

3. monitoring the water temperature in the exit plenum by means of thermocouples distributed through the plenum, and

4. requiring appropriate flow restrictions on experiments placed within the reflector retaining wall and being certain that all reflector positions are filled previous to start-up.

Following shutdown, boiling in these regions is of concern only in the sense that some damage could conceivably result from the higher temperatures. To prevent boiling during shutdown, a flapper valve is provided which opens automatically upon loss of the pool pump. When open, this valve permits pool water to circulate through the three regions by natural convection.

3.6.1.3. Density Reduction Due to Radiolytic Gas Production

An estimate of the density reduction due to production of radiolytic decomposition gases suggests that no problem exists in this area.

Assuming an effective G value in all regions of 1 hydrogen molecule per 100 ev absorbed, and assuming immediate appearance of the evolved H_2 and O_2 as void, the maximum density reduction occurs in the island. At a velocity of 2.5 ft/sec, and at 10 Mw, the average density reduction due to both hydrogen and oxygen is 0.003 in this region. A somewhat higher velocity would be required to reduce the residence time in the island sufficiently to cause the gas production in the core to compensate in reactivity for the island production.

The design velocities in all regions should be high enough to ensure that any gases or steam bubbles are swept downward out of the regions.

3.6.2. Shearing of Beam Ports

A heavy object dropped from the crane could shear a beam-port extension and cause loss of pool water. Bulkheads or shutoff valves could be provided in the beam tubes, but a bulkhead would prevent deep insertion of a collimator or experiment, and such insertion beyond a valve position would obstruct valve closure. The beam port shutters may serve as partial shutoff valves, but they function only when the collimator or plug is not present. Since this is the condition which causes the highest rate of water loss, they do serve to reduce the maximum rate.

Provision is made to pump the contents of the demineralized storage tank into the pool in the event of pool-water loss. If the leak cannot be compensated by this means, raw water would be dumped from a large hydrant at 1,000 gpm, a rate calculated as adequate to retain more than 3 ft of water above a completely severed 6-in. beam port with its collimator blown clear. Reduction of pool-water level also scrams the reactor.

3.6.3. Rupture of Pool Loop

A serious leak in the lower plenum, or failure of the valve on the plenum to close, would cause the flow through the test hole, reflector, and rod gaps to be reduced. This condition would be annunciated by signals derived from thermocouples distributed in the lower plenum. These thermocouples would be so distributed as to minimize the chance that mixing of leakage water would obscure the higher temperature of water flowing downward into the plenum. A high reading from any thermocouple would generate an alarm signal. Spare thermocouples would be installed initially to eliminate the need for remote replacement of defective units.

A break in the pool loop external to the shield would cause a rate of loss less than that from a sheared beam port - the line is 5 in. and loss from both ends is precluded by the check valve on the return line. The pump flow is low relative to that which could result from the gravity head; hence in the event of a break beyond the pump, which might leave the pump running, the rate of loss should still be less than that from a sheared beam port.

The reactor is scrammed either by failure of the pool pump or by reduced pool level resulting from the water loss.

Sufficient time should be available following the low-level alarm and scram to trip the pump and close the pool stop valve before serious loss of pool water occurs. The demineralized storage tank and the large raw-water hydrant are also available to compensate the loss.

In the event that the pool is drained and the heat exchanger within the pool is exposed, the operation of the convective loop is impaired. However, the reactor-loop pump continues to run following the scram and the reactor loop removes the decay heat. Also, the heat production in the reflector and rods due to fission-product decay gammas should not cause meltdown of these members, even with total loss of water.

3.6.4. Rupture of Reactor Loop

Protection against loss of core water resulting from rupture of the reactor coolant loop is provided by the check valve on the inlet line and by the invert loop on the exit line. A vacuuum-breaker valve at the top of the invert loop provides a siphon break, and a manually-operated stop valve is installed on the exit line.

All three valves are located within the shield wall or at its outer face so that a line rupture between the core and the valves is improbable unless it occurs within the pool.

A rupture anywhere within the pool causes the reactor to scram and causes the loop to drop to pool pressure. Loop water is not lost because pool pressure holds the vacuum-breaker valve closed and the external loop is presumably still intact.

Regardless of the location of the internal break, the shutdown core is cooled adequately by the pump flow or by the convective loop within the pool, depending on whether or not the pump trips.

A rupture external to the shield also causes the reactor to scram, and it activates the siphon break at the top of the invert loop. The downleg of the invert drains, and a head of water equal to the elevation of the invert is retained in the loop within the pool. Loss of flow resulting from the break activates the convective loop within the pool, and the shutdown core is cooled adequately.

In the event of activity release accompanying the conditions which caused the rupture, the manual stop valve can be closed to prevent activity leaking out the external break via the bleed line on the convective loop and the drained downleg.

3.6.5. Pump Failure

Failure of the reactor-loop pump causes scram and activates the convective loop which is adequate to remove the shutdown heat. An emergency pump is not required.

Failure of the pool-loop pump causes scram and causes the valve on the lower plenum to open, permitting pool water to circulate up through the reflector regions.

Failure of a secondary pump leads to higher water temperatures in the core and if these temperatures are excessive, they cause automatic run-in of the rods. In the event of total loss of the heat sink, the pool absorbs the shutdown heat, primarily through the convective loop.

3.6.6. Power Failure

The reactor is scrammed. Pump failure due to power loss or any other reason is handled adequately by the cooling system. Emergency power is not required for any of the pumps.

3.6.7. Test-Hole Usage

As discussed in Section 5.7.2. movement of samples in the test hole will be prevented during operation. Further restrictions with regard to size, material, and cladding will be developed and enforced to ensure adequate sample cooling and to avoid contamination of the pool.

3.6.8. Flooding of Beam Ports

The calculation based on the beam-port volume or water homogenized in the graphite indicates a small reduction in reactivity upon flooding the beam tubes. Despite the uncertainty of this calculation it may be concluded that the effect of flooding or emptying a beam port extension will be small.

3.6.9. Sample Movement

The results of the reactivity calculations performed for various compositions of the outer reflector suggest that reactivity changes resulting from motion of experimental absorbers in the graphite or beam ports will be minimized if the 3-in. layer of BeO remains intact. It is not intended to move this layer nor to locate experiments in it.

3.6.10. Refueling

Appropriate grappling tools, wrenches, and underwater lights will be provided to facilitate replacement of fuel elements. Although the fuel elements drawings indicate only a lifting bar, the depth and clearances on the vessel may make a rigid connection advisable. The fuel-handling tools will be specially designed for a suitable connection and they may have to incorporate a light to illuminate the interior of the vessel.

For refueling, the control-rod extensions would be disconnected and moved back to the pool wall by means of the traveling bridge, and the pool level would be lowered. The test-hole seal and the vessel lid would then be removed. Fuel elements would be transferred one at a time to the temporary storage racks on the pool wall. They would be transferred to the fuel-element storage wall only after the pool level had been raised back to its operating level.

Transfer of elements to shipping casks would be performed under water with the cask located on the shelf behind the weir. Transfer would be made only after such decay time that a horizontal element can be cooled adequately by heat transfer to air. Thus, in the event of an accident involving a dropped cask, the fuel elements would represent only a direct gamma hazard and the fission products would not be released. Analogy with tests performed on ORR fuel elements indicates that several days cooling time will be required following 10-Mw continuous operation.

All storage racks and casks will be designed to be safe with regard to criticality.

3.7. References

(1) Internuc 23, "An Advanced Engineering Test Reactor," March 15, 1958.

(2) IDO-16666, "Proposal for an Advanced Engineering Test Reactor," March 17, 1960.

(3) ORNL CF-59-2-65, "High Flux Isotope Reactor Preliminary design Study," February 27, 1959.

(4) Internuc 22, "High Flux Reactors for University Research," March, 1958.

(5) Nuclear Safety, March 1960, p. 35.

4.0. DESCRIPTION OF AUXILIARY SYSTEMS, FACILITIES AND BUILDINGS.

4.1. The Reactor Building

The reactor building will house the reactor, the operating staff, and experimental users. A control room will be provided from which the pool surface as well as the beam-hole floor will be visible. There will be three offices, a small laboratory, an equipment storage room, and an observation balcony for visitors to view operation of the reactor facility. There will be an elevator for pedestrian and small equipment transfer from one floor to another, as well as an overhead crane system of ten ton capacity (see Figures 4.1-4.4).

The control room of the reactor will house the instrumentation and control console. It will overlook the reactor pool in such a manner that the reactor operator has visual contact. The control console will also overlook the beam-hole floor allowing the reactor operator a view of the various experiments being conducted. A closed circuit television system will be installed for monitoring various experiments.

There will be a small laboratory for chemical and radioactivity analysis on the coolant water systems for the reactor.

Offices will be available within the reactor building for the reactor staff. Office facilities will be provided for the reactor supervisor, the reactor health physicist, the associate reactor supervisor, and a secretary. The offices of the secretary and reactor supervisor will serve as a collection point for records related to reactor operations.

A storeroom is provided for small electronic supplies necessary for routine maintenance of the reactor control systems.

There will be storage space provided in one wall of the beam-hole floor for the decay-storage of beam-port apparatus that has been removed from the reactor. Each of these storage vaults will be serviced by the off-gas system so that any radioactive off-gassing will be filtered, diluted, and disposed of through the stacks. These storage vaults will be closed with steel and lead covers to provide adequate shielding.

[redacted]

BELOW GRADE LEVEL (BEAM HOLE FLOOR)

SECOND FLOOR PLAN

UNIVERSITY OF MISSOURI
REACTOR LABORATORIES
COLUMBIA MISSOURI
CORNELIUS L. T. GABLER A. I. A.
AND ASSOCIATES ARCHITECTS
DETROIT MICHIGAN
SCALE 1/32" = 1'-0" JANUARY 15, 1961

0 5 10 15 20 25 50 75 100

FIGURE 4.1

[redacted]

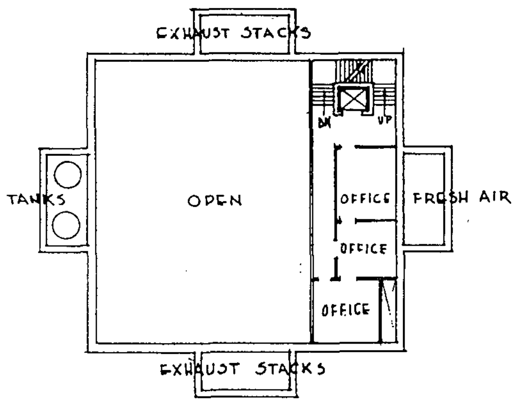
DOCK

FIRST FLOOR (GRADE) FACILITIES

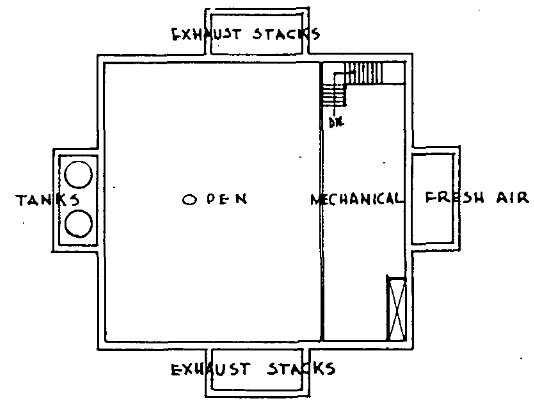
UNIVERSITY OF MISSOURI
REACTOR LABORATORIES
COLUMBIA MISSOURI
CORNELIUS L. T. GABLER A.I.A.
AND ASSOCIATES ARCHITECTS
DETROIT MICHIGAN
SCALE 1/32" = 1'-0" JANUARY 15, 1961

0 5 10 15 20 25 50 75 100

FIGURE 4.2



THIRD FLOOR PLAN



FOURTH FLOOR PLAN

FIGURE 4.3

UNIVERSITY OF MISSOURI
REACTOR LABORATORIES
COLUMBIA MISSOURI
CORNELIUS L.T. GABLER A.I.A.
AND ASSOCIATES ARCHITECTS
DETROIT MICHIGAN
SCALE 1/32" = 1'-0' JANUARY 15, 1961

0 5 10 15 20 25 50 75 100

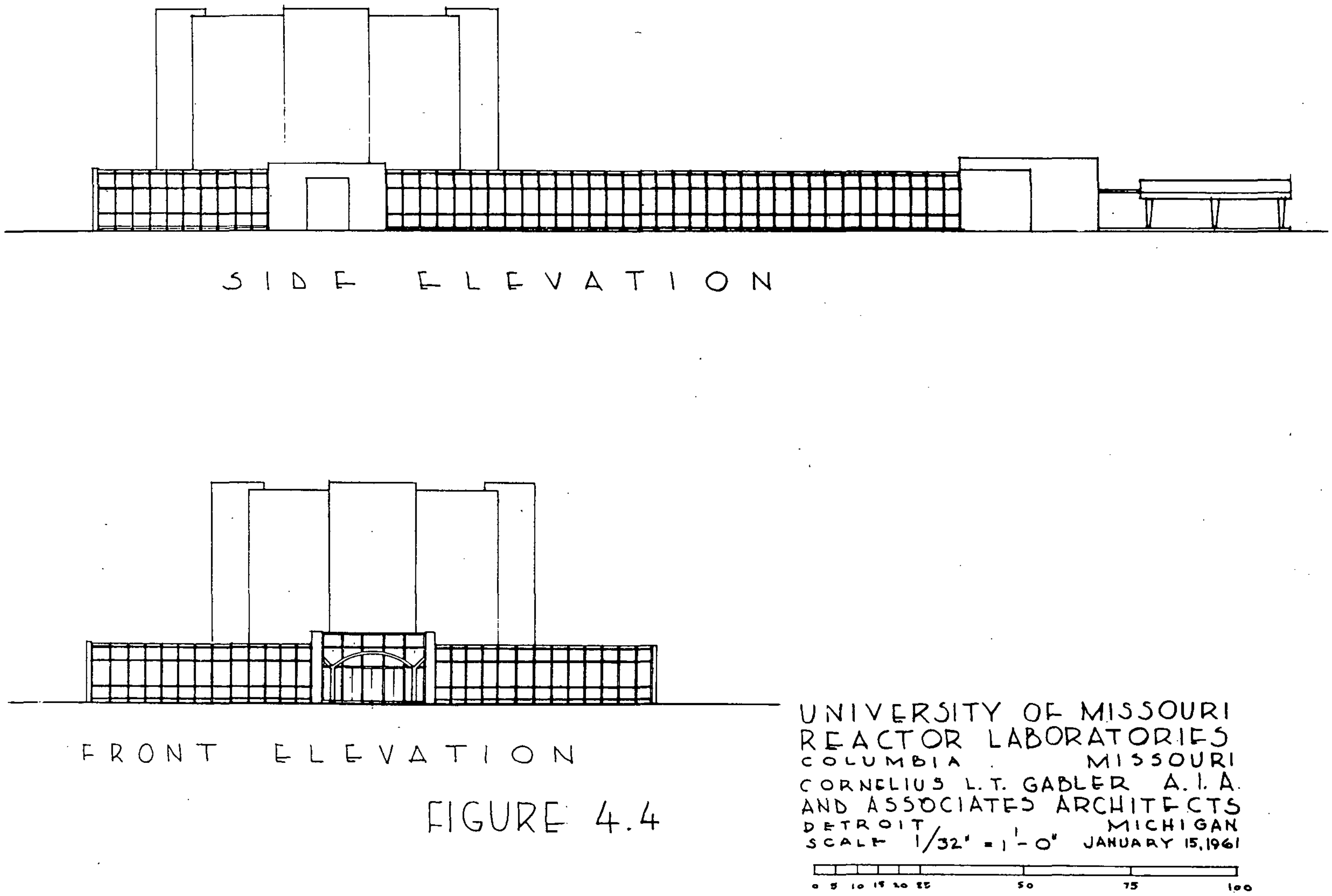

SIDE ELEVATION
FRONT ELEVATION
UNIVERSITY OF MISSOURI
REACTOR LABORATORIES
COLUMBIA MISSOURI
CORNELIUS L. T. GABLER A. I. A.
AND ASSOCIATES ARCHITECTS
DETROIT MICHIGAN
SCALE 1/32" = 1'-0" JANUARY 15, 1961
0 5 10 15 20 25 50 75 100

FIGURE 4.4

4.1.1. The Structure

The reactor building will be approximately 60 ft x 60 ft square of poured concrete as a monolithic structure. The exterior walls will be finished with a brick veneer. There will be attached to the walls two pilaster support columns on each of the outside walls to achieve the requisite structural strength.

The interior wall of the structural concrete, making up the reactor building, will be coated with a spray-on plastic finish. This spray-on plastic finish is catalyzed at the time of application from a spray-gun nozzle and hardens into a layer of impermeable plastic. The coating will withstand limited cracking of the structure without rupture, and it will form a gas-tight envelop on the interior of the concrete. This type of coating is presently being used to line structures to make them impermeable to water leakage. All of the internal surfaces of the concrete shell will be finished with this plastic.

4.1.2. Penetrations and Closure

Penetrations through the concrete structure will be limited to sealed electrical inlets, pedestrian and truck entry doors, and inlet and exhaust air ducts. All cooling-system water lines, heating steam lines, gas, air, and water will be brought in through two 4.6-foot-deep water legs, one of which will form a seal between the building and the adjacent laboratory structure, and another between the reactor building and the adjacent cooling equipment room. These water legs will function as a seal to 2.0 psi over-pressure in the building. In the event of over-pressure exceeding 2 psi the water would be elevated out of the water seal, partially flooding the floor in the heat exchanger room and then flowing back into the water seal to reclose the building after the release of the over-pressure. The water that is used to seal these legs will act as an absorbent of any radioactive off-gas in the event of a fuel rupture. This type of water seal must be replenished at intervals to make up for evaporation losses. It is proposed that a layer of plastic would be floated on the top of these water seals to minimize evaporation losses.

Water, gas, compressed air, and steam will enter the reactor building through the water seal between the building and the laboratory area. The water, gas, and compressed air lines will, of necessity, be under more than 2 psi of pressure to deliver these utilities to the user. In addition to the protection from contaminant escape afforded by this pressure, each of these service lines will be equipped with a check valve to further reduce the probability of back leakage.

The steam line, entering the reactor building through the water seal, will serve as a source of heat for the building. This will be a completely closed loop consisting of steam line in, heat exchanger to air, and a steam condensate return line through the water seal.

The pedestrian entry, at second level of the reactor building (Figure 4.2), will be closed by a single door which closes against an inflatable gasket. This door will be closed and latched from the outside with provisions to unlatch it either from the outside or the inside, and a gasket inflated to eliminate leakage around the door. There will be, as a part of the latching mechanism, an over-pressure relief device such that the door can be opened from the inside in the event that an individual is locked in the building.

The truck entry is located below grade and is serviced by a 15-ton hydraulic elevator (Figure 4.1). It will be closed against a solid rubber gasket at all times during reactor operation. This door will be under strict supervisory control and will be used for entering large pieces of equipment only during reactor down time.

The inlet and exhaust air ducts for building ventilation and air conditioning will penetrate the containment building at the fourth-floor level (Figure 4.3) from the mechanical area into the "fresh-air" plenum formed by the two pilasters. Two large valves (approximately 5 foot diameter), actuated by a compressed air cylinder, will be held in a normally open position. These valves will close (by gravity) when the air pressure to the operating mechanism is released as a result of any reactor scram or a building or area alarm. The valves will close against solid rubber gaskets.

4.1.3. Leak Rate and Testing

The building will be designed and constructed to insure a leak rate of less than 1% of contained volume in any twenty-four-hour period.

The measured accumulated leakage for a number of six-hour periods will be extrapolated to furnish the leak rate for 24 hours at 2-psi over-pressure. The pressure differential will be adjusted to 2 psi at the start of each one-hour interval. This will maintain a more uniform pressure differential than would be possible if the pressure were not returned to the starting value.

The test data will consist of the volume of air measured by a dry-flow meter (at 2 psi over-pressure) needed to return the reactor containment to a 2 psi over-pressure. This may be either added air or removed air, depending upon the temperature of the containment vessel. The temperature of the containment vessel will be recorded from 10 thermocouples located throughout the vessel. Barometric pressure will also be recorded.

The leakage rate will be calculated as the

$$\frac{\text{added volume of air/24 hours}}{\text{total volume of air}}$$

The added volume of air will be corrected to 2 psi at the temperature of the start of each of the one-hour intervals and the corrected data for the six-hour period will be extrapolated to a 24-hour leakage rate. The test will be conducted during the time of optimum temperature equilibrium, which is expected to be from 12 P.M. to 6 A.M.

The water traps will be checked during the test to ascertain that a pressure build-up of greater than 2 psi will cause air to be released through the water trap.

4.1.4. Containment Philosophy

The building will be designed and constructed in a manner such as to minimize the leak rate to a value of less than 1% for twenty-four hours. This containment and retention of gases will be achieved by means of a plastic inner liner applied to the concrete structure.

The concrete structure will give the building the strength necessary to resist a 2-psi internal over-pressure, while the plastic envelope provides a gas-tight enclosure. Plastic coatings will be applied as required to seal leaks or to attain the specified leak rate.

The utilization of the water-seal feature will permit entry of various utilities and supplies without requiring that each utility and supply be passed through a bulk-head type seal. The water leg will also serve as an over-pressure relief in the event of an incident and further, will serve as a scrubber should there be a release of fission products to the interior of the building. The water seal has the added advantage of renewing itself in the event of momentary rupture.

Reviewing the attributes of a double-door air-lock type entry system for pedestrians, it becomes obvious that such a pedestrian entry, though faithfully maintaining gas-tight integrity of the building, does not permit the rapid escape of the personnel contained within a building. Those personnel airlocks which are suitable for a reactor of this size will handle 12 or 15 people at a single pass. Any number of people greater than this will be confined in the reactor building for a variable period, depending on the number of people and the operating time of the airlock. It is felt that the immediate evacuation of all personnel from the reactor housing is of as great importance as is the instantaneous attainment of a seal.

It is difficult to postulate an accident in which an instantaneous release of fission products would occur. In most accident analyses there is a time delay between the accident, the over-heating, the fuel melt-down, and the release of fission products to the containment vessel. In light of these considerations it is felt that a simplified means of exit from the building, together with a gas-tight single-door closure constitutes an adequate containment vessel penetration for pedestrian exit and entry.

4.1.5. <u>Waste Disposal</u>

All radioactive liquid waste (or suspected waste) will drain from various drains within the reactor building to a sump located in the reactor building floor. From this sump these drains will be pumped through a pipe leading through the water-leg seal into one of three 5,000-gallon storage tanks located in the adjacent laboratory building. This particular waste-disposal system will handle all runoff wastes from the floor drains, from a drainage trench that runs around the base of the reactor, from the floor drain in the chemistry laboratory hood, and the drain lines from the various beam ports.

There will be a second disposal sump and drain system running from non-contaminated-water waste lines within the reactor building. These wastes will be lead to a non-radioactive sump and pumped from the sump through the water seal into the sanitary sewer drain line located within the adjacent laboratory building. A description of the techniques for liquid waste treatment is presented in section 4.2.6 on the laboratory building.

There will be an off-gas exhaust system servicing the off-gases from the pressurizer, the vestibules of the beam ports, the thermal column, the pneumatic blower, and the hood in the chemistry laboratory. This exhaust duct will lead to an absolute filter box, to a quick-closing and positive-sealing valve, and then through the containment shell wall. The duct will enter the mechanical equipment room in the vicinity of the exhaust fans. The exhaust from the system will be monitored and discharged through a stack. A description of the off-gas monitoring and discharge is given in Section 4.2.7.

4.2. The Laboratory Building

Surrounding the reactor building on four sides will be a complete laboratory facility for utilizing radioactive materials generated in the nuclear reactor. These facilities will include a hot cell, a cobalt-60 gamma irradiation facility, an x-ray unit, various radiochemistry laboratories, and supporting facilities such as machine shop, electronic shop, small animal rooms, offices, conference room, library, and lockers for laboratory personnel. The laboratory building will be a one-story structure built in the form of a square enclosing the reactor. A diagram illustrating the preliminary floor plan is presented as Figure 4.2.

The laboratory structure will have an exterior of brick and glass. Interior walls will be formed with cement block and the roof will be constructed with prestressed concrete beams. The laboratories will be arranged in the form of a "U" around a central mechanical equipment distribution area with utilities fed from this mechanical equipment area to each of the various laboratories.

At the front of the building, research users as well as the staff of the laboratories will enter through the lobby, pass from the lobby into the corridor, and to the Health Physics equipment area where they will procure their film badges and dosimeters. From this area they will go to the lockers, and from the lockers to the various laboratories or the reactor.

A portion of the total laboratory area will be excavated (Figure 4.1). This excavated area will surround the reactor building on three sides. To the back of the reactor, and below grade, will be the cooling equipment room for the reactor. On one side of the reactor building there will be a bank of ion-exchange resin columns for water purification for the reactor, a 15-ton freight elevator servicing the beam-hole floor, the hot cell, and the gamma irradiation facility. To the front of the reactor building, and below the laboratory mechanical equipment room, will be located the liquid-waste storage tanks and treatment equipment. There will also be floor space in this area for the temporary storage of active waste cans and contaminated objects.

The regenerant solutions for the ion-exchange columns will be mixed in an area adjacent to the cooling towers located outside of the laboratory structure. Regenerants will be pumped from the mixing tanks on demand to the various ion-exchange columns. The spent-liquid regenerant solutions (wastes) will be drained from the ion-exchange columns to a sump in the floor from which they will be pumped into the liquid-waste disposal tanks.

4.2.1. The Hot Cell

A single hot cell equipped with lead glass windows and a pair of model 8 manipulators will be installed in the laboratory. This hot cell will be constructed to handle 10,000 curies of cobalt or equivalent. Minimum facilities will be provided for research in the hot cell since there is not a well-developed program in existence for immediate use of this cell.

A number of sodium-vapor lamps on swing-out brackets will be installed to be pulled out for use as monochromatic light sources. A system of high-intensity fluorescent light fixtures will be installed as valance or ceiling-mounted lights within the cell. A half-ton electric crane will be installed within the cell. Penetrations for the insertion of experimental apparatus will lead from the face of the cell to the interior. All utilities, such as water, gas, air, and vacuum, will terminate on the face of the cell, and will be lead through a penetration port for use within the cell.

4.2.2. The Gamma-Irradiation Facility

Adjacent to the hot cell, and interconnected with it by means of a water canal, will be a cobalt- or cesium-irradiation facility. The walls of this radiation facility, including the entrance maze, will be constructed of concrete and have a thickness adequate to provide protection from a radiation source of 10,000 curies of cobalt or its equivalent. The interior room of the gamma-irradiation facility will be approximately 10 x 10 foot square with a ceiling height of not less than 13 feet. The source will be contained within a water canal located in the floor of the gamma room and extending beneath the wall into the adjacent hot cell. Source movement for irradiation purposes will be by means of an elevator mechanism bringing the cobalt source from below the water up into the room. The elevator mechanism will be coupled to the operation of electrical interlocks to eliminate the possibilities of unauthorized entry or use.

4.2.3. X-Ray Facility

To provide a total spectrum of radiation for research use, a 270-kev x-ray unit will be installed in the laboratory building. This x-ray unit will be a ceiling mounted device capable of continuous operation at 20 milliamps and 270 kev. The x-ray unit will be used for irradiating biological materials, for standardizing radiation doses, and for studying radiation effects. The x-ray tube will be equipped with a beryllium window to permit extremely high dose rates of soft x-rays as well as a filtration system to permit hardening of the x-ray beam for utilization of high-energy x-rays. It is felt that an x-ray unit of this type is essential as a "base standard", since a great deal of the existent literature on the biological effects of radiation is based upon a 250-kev x-ray dosage.

The power supply and control panel for this x-ray unit will be located in an adjacent room. The room housing the x-ray unit will be constructed in accordance with current radiological safety practices, and operation of the x-ray equipment will be by authorized personnel only. The door entering the x-ray unit will be interlocked with the control panel such that unauthorized entry will instantaneously shut down the x-ray unit.

4.2.4. Laboratories

A minimum of two radiochemistry-type laboratories will be installed. These laboratories will be designed for general radiochemical work, not to the demands or needs of any specific individual or any specific research program.

(In all instances the laboratories making up this structure will be designed for fields of use rather than for individuals presently on the campus of the University of Missouri.)

The general content of the chemistry laboratories would be laboratory bench work with at least 10 foot of open wall for the installation of specialized equipment as required. Each laboratory will contain a six foot hood. These hood installations will be exhausted into the mechanical equipment space immediately behind each laboratory where the exhaust will enter an absolute filter box. The location of these filter boxes within the mechanical equipment room will enhance the ease of changing of dirt-laden (and possibly contaminated) filters. The laboratories will be equipped with safety showers and chemical-type fire extinguishers. All laboratories will be maintained at a negative pressure with respect to the corridors.

A pneumatic-tube system will bring samples from the vicinity of the core of the reactor to selected laboratories in the laboratory building. A single pneumatic tube will be used to service adjacent laboratories by means of a switch-box assembly such that there will be a terminal in each of two laboratories.

A general engineering research laboratory will be provided. This laboratory will be fitted with necessary furniture, and it will be provided with a depressed floor area at one end to permit erection of columns or structures as long as 26 feet. Entry to this two-story area will be by means of a door from the below-grade floor and by means of a ladder from the engineering laboratory.

A general physics laboratory will be constructed. This laboratory will have a six foot fume hood as well as laboratory bench work. The furniture layouts and installed equipment will be developed in cooperation with the Physics Department.

An agriculture and a general biological research laboratory will be installed adjacent to each other. These laboratories will have the normal furniture and fume-hood complement comparable to that in the radiochemistry laboratories. These two laboratories will be serviced by a pneumatic tube from the reactor.

A minimum of two animal rooms will be provided for housing experimental animals. These animal rooms will be located adjacent to the mechanical equipment area where they will have their own ventilation and air conditioning system to minimize the problem of odors. A cage-washing room and a storage room will be installed between the animal rooms.

Two counting rooms have been designated to service the experimenters using the various laboratory facilities. One of these counting rooms will be for general use and will house a multichannel analyzer as well as general scintillation, geiger, and proportional counting equipment. The second counting room will be used to house special counting equipment such as a liquid scintillation counter, a vibrating-reed electrometer, and a four-pi low-background beta-counting assembly.

4.2.5. Supporting Facilities

A machine shop will be provided for the laboratory and reactor facility. The machinist supervising this shop will assist in the maintenance and preparation of reactor components, and in the construction of specialized pieces of equipment required for research utilization of the reactor. Adjacent to the machine shop there will be located a storeroom for parts, supplies, and stock used in the machine shop.

An electronic shop will be located on the opposite side of the storeroom from the machine shop. A full-time electronic technician will supervise and operate this shop for the construction of research equipment to be used by the experimenters as well as by the laboratory-reactor staff. The electronic technician employed to supervise the shop will also act in the maintenance, repair, and calibration of reactor control systems, the monitoring systems, and electronic experimental equipment.

The lobby will be at the front of the building in a position to control entry to the facilities. A receptionist-secretary will be on duty in the lobby to register technical and non-technical guests. There will be costumers provided for coats and hats within the lobby entry. The receptionist will also serve as a secretary and call director for telephone communications. There will be a master control for the intercommunication system located in the lobby at the desk of the receptionist.

Opening off the lobby will be the offices of the laboratory director and the laboratory supervisor. Adjacent to these offices, and to the lobby, will be a room housing duplication equipment. This room will also serve as a record-storage area.

Adjacent to the offices there will be a conference room capable of seating approximately twenty-five occupants. This conference room will be equipped with a chalk board along one wall, a pull-down screen for the showing of slides, a table, and a number of chairs.

On the opposite side of the lobby, space will be provided for the installation of a library facility. This library will be used to supply current publications on reactors, radiation, isotope utilization, and other aspects of radiation research. Adjacent to the library there will be provided a small lounge area for the use of the staff.

Offices for research personnel will be provided as two-man rooms adjacent to each of the laboratories. These offices will have one wall windowed such that the occupant may view activities within his particular laboratory. The offices will open off the corridors adjacent to the doors entering each of the laboratories, such that the occupant of the laboratory must leave his lab and re-enter his office. In this manner the practice of having books and desks within the isotope laboratory will be discouraged.

4.2.6. Liquid-Waste Treatment Facilities

Three five thousand gallon liquid-waste storage tanks will be installed below grade in the laboratory building. The location of these liquid-waste storage tanks is behind the hot-cell installation (see figure 4.1). The tanks will be constructed of a non-corrosive material or they will be internally lined with a non-corrosive lining. All radioactive liquid waste from the reactor building, from the ion-exchange columns, and from the various laboratories will be drained to one of these three tanks. Each tank will have a six inch top fill, a three-quarter inch bottom sampling port, a one-half inch compressed air mixing line, a two-inch vent leading to the contaminated off-gas system, a six-inch overflow, a two-inch drain at least six inches off the bottom, and a two-inch bottom drain.

As a single tank is filled with contaminated waste, the flow will be diverted to the number 2 tank in the system of three. At this time the waste accumulated in the number 1 tank will be sampled to determine the amount and identity of any radioactive contaminants. If these contaminants are of short half-life, the waste in the number 1 tank will be stored for a period of time to permit radioactive decay. If the contaminants within this tank are of longer-lived activities, or if there is a need for immediate treatment, a flocculating agent will be added to the tank. Mixing will be attained by means of compressed air, and the waste, together with the formed floc will be pumped through a "Cuno" filter system.

There will be two Cuno filter assemblies connected in parallel so that the flow may be diverted from one to the other as the floc is removed and the back pressure builds up.

The decontaminated waste will be pumped to the third (empty) tank, accumulated in this tank, then sampled to ascertain the quantity and identity of any radioactive contaminants remaining in the liquid waste. In the event that the contaminants remaining exceed maximum permissible dump levels, the waste will be pumped through an ion-exchange resin bed for further decontamination and returned to storage tank number 1. They will again be sampled and if still found to be radioactive they will be confined to fifty-five gallon drums for long-term storage. In the event that these liquid wastes are at less than maximum permissible dump levels, the waste will be pumped from the tank to the sanitary sewer.

Pumping from one tank to another and eventually from one of the tanks to the sanitary sewer system will be by means of a "Milton Roy" (or equivalent) variable-capacity pump. Two of these pumps will be installed to cover flow ranges from 0 to 313 gallons per hour, and from 100 to 690 gallons per hour.

Each of the liquid-waste storage tanks will be equipped with a level indicator gauge with the read-out device located in the vicinity of the hot cell. The level indicating gauges will have a scale reading from 0 to 5,000 gallons and an electric-trip mechanism to warn when the tanks are filled.

In no instance will radioactive liquid wastes be pumped to the sanitary sewer system unless the radiation level displayed is less than current permissible dump levels. In all instances the disposal of liquid wastes will take place during the normal eight-hour work day and this disposal will be accompanied by dilution from the sanitary sewer facilities of the laboratory and reactor.

4.2.7. Off-Gas Monitoring and Disposal Facilities

Many of the laboratories within the laboratory building will be serviced with isotope-type fume hoods. The exhausted air from these fume hoods will consist of 50% conditioned room air and 50% outside make-up air. This air will be exhausted from the fume hoods into a filter box located in the adjacent mechanical equipment room. The filter box will contain a fiberglass type pre-filter, followed by a "Cambridge" (or equivalent) absolute filter. The exhaust gases from these filter boxes will be directed through duct work to exhaust fans, and from the fans to the exhaust stacks. The exhaust stacks will be housed in two of the pilasters making up the structure of the reactor building (Figure 4.1 and 4.3). A scintillation crystal radiation detector will be installed within each of the two exhaust stacks. This crystal will be equipped with a thin aluminum window to permit detection of both beta and gamma radiation. The signal from the scintillation detectors in the two stacks will be taken to a dual-count-rate meter and from the count-rate meter to a two-channel recorder.

One stack will service the exhaust waste from the hot cell, the vents on the waste-disposal storage tanks, and from those laboratories making up the wing of the lab building above the hot cell. The second stack will dispose of the off-gases from the reactor building (which will consist of the exhaust from the pneumatic-tube system, the vestibules of the various beam ports, the pressurizer off-gas, and from the thermal-column vestibule) as well as the off-gases from the filter boxes for the various laboratories in the wing of the laboratory building opposite to the hot cell.

All off-gas will be disposed of through two stacks terminating above the roof of the reactor building, and approximately forty feet above the roof of the laboratory facilities. Each discharge stream will be diluted with 6,000 cfm of building air. A model will be constructed and smoke tests made to ascertain that an adequate flow and dilution pattern is achieved with this stack arrangement.

The largest activity source from the reactor building is the A^{41} discharged from the pneumatic tubes and from the empty volume of the beam ports. With continuous 10-Mw reactor operation and continuous air flow through the exposed volumes, the discharge rate of A^{41} is:

$$Q_c = \Sigma_a \bar{\Phi}_{th} \lambda V = 2.66 \times 10^7 \frac{\text{dis}}{\text{sec}^2} = 718\ \mu c/\text{sec}$$

where $\Sigma_a = 1.32 \times 10^{-7}\ \text{cm}^{-1}$ for A^{40} in air

$\lambda = 1.07 \times 10^{-4}\ \text{sec}^{-1}$

$V \approx 2.04 \times 10^4$ cc total volume

$\bar{\Phi}_{th} = 9 \times 10^{13}$ at 10 Mw

The maximum ground concentration from a stack h meters in height, discharging at the rate Q_c is:

$$\chi_{max} = \frac{C_z}{C_y} \quad \frac{2\ Q_c}{e\pi\ \bar{u}h^2}$$

If $C_z = C_y$ and the wind speed, $\bar{u}$, is 1 meter/sec, the height required to reduce χ_{max} to the tolerance value of $2 \times 10^{-6}\ \mu c/cc$ is found to be h = 9.17 meters or 30.1 feet.

If the exposed volume is discharged intermittently so that appreciable decay occurs before discharge, the time-averaged dose rate will be less than with continuous discharge. If the argon is saturated and then released instantaneously, the total activity release is given by:

$$Q_I = \Sigma_a \bar{\Phi}_{th} V$$

and the maximum total integrated dose is given by:

$$TID_{max} = \frac{C_z}{C_y} \quad \frac{2\ Q_I}{e\pi\ \bar{u}h^2}$$

Thus the dose from a total discharge of saturated air from a 30 foot stack would be $1/\lambda$ times the dose rate corresponding to χ_{max}, or 2.6 hr times about 1.4 mr/hr = 3.6 mr.

The proposed stack provides a dispersion factor four times greater than would a 30-ft stack. Building interference is not thought to represent a problem because the stack velocities resulting from 6000 cfm of diluent air should cause the effective stack height to be considerably greater.

4.3. Utilities and Services

The various services to the reactor site will be brought in through a tunnel leading from the site to a point on Stadium Road, approximately 1600 feet due north of the site. These utilities will consist of steam and a steam-condensate-return line, gas, sanitary sewer, water, electricity, and phone lines. The present location of each of these utilities is described in the paragraphs which follow.

There is a steam line which presently terminates approximately 200 feet southeast from the Medical Center, adjacent to the south side of Stadium Road. This is a 12-in. line (and condensate return) which is capable of handling the additional loading from the research reactor facility.

Water is available approximately 800 feet to the northeast of the site from a deep well, with an inter-connection to three other university wells to assure continual pressure. At the same location there is an underground duct carrying electrical power for the well with a spare duct available for the lines necessary to bring in electrical power to the research reactor facilities.

The sanitary sewer runs parallel to and on the south side of Stadium Road, approximately 1500 feet due north from the site. The storm sewer runs from northeast to southwest approximately 1000 feet north of the reactor site.

Gas is available from a 6-in. high-pressure distribution line on the north side of Stadium Road.

Steam, gas, sanitary sewer, and telephone communications will come in through a tunnel from Stadium Road to the reactor site. There will be a right-angle branch off from this tunnel at a point approximately 500 feet north of the reactor and extending 300 feet to the east. This feeder to the main tunnel will bring water and electricity into the tunnel and from there to the site.

Phone conduits presently terminate at a point approximately 200 feet south of the Medical Center. These conduits will be extended to the east about 300 feet where they will enter the tunnel.

The University of Missouri has its own electrical power-generating equipment located in the Physical Plant Department on the corner of Stewart Road and 5th Street. This electrical-generating equipment is tied in to the City generating equipment such that a failure of either one results in a partial carrying of the load by the other. In addition to this dual supply, there is an emergency generating system consisting of two 500 kilowatt diesel-driven units located at the Physical Plant Department. These diesel units are kept on a stand-by basis and used to provide emergency power to certain buildings on the campus in the event of failure of both the University and the City generating systems. The two 500 kilowatt diesel generators are connected through a distribution panel at the Plant Department to pick up emergency loads as required; first priority is given to the University Hospital whose maximum loading is 600 kilowatts. The Research Reactor Facility would have a high-priority rating for this source of emergency supply.

5.0. ORGANIZATION AND ADMINISTRATION OF THE FACILITIES

5.1. General Description

The reactor and laboratory facilities, described in detail in prior sections of this report, will be available to any faculty member or graduate student interested in pursuing research involving radiation, radioisotopes, or the reactor. The diverse research programs initiated will be coordinated and health and safety supervision provided by the permanent staff employed to operate the facilities. The research staff, composed of faculty members and graduate students, will be semi-transient, since no permanent assignment of space or facilities will be made. Administration of the reactor and laboratory will be divorced from the research programs to eliminate the possibility of a compromise in safety for the sake of experimental expendiency.

The research reactor facilities staff is divided into two groups. One group, the reactor-operating group, has responsibility for the daily operation and maintenance of the reactor. The second, the laboratory-operations group, has responsibility for the supervision and maintenance of the laboratory facilities associated with the reactor. Coordination and overall adminstration of these groups is performed by the director of the project. A table of organization for the reactor facility is presented as Table 5.1. It is estimated that the staff requirements for operation of these facilities on an eight-hour day, five-day week, will be twelve to sixteen people. These people will provide operation and supervision of the facilities for all research personnel, expected to number from thirty to sixty people.

The research reactor facility fits within the organizational structure of the University of Missouri as a separate entity (Table 5.2). The various operating and service groups of the University can be divided into three major categories. One of the major categories is the academic departments, each of which is supervised by a dean. A second group of operating subdivisions within the University are classified as academic service organizations. This group includes the University library, television station, University Press, etc., and will include the Research Reactor Facilities. Each of these departments has an assigned director who is responsible for the administration of his particular division. The third category is termed non-academic services and include such things as physical plant, budget, construction, auditing, etc.

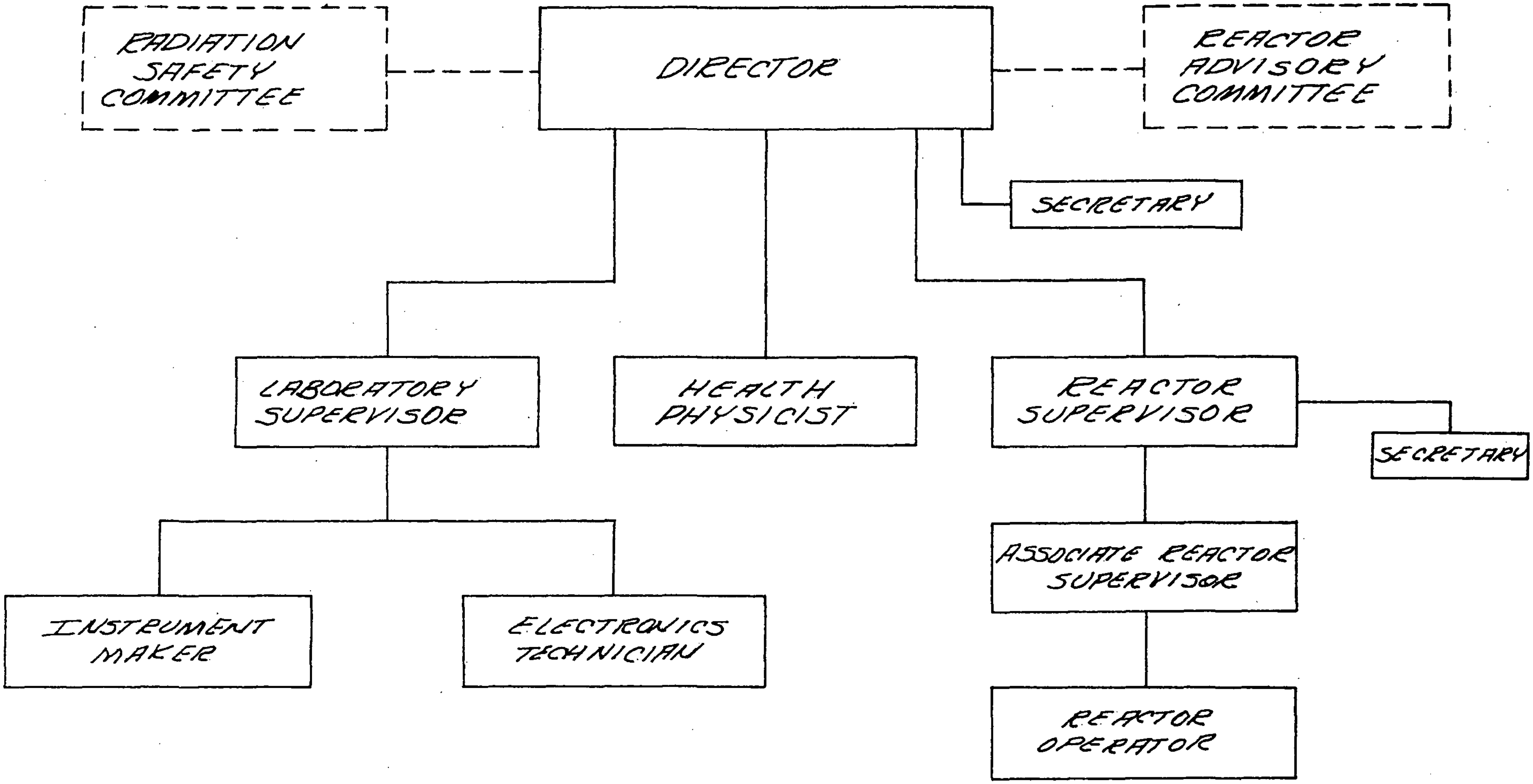


TABLE 5.1 PROPOSED TABLE OF ORGANIZATION FOR THE UNIVERSITY OF MISSOURI RESEARCH REACTOR FACILITIES

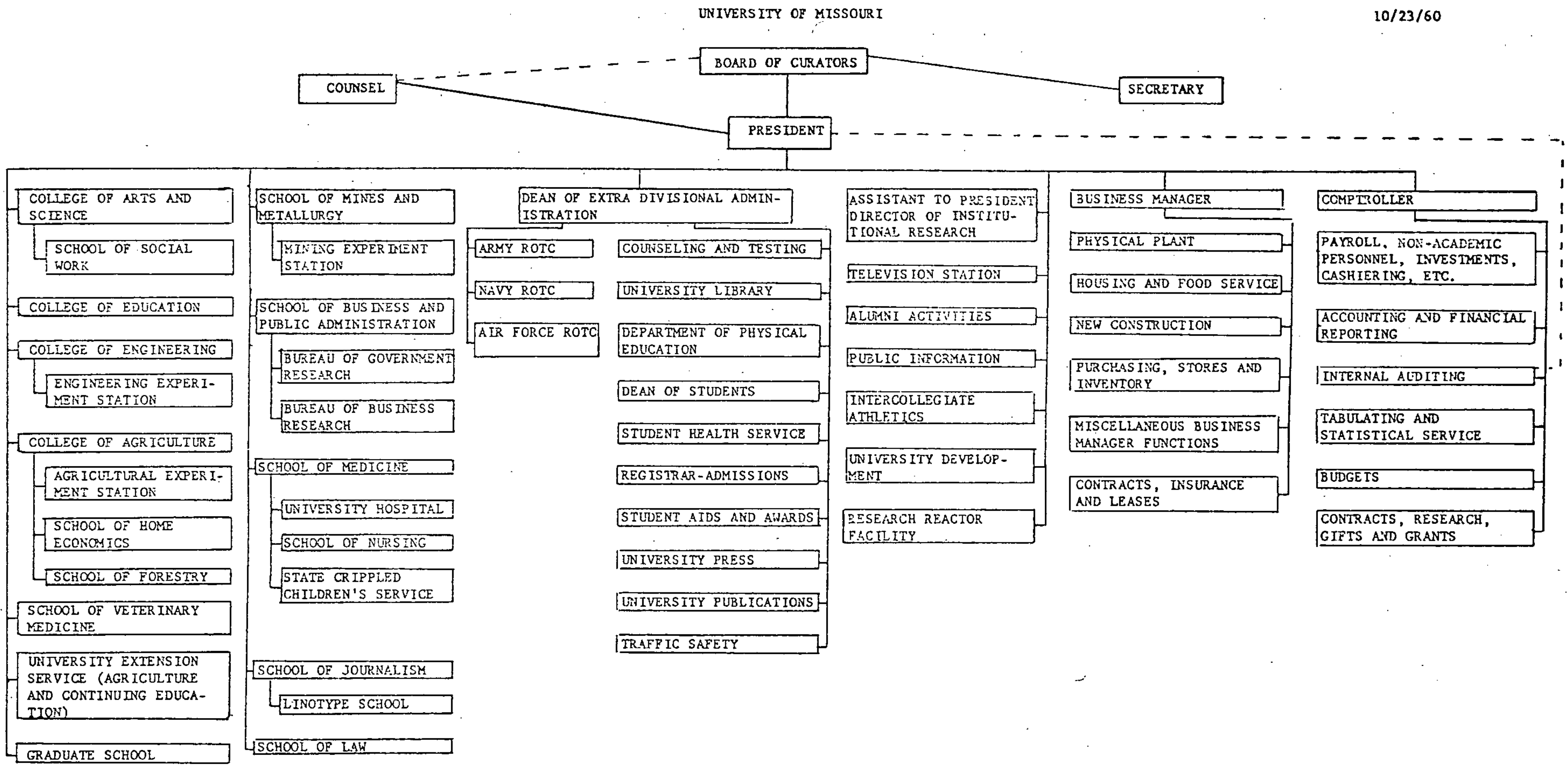


Table 5.2 Organization of the University of Missouri

5.2. Organization

5.2.1. The Director

The Director of the Research Reactor Facilities will report directly to the President of the University. The Director will have overall responsibility of the reactor and all associated laboratories. He will supervise, through his assistants, the operation, as well as the utilization and maintenance of these facilities. He will prepare and administer the budget, he will negotiate license amendments, fuel procurement, and any industrial contract arrangements for use of the reactor facility. He will interview, negotiate with, and employ the necessary staff for operation of these facilities. He will be available to all faculty members for consultation on research proposals for the utilization of the reactor or associated laboratory facilities.

5.2.2. The Reactor Advisory Committee

The Reactor Advisory Committee will function during the designing, planning, and construction stages as a review and advisory committee. As construction progresses and is completed their function will be altered. After construction is completed they will serve 1) as a review committee for reactor operations, 2) in the review of unusual experimental proposals to evaluate the hazards associated with these proposals, 3) as an advisory group for the selection of major staff personnel, and 4) in the disbursement of a limited amount of research monies to faculty research people. Appointments to this committee will be for a period of one year. This committee will meet once monthly to review reactor operations and consider special problems.

This committee will consist of the Director of the laboratory facilities, the Supervisor of the reactor, the reactor Health Physicist, and five faculty members appointed by the President of the University. The Director of the laboratory facilities will act as chairman of this group.

5.2.3. The Reactor Supervisor

The Reactor Supervisor will have responsibility for operating the reactor in a safe, reliable, efficient, and economical manner, and in a manner such as to make the fullest possible services available to experimenters.

He will be responsible for instructing personnel in their duties, for issuing the operating schedule, for personally supervising non-routine reactor operation, for establishing a procedure for the accommodation of experiments, for the review and processing of applications for radiation previously approved by the advisory committee, for advising on safety aspects of proposed experiments, and for the provision of information to technical visitors regarding the reactor facility and its operation. He will also be responsible for the safety of experimenters, students in laboratory courses, and students pursuing graduate research. He will report to the Director of the laboratory facilities on all phases of reactor operation.

5.2.4. The Laboratory Supervisor

The Laboratory Supervisor will supervise the activities being conducted in the various laboratories associated with the reactor. He will maintain film badges, he will teach and supervise the use of the hot cell, the gamma facility, and the pneumatic tubes. He will assist experimenters with 1) the design of their experiments, 2) the selection of proper instrumentation, and 3) the evaluation of radiation hazards. He will supervise the decontamination, packaging, and disposal of radioactive waste. The laboratory supervisor will have a background in Health Physics.

5.2.5. Reactor Health Physicist

The Reactor Health Physicist will report directly to the Director of the Research Reactor Facility. He will be a member of the reactor-operating group, but will be empowered to override the reactor supervisor in questions of reactor safety. His duties shall include routine evaluation of water-contamination levels in the primary loop and in the pool-water system. He shall be responsible for day-to-day surveys of experiments associated with the reactor. He shall be responsible for the evaluation of shield effectiveness on beam-hole experiments. He shall serve as an advisor for the design of experiments to be conducted in the reactor facilities. He will perform daily checks of all the permanently-mounted radiation-monitoring facilities associated with the reactor, including the fission-product water monitor in the primary loop, the stack monitors, and the area-survey monitors.

5.2.6. Reactor Operators

Two reactor operators will be employed for eight-hour day operation of the reactor. These reactor operators will be selected such that one has a background in electronics work and the second has a background in machine shop techniques. The two operators will be trained by members of the staff and by attendance at an operator's training course. They will serve on a four-hour on, four-hour off schedule, and each will have available a student or technician to assist during his four-hour operation stint. During their four-hours off operation, these reactor operators will perform routine maintenance functions under the general supervision of the machinist or the electronics technician in charge of the shops.

There will be two shops established for the maintenance of the reactor and laboratory equipment, as well as to meet the needs of research personnel utilizing these facilities. The shops will be under the direct supervision of the Laboratory Supervisor. He will approve expenditures and make periodic evaluations of operations. The shop personnel will construct, or supervise the construction of, complex pieces of research equipment. They will also provide routine maintenance as required for the reactor and all instrumentation in the various laboratories associated with the reactor.

5.3. Training of Staff

The organization outlined in the preceding paragraphs constitutes an adequate staff for the operation and maintenance of the facilities. Of the members of this staff, the University would propose to hire, in addition to the Director, A reactor Supervisor and a Laboratory Supervisor who have had past experience with a megawatt power level research reactor. The University will seek, for the position of laboratory supervisor, an individual who has had experience with a research reactor, and who also has a background in health physics. The reactor supervisor will preferably be an individual who has supervised a research reactor in a university environment and whose background is strong in electronics and reactor control systems. The Director, who is presently on the University of Missouri project, has had experience with a research reactor at the University of Michigan, and prior to that, had health physics experience at Oak Ridge National Laboratory and at the University of Michigan. These three people, the Director, the reactor supervisor, and the laboratory supervisor, will constitute the nucleus of the staff. They will have experience in the operation of research reactors and research reactor facilities.

The reactor operators will be hired and a teaching and training program instituted at the University of Missouri. A machinist and an electronic technician will be hired before the completion of the construction of the reactor facilities. In this manner, these two individuals can procure valuable experience in the final construction and installation of machinery, as well as with the electronics associated with the control of the reactor. The machinist and electronic technician will be trained in the first group of reactor operators to permit them to procure an operator's license so that they might be better prepared to do maintenance on the reactor. In addition to the machinist and electronic technician, two reactor operators, would also take part in the first teaching and training program for operators.

The training program for operators will consist of a series of lectures on reactors, reactor safety, general health physics, reactor control systems, and postulated reactor malfunctions. These lectures will be coupled with a series of experiments which will demonstrate the utilization of health-physics equipment, the operation of components of the reactor systems, and the operation of the reactor controls. This program of training should prepare these people for examination for the operator's license.

If it seems advisable, an operator may be sent to Rolla, Missouri, where the University will have a ten-kilowatt teaching and training reactor in operation. This individual would spend six weeks at Rolla, procuring the fundamentals of reactor operation prior to criticality of the research reactor.

An alternate technique for procuring one trained operator prior to criticality, would be to send one individual to the University of Michigan, where he could take a two-month training course in reactor operation, and serve as an apprentice to one of the experienced operators on the Ford Nuclear Reactor.

5.4. Research Personnel

Personnel in attendance at the University of Missouri to work with the reactor or with the reactor staff for research or for training purposes, shall have either student or guest status depending on their qualifications as determined by the University of Missouri Admissions Office. All persons participating in training or research activities with the reactor shall appear on the official rolls of the University. In every instance, these people will be provided with the same supervision and radiation protection devices as full-time faculty employees of the University of Missouri.

5.5. Transient Visitors

It is expected that the Nuclear Reactor Facility will be an attraction for social and professional organizations in the State of Missouri. It is intended that the reactor and related laboratories will be designed in such a manner that these guests can be handled. A lobby will be provided which will receive 50 guests, permitting the registration and handling of these people and the storage of their coats and hats. The various laboratories will have windows so that it will be possible for visitors to view operations in these rooms.

An open-house will be held once each year to familiarize the townspeople of Columbia with the operations and activities being carried out in this facility. For other groups of visitors (Cub Scouts, Girl Scouts, various fraternal organizations, science classes, etc.) there will be one afternoon of each week open for visitors between the hours of 1 and 5 P.M. These non-technical visitors will be guided throughthe facility by a trained student guide. This guide will be equipped with the normal personnel-monitoring equipment and he will be instructed to stay with the group of 12 to 25 people assigned to him. He will take the group on a pre-assigned path through the facilities. Variations would be made in this path to insure that no visitor enters an area in which there is loose contamination or a possibility of a radiation exposure. In instances where there is experimental work being carried out on the beam-hole floor, those areas which represent exclusion areas to visitors would be roped off to eliminate the possibility of a visitor inadvertently entering.

Technical visitors will be accompanied through the facilities by members of the staff. Those technical visitors desiring to view a particular experiment or radiation facility will be equipped with film badges at the front desk in the lobby of the laboratory facility. Technical visitors who have no desire to view the facilities but who wish only to speak to members of the staff or use the library facility, will not be required to have a film badge for intermittent visits. If it is their intent to work in the library or any of the facilities for an extended period of time, they must comply with the established regulations for use of the facilities.

5.6. Policy with Regard to Reactor Utilization

The research reactor facilities shall be available for research utilization by any member of the faculties of the University of Missouri, including the School of Mines and Metallurgy, and of the Universities comprising the Mid-America Association of State Universities. Priorities for the use by the faculty members of any specialized facilities on the reactor shall be established by the simple technique of "who asked first." In the event that questions arise as to the advisability of such a priority assignment, these questions will be negotiated with the Reactor Advisory Committee, and their findings willl be final.

All reactor utilization by faculty members will be subject to the supervision of the Reactor Supervisor, and the Reactor Health Physicist. Their decisions and recommendations having to do with radiation safety or safety of reactor operation may be negated only by appeal to the Reactor Advisory Committee. With respect to faculty, graduate students, other educational institutions and industrial contract research, the priority for use shall be as follows:

1) Faculty and graduate students of the University of Missouri and of other universities in the Mid-America Association of State Universities.

2) Faculty and graduate students of other educational institutions in the state.

3) Faculty and graduate students from out-state educational institutions.

4) Industrial contract research.

Graduate students, under the direct supervision of a faculty member of the University of Missouri, will be urged to utilize the research reactor facilities in their research programs. Space and radiation facility assignments will be made by the Reactor Supervisor in consultation with the chairman of the graduate student's research committee (or his advisor). Routine irradiations for graduate students who have demonstrated competency in the handling of radioactive materials will be performed without the approval of his faculty advisor.

The basic necessities for research will be provided by the reactor staff. These necessities will consist of the necessary utilities for driving the experiment, the handling tools associated with the reactor for the transfer or handling of intensly radioactive materials, and radiation experiment supervision by the reactor staff. There will be a stockroom in the laboratory building from which the experimentor may requisition glassware, chemicals, and other assorted items of research equipment. Specialized equipment for the utilization of the reactor in a research program may be constructed in the facility shops or it may be provided by the department sponsoring the particular graduate student.

Faculty members or graduate students from other educational institutions in the state may utilize the research reactor facilities upon written request to the Director of the laboratories. They will be subject to the same supervision and control as members of the University of Missouri faculty. While in attendance in the reactor facility, they will be registered at the University of Missouri as either students, guests, or trainees. This registration must be established through the University Admissions Office.

Whenever possible the research reactor and its associated facilities will be available to industrial users from the State of Missouri. Industrial users shall make application to the Director of the facilities for the use of these facilities. This application should include a detailed description of the proposed experiment, together with a suggested time schedule and a listing of the necessary equipment that it is expected would be furnished by the University of Missouri. The Director of the laboratory facilities will provide interested Missouri industries with a price schedule for reactor services. This price schedule will reflect costs to the University for the operation of the research facilities. It should be pointed out that the industrial users are members of the priority group, which is to say that all educational research activities take precedence over industrial research. It should further be pointed out that the University of Missouri, as an educational institution, is interested primarily in the procurement of industrial research which will further the teaching and academic research programs.

5.7. Experimental Programs

The research reactor and its associated laboratory facilities will be available to any faculty member or graduate student interested in pursuing research involving radiation, radioisotopes, or the utilization of the reactor. Space assignments will be made on the basis of order of application and degree of hazard associated with the proposed program. No permanent assignment of space will be made. No laboratories are being designed for specific individuals. In every instance, an effort is being made to satisfy the requirements of a cooperative research program fitting the needs of the total University. The research staff occupying these facilities will be semi-transient, leaving the laboratory and returning to their own departments when they complete their research programs.

Office space for research personnel will be provided in areas immediately adjacent to the various laboratories. Adequate shop facilities for the construction of special research equipment will be provided in the laboratory building adjacent to the reactor.

It is impossible to visualize the exact nature of the programs which will be undertaken with the installation of these facilities. A general survey has been made of all potential users of the reactor presently on the campus. The evaluation of the proposals submitted indicates interest in the following areas of research:

Radiobiology;
Treatment of malignant diseases with neutrons and with radioisotopes;
Research with isotopes in soils;
Food preservation studies;
Genetic effects of neutrons and gamma radiation;
Investigations in plant and animal metabolism;
The applications of tracers in drug studies;
Neutron diffraction investigations;
Investigations in solid-state physics;
Radiation and tracer studies in sewage treatment;
Radiation damage to structural materials; and
Activation analysis.

In many instances the proposed programs of investigation have been very carefully outlined, and in others the proposals were much less specific. The distribution of these proposals indicated the research strength which the University of Missouri possesses in the fields of medicine and agriculture. It appears that there will develop at this facility a very strong research program in these two fields.

In light of the difficulty of foreseeing the future research programs it appears more pertinent to describe the various experimental facilities which will be provided as a portion of this research reactor and the general techniques which willbe followed in the utilization of these facilities. This procedure is followed in the paragraphs which follow.

5.7.1. General Comments on Facility Use

Simple (routine) irradiations will be performed with a minimum of formality. To illustrate: The utilization of the pneumatic tubes will be preceded by the filling out of a form indicating the identity, the weight, and the cross section of the particular material it is desired to irradiate, together with the experimenters name, pneumatic tube number, power level, and irradiation time. Samples to be irradiated in insert baskets will utilize the same form. The reactor operator will indicate on this from the position of the sample in the reflector irradiation positions adjacent to the reactor.

The experimenter must secure approval from the Radiological Safety Officer of the University of Missouri for the irradiation of materials to be removed from the confines of the laboratory to existent laboratories on the campus. At the conclusion of reactor irradiation, the Radiological Safety Department will be notified, and they will remove the sample to its laboratory location on the campus.

Safe use of the reactor requires that more complicated experiments be approved by the Reactor Advisory Committee. It is the prerogative of the Reactor Supervisor to make the decision as to whether committee review is required or not. The Reactor Advisory Committee will meet once each month to consider routine operations as well as special requests for irradiation services, and to make recommendations pertaining to the procedures to be used on these extraordinary irradiations.

5.7.2. Center Test Hole

The center test hole or flux-trap facility can provide a large reactivity effect if used without proper restrictions and supervision. Consequently, use of this facility will be subject to a high degree of administrative control to minimize the possibility of either inserting a sample which would increase reactivity, or inserting a sample of high poison content and then removing it during reactor operation.

To forestall the possibility of either of these happenings, samples must be inserted or removed only during reactor shutdown and means will not be provided to move samples during operation.

The number and volume of samples will be limited mechanically by the design of the center-hole liner or canister which holds the samples to be irradiated. This liner consists of a hollow aluminum tube assembly which will be inserted in the center test hole previous to startup. The assembly will essentially fill the test hole and will seat at the bottom of the test hole and be held in place by the flow. One or more short sections of the tube will be equipped with transverse screens which form sample compartments to fix the sample position, yet allow adequate cooling. The tube assembly will be perforated throughout its length to facilitate mixed flow. It will have a screen at the top to preclude the possibility of dropped objects entering the center test hole while the assembly is in place.

All sample insertions into the flux-trap position will be subject to previous review by the reactor supervisor. The reactor supervisor will supervise insertion and removal of all samples during shutdown, and he will see that either a canister is in place, or that a screen is provided over the test hole previous to startup.

A form, similar to that utilized for pneumatic tube irradiations, will be used to describe the sample which the experimenter wishes to have inserted. Applications will be reviewed carefully with regard to size and material to predict reactivity effects, and with regard to size and cladding or containment to insure adequate cooling by pool water and to avoid contamination of the pool.

Standards for approval will be established on the basis of subsequent reactivity studies. Applications involving extraordinary conditions will be referred to the Reactor Advisory Committee.

Alternate means for usage of this facility would be provided only after operational experience has provided complete data on this facility, and after receiving AEC license amendment approval of the suggested revision to the facility.

5.7.3. Beam-Port Facilities

Six beam-port facilities are provided, as described in Section 3.2.5.2. Four beam ports are "radial" and two are "tangential" in the sense that they terminate below the core and do not view the reactor core directly.

The primary usage of the beam ports will be for neutron and solid-state experiments. Neutron beams can be brought out of the reactor through filters and collimators installed in the tubes. Experiments, including solid-state structure evaluations by neutron diffraction, cross-section measurements, optical properties of neutrons, and n-gamma reactions, will be conducted using neutron beams extracted from these tubes.

When the beam ports are used with collimators and filter assemblies, surveys of each experimental installation will be made to ascertain that the beam catchers and shielding barricades are adequate to control radiation hazards. In addition to the routine surveying techniques, there will be permanently installed radiation monitors at various locations on the beam-hole floor to detect any radiation leakage from beam-port experiments.

The tangential beam ports will be used in neutron beam work in which it is desired to minimize the fast neutron and gamma-ray background. The beam-port plug assemblies will be identical to those used in the radial beam ports. The experimental work performed in these tangential facilities will be comparable to that in other beam ports.

In some instances the beam ports will be loaded with long-term irradiation experiments. These experiments will be contained in an experimental can which is an extension of a modified shield plug. These modified shield plugs will have helical conduits through them to admit electrical leads and utility lines to the experiment can. A water line will be put through all shield plugs, whether for experimental use or shielding. It will be possible to flood the port, or to provide cooling water to an experiment, by means of this water line.

It is pertinent to point out that all changes in beam-port experiments, and intentional flooding or draining of the beam tubes, will be performed only when the reactor is shut down. All beam-port experiments are semi-permanent in nature, and in no instance could the contained experimental apparatus be withdrawn rapidly.

In the event that experiments are proposed of a loop type, the Reactor Advisory Committee will review these experimental proposals and make recommendations regarding the critical parameters which must be monitored and recorded, such as temperature, pressure, or coolant flow, and which measurements, if any, should be used to activate an annunciator or the reactor scram system. Spare inputs to the reactor scram system and circuits leading to the beam ports will be provided for this purpose. All design and operation procedures for loop experiments must be reviewed by the Reactor Advisory Committee.

A number of beam-port apparatus storage holes will be provided in one wall of the reactor building. These storage holes will be shielded with concrete and closed with lead and steel covers. Transfer of radioactive apparatus from a beam port to the storage hole will be made in a movable shielded coffin.

5.7.4. Shield-Plug Facility

There will be a shield-plug facility provided which will ultimately be used as a graphite-filled thermal column. The initial installation will consist of a cavity in the shield wall filled with barytes blocks and a small beam-tube penetration to the aluminum tank liner of the reactor pool. The shield blocks would eventually be removed and replaced with graphite stringers to construct a typical thermal-column facility.

In light of the present trend amongst various university reactors toward minimum use of a thermal-column facility, it is proposed not to install the graphite initially. Further, the University of Missouri medical group may, at a later date, wish to install shields and shutter assemblies to permit irradiating large animals for the investigation of neutron beams on tumors.

Then the initial installation of barytes-concrete shield blocks offers no problem other than the requirement that this section of the shield be adequately surveyed at start-up to ascertain that there are no radiation leakage paths. The cavity containing the shield blocks will be lined with aluminum plate and the inner end of the cavity will be the 3/8" aluminum pool-liner wall. The shield-block cavity will be closed with a shield door on the outside face.

When detailed plans are formulated for installing either the requisite equipment for medical irradiations or the graphite stack for a thermal-column installation, an addendum to the initial license will be submitted.

5.7.5. Pneumatic Facilities

The reactor will be equipped with four pneumatic tubes which terminate in graphite-reflector positions external to the core. These tubes will be used for experiments which require very short exposures of precise duration, as well as those irradiations which require rapid sample recovery. These systems will provide neutron irradiation of samples for activation analysis.

Procedures for insertion of a sample will begin with completing a form requesting a service irradiation. This form will be submitted to the reactor supervisor, or, if it is a repeat of a previous irradiation, to the reactor operator, the reactor supervisor will review and initial the form, and deliver it to the reactor operator.

When the reactor is at the power requested for the given irradiation, the reactor operator will notify the investigator that he may send his sample from the sending-receiving terminal in the laboratory into the reactor. This transport will be performed by actuating a switch in the laboratory at the pneumatic-tube terminal. This switch is active only after the reactor operator has energized the blowers to operate the pneumatic system. Thus, ultimate control of sample insertion via the pneumatic tubes rests with the reactor operator.

Samples for irradiation in the pneumatic tubes will be placed in special aluminum or plastic carriers and drawn into position within the reactor reflector by a low-pressure pump. The system will be fast, permitting of immediate chemistry in the hoods to which the samples will be delivered after irradiation. The positioning of these terminals in the hoods of selected laboratories will minimize the possibility of particulate contamination being released, as might be the case from a central terminal box. This pneumatic-tube system will see heavy use in the study of short-lived isotopes and for routine irradiations of small sample volumes. The position and size of the pneumatic tubes is such as to preclude any important effect on reactivity due to the rapid insertion and removal of samples.

Personnel who desire to use the pneumatic-tube systems will be instructed in their use by the laboratory supervisor. In initial operations the laboratory supervisor will observe the utilization of these facilities and see thatthe necessary radiation protection precautions are followed. Research personnel will be instructed regarding the useful life of plastic containers subjected to irradiation. They will be requested to maintain a record of the time of exposure of plastic containers and dispose of the container prior to its known failure point.

5.7.6. Reflector-Irradiation Baskets

There will be a total of eleven graphite reflector elements which will be removable and replaceable with modified reflector elements containing samples for irradiation. These removable reflector elements are positioned in the graphite reflector outside of the beryllium-oxide reflector row. They will be used to introduce samples into a relatively high thermal neutron flux for longer periods of time than would be normal for the pneumatic-tube system. Samples for study of radiation effects as well as longer-lived isotopes will be produced in this facility.

In the event that it is desired to irradiate a relatively large sample, the reflector elements would be removed and flow-restricting plugs would be inserted in the positions normally occupied by the reflectors and the large sample could be suspended into this region.

Samples to be inserted into the irradiation baskets and to be positioned in the reflector positions, will be turned over to the reactor operator, who will assign an irradiation position and denote this radiation position on a board as well as in the reactor log book. A form will be filled out by the experimenter and approved by the reactor supervisor prior to putting the sample into the reactor. This form will be similar to that used for pneumatic-tube irradiations.

The graphite-reflector positions occupied by these irradiation baskets are far-enough removed from the core such that the entry of a large poison sample will not materially affect the reactivity of the reactor. All entry and removal of these irradiation baskets will be performed during shutdown of the reactor. In no instance will these reflector elements, or the irradiation baskets positioned in place of the reflector elements, be moved during reactor operation.

5.7.7. Gamma-Exposure Facility

In the biological shield wall, in that end of the reactor pool used for the storage of spent fuel elements, there is a cavity approximately 3 x 3 foot square extending through the shield wall up to the tank lining. Within the pool and adjacent to the tank liner, there is a fuel element storage rack designed to hold fuel elements in a position immediately adjacent to this cavity. A shield-block insert would be designed at a later date, consisting of either a turntable or a rolling-ball sample-insertion arrangement such that samples might be placed adjacent to the spent fuel elements for intense gamma irradiation. The utilization of these spent fuel elements in this manner imposes no restrictions on reactivity addition or removal from the reactor, since the fuel elements would be in a non-critical configuration while stored in the gamma-irradiation position.

Initially, this cavity will be filled with barytes-concrete blocks and not utilized as a gamma-irradiation facility. Only after some operating time at high power will it be possible to utilize the gamma facility. At that time there will be designed and installed the gamma-irradiation equipment to permit sample insertion from the face of the biological shield to the rear of this cavity and adjacent to the spent fuel element storage rack.

The use of this spent fuel element gamma-irradiation facility will be under the supervision and control of the laboratory supervisor. The storage of the fuel and the history of that fuel in storage in this position will be supervised and logged by the reactor supervisor in the reactor log book.

5.8. Radiation Monitoring

5.8.1. Personnel

The reactor and laboratory facility will be staffed with a permanent group of from 12 to 16 people. Each member of this staff will be required to wear a beta-gamma and neutron film badge which will be changed on a bi-weekly schedule.

In the event that staff members are undertaking operations such as transfer of fuel, unloading of beam ports, transfer of high-level sources in the hot-cell facility, and any other operation which might involve an above-normal hazardous condition, they will be equipped with pocket dosimeters. Pocket dosimeter's will not be utilized on a day-to-day basis since there is not sufficient justification to go to this length in monitoring of personnel.

All experimenters carrying out research within the reactor and laboratory facilities will be equipped with film badges which will be changed on a bi-weekly schedule. A film badge pick-up station will be established adjacent to the entry and all experimenters will be responsible for picking up and leaving their own badges at the beginning and termination of each day.

All visitors will be required to enter the laboratory-reactor facility through the lobby. When they come into the lobby they will be registered and accompanied through the facility by a guide (usually a graduate student) whose function it is to see that they are escorted only through those areas that are open to visitors. The visitors will not be equipped with personnel-monitoring devices, but the escort will wear a film badge just as all other members of the staff wear film badges. In the event that technical visitors are received who desire to inspect, in some detail, certain of the facilities or experiments these technical visitors will be provided with a film badge by the receptionist in the lobby. This film badge will be issued in their name and processed, together with all other badges utilized by the facility, on a bi-weekly schedule.

All film-badge reports will be returned from the processor to the Radiological Safety Department of the University of Missouri. Having received the reports, the Radiological Safety Department will submit to the reactor facility staff a summary of all exposures to personnel in this area. These reports will be issued every two weeks and will be kept as a permanent record in the reactor facility, as well as in the offices of the Radiological Safety Department. In those instances where pocket dosimeters are utilized for the surveillance of experimental work, the readings from the pocket dosimeters will be entered into a personnel file on that particular individual after the completion of the procedure.

5.8.2. Area Monitoring

Area monitoring, in addition to that accomplished by fixed instrumentation described elsewhere, will be done within the laboratory facilities by a daily-smear-survey technique in which the health physicist smears critical areas each morning between the hours of 8 and 9 A.M. These smears will be counted to ascertain that no contamination is loose from operations of the preceding day. The reports from these smear surveys will be kept by the health physicist as a permanent record.

Additional health physics surveillance will be accomplished by providing constant health-physics supervision of all experimental work carried on both in the reactor and in the various laboratories associated with the reactor. The health physicist will be responsible for the review of proposed experiments to see that all safety criteria for these experiments are adequately met. He will perform daily tours through the various laboratory facilities and through the reactor to maintain adequate supervision at all times. He may prescribe special shields or special experimental techniques to reduce the radiation hazard. The health physicist is empowered to shut down any operation at any time if he feels that this operation is in any way hazardous to the experimenters, to the staff, or to the community.

A monthly report of reactor operations will be prepared by the reactor staff. In addition to the summation of experimental utilization of the reactor, including operating times, shut downs, fuel changes, fuel arrangements, etc., there will be a section in this monthly report describing any extraordinary health-physics problems or incidents.

5.8.3. Environmental Monitoring

A limited environmental-monitoring program will be initiated approximately one year prior to completion of construction of the reactor-laboratory facilities. Samples of vegetation, soil, water, and air will be collected from selected spots within a radius of about

5 miles around the proposed reactor site. The samples will be analyzed for radioactivity and the results will be compiled in the form of a permanent record permitting comparisons at a later date to ascertain that the reactor has not contributed to the environmental radiation levels.

In addition to the study initiated prior to completion of construction, samples of environmental materials will be taken at six-month intervals after the reactor has gone critical. The sampling points selected for the initial study will be utilized on a continuing basis for the collection of samples after criticality is attained. The information obtained as a result of these studies will be incorporated in a yearly report summarizing reactor and laboratory operations. Grab samples for environmental studies will be supplemented by the records from the monitoring systems permanently installed within the laboratory facility.

5.8.4. Facility Monitoring (Installed Systems)

A permanently-mounted area monitoring system is included in the design of the reactor. This system consists of separate ion-chamber detectors, each one located at a strategic spot for radiation detection. Each chamber will feed a signal to a multi-point recorder located in the reactor control room. Some of the locations which will be covered by this system are as follows:

(1) There will be three chambers on the beam-hole floor.

(2) There will be a chamber in the reactor building exhaust-air plenum.

(3) There will be a chamber positioned above the reactor pool.

(4) There will be a chamber positioned to monitor the area adjacent to the cooling equipment room.

These remote monitoring stations will provide the reactor operator with a constant record of radiation levels throughout the facility. This system is illustrated in Figure 5.1.

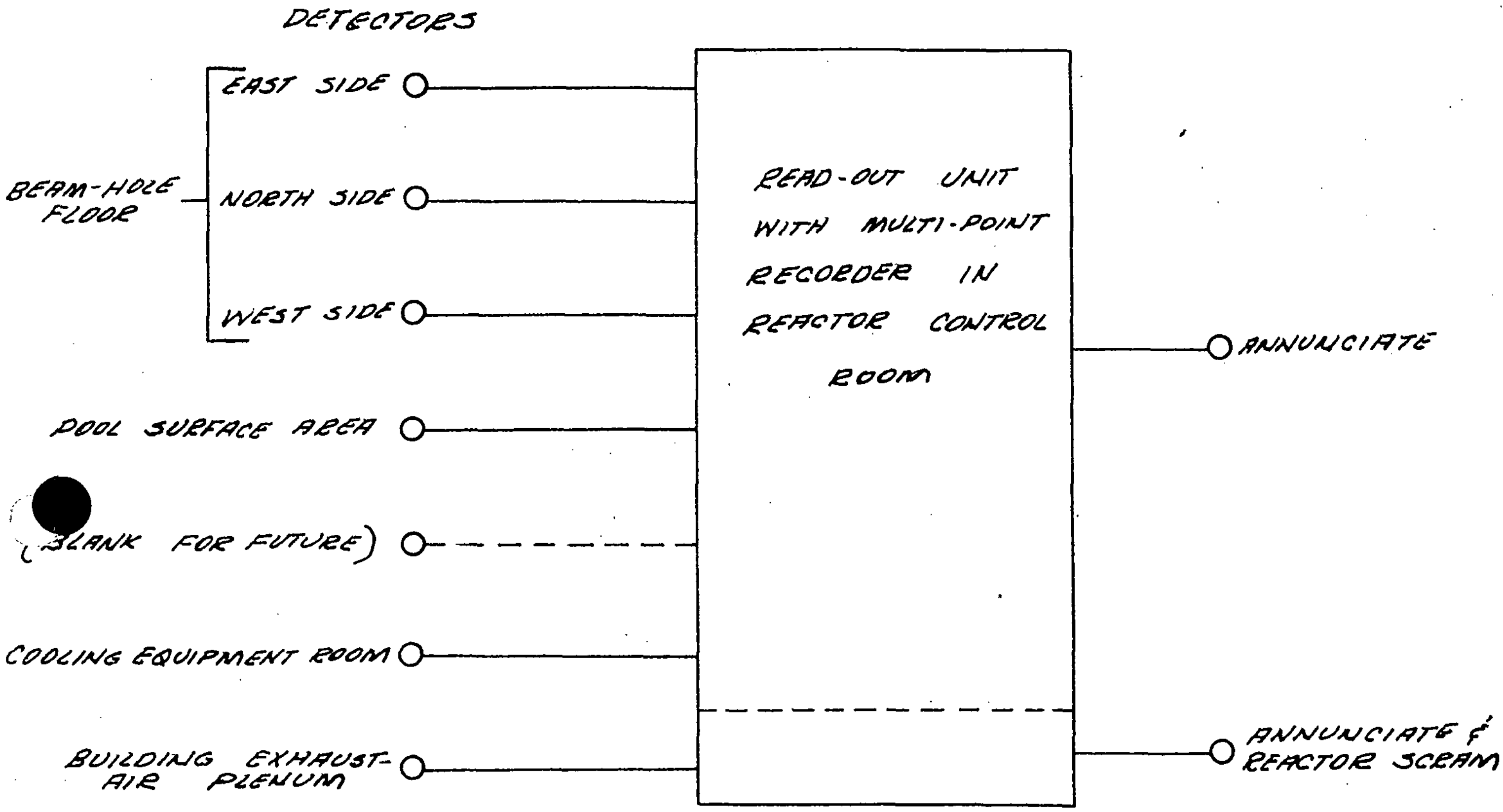


FIG. 5.1 AREA MONITORING SYSTEM

In addition to the ion-chamber air monitor located in the reactor building exhaust-air plenum, there will be a thin aluminum-cased scintillation-counter assembly located in each of the two stacks to monitor the effluent air from the hood exhausts in the reactor and the laboratory facilities. These off-gas monitors will operate in parallel to provide a constant record of off-gas contamination levels in each of the two stacks. This system will provide maximum sensitivity for both beta and gamma air-borne contaminants. The records from the recorder will be stored and kept for a limited period. The system envisioned for this function is illustrated in Figure 5.2.

A fission-product monitoring system will be installed in the primary (pressurized) coolant loop of the reactor. The system proposed is sketched in Figure 5.3. The detector is located in a dry well in a small by-pass monitoring loop. It is realized that this unit will respond to the contaminants induced by neutron interactions with the primary-loop water and that a "live background" will exist during operation of the reactor. It is proposed that the pulse-height analyzer will be adjusted such that a response over background would be produced by fission-product contamination.

The secondary cooling loop should always be "clean" water. It is possible to foresee a condition of leakage from the pressurized side of the heat exchangers into the secondary loop. In the event that it is necessary to waste water from the cooling towers to the sanitary sewer system, it is essential that this water is not contaminated. A secondary-loop monitoring system will be installed to provide a continuous evaluation of the activity of this water. This monitoring system is illustrated in Figure 5.4.

A semi-portable continuous monitor for air-borne particulate radioactivity will be available in the reactor facility. A Tracerlab Model MAP-1 or equivalent is proposed. This unit will be moved from one location to another to meet the needs of particular experimental procedures.

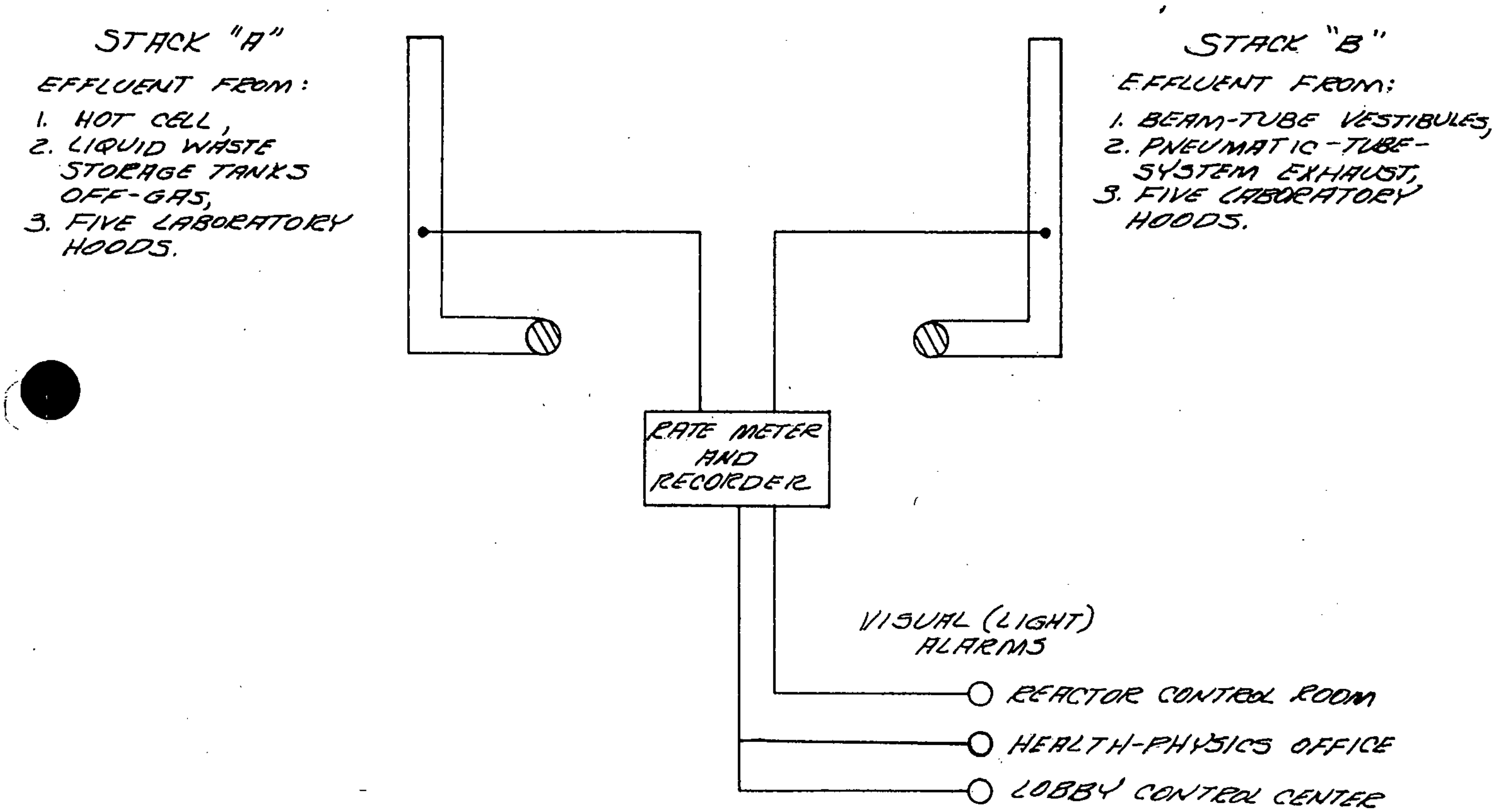


FIG. 5.2 EXHAUST-STACK MONITORING SYSTEM

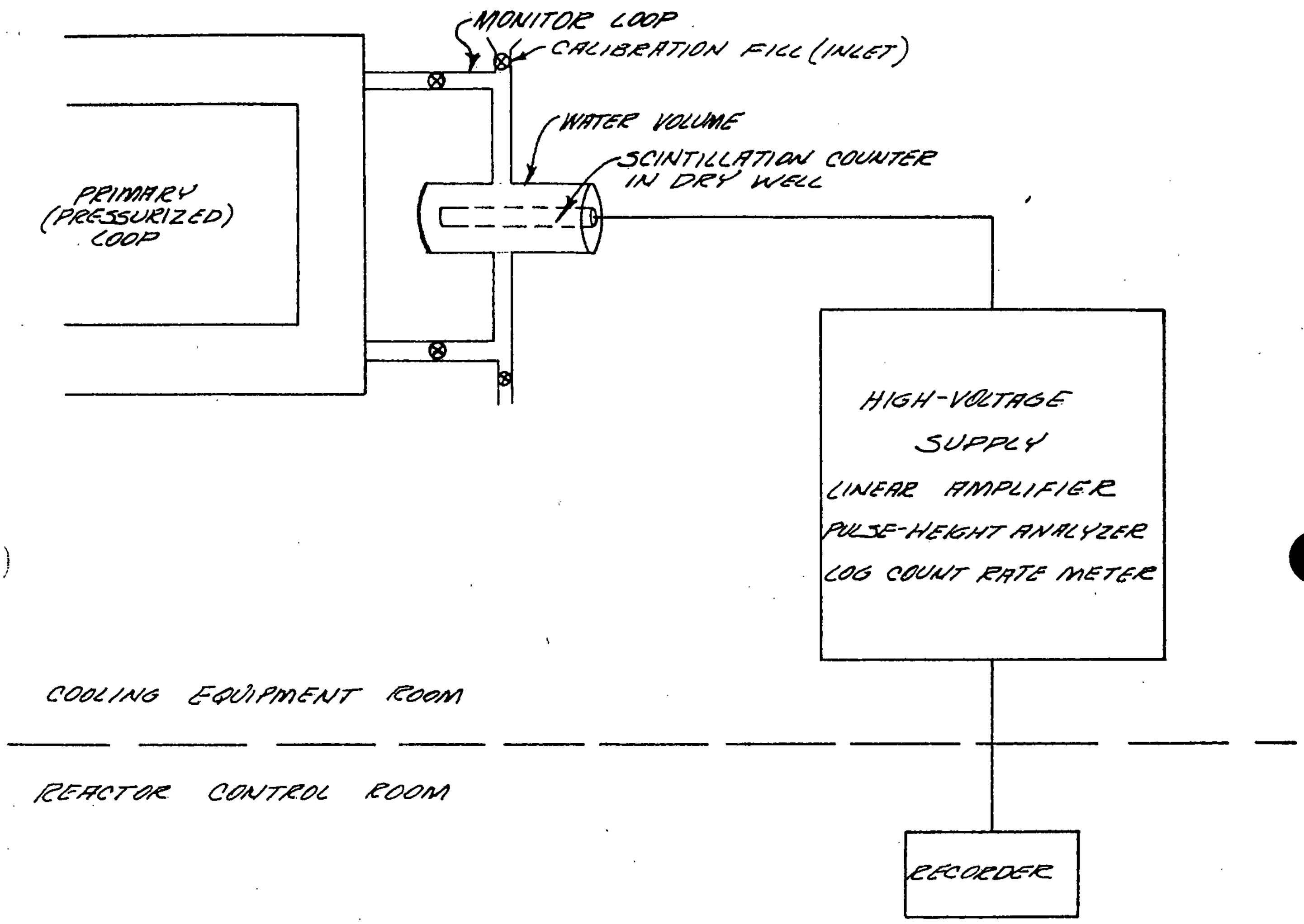


FIG. 5.3 PRIMARY LOOP (FISSION PRODUCT) MONITORING SYSTEM

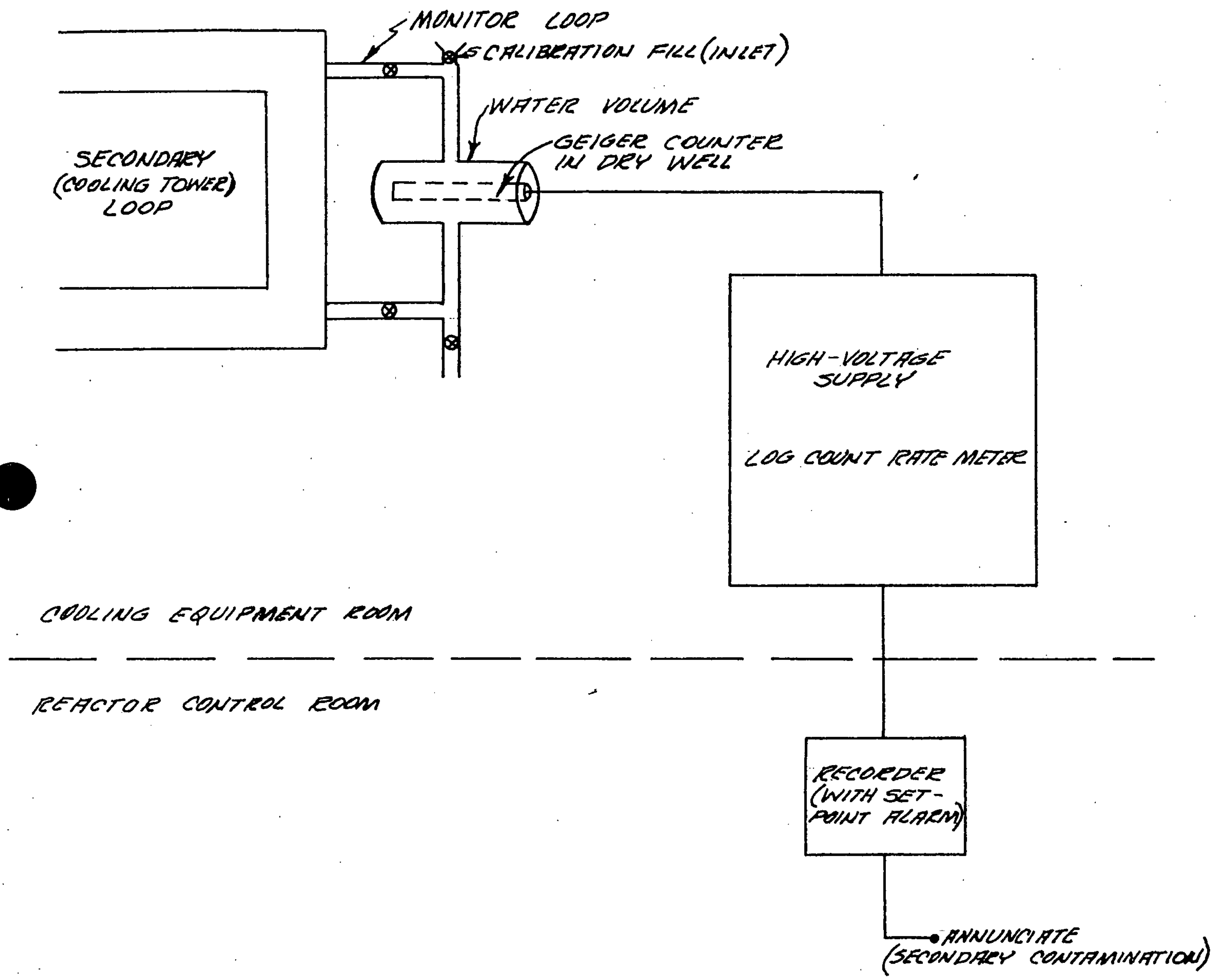


FIG. 5.4 SECONDARY LOOP MONITORING SYSTEM

5.8.5. Radioactive Liquid-Waste Control

It is planned that all contaminated waste water will be routed to one of three 5,000-gallon storage tanks. When one storage tank is filled with active liquid waste, the waste will be monitored, by means of a grab-sample procedure, for alpha, beta and gamma contamination. If the waste displays contamination levels in excess of the established (latest) permissible dump values, a procedure for decontamination will be initiated.

A flocculating agent will be added to the waste tank and mixing accomplished by means of compressed air. When the floc has formed, the waste solution will be pumped through a filter system and returned to one of the empty tanks. The waste solution will then be monitored to ascertain the identity and quantity of contaminants. If it displays contamination levels lower than permissible dump values, it would be pumped, with dilution, to the sanitary-sewage system.

A variable-capacity-pump installation permits the discarding of the waste with a selected amount of dilution. The pumps are capable of pumping rates from one gallon per hour to 600 gallons per hour. If the wastes are not below permissible dump value levels following the flocculation treatment, ion-exchange columns will be available for further decontamination of the liquid waste.

The filters, containing the radioactive waste extracted from the waste solutions, will be removed from the filter holder and incorporated in a one-cubic foot block of concrete. These concrete blocks will be stored within the laboratory facility until sufficient wastes are accumulated to justify a shipment to a burial area.

5.9. Emergency and Evacuation Procedures

The emergency procedures outlined in the pages which follow have been prepared in an effort to minimize panic, property damage, and injury to people should a reactor accident, radiation incident, or fire take place in the reactor building or radiation laboratories. These procedures will be subject to review and revision in the future to maintain the most effective operational plan.

The University of Missouri has formulated, and disseminated to its staff, a very comprehensive, "Campus Disaster Plan." This Plan includes procedures for radiological detection specifically related to nuclear warfare. The Plan calls for the formation of four (4) radiological detection and warning teams. These teams will be formed and trained as a portion of the program for the development of adequate emergency and evacuation procedures related to the reactor. Further, the emergency and evacuation plans for the research reactor facility will be integrated into the "Campus Disaster Plan," in a manner such as to use the established chain of command, communications, transportation, and related services.

These emergency procedures are divided into two cases as a function of the degree of hazard associated with a particular incident or accident. The first section describes a <u>building emergency</u>, and would include an event of magnitude such as to produce a situation of danger to the inhabitants of the reactor or laboratory building. This would include a fire in these facilities. The second case of accident will be termed an <u>area emergency</u> and is defined as a reactor or radiation accident which, in the opinion of the senior staff member present in the facilities, may produce or is producing a situation of danger to the inhabitants of buildings in the vicinity of the reactor-laboratory facilities.

A summary of emergencies and actions is presented as Table 5.3. Table 5.4 gives some definitions pertinent to the paragraphs which follow on emergency procedures

5.9.1. Building Emergency Procedures

When

These procedures will be put into effect immediately after an accident in the reactor building or in the radiation laboratories. If the accident is contained within the building, or in the event that there appears to be imminent danger, as foreseen by any of the staff members or any individual experimenter, a building emergency may be initiated. A building emergency may be changed to an area emergency at the discretion of the senior staff member present in the facilities.

TABLE 5.3

SUMMARY OF EMERGENCIES AND ACTION

1. Stack Monitor High - Visual alarm - investigate - persists shutdown reactor.

2. Area Building Monitors High - Audible alarm and visual display - investigate and correct.

3. Reactor Building Exhaust-Air Plenum High - Audible (building) alarm and reactor scram - evacuate and follow procedures.

4. Reactor or Radiation Accident - Building alarm - evacuate to assigned area - initiate emergency procedures and notify Radiological Safety Officer.

5. Reactor Catastrophe - Building alarm and area alarm - evacuate to assigned areas - initiate emergency procedures, notify Campus Disaster Relief Director and the Radiological Detection and Warning Groups.

TABLE 5.4

SOME DEFINITIONS PERTINENT TO THE EMERGENCY PROCEDURES

Stack Monitor - The radiation detection system (including ratemeter, recorder, and alarm) which continuously monitors the exhaust air from the laboratory hoods, the hot-cell installation, the pneumatic-tube-system exhaust, the exhaust air from the beam-tube vestibules, and the exhaust air from the coolant-loop pressurizer.

Area Building Monitors - A system of five (or more) ion-chamber monitors located at various points in the reactor building, each of which reports to a read-out-recorder system in the control room.

Reactor Building Exhaust-Air Plenum Monitor - An ion-chamber detector unit, which is located in the reactor building normal-air exhaust, which reports readings to the read-out-recorder system in the control room, and which is capable of initiating a reactor scram signal on a high radiation reading.

Reactor or Radiation Accident - An event of lesser magnitude than a catastrophe, which might produce, or is producing, a situation of danger to the inhabitants of the reactor or laboratory building. This would include fire.*

Reactor Catastrophe - This is a reactor or radiation accident which, in the opinion of the senior staff member present in the facilities, may produce or is producing a situation of danger to the inhabitants of buildings in the vicinity of the reactor-laboratory facilities. The senior staff member will initiate the area alarms in these buildings.

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* An effort will be made to have only one alarm system for evacuation, independent of the origin of the alarm.

Who Initiates Building Alarm

A building alarm may be initiated by any staff member or any experimenter working in the reactor or radiation facilities. When the individual initiates this alarm, he must have an adequate reason for evacuating the facilities. The building alarm may be initiated by the senior staff member at periods other than accidents to familiarize the staff with the evacuation procedures and to provide the various staff members with practice in these procedures.

Building Alarm System

The building alarm system shall consist of a number of horns positioned throughout the reactor building and the laboratories in a manner such that they are audible to all experimenters, visitors, and staff members. This building alarm is controlled by a push button located on the reactor control panel, and from a push button located at an emergency control panel in the lobby of the laboratory building. Energizing the building alarm automatically scrams the reactor, and closes the building damper system.

Evacuation Procedures

Upon hearing the alarm all staff members not assigned to specific tasks, all research people, all students, and visitors shall proceed to the designated shelter area. There will be signs in the corridors and in the laboratories indicating directions to the assigned shelter area. The experimenters in the various laboratories should leave their laboratories immediately after extinguishing any open flames and otherwise securing their equipment. All personnel shall move quickly but need not run.

Staff Responsibilities and Assignments

After initiation of the building alarm system, the following procedures and assignments shall be executed by the various staff members of the reactor and laboratory.

Laboratory Director

1. Shall proceed immediately to the lobby of the laboratory building and assume the duties of emergency director. In the event the laboratory director is not present on the premises, the chain of command shall be as follows:

A. Senior Health Physicist
B. Reactor Supervisor.

2. The emergency director shall ascertain the availability of staff members required to execute the emergency procedures, and in the event that there has been a shift of responsibility of the emergency director, or in the event that other members assigned to specific duties are absent, the emergency director shall assign substitutes as required.

3. He shall instruct the radiaction physicist as required before dispatching him on a survey of the laboratory area.

4. With the aid of the reactor and laboratory supervisor, he shall establish that the entire laboratory and reactor facility is evacuated and secured as required by these procedures.

5. He shall ascertain, at the earliest possible time, the cause of the alarm and the magnitude of the incident.

6. He shall immediately notify the radiological safety officer or his associate.

7. After the emergency is terminated, he shall direct the procedures necessary to place the facility back in routine operation.

Reactor Supervisor

1. He shall proceed to the lobby control center immediately, and in the absence of the laboratory director and the senior health physicist assume the duties of the emergency director.

2. He shall assist the emergency director in establishing that the facility is evacuated and secured.

3. He shall aid in evaluating the cause of the alarm and the magnitude of the incident.

4. He shall receive and interpret reports from the reactor operator on duty at the time of the incident.

Laboratory Supervisor

1. He shall check that the total laboratory is evacuated and make an observation of the stack monitor to ascertain that no radioactive materials are being released from the building.

2. He shall proceed to the lobby control center.

3. He shall assist the emergency director in establishing that the facility is evacuated and secured.

4. He shall aid in evaluating the cause of the alarm and magnitude of the incident.

5. He shall receive and interpret reports from the radiation physicist and the electronics technician.

Associate Reactor Supervisor

1. He shall proceed immediately to the lobby control center and assist the emergency director as directed.

Senior Health Physicist

1. He shall proceed immediately to the lobby control center and assume the duties of emergency director if the laboratory director is not present.

2. After receiving instructions from the emergency director he shall proceed through the hot laboratory area and check that the area is evacuated and determine if any abnormal conditions exist.

3. He shall return to the lobby control center and report his findings to the emergency director.

Instrument Maker

1. He shall proceed immediately to the motor control center and turn off the building air in-take fans.

2. He shall report to the lobby control center for further instructions from the emergency director.

Electronics Technician

1. He shall proceed immediately to the lobby control center and act as an assistant to the health physicist. In the absence of the health physicist, he shall perform his duties (other than acting as emergency director).

2. He shall accompany the health physicist in a tour of the laboratory area to determine that the personnel have been evacuated and to ascertain the magnitude of the accident.

Reactor Operator

1. He shall check that the reactor has been scrammed as indicated by the appropriate instrumentation.

2. He shall turn all shim-rod-drive switches to the insert position.

3. He shall turn off the pneumatic-blower system.

4. He shall check that all personnel have evacuated the reactor control room and the reactor offices and shall then proceed down to the second level and evacuate personnel from this level.

5. He shall then exit from the second floor level and remain in the corridor to the laboratory building adjacent to the door entering the reactor, prepared to close and clamp this door after clearance of all personnel from the beam-hole floor.

6. At the completion of his duties, which shall include taking a report from the assistant reactor operator (whose duties are defined in the next paragraphs) he shall proceed immediately to the lobby control center and report to the reactor supervisor.

Assistant Reactor Operator

1. He shall check that all personnel have been evacuated from the beam-hole floor.

2. He shall follow the last person leaving the beam-hole floor to the second floor level exit and, upon clearing the door, shall report to the reactor operator who shall then close and clamp this exit door.

3. He shall proceed to the lobby control center and report to the reactor supervisor.

Receptionist-Secretary

1. If not already present in the lobby control center, she shall proceed immediately to this center.

2. At the direction of the emergency director she shall contact the radiological safety officer.

3. At the direction of the emergency director, she shall contact any other auxiliary organizations which have been made aware of these emergency procedures.

4. She shall remain in the lobby control center and await instructions from the emergency director.

<u>Secretary to Reactor Supervisor</u>

1. She shall proceed immediately to the lobby control center, and report any absentees on the reactor staff to the reactor supervisor so that substitutions may be made.

2. She shall remain in the lobby control center and await further instructions.

<u>Installed Alarms</u>

There shall be installed in the reactor and laboratory building a horn system such that it is audible to all people present at any locality in the building. The system proposed will be initiated by a signal from the reactor control console, from the lobby control center, or from the health-physics office. This system is outlined in Figure 5.5.

<u>Communications</u>

There shall be an adequate telephone system interconnecting the lobby control center with all offices within the reactor and the laboratory facilities. There shall be installed a public-address system which will be audible to all members of the staff as well as experimenters using the facilities, and a third communication system consisting of intercommunication units located at various strategic spots within the laboratories and within the reactor. This intercommunication system shall have a master control panel at the reactor console and in the lobby control center.

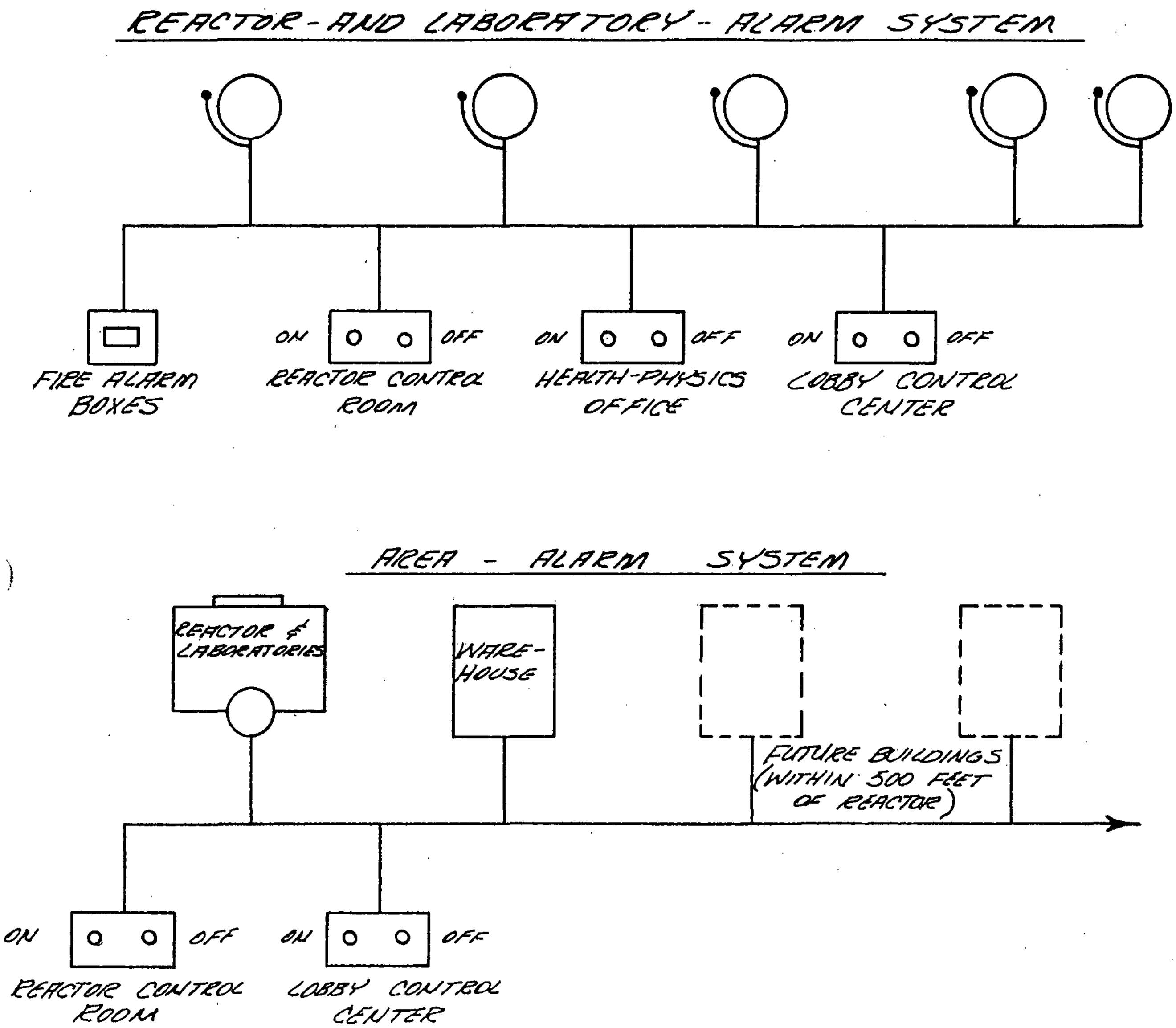


FIG. 5.5 AUDIBLE ALARMS, MASTER PLAN

The normal telephone system shall be used to procure additional help in the event that there is an injury sustained or in the event that additional disaster units need to be notified.

Re-entry Equipment

There shall be available in or immediately adjacent to the lobby control center a minimum of one ion-chamber-type survey instrument, one G.M.-Type survey instrument, and one self-contained respirator unit. These units shall be permanently stored in this locality and shall be tested to ascertain operational capability on a monthly test program.

Emergency Preparation

1. These procedures shall be reviewed at intervals of six months to ascertain that they function properly and that all personnel are aware of their responsibilities.

2. A series of training sessions and drills shall be initiated to insure the most efficient emergency action.

3. A list of all emergency equipment and its location will be prepared and distributed to all staff personnel.

4. There shall be specific equipment for emergency use only and strict maintenance to this usage will be enforced.

5.9.2. Area Emergency Procedures

When

An area emergency may be initiated at the discretion of the emergency director present in the research facilities. The chain of command for the emergency director is outlined in the preceding building-emergency procedures. In most instances the laboratory director will be the senior staff member present and will assume the duties of emergency director.

Area to be Evacuated

In the event of an area emergency all those buildings within 500 feet of the reactor facility shall be evacuated. This evacuation will be initiated by an alarm system located in each of these buildings. The occupants of these buildings shall be subjected to an emergency (dry-run) evacuation at least twice a year to familiarize them with the procedures which they must follow.

Suitable signs designating the shelter area will be posted in each building within the 500 foot radius. These signs will define the meaning of the alarm system and they will present directions to the shelter area. They will also provide directions pertaining to the conduct of the evacuees upon arrival at the shelter area.

Installed Alarm

A system of horns will be installed in any building which is constructed within a radius of 500 feet of the reactor building. This emergency evacuation alarm system shall meet the same criteria as the building-alarm system within the radiation facilities. The area-alarm system may be initiated from a control button located in the lobby control center of the radiation facilities. The proposed area-alarm system is outlined in Figure 5.5.

Staff Responsibilities

Immediately after initiating the area alarm the emergency director shall carry out the following activities in the order presented here:

1. The emergency director shall immediately dispatch members of the reactor-laboratory staff to each building evacuee group to act as group leaders and guide these people into the transports for evacuation.

2. The emergency director shall dispatch a laboratory survey team to the down-wind area to make initial radiation surveys and an immediate report on down-wind radiation levels. This report will be made to the lobby control center by telephone at their earliest convenience.

3. The emergency director shall call the Physical Plant Department and they shall take the following steps:

A. They shall provide transport of evacuees from the designated shelter area to the gymnasium.

B. The Physical Plant radio operator shall call in all radio control cars and trucks from their various assignments on the campus.

4. The emergency director shall alert the four radiological defense and warning survey teams and inform them to assemble at the radio control room in the Physical Plant Department.

5. In the event that radiation levels are reaching intolerable limits in the laboratory control center the emergency director shall order the evacuation of the laboratory control center and a new control center will be established in the radio control room of the Physical Plant Department.

6. The emergency director shall dispatch the survey teams which have assembled at the Physical Plant Department making an assignment of a designated area for survey to each team, and informing them of the type of emergency and expected hazards.

7. The emergency director shall receive survey-team reports and evaluate this data and make a decision with regard to the necessity for evacuation of other nearby buildings.

8. If the survey-team reports indicate the need, the emergency director shall require that additional vehicles be assembled for further evacuation as required.

9. If the situation develops to the point requiring the action specified in No. 8 above, the emergency director shall relinquish control and responsibility to the Campus Disaster Director who will initiate campus disaster procedures as

required to deal with the emergency. In the event the Campus Disaster Director takes control of the emergency procedures, the aforementioned emergency director shall serve as an aid and advisor to the Campus Disaster Director in the evaluation of radiation survey reports.

10. The Campus Disaster Director shall alert additional auxiliary units as required to cope with the circumstances. These additional auxiliary units may include the local Civil Defense group, the Sheriff's office, the Columbia City Police, the Columbia City Fire Department, and other auxiliary units as deemed necessary.

Communications

These area-emergency procedures depend heavily upon the existent and normal telephone services for communication between the laboratory facility lobby control center and the Physical Plant Department. It is proposed that there will be not less than three separate phone lines into the laboratory control center, one of which will be kept open for the report from the laboratory survey team, the second which will be utilized for communications with the radio control center of the Physical Plant Department, and the third which will be kept open and used for the notification of additional survey teams and auxiliary services as required.

The University of Missouri Physical Plant Department has installed a communication system consisting of a radio control center and various two-way units mounted in trucks and cars used by the Physical Plant Department personnel. These vehicles are available in the event of an emergency and may be recalled by radio to the Physical Plant Department. They would be used in the transport of the radiation survey teams and the results of their surveys would be radioed back to the control center.

Survey Equipment

Members of the four radiological defense warning teams will be recruited from faculty members presently conducting research with radioisotopes on the campus of the University of Missouri. These teams will utilize the portable survey equipment which they presently have in their laboratories for making general radiation surveys. The equipment available to these various teams will be increased as required to provide each team with a minimum of one G.M.-Type low-level survey meter and an ion-chamber type high-level survey meter. In the event these teams are assembled at the Physical Plant Department they will bring this survey equipment with them, and utilize it in their emergency activities.

6.0. HAZARDS ANALYSIS

6.1. Introduction

A maximum credible accident will be defined later on the basis of a detailed analysis of various conceivable accidents. The analysis and the consequences of this maximum credible accident will be presented in the Final Hazards Report.

This section describes the consequences of a maximum hypothetical accident which are intended to exceed in severity the consequences of the maximum credible accident, and discusses hazards considerations with respect to the reactor and its enclosure which influence the extent of the accidents.

Particularly significant to the hazards evaluation is the intended use of the facility:

1. This reactor will be used exclusively for research programs conducted by faculty and graduate personnel. It will not be used for reactor experiments nor for student training.

2. No particularly hazardous types of irradiation or experiment are intended, such as fuel-element testing or circulating-fuel experiments. Initial usage of the center-test-hole or flux-trap facility will be limited to experiments which can be moved or altered only upon reactor shutdown. Means for alternate usage of this facility would be installed only upon the approval of an addendum to the Final Hazards Report.

6.2. Reactivity Considerations

At 10 Mw, the reactivity worth of xenon is considerably greater than in low-powered pool reactors. With continuous operation, the equilibrium Xe and Sm are computed to be worth about 4.7% in reactivity; with cyclic operation, 8 hours on and 16 off, the maximum value is 5.4%.

In general, the reactivity worth of equilibrium xenon tends to be greater for a flux trap than for an equivalent solid core because of the higher neutron leakage, and it tends to be less because of the reduced poisoning, $\Sigma xe/\Sigma u$, caused by the higher fuel loading. The xenon behavior of the reference 6-Kg flux-trap design was compared with that of the 3.5-Kg solid core examined previously for comparison of beam-tube currents (Section 3.3.6.2), with both reactors operated at 10 Mw. The reactivity associated with equilibrium xenon was found to be somewhat less for the solid core. However, the reactivity that must be overridden after 16-hour shutdown during saturated cyclic operation is much greater for the solid core. This is due primarily to the iodine-decay term which accounts for most of the xenon present at cyclic start-up. Although more xenon is consumed during operation at the higher flux level associated with lower fuel loading and less xenon is then present to decay, the iodine content at shutdown is independent of flux. At 16-hour shutdown, the xenon concentration is largely due to iodine decay, hence the total abundance of xenon at start-up is relatively insensitive to fuel loading. Since the poisoning and the reactivity worth associated with a given concentration are inversely proportional to the fuel loading, the start-up reactivity increment is greater for the solid core.

It is qualitatively concluded on this basis that the reactivity requirements of the reference design are typical of fully-enriched Al-H_2O reactors operated at 10 Mw. They are less than for an equivalent solid core when operated on an 8-hour-per-day basis at 10 Mw.

The total reactivity requirement for 10-Mw, 8-hour-daily operation may be summarized as follows:

Sm and maximum cyclic Xe	.054
40-day continuous fuel burn-up (529 gm U^{235}) plus fission products	.021
Temperature	.004
	.079
Excess available for experiments with 6-Kg loading	.038
Total with 6 Kg	.117

It should be noted that the experimental requirements have been rather arbitrarily selected; the required value and the required fuel loading will be determined in final design.

Comparison with reactivity breakdowns from similar reactors is sometimes uncertain because of differences in presentation and methods of calculation; for example, the spatial variation of fuel burn-up and xenon build-up are not always considered. However, it may be concluded qualitatively that the required reactivity is typical of the power level and is not unique to the reactor type.

The only unusual reactivity characteristic concerns the void coefficient, which is strongly negative within the annular core, and positive but lower-valued in all other regions. The consequences of this effect and the required safeguards are discussed in Section 3.6.1. With proper design and adequate safety circuitry, it does not appear to represent additional hazard.

6.3. Containment

6.3.1. Reactor Enclosure

The reactor building and its penetrations are described in Section 4.1. The concrete walls of the building will be designed to withstand at least 2-psi differential pressure, and a plastic coating will be provided on the interior of the building to minimize the leak rate. Tests will demonstrate that no more than 1% of the contained volume will leak out in 24 hours.

Two water seals are provided where the coolant lines penetrate the enclosure before entering the coolant equipment room and where the major utility lines enter the laboratory building. Both the lab building and the coolant equipment room are vented to the atmosphere so that a differential pressure of more than 2 psi in either direction across the enclosure walls will cause the pressure to be relieved through the water legs. The water will then flow back, replacing the seal.

The seals are provided to avoid rupture of the enclosure in the event of pressure differentials in excess of 2 psi. This value is thought to exceed the overpressure which could result from a nuclear accident, in which case no leakage would occur through the seals. If this value were exceeded, a quantity of air would be transported through the seals during the pressure relief. Some degree of scrubbing action might accompany this release depending on how quickly the pressure built up. Essentially, the building is intended to withstand all pressure rises and also to prevent any release at the pressures anticipated from a nuclear incident.

Detailed attention will be given to the design of all penetrations of the enclosure to ensure that its integrity is not violated. The pedestrian entry will be sealed by an inflatable gasket and its closure will be a responsibility of the reactor supervisor. The truck entry will be sealed against a rubber gasket with manually-operated clamps and will be interlocked with the reactor control system to prevent reactor operation with the door unsealed.

All air and water lines entering the reactor building will have stop and check valves. All electrical lines will enter the building through special gas-tight seals.

The ventilation inlet and outlet ducts have quick-closing, positive-sealing, flapper valves operated by compressed air. These valves are automatically closed on (1) a reactor scram, or (2) a building alarm, or (3) an area alarm signal.

6.3.2. Pressure Loading on Containment

It is difficult to envision a pressure build-up within the containment, because the maximum design temperature of the bulk water in the reactor is only 160°F. In the case of loop ruptures, the core water would still be subcooled relative to atmospheric pressure, and the water would not flash to steam as in the case of a highly-pressurized water reactor.

In the event of a runaway leading to steam formation and loop rupture within the pool, the pool water would probably quench any steam released. It is noted, for example, that the 1-foot diameter column of water immediately above the 1-foot diameter reactor vessel could alone absorb 108 Mw-sec of energy before reaching 212°F. Experiments described recently by Whelchel and Robbins (1) strongly suggest that all the steam would be quenched and that the containment pressure would return to atmospheric a few seconds after a sudden energy release within the pool.

The magnitude of energy release that could credibly be expected and the associated pressure-impulse loading have not been examined; however, the energy release necessary to cause a 2-psi equilibrium loading has been considered, with all heat sinks neglected.

With initial conditions of 75°F air at 50% relative humidity and pool water at 100°F, and with saturation of air assumed for the final condition, the amount of steam that must be released from the pool into the 240,000-ft^3 building to cause a 2-psi rise corresponds to 1040 Mw-sec. This is much greater than the maximum energy release of 135 Mw-sec observed in the BORAX experiments. The final equilibrium temperature within the containment would be only 113°F.

6.3.3. Effects of Possible Al-H_2O Reaction

There exists a possibility that a fuel meltdown will cause molten aluminum and water to react, releasing considerable amounts of energy. The conditions under which such a reaction can occur are discussed in much of the literature pertaining to reactor hazards.

The occurrence of a rapid reaction requires first that the metal be present in finely divided form, in droplets 500 microns or less. Also, it is necessary that the Al temperature be above its melting point. Epstein (2) observes that for many metals, including Al, the required temperature is near the boiling point of the metal at 0.15-mm pressure; i.e., at more than 2100°F for Al rather than at its melting point of 1950°F.

The fuel plates contain 29.4 Kg of aluminum, over half of which is cladding. If this quantity of aluminum were to react completely with water, it would produce 441 Mw-sec of energy and release about 1340 ft^3 of hydrogen at standard conditions. If the hydrogen were completely recombined, an additional 523 Mw-sec of energy would be released.

The total energy release of 964 Mw-sec is less than the corresponding energy in steam which would have to be released to cause 2-psi equilibrium pressure.

Total reaction of the aluminum and the evolved hydrogen, however, is an excessively conservative assumption. A more realistic assumption might be that 10% of the metal reacts, consistent with the assumption of 10% meltdown used in the dosage calculations (section 6.5).

From these general observations, it is concluded that the assumption of 2-psi equilibrium pressure is adequate. The explosive effects of chemical reactions have not been considered and final hazards studies might consider the impact loadings on the containment and within the pool.

6.4. Maximum Hypothetical Accident

Most of the conceivable fault conditions, and the safeguards which prevent their causing fission-product release, are discussed in Section 3.6. The accident-analysis section of the Final Hazards Report will determine the severity of accidents which may be expected and which can be tolerated in the event of failure of these safeguards.

To establish a maximum hypothetical accident or its consequences previous to such study, it is convenient to examine three general types of accident with regard to their possible release of fission products.

1) loss of pool water
2) reactor-coolant-loop rupture
3) fuel meltdown not provoked by reactor-coolant-loop rupture.

6.4.1. Loss of Pool Water

The reactor pool is cooled by a circulating loop separate from the reactor-coolant loop. Loss of pool water could conceivably occur through rupture of this pool loop or by shearing of a beam port, with simultaneous failure of the means to make up the water. In either case the reactor would be scrammed automatically before appreciable loss of pool water occurred.

The operation of the convective loop within the pool would be impaired if its heat exchanger were exposed, but this loop is essential and active only when the reactor-loop pump fails. The pump is not tripped upon scram, and it should continue to run, removing the decay heat.

Calculation indicates that, at 6 minutes after shutdown with the time required for the water level to drop from the refueling level to the top of the core following a clean break of one 6-in. beam port, laminar heat transfer to air is adequate to prevent meltdown of the exposed rods or reflector pieces. Thus loss of pool water would result only in a direct radiation hazard from the empty pool, and would not cause rod meltdown or rod loss followed by a power excursion.

6.4.2. Reactor-Coolant-Loop Rupture

The consequences of a rupture of the reactor loop are discussed in Section 3.6.4. The reactor can be started only at pressure, and pressure loss due to rupture causes scram. In the event of an external break, the invert loop and siphon break maintain water over the core and the convective loop within the pool is activated automatically to remove the decay heat. With an internal break, the decay heat is removed adequately, either by the reactor loop or by the convective loop, depending on whether or not the pump trips, activating the convective loop.

Loop rupture alone should therefore not lead to fuel melting and fission product release.

6.4.3. Fuel Meltdown not Provoked by Reactor-Coolant-Loop Rupture

Boiling burnout and partial fuel melting, resulting in release of fission products to the reactor coolant could conceivably occur if safeguards were ineffective and if sufficiently severe accident conditions are postulated.

Release of fission products to the building or to the pool would require also that the reactor-coolant loop be ruptured. Such a rupture might conceivably accompany a runaway severe enough to cause fuel meltdown, because of the associated pressure rise in the loop.

If the break were external to the pool, loop pressure would be lost and the siphon break and convective loop would be activated, as in the case of an accident initiated by loop rupture. Fission products could escape through the break, into either the reactor building or the coolant equipment room, by rising through the liquid level at the siphon break and passing down through the drained downleg. This path is closed whenever the operator closes the manual stop valve at the shield face.

With an internal break, fission products would enter the building only by rising through the pool water.

With both internal and external breaks, the pool level would drop to the level of the siphon break with the combined effects noted above.

In all cases, little activity should escape from the loop via the vacuum-breaker valve on the invert loop, because this valve opens only to permit inleakage.

Particularly in the case of the internal break, the hazard of fission-product release would be reduced by the tendency of iodine and solid fission products to be entrained in water. In the case of the external break the fission products would also have to pass through a head of water to reach the drained downleg.

In each case the extent of fuel melting should be limited to that caused by the power excursion which initiated the melting, because subsequent loop rupture does not cause the core to be exposed.

6.4.4. Definition of Maximum Hypothetical Accident

1) The reactor is assumed to have operated continuously at 10 Mw for 40 days.

2) It is assumed that certain undefined run-away conditions occur which cause partial meltdown of the fuel and subsequent rupture of the reactor-coolant loop.

3) Ten per cent of the fuel is assumed to melt. This assumption is somewhat arbitrary previous to detailed accident analysis, but it seems unlikely that a larger fraction of the core would melt, except through loss of core water. Total loss of core water, of course, could cause essentially complete meltdown, but this condition is effectively precluded by the design of the coolant loop.

The extent of meltdown would be limited in a power excursion by moderator expulsion and, eventually, by fuel disassembly. The power densities and temperatures are greater at the bottom of the core because the directions of rod insertion and coolant flow are downward; consequently melting should begin near the bottom of the core, and some fuel may be expected to drop downward and reduce reactivity before extensive melting occurs.

4) It is assumed that the fraction of the fission products released from the molten fuel is given by the following percentages, as recommended by AEC personnel during discussion of the informal hazards information presented previously:

Noble gases	-	75% of total
Halogens	-	25% of total
All others	-	1% of total

The halogen-release fraction indicated above is about 4 times greater and the total gamma-energy-release rate is 2 to 3 times greater than that derived from the maximum release rates observed by Creek, Martin, and Parker (3). The relevant experiments involved controlled melting of disks punched from MTR fuel plates which are essentially identical to the fuel elements proposed for this reactor. The experiments included both rapid melting (2 minutes at 700°C) and slow melting (800°C in air for 1 hour).

5) It is assumed that all fission products released from the molten fuel are dispersed immediately

within the reactor building. The entrainment of iodine in the water is neglected in these maximum hypothetical conditions, but the dose reduction resulting from such entrainment is estimated on the basis of experimental information and included in the results.

6) The reactor building is assumed to leak at the constant rate of 1% per day without regard to variation of internal pressure. Rupture of the containment is not considered.

6.5. Consequences of Maximum Hypothetical Accident

6.5.1. Fission-Product Sources

The total fission product gamma activity is presented in Table 6.1 according to energy and time after reactor shutdown, following 40-day continuous operation at 10 Mw. These values are based on the data of Mr. M. R. Smith (4) which were obtained by integrating the fission-pulse data of Putnam and Knabe (5).

In the latter work, the results of spectral measurements performed at short times after fission were used in conjunction with the total spectral data obtained by calculation of individual species of fission products. By this method the gross activity can be determined down to short times following release or shutdown.

TABLE 6.1

TOTAL-DECAY GAMMA ACTIVITY

Energy Group	Energy Mev	Activity, 10^{16} Mev/sec					
		Seconds After Release					
		10^0	10^2	10^3	10^4	10^5	10^6
1	0.1-0.4	11	6.7	4.5	2.6	1.23	0.33
2	0.4-0.9	49	30	19.7	11.2	5.0	1.58
3	0.9-1.35	36	20	9.5	3.0	1.2	0.53
4	1.35-1.8	32	18	9.8	4.3	2.0	1.1
5	1.8-2.2	21	11	5.9	2.6	0.28	0.017
6	2.2-2.6	17	8.7	4.0	0.95	0.20	0.080
VII	> 2.6	38	11	2.2	0.12	0.0075	0.0047
	Totals	204	105	55.6	24.8	9.92	3.64

The total activity of those noble gases and halogens which contribute significant gamma activity or ingestion hazard are tabulated in Table 6.2. These data are obtained from the curves of ORNL-2127 (6) and include the effects of burn-up of individual nuclides. The thermal flux is assumed to be 5.56 x 10^{13} and the number of U^{235} atoms is 14.37 x 10^{24}.

TABLE 6.2

TOTAL GAMMA ACTIVITY FROM NOBLE GASES AND HALOGENS IN 10^{16} MEV/SEC

	Seconds After Release	Gamma Activity, 10^{16} Mev/sec					
		10^0	10^2	10^3	10^4	10^5	10^6
Iodine	131	.317	.317	.317	.314	.287	.117
	132	2.274	2.274	2.274	2.274	1.93	.204
	133	.847	.847	.838	.772	.235	----
	134	2.766	2.705	2.218	.304	----	----
	135	3.106	3.106	3.019	2.323	.171	----
	136	.202	.085	----	----	----	----
Bromine	83	.007	.007	.007	.002	----	----
	84	.813	.813	.813	.016	----	----
	87	3.237	1.00	----	----	----	----
Total Halogens		13.569	11.154	9.486	6.005	2.623	.321
Krypton	88	.244	.244	.244	.12	----	----
Xenon	133	.163	.163	.163	.163	.163	.042
	133m	.002	.002	.002	.002	.002	----
	135	.061	.061	.061	.12	.09	----
	135m	.302	.302	.302	.21	.018	----
	137	2.28	1.60	1.14	----	----	----
	138	2.22	2.22	1.11	.002	----	----
Total Noble Gases		5.272	4.592	3.022	.617	.273	.042

Use of these curves required a composite correction factor of 0.682 to account for reduced thermal sion cross section and for epithermal fission. It should be noted that the total activities in Table 6.2 are probably underestimated at times less than 10^3 seconds because of short-lived activities for which decay schemes are unknown.

It is assumed that 10% of the fuel melts down and that 75% of the noble gases, 25% of the halogens, and 1% of all other fission products present in the molten fraction are released immediately to the containment.

The energy spectrum of the total fission-product gamma activity is available from Table 6.1. For calculation of external gamma exposure, it is assumed that the gamma energy spectrum of both the halogens and noble gases is identical to that of the total gamma activity at each time of interest. This assumption was justified for the composition of the fission products assumed for the preliminary cloud calculations submitted in the informal presentation. The presently assumed composition is largely halogens and noble gases; hence, the assumption is less valid. However, spot calculations based on the dominant halogen and noble gas activities indicate that little error is introduced in the cloud calculation by this assumption, primarily because the dose rate per unit energy flux is nearly independent of energy. The dose from the containment is more sensitive to the spectrum chosen, but the error again appears small because of the low attenuation factors involved.

Accordingly, an effective percentage of the total gamma activity is determined at each time after release, based on the total Mev/sec for each class of fission products and their assumed release rates. For example, at 10^5 seconds, when this effective fraction is a maximum, the halogen energy fraction is .264 of the total, and the noble gases represent .028 of the total (see Tables 6.1 and 6.2). The effective fraction of the total energy release is then:

Noble gas equivalent	=	(.028 x .75 x .1 meltdown)
+ Halogen equivalent	=	(.264 x .25 x .1 meltdown)
+ All others	=	(.708 x .01 x .1 meltdown)
Effective Fraction		.0094

The effective fraction at each time decade is applied to the total gamma energy available, given by Table 6.1, to obtain the gamma activity present in the containment as a function of time and energy.

The method tends to underestimate the initial dose rates because of the underestimate in Table 6.2 at short times. However the time integrated doses are not affected significantly.

6.5.2. Direct Gamma Radiation From Containment

The walls of the reactor building are of ordinary concrete and they have a minimum thickness of 12 inches.

Dose rates and time-integrated doses were first computed for an observer at the exterior face of the building, with the assumed fission-product source uniformly distributed within the building. With the containment regarded as an infinite slab 60 ft thick, shielded with one foot of concrete, the observer receives 59 r/hr at 100 sec after release and 53 r during the first two hours. With iodine eliminated by entrainment, these values would be reduced to 40 r/hr at 100 sec and 23 r at 2 hours. The estimate is conservative in that the effective source volume would probably include only the space between the building wall and the pool structure. The dose under these conditions might be one-third as great.

At large distances the containment is regarded as a point source shielded by one foot of concrete. With air attenuation and appropriate buildup factors applied to each energy group, the dose rates and time-integrated doses are as shown in Figure 6.1. The dashed line indicates the dose reduction if all iodine were removed by entrainment in the pool water.

The radiation 1000 feet distant from the reactor would be less than that at 300 feet by a factor of about 30.

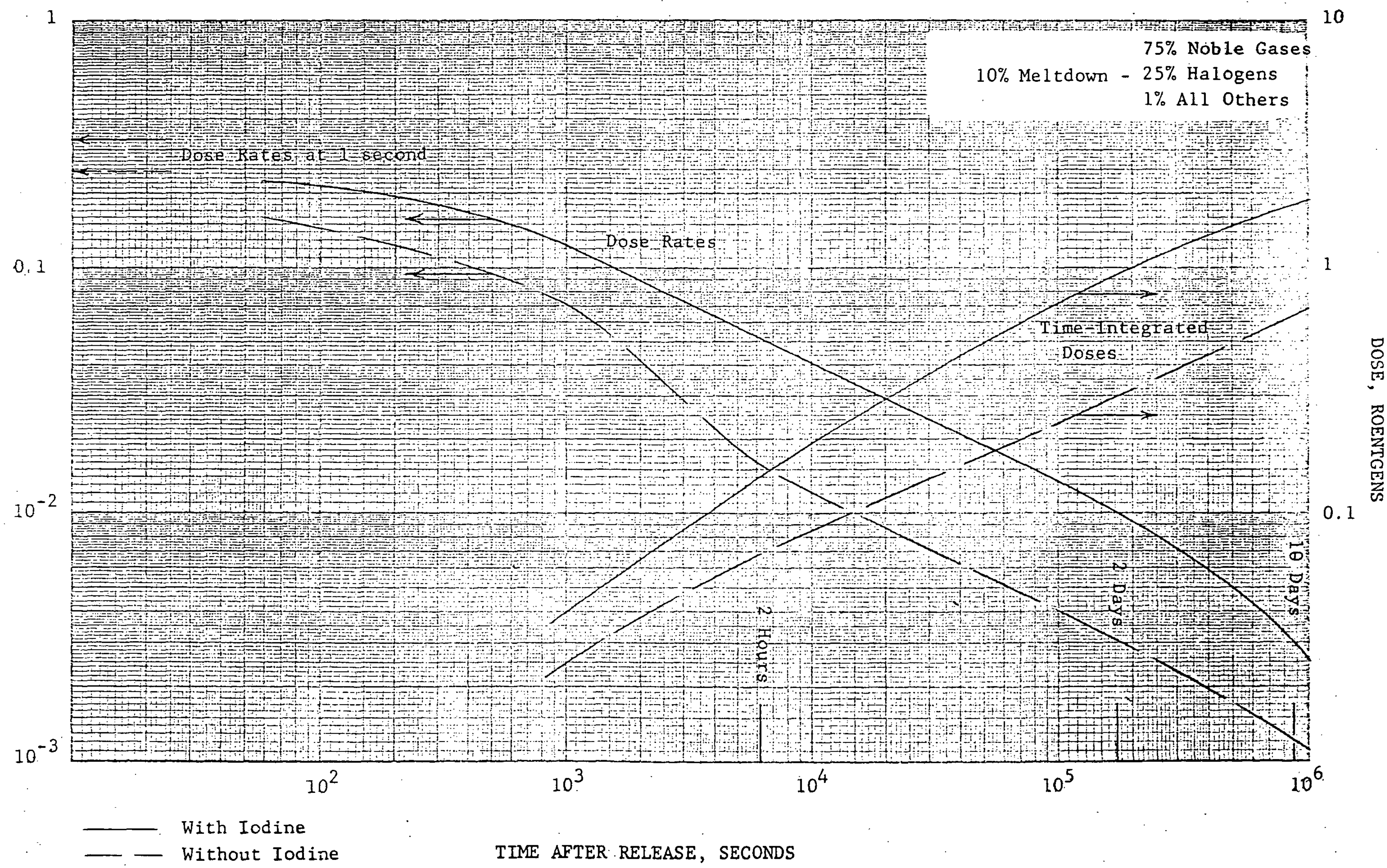
Shielding by the laboratory building, which envelops the lower third of the reactor building, has been neglected.

6.5.3. Dosage Resulting From Containment Leakage

The fission-product source described in Section 6.5.1 is assumed to be distributed uniformly within the

FIGURE 6.1

DIRECT GAMMA DOSE RATE AND DOSE AT 300 FEET FROM CONTAINMENT



containment, and the containment is assumed to leak at the uniform rate of 1% of its volume in 24 hours. Rupture of the containment is not considered.

Cloud gamma dosage and thyroid dosage resulting from iodine ingestion are determined at downwind distances of 300 feet, 1000 feet, and 5000 feet.

6.5.3.1. Diffusion Calculations

The activity is assumed to diffuse as from a continuous point source on the ground, but with corrections to include the effects of building eddies and wind variability. The downwind concentration on the plume centerline is given by Sutton's equation for a continuous point source, multiplied by appropriate reduction factors:

$$\chi = \frac{2\,Q}{\bar{u}\,\pi\,C_y\,C_z\,x^{2-n}}\,C_1\,C_2\,C_3$$

Where χ = centerline concentration, μc/cc or Mev/cc-sec,

Q = release rate, μc/sec or Mev/sec^2,

$\bar{u}$ = wind speed, m/sec,

C_y, C_z = horizontal and vertical diffusion coefficients,

x = distance downwind in meters, and

n = stability parameter.

Two wind conditions are considered, as described in Table 6.3, but the results presented are primarily for the case of inversion.

TABLE 6.3

METEOROLOGY PARAMETERS

	Inversion	Lapse
$\bar{u}$, m/sec	1	6
n	0.5	0.25
C_y, $(m)^{n/2}$	0.4	0.25
C_z, $(m)^{n/2}$	0.07	0.25

Dilution by Building

The additional downwind diffusion caused by the plant structure is included by a correction recommended by the U.S. Weather Bureau staff for a similar application:

$$C_1 = \frac{\pi C_z C_y x^{2-n}}{\pi C_z C_y x^{2-n} + A/2}$$

where A is the minimum projected cross-sectional area of the structure, taken as 465 square meters.

It is noted that this correction gives essentially the concentration from an equivalent disk source having the same total source strength spread uniformly over its area A. A better analogy might be a half-disk, also having area A, in which case the term A/2 in the equation should be replaced by A and greater dispersion is indicated.

Wind Variability

Observation and photographs indicate that for a period of time beyond a few minutes there is an inherent variability of wind direction, whereas Sutton's equation yields essentially instantaneous downwind concentrations. At the suggestion of the Special Projects Group of the U. S. Weather Bureau, previous hazard studies have included a concentration reduction factor for this variation in wind direction (7). This is being proposed as standard by Subcommittee N-6 of the ANS, and it has support from experiments (8).

Assuming Gaussian distribution of the wind in any 45° sector with a ratio of occurrence at centerline to edge of 1000-to-1, the following correction factor was derived and used:

$$C_2 = \left[1 + \frac{x^n \tan^2 (\pi/16)}{C_y^2 \ln 1000}\right]^{-1/2}$$

This factor is 0.65, 0.50 and 0.38 at 300 feet, 1000 feet, and 5000 feet, respectively, for the inversion condition.

Time of Flight Decay

The activity concentration at each downwind station is reduced by exponential decay for individual iodine activities, or displaced according to curves for the bulk activities, to include decay during the time of cloud transit, $t = x/\bar{u}$. This represents correction factor C_3.

6.5.3.2. External Gamma Dosage

The average plume centerline concentrations are determined for each energy group based on the energy sources of Section 6.5.1 and a 1%/day leak rate, and using the cloud dilution calculations described in Section 6.5.3.1. The average centerline concentration is assumed to persist uniformly throughout the hemisphere above the observer and build-up is neglected. The dose rate from each energy group is given by:

$$r/hr = \frac{C\chi}{2\mu}$$

where χ = concentration, Mev/cc-sec

μ = energy absorption coefficient for air, cm^{-1}

C = conversion factor; energy flux to dose rate

The downwind doses and dose rates are presented in Figures 6.2 and 6.3 for the case of inversion conditions. Neglect of the iodine activity causes the dose reduction indicated by the lower curves in Figure 6.2.

FIGURE 6.2

EXTERNAL GAMMA EXPOSURE FROM CLOUD AT 1000 FEET

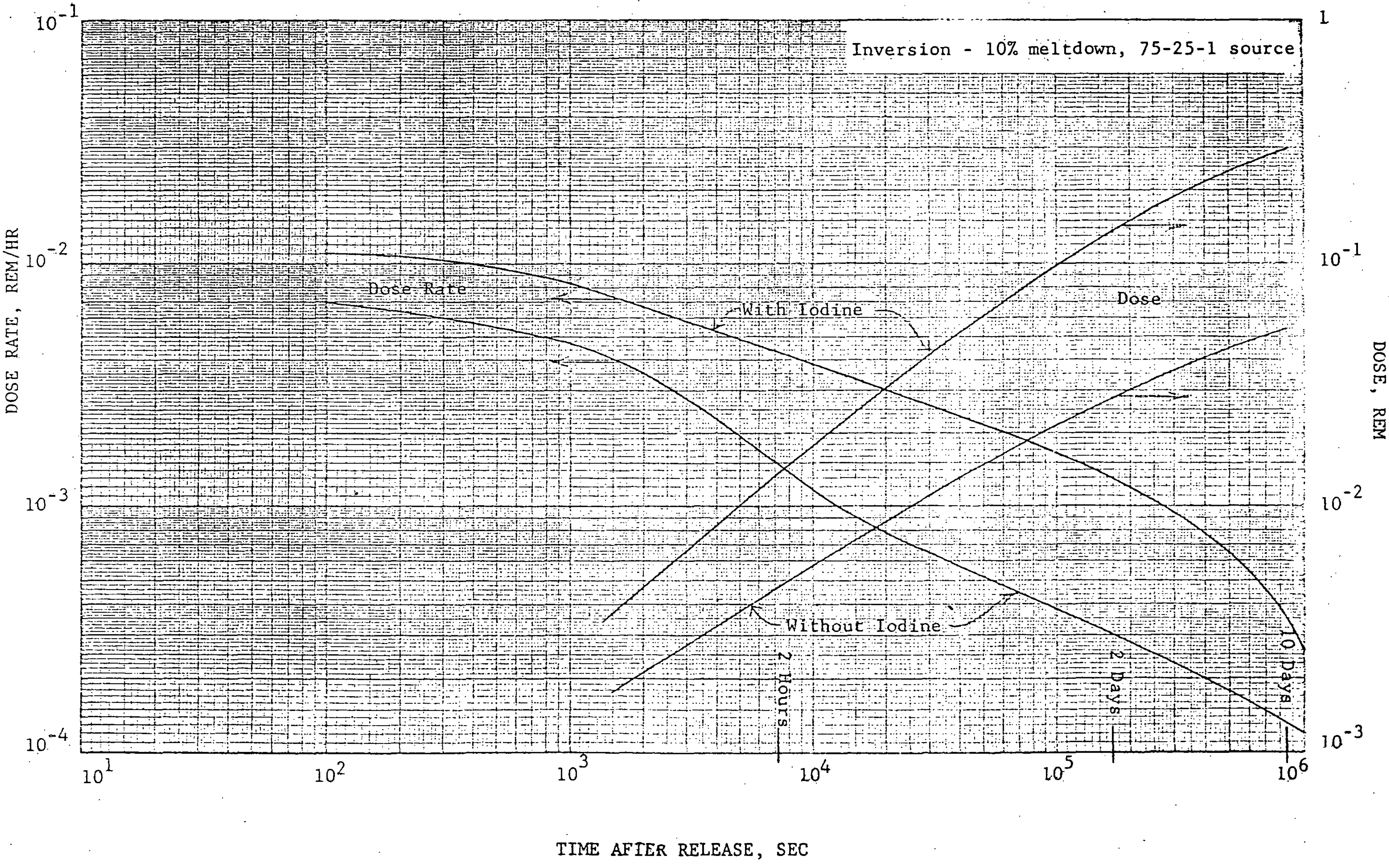
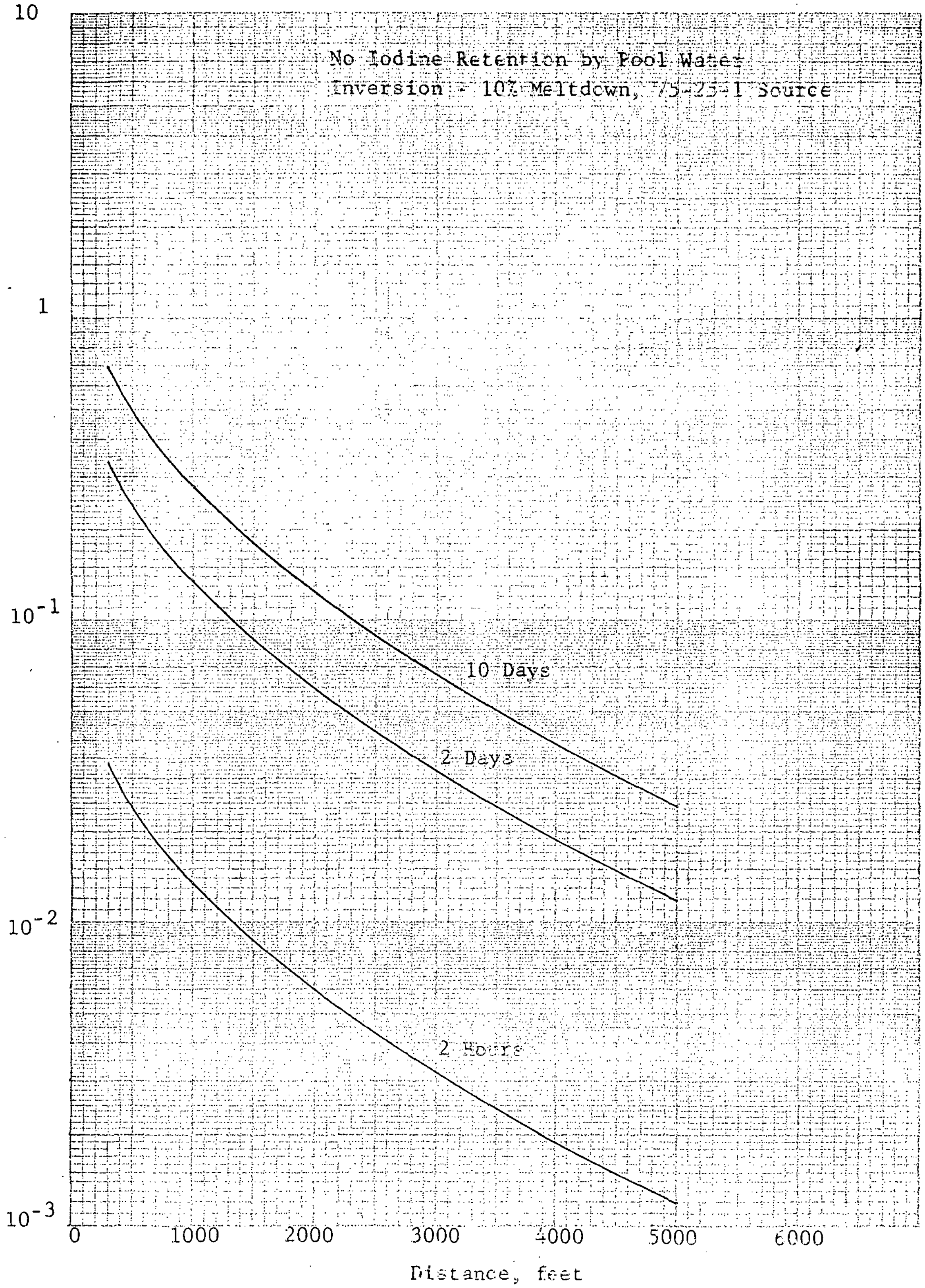


FIGURE 6.3

External Gamma Dose from Cloud



It is noted that the assumption of a uniform cloud having the average centerline concentration overestimates the gamma dose, particularly at short distances and under stable wind conditions.

6.5.3.3. Iodine Ingestion

Of the ingested isotopes, only iodine is considered because the iodine dose to the thyroid is generally the most severe of the doses calculated.

All the iodine activities of Table 6.2 are included, and the average plume centerline concentrations are determined by the methods of Section 6.5.3.1. For each integrated time of exposure, the initial concentrations are reduced to correspond to the average value of the exponential decay during the time of exposure. Dose rates are then determined by the specific exposure values (mrem/inhaled microcurie) presented by Burnett (9) for each iodine activity. A breathing rate of 500 cc/sec is assumed.

The resulting dose rates and doses are plotted in Figures 6.4 and 6.5 for the case of inversion. The effect of iodine retention by the pool water is not included in these figures.

6.5.3.4. Summary of Calculated Dosage

The assumptions used throughout the calculations may be summarized as follows:

1) 10% fuel melting
2) 75-25-1 release of noble gases, halogens, and other fission products from the molten fuel
3) building leakage at 1% per day
4) inversion weather with 1 m/sec wind

Reduction factors in the cloud calculations include decay during the time of flight, added diffusion by the building, and a small correction for wind variability (0.65 to 0.38 with inversion).

The initial dose rates and the time-integrated doses are tabulated in Table 6.4, where reduction due to iodine retention by the pool water is neglected. These values represent the consequences of the maximum hypothetical accident.

FIGURE 6.4

Thyroid Exposure at 1,000 Feet Due to Ingested Iodine

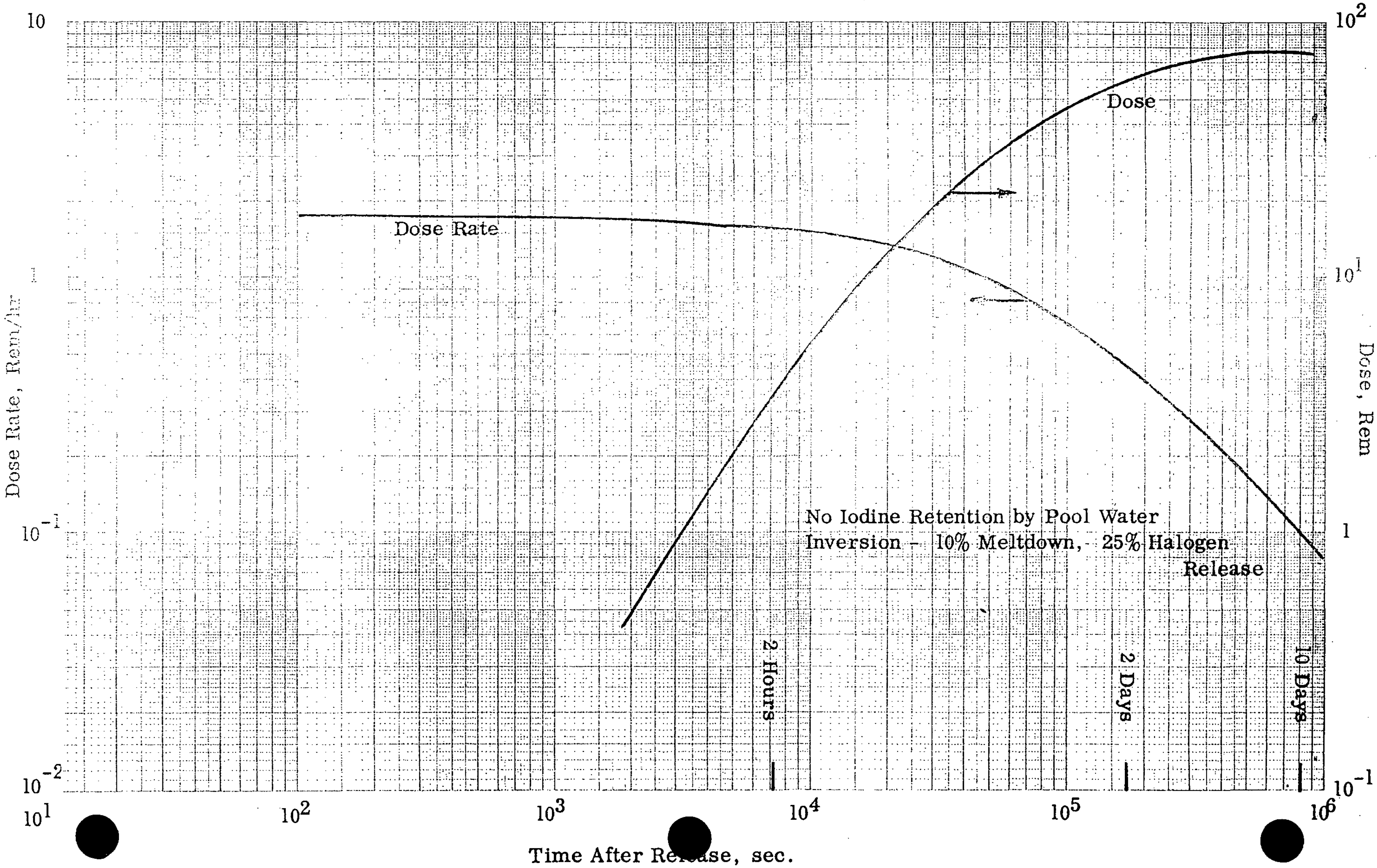


FIGURE 6.5

Thyroid Exposure

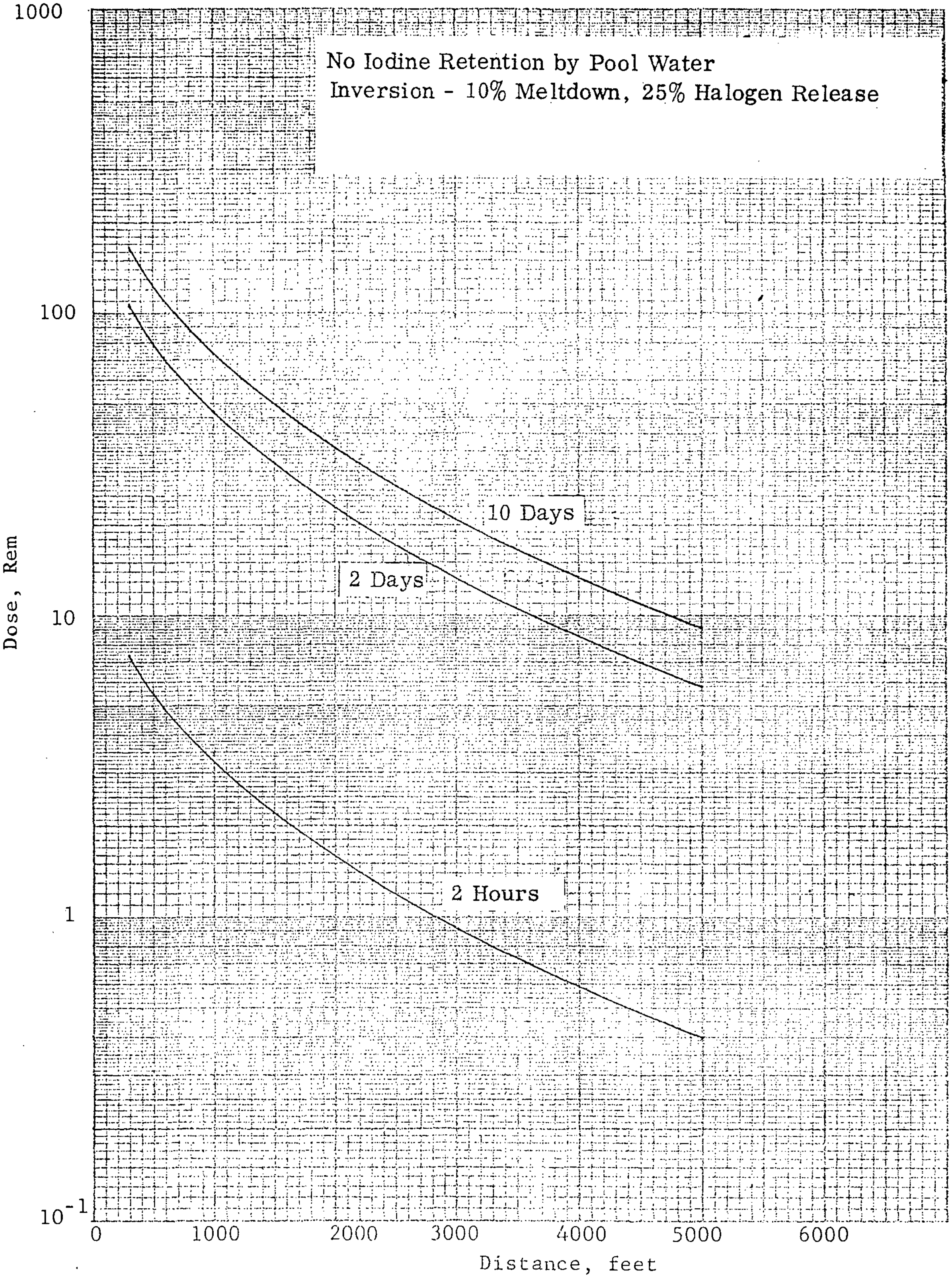


TABLE 6.4

DOSAGE SUMMARY WITHOUT IODINE RETENTION BY THE POOL WATER -INVERSION CONDITIONS ASSUMED

Downwind Distance	300 feet	1000 feet	5000 feet
Initial *Dose Rate, rem/hr			
Direct γ	0.22	0.008	----
Cloud γ	0.027	0.011	0.001
Thyroid	3.92	1.73	0.214
2-Hour Dose, rem			
Direct γ	0.14	0.005	----
Cloud γ	0.033	0.014	.001
Thyroid	7.32	3.24	0.40
2-Day Dose, rem			
Direct γ	0.94	0.034	----
Cloud γ	0.336	0.135	0.012
Thyroid	106	47	5.80
10-Day Dose, rem			
Direct γ	1.8	0.064	----
Cloud γ	0.692	0.279	0.025
Thyroid	166	73.7	9.09

* Direct gamma dose rates at 100 sec; others at 100 sec plus time of flight.

Under lapse conditions, the cloud gamma and thyroid doses indicated in Table 6.4 would be reduced substantially. Including the changes due to different time of flight and wind variability, the lapse values would be less by factors of 0.100, 0.0047, and 0.0021 at 300, 1000, and 5000 feet, respectively.

6.5.4. Retention of Iodine in Pool Water

The effectiveness of the pool water in retaining the iodine nuclides can be estimated from the results of experiments described by Whelchel and Robbins (1). In these experiments Xe, Kr, and I were released under pressure to a pool with the discharge very near the surface of the pool. The fraction of each element which escaped to a containment above the pool was then determined. A large fraction of the noble gases escaped, but the pool water was found to retain nearly all the iodine.

Shortly after release, the fraction of Iodine in the containment was 5.8×10^{-7}; five hours later, it was 7.2×10^{-6}.

Assuming that the mode of release was a pulse followed by a constant release rate from the pool, the transient fraction of iodine in the containment can be described by the following equation:

$$\frac{dC}{dt} = R - \alpha C$$

where C = fraction of iodine in containment at time t

$$R = \text{release rate from pool} = \frac{7.2 \times 10^{-6} - 5.8 \times 10^{-7}}{5\ \text{hr}} =$$

$$13.24 \times 10^{-7}\ \text{hrs}^{-1}$$

$$\alpha = \text{release rate to atmosphere} = \frac{.01}{24\ \text{hr}} = 4.17 \times 10^{-4}\text{hr}^{-1}$$

If, at time zero, the fraction of iodine in the containment is just that in the pulse: $Co = 5.8 \times 10^{-7}$, the solution is:

$$C = \frac{R}{\alpha}(1-e^{-\alpha t}) + Co\ e^{-\alpha t}$$

$$= 3.18 \times 10^{-3}(1-e^{-\alpha t}) + 5.8 \times 10^{-7}\ e^{-\alpha t}$$

Accordingly, the maximum iodine content would increase with time, approaching 3.18×10^{-3} of the total. Ten days are required to reach one-tenth of this value.

The build-up with time is offset by the iodine decay. For example, the composite effect of decay is such that the total dose from time zero to infinity is reduced for each iodine activity by the factor:

$$\frac{R}{\alpha}\left[\frac{\alpha}{\alpha + \lambda}\right]$$

where λ is the decay constant for the particular activity.

This factor varies from 3.3×10^{-4} for I^{131} to 4.6×10^{-5} for I^{136}. Consequently the thyroid dose is negligible if the iodine is released through the pool water and provided the effectiveness of retention is equal to that observed in the experiment.

The consequent reductions in dosage are indicated in Table 6.5 where all contributions from iodine are neglected.

TABLE 6.5

DOSAGE SUMMARY-INVERSION CONDITIONS-NO IODINE

Downwind Distance	300 feet	1000 feet	5000 feet
Initial Dose Rate*, rem/hr			
Direct γ	0.145	0.005	-----
Cloud γ	0.018	0.007	0.001
2-Hour Dose, rem			
Direct γ	0.068	0.002	-----
Cloud γ	0.012	0.005	0.001
2-Day Dose, rem			
Direct γ	0.30	0.011	-----
Cloud γ	0.071	0.028	0.003
10-Day Dose, rem			
Direct γ	0.64	0.023	-----
Cloud γ	0.137	0.054	0.006

* Direct gamma dose rates at 100 sec; others at 100 sec plus time of flight.

6.6. Conclusion

A great deal of information has been presented in the various sections making up this report. All of this information is pertinent to the evaluation of the hazards associated with construction of the research reactor facility at the University of Missouri. In the paragraphs which follow an attempt will be made to summarize portions of this data and inter-relate the various conditions which prevail on the selected reactor site.

Considering first the climatology and meteorology coupled with the population-density study, it is pertient to point out that under the worst conditions which could exist (a stable atmosphere) it is to be expected that released contaminants from the reactor facility would drift to the south and southwest. It is this area which has a very low population density. During unstable atmospheric conditions the probability is that released contaminants would drift north or northwest until they experience a rise to higher altitude, at which time they would be expected to drift toward the east. The area to the north of the selected site contains a higher population density, but it is also an area owned, out to a distance of one mile, by the University of Missouri. As such, it is subject to more rigid control than would be a residential area.

The region surrounding Columbia, Missouri is one of low tornado probability. Therefore, it is proposed that construction practices advocated by the facility architect would be adequate for tornado protection.

The seismic history of the mid-Missouri region indicates that the probability of seismic damage is very low in the Columbia area. A summary of the seismic history has been presented and reveals that in only one instance has there been a seismic disturbance which would have produced extensive damage in the Columbia area (had there been buildings at this point).

The present water level in wells supplying the City of Columbia and the University of Missouri is located approximately 200 feet below the ground surface. In all instances City and University water supply is taken from deep wells of 1000 or more feet in depth. The probability of these wells being contaminated from the surface is extremely low.

Any contaminants which might occur as fallout following a reactor accident would be carried by surface water runoff to the Hinkson Creek. The Hinkson Creek drains through the Perche Creek into the Missouri River. The nearest municipal user of the Missouri River for a drinking water supply is Jefferson City, Missouri, located 42 river miles from the reactor site. It has been calculated that there would be a minimum dilution factor of 10^5 available, which, when coupled with the decay in transit and absorption-precipitation phenomena, minimizes this hazard.

The utilization of the area immediately surrounding the selected site is visualized as future research buildings, a storage warehouse for University supplies, and possibly a central shop facility. Under no circumstances would another building be constructed within 400 feet of the reactor. If, at a future date, buildings are constructed within 500 feet of the reactor facility, these buildings will be placed on the area alarm system and evacuated in the eventof a reactor accident.

Table 6.4 indicates that the total integrated dose after a 10-day occupation period at 300 feet from the reactor would be 169 rem of which 166 is thyroid dose. It is difficult to conceive of circumstances whereby a person would accrue this dosage, since evacuation of the area would have occurred in a much lesser amount of time than 10 days. It is apparent from this hypothetical accident study that the maximum doses accrued by people external to the reactor facility would, in all cases, be less than a lethal dose, and if evacuation proceeded with dispatch there would be no exposures in excess of the emergency exposure value of 25 rems.

6.7. References

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