SAFETY ANALYSIS REPORT FOR THE UNIVERSITY OF MISSOURI-ROLLA REACTOR

REVISION 1

SEPTEMBER 27, 1984

SAFETY ANALYSIS REPORT

FOR

THE UNIVERSITY OF MISSOURI-ROLLA REACTOR

License Number R-79

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Rolla, September 27, 1984

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1. INTRODUCTION

This Safety Analysis Report describes and analyses the University of Missouri-Rolla Reactor (UMRR) and its associated facilities. It was written as a part of the documentation required for the UMRR license renewal. Preceding this report, two documents, (1) and (2), were submitted in the past to the regulatory agencies at various stages of the facility development.

The UMRR began to operate in December 1961. At that time it was licensed for the power level of 10 kW. In 1967 an amendment was granted to increase the maximum power to 200 kW. The average yearly thermal output is about 10 MW-hrs. The reactor is operated by a professional staff within the School of Metallurgy and Mines of the University of Missouri-Rolla.

The UMRR is used for training of nuclear engineering students and other engineering and science students. It is also used for research by the University faculty, their graduate students, and staff. The UMRR is made available to users from outside the University under suitable contract arrangements, e.g. to the electric utilities for the reactor operator training. Students and instructors from other colleges and universities in the Midwest use the reactor under the Reactor Sharing Program funded by the Department of Energy. More details on the uses and programs at the UMRR are given in (3) and (4).

2. FACILITY SITE

The most important demographic and natural factors pertaining to the site of the UMRR are discussed in this chapter. The analysis is based largely on the Preliminary Hazards Evaluation (1) and Hazards Summary Report (2). Updated data are used throughout this chapter, when they were available, especially in Sections 2.1 and 2.2.

2.1 Location

The reactor site is on the east side of the campus of the University of Missouri-Rolla. The campus is located in Rolla, Missouri, about 80 km (50 air miles) southeast of Jefferson City, Missouri. Rolla is located about 161 km (100 miles) southwest of the city of St. Louis, Missouri, and about 290 Km (180 miles) southeast of Kansas City, Missouri (see Figure 1).

In addition to being the home of the University, Rolla is headquarters for the Missouri Geological Survey. A United States Bureau of Mines research division is also located in Rolla, as are important Topographic Mapping and Water Resources divisions of the United States Geological Survey.

The country side near Rolla is largely hilly and rolling. Where land is cleared, the farms are largely devoted to handling beef and dairy cattle. Many farmers also raise hogs, chickens, and turkeys. Grape orchards are locally important east of

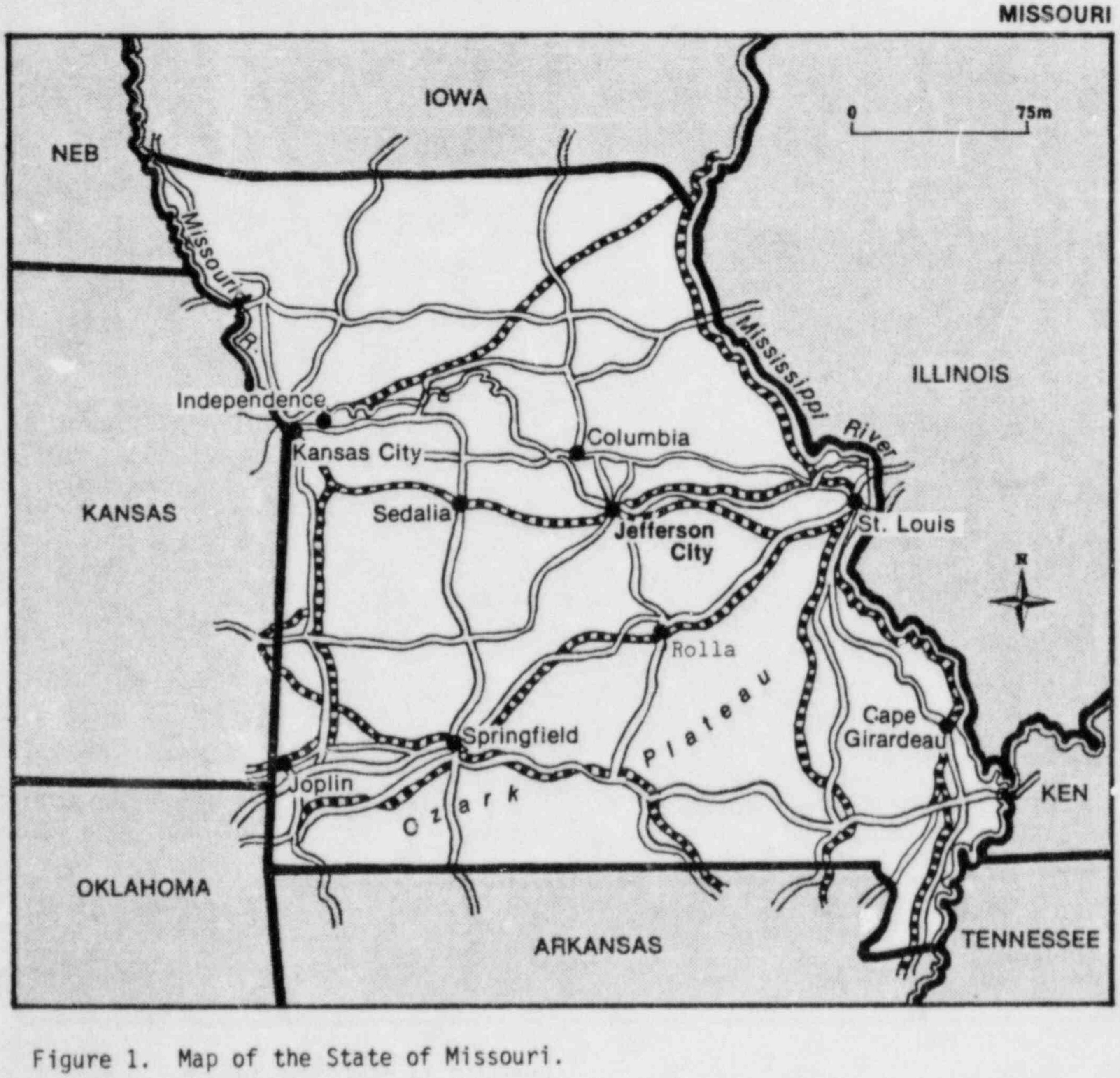


Figure 1. Map of the State of Missouri.

Rolla, especially near the town of Rosati.

The land surface is too rough in most areas for intensive agricultural practice. This accounts for the absence of large population centers with the possible exception of Fort Leonard Wood.

2.2 Demography

The reactor site (Figure 2) is pinpointed in the center of the concentric circles, east of the building which now houses Metallurgical Engineering and Ceramic Engineering. The innermost circle is scaled to a radius of 100 m (300 ft), the next to a radius of 500 m (1500 ft), the next to a radius of 1 km (0.6 mile) and the outermost circle to a radius of 2 km (1.25 mile). Rolla now has a population of about 13,300. Inspection of Figure 2 would indicate that about 14,500 people normally live within a radius of 2 km (1.25 mile) of the reactor site. The University personnel, including students and staff, totals about 8,000. During school hours about 8,000 people would normally be within one-eighth mile of the reactor site. During working hours about 9000 people would be within one-quarter mile of the site, and about 10,000 within 1 km (0.6 mile).

About 40 km (25 miles) southwest of Rolla, Fort Leonard Wood has about 21,500 military personnel in training. Near to the Fort, Waynesville has a population of about 3,000.

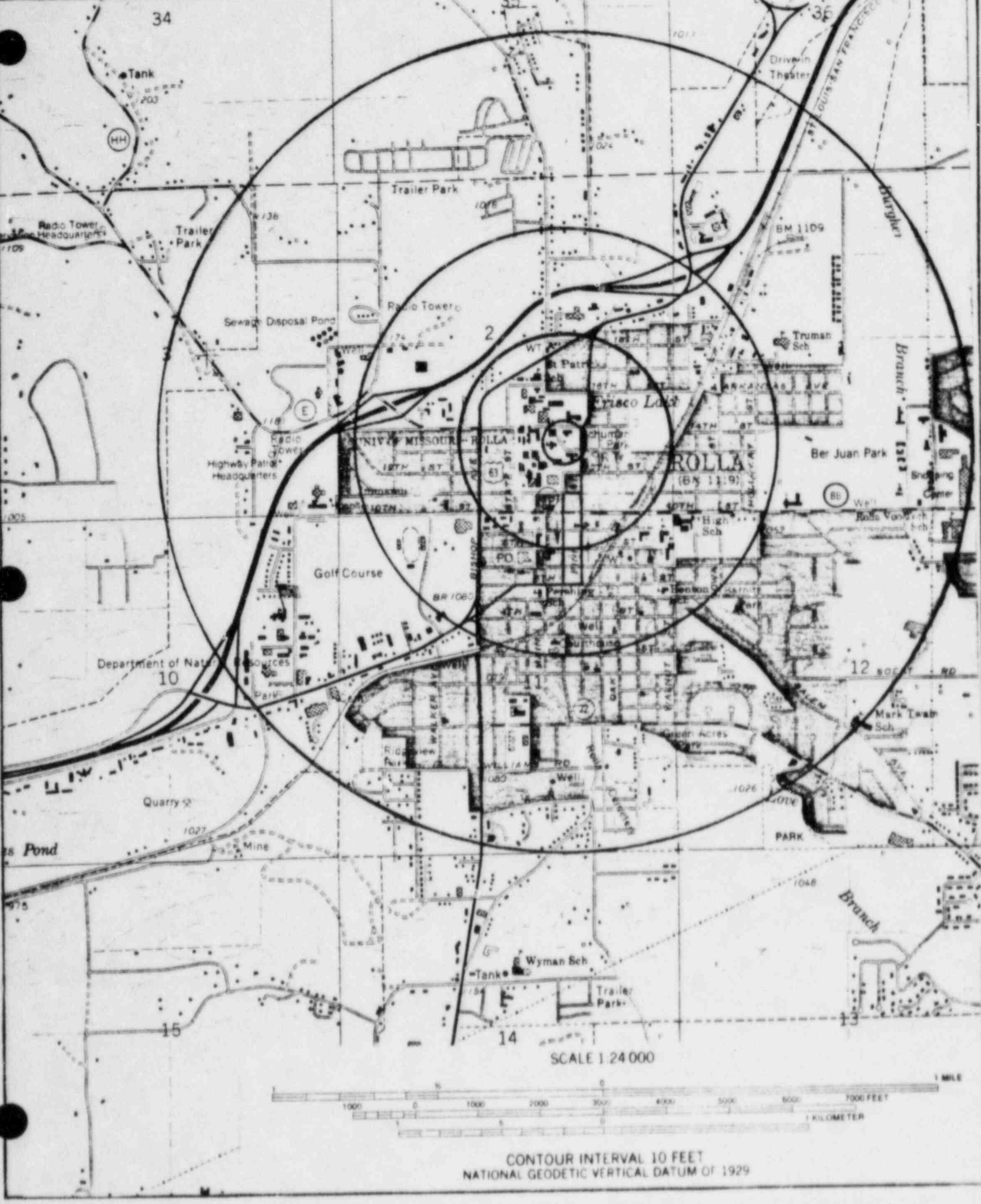


Figure 2. Map of Rolla area.

Population centers within 40 km (25 miles) of Rolla, with distance and direction from Rolla, are tabulated in Table I. Cumulative population distribution within the 40 km (25 miles) zone is given in Table II.

2.3 Geology and Seismology

2.3.1 Geology

Rolla is located toward the northern edge of the Ozark uplift. The sedimentary rock section in the Rolla area averages about 510 m (1700 ft.) in total thickness. This section consists largely of Paleozoic dolomites and magnesium limestones, but with some sandstone and shale members (Figure 3). The Cambriam Lamotte formation, a basal sandstone, usually is encountered in deep wells. The Lamotte uncomformably overlies pre-Cambrian metamorphic and igneous rocks.

The geographical center of the Ozark uplift lies to the southeast of Rolla. Consequently, the regional dip in the Rolla area is toward the northwest, with a very gentle gradient of less than 1 . In places, however, sink structures, developed in the Gasconade, Roubidoux, and Jefferson City formations (Figures 4 and 5) cause high local dips and even faulting.

The sink structures were caused by collapse of old solution channels in the carbonate rocks. Surface exposures of sink structures at Rolla ordinarily show solidly compacted fillings

Table I. Population Centers within 40 km (25 miles) of Rolla

Name of Town	Population	Distance from Rolla (km)	(miles)	Direction from Rolla
Fort Leonard Wood	21,500	40	25	Southwest
Salem	4,500	40	25	Southeast
St. James	3,300	16	10	East Northeast
Waynesville	2,900	40	25	Southwest
Cuba	2,100	32	20	East Northeast
Steelville	1,500	35	22	East
Newburg	1,200	16	10	Southwest
Dixon	1,400	32	20	West
Belle	1,200	37	23	North
Bland	700	40	25	North
Vienna	600	29	18	Northwest

Table II. Cumulative Population Distribution within 40 km (25 miles) of Rolla

	Population
Within 2 km (1 mi) radius	14,500
Between 2 km (1 mi) radius and 4 km (2.5 mi) radius	300
Between 4 km (2.5 mi) radius and 8 km (5 mi) radius	1,000
Between 8 km (5 mi) radius and 16 km (10 mi) radius	7,000
Between 16 km (10 mi) radius and 40 km (25 mi) radius	41,000
Total population within 40 km (25 mi) radius	63,800

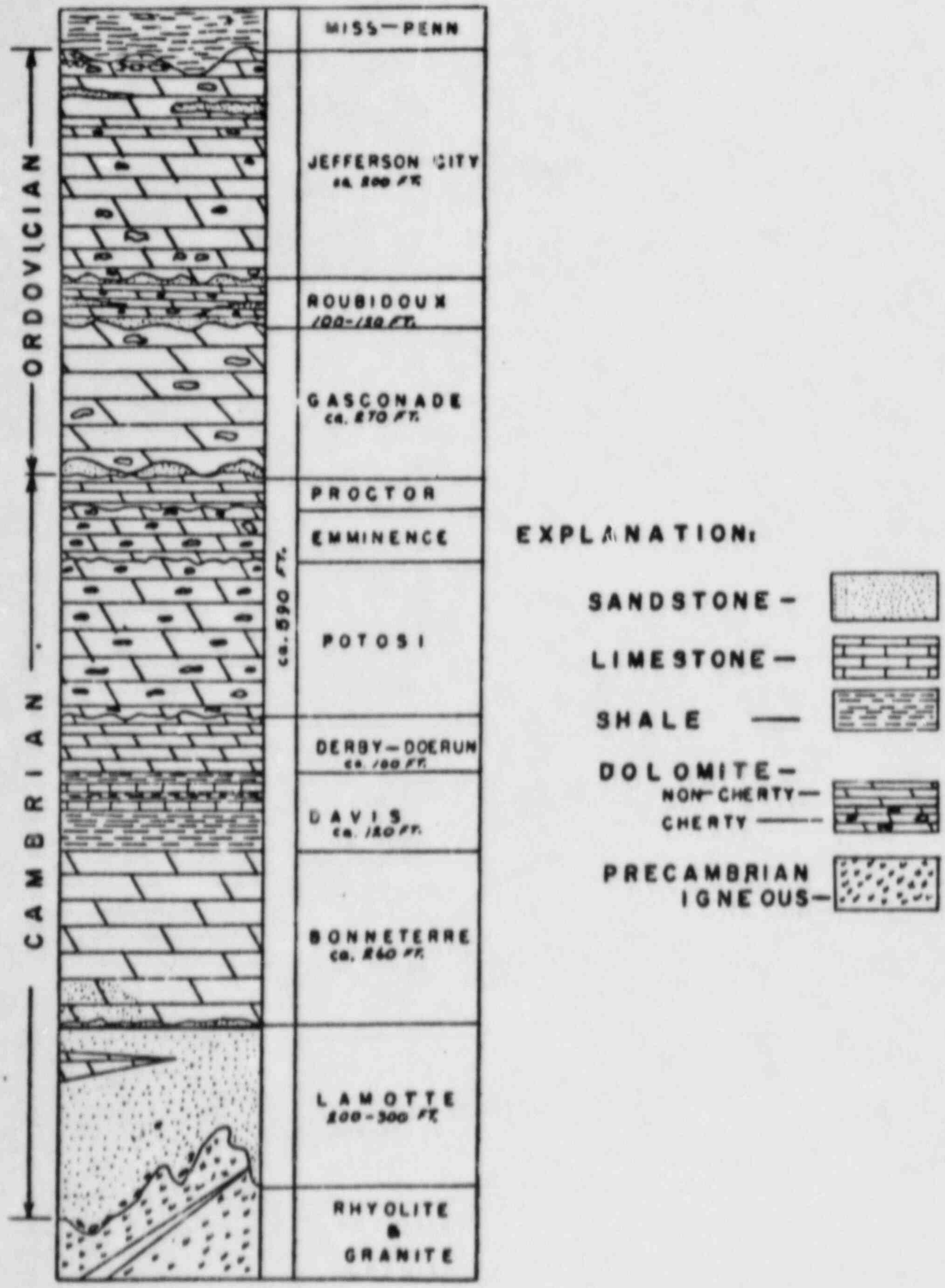


Figure 3. Geologic column, Rolla area, Phelps Co., Missouri.

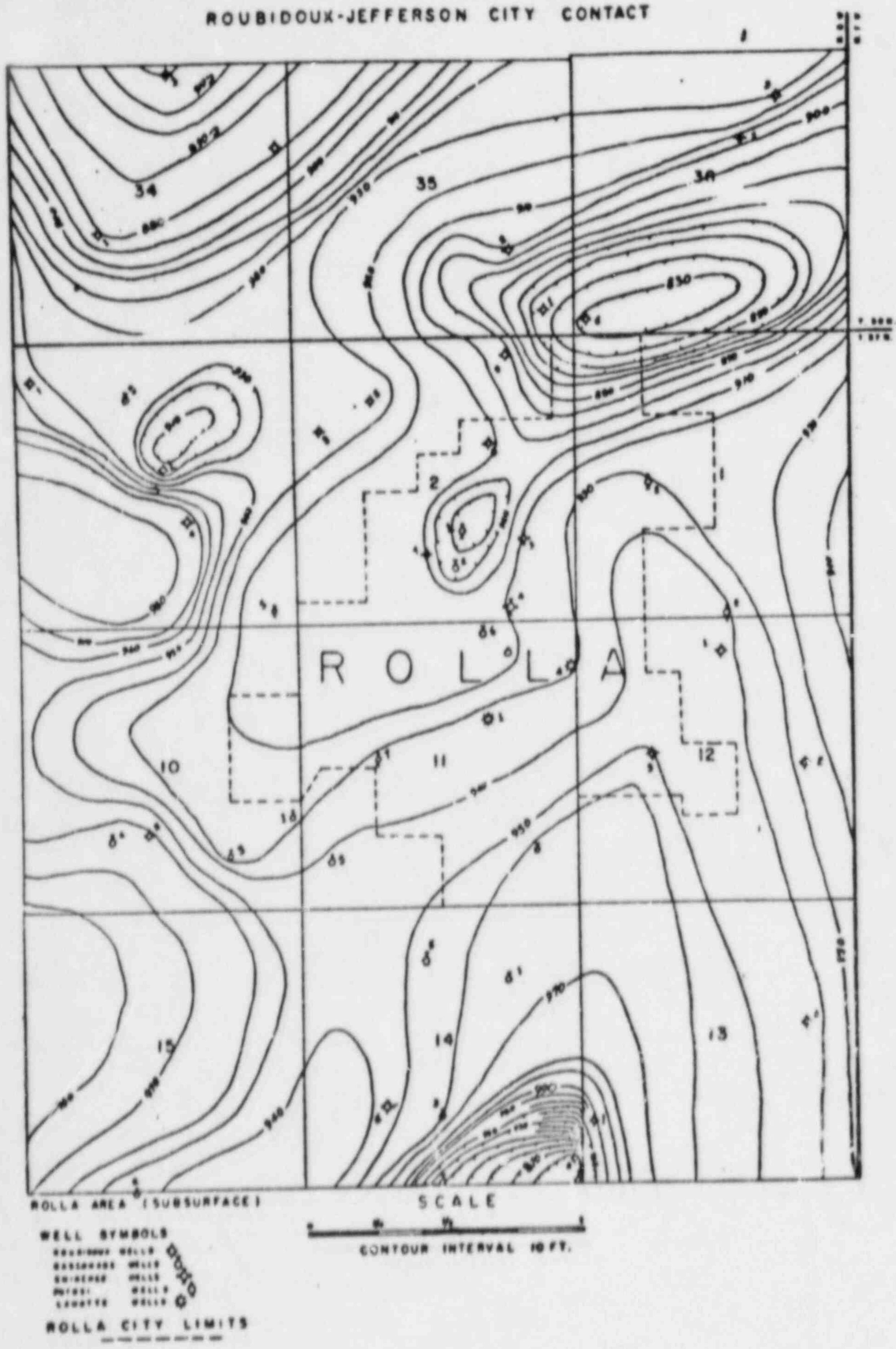


Figure 4. Subsurface contour map.

Figure 5. Geological map of the Rolla area.

of clay shale and sandstone of Pennsylvanian age.

Soils developed on surface exposures in the Rolla area are predominantly of the silty loam type. In flood plains and channels of larger streams, such as the Dry Fork, deposits of almost pure quartz sands are locally developed.

2.3.2 Seismology

Examination of the Bulletins of the Seismological Society of America for the period 1925-54, selected papers on the seismic history of Missouri, and others on the regional distribution of seismic disturbances revealed that, although the state of Missouri lies within a relatively inactive area, it contains six districts that can be classed as minor seismic districts. These districts have been named the New Madrid, St. Mary's, St. Louis, Hannibal, Springfield, and Northwestern districts. Rolla does not lie in any of these districts but is situated approximately in the center of a square formed by connecting the Springfield, St.Mary's, St. Louis, and Northwestern districts. There has been no recorded instance of an earthquake focus occurring in or adjacent to the town of Rolla in at least the last 140 years. It seems reasonable to assume, on a basis of its past seismic history and because it does not fall in one of the known seismic districts in Missouri, that it will most probably not be the focus for an earthquake in the near future.

A review of the seismic history of Missouri, shows that the first recorded instance of seismic activity was in 1811-12. A series of earthquake shocks (now called the New Madrid series) occurred over a period of more than one year with some 1,874 individual shocks being reported. The affected area included S.E. Missouri, N.E. Arkansas, Western Kentucky and Tennessee. These shocks were unequaled in number, continuity, area affected, and severity by any earthquakes in the United States in historic time. Visible surface effects covered an area of 1.3x10 square km (5x10 square miles) and felt motion occurred in an area of one million square miles. From Indian legends and public accounts it would appear that this area has an earthquake history prior to 1811, but nothing of that magnitude. The data in the attached table (compiled from the "Seismological Notes" in the SSA Bulletins, and from a paper by Ross R. Heinrich on seismic activity in Missouri) lists the recorded earthquakes originating in Missouri from 1811-1954, with date, probable place of origin (or reporting point closest to focus), and intensity in terms of the Wood-Neumann scale. It is evident from this list that Missouri is a fairly active minor seismic area, with fairly frequent minor shocks and occasional large ones.

As previously mentioned, Heinrich and other investigators have divided Missouri into six seismic districts. The New Madrid district is made of portions of five states, Missouri, Arkansas, Illinois, Kentucky, and Tennessee. The Missouri section of the seismic zone is made up of Pemiscot, Dunklin, Mississippi, New Madrid, Stoddard, Scott, and part of Butler Counties. The earthquakes originating in this seismic district tend to occur along a line connecting New Madrid, Charleston, and Caruthersville, strongly suggesting basement faulting along this line. Approximately 60% of the seismic activity in Missouri has originated in this district.

The St. Mary's district is confined to Perry, Ste. Genevieve, St. Francois, and parts of Iron, Washington, Franklin, and Jefferson Counties. This district is on the northeastern flank of the Ozark uplift and is traversed by a line of northwesterly trending faults. About 25% of the seismic activity originating in Missouri occurs here.

The remaining 15% of the seismic activity originating in Missouri in the past has been divided between the four remaining districts: St. Louis, Hannibal, Springfield, and Northwestern. Frequency of earthquakes in any given seismic area cannot be predicted on any periodic basis. This is, indeed, a very controversial question among seismologists. Many such attempts have been made to demonstrate periodic frequencies, but most have proved negative. Heinrich has estimated, however, that as an average 4 earthquakes per year in Missouri (provided results are tabulated for at least a ten year period) could be expected. With considerably more confidence, it can be said that these earthquakes would be expected to be confined to the six seismic zones (focus, that is) and that 60% will occur in the New Madrid district, 25% in the St. Mary's district, and 15% will be spread throughout the remaining four districts.

The intensities of Missouri earthquakes has ranged from a minimum of I on the Wood Neumann scale to the maximum recorded for any earthquake; however, 85% since 1811 have been of slight to moderate intensity. Of the remaining 15% only 7.5% were strong enough to do considerable damage, and almost all of these earthquakes originated in the New Madrid district. Occurrence and intensity of earthquake activity in Missouri since 1811 is shown in Table III.

From the above considerations it would seem that Rolla should be reasonably secure from the prospect of earthquake damage. The probability is against the occurrence of an earthquake focus in or near Rolla and the intensity of any earthquake shocks felt in Rolla from seismic activity in one of Missouri's seismic districts would not normally be expected to be in excess of IV on the Wood-Neumann scale and would probably be considerably less.

2.4 Hydrology

2.4.1 Ground Water

Wells furnishing water for the city of Rolla are cased for varying depths from the surface. Danger of contamination of city water supplies from any possible escape of radioactive liquids at the reactor site seems to be very slight. Dilution

Table III.

Occurrence and Intensity of Earthquake Activity in Missouri Since New Madrid Shocks of 1811-1812 (*violent enough to cause damage)

Date	Place	Intensity	Remarks
* 1811-1812	New Madrid	XII	See Text
July 25, 1816	New Madrid	III-IV	
April 11, 1818	St. Louis	III-IV	
Sept. 2, 1819	New Madrid	III-IV	Also felt in St. Louis
Sept. 16, 1819	Cape Girardeau	III-IV	
Nov. 9, 1820	Cape Girardeau	(?)	
July 5, 1827	St. Louis	IV	
Aug. 14, 1827	St. Louis	III	
* June 9, 1838	St. Louis	V	
* Jan. 4, 1843	New Madrid	IX	One of the most severe in Missouri history
Feb. 16, 1843	St. Louis	(?)	
Mar. 26, 1846	New Madrid	II-III	
* Oct. 8, 1857	St. Louis	VII	
* Aug. 17, 1865	New Madrid	VII	
July 8, 1872	Western Missouri	III	
Nov. 8, 1875	Kansas City	III	
Sept. 25, 1879	Gayoso	III	
July 13, 1880	Gayoso	(?)	
* July 20, 1882	Charleston	V	
July 28, 1882	Ironton	(?)	
* Sept. 27, 1882	Mexico	VI	Covered area 250 x 160 mi.
* Oct. 14, 1882	Eastern Missouri	V	
Nov. 15, 1882	St. Louis	III	
* Jan. 11, 1883	New Madrid	V	
* Dec. 5, 1883	Rovenden Springs	VII	
Feb. 15, 1884	Caledonia	III	
Feb. 21, 1885	Carthage	III	
Aug. 31, 1886	Eastern Missouri	II	Effect of destructive earthquake at Charleston, S.C.
Oct. 18, 1895	New Madrid	II	
* Oct. 31, 1895	Charleston	VII-IX	Felt as far as New Mexico
Dec. 2, 1897	Kansas City	III	
June 14, 1898	New Madrid	III	
* Jan. 24, 1902	St. Louis	VI	Two severe shocks strongly felt in "Lead Belt"
* Oct. 4, 1903	St. Louis	V	
* Nov. 4, 1903	New Madrid	VI	Felt in 8 states
Nov. 24, 1903	New Madrid	II-III	
Nov. 25, 1903	New Madrid	II	
Nov. 27, 1903	New Madrid	III	
* Aug. 21, 1905	Mo., Ind., Ky, Tenn	VI	Considerable damage in St. Louis
Feb. 23, 1906	Anabel	II	
Mar. 6, 1906	Hannibal	IV	

Table III. (cont.)

Date	Place	Intensity	Remarks
July 4, 1907	Bismark	IV	
Nov. 10, 1907	St. Louis	IV	
Nov. 12, 1908	Sedalia	IV	
* Oct. 23, 1909	Cape Girardeau	V	
* Feb. 28, 1911	Kenwood Springs	IV	
Apr. 28, 1915	New Madrid	IV	
May 21, 1916	New Madrid	IV	
* Apr. 9, 1917	St. Mary's	VI	Considerable damage
May 9, 1917	Hendrickson	III-IV	
June 9, 1917	New Madrid	IV	
July 1, 1918	Hannibal	IV	
* Oct. 15, 1918	New Madrid	V	
May 26, 1919	New Madrid	(?)	
Feb. 28, 1920	Springfield	IV	
* May 1, 1920	St. Louis	V	No shock felt in Columbia
Oct. 3, 1920	Harrisonville	III	
Jan. 9, 1929	New Madrid	IV	
* Mar. 22, 1922	New Madrid	V	Slight damage
Mar. 28, 1922	Popular Bluff	III	
* Nov. 26, 1922	St. Louis	V	Some damage in St. Louis
* Oct. 28, 1923	New Madrid	VII	
Dec. 31, 1923	New Madrid	IV	
Mar. 2, 1924	New Madrid	IV	
July 30, 1925	Kansas City	(?)	
Oct. 27, 1926	Popular Bluff	IV	
Dec. 13, 1926	Perma	III	
Feb. 1, 1927	Jackson	IV	
Feb. 3, 1927	Popular Bluff	IV	
* May 7, 1927	New Madrid	VI	Some damage
Mar. 17, 1928	St. Louis	I	
Apr. 15, 1928	New Madrid	III	
May 31, 1928	New Madrid	IV	
Feb. 26, 1927	Arcadia	IV	
Apr. 2, 1930	Caruthersville	IV	
May 28, 1930	Hannibal	IV	
Aug. 8, 1930	Hannibal	IV	
Sept. 1, 1930	Perma	IV	
Dec. 23, 1930	St. Louis	IV	
Apr. 6, 1931	St. Louis	III	
July 18, 1931	New Madrid	IV	
Aug. 9, 1931	Kansas City	IV	
Dec. 17, 1931	St. Louis	II	
Mar. 17, 1933	Poplar Bluff	IV	
July 13, 1933	St. Mary's	III	
Aug. 3, 1933	St. Mary's	IV	
Oct. 24, 1933	Cape Girardeau	(?)	
Nov. 16, 1933	Grover	IV	
Apr. 17, 1934	St. Mary's	III	
May 15, 1934	St. Mary's	III-IV	
July 2, 1934	Pemiscot County	III	

Table III. (cont.)

Date	Place	Intensity	Remarks
* Aug. 19, 1934	Charleston	V	
Jan. 30, 1935	Pawnee	III	
Feb. 16, 1936	Hayti	IV	
Oct. 20, 1936	New Madrid	I	
Oct. 31, 1936	S. E. Missouri	I	
Jan. 30, 1937	Caruthersville	III	
Mar. 18, 1937	Perryville	III	
Oct. 5, 1937	New Madrid	III	
Jan. 16, 1938	Perryville	III	
Mar. 16, 1938	New Madrid	(?)	
Sept. 28, 1938	Malden	III	
Apr. 15, 1937	New Madrid	(?)	
Feb. 4, 1940	Cape Girardeau	III	
Dec. 27, 1942	Maplewood	(?)	
Jan. 15, 1945	Little Saline Creek	IV	
May 15, 1946	Doniphan	III	
* June 29, 1947	St. Louis	V-VI	Some damage
Dec. 1, 1947	Little Black River	II-III	
Feb. 8, 1950	Lebanon	IV	
Sept. 11, 1953	St. Louis	(?)	Slight
Feb. 2, 1954	Poplar Bluff	IV	Felt over wide area of S. E. Missouri

by ground water would also be a mitigating factor.

Ground water is restricted to aquifers. In order of decreasing importance with respect to wells bottomed in them, these are the Roubidoux, Gasconade, Potosi, Jefferson City, Eminence, and Lamotte formations illustrated in Figure 4.

The Roubidoux sandstones and the Gasconade formation outcrop in stream channels which drain the reactor site toward the east (Figure 5). Livestock drinking from the surface water drainage would be more directly exposed than would the human population which depends largely on water from drilled wells.

a. Roubidoux Formation. The most important water bearing formation in the area at the present time is the Roubidoux. Dolomite is the most abundant lithologic type, although locally the formation is composed largely of sandstone and chert. The sandstone in the Roubidoux formation usually occurs in two beds separated by cherty dolomite. In some locations one or three sandstone beds may be present.

Of the fifty-five water well logs studied, twenty-six wells bottom in the Roubidoux. These yield from one to twenty-five gallons per minute. The depths of the Roubidoux wells range from 43 m (142 ft) to 132 m (440 ft) and average nearly 90 m (300 ft). Most of the wells bottom in the sandstone, but some bottom in the dolomite, usually only a few feet below the sandstone.

The static water levels in the Roubidoux wells, as recorded by the Missouri Geological Survey well logs, are highly variable from well to well. The Roubidoux-Jefferson City contact in well number 2 in section 14 lies 41 m (137 ft) below the same contact in well number 1 in section 13, which is less than one-quarter mile distant. The slope of approximately 12 degrees between the two contact points is three times greater than the static water level slope. This indicates circulation of water between the two points in the sandstone. Other wells show greater static water slope compared to structural slope. This indicates that the hydrologic properties of this aquifer are not uniform laterally.

b. Gasconade Formation. Second in importance as an aquifer, insofar as the number of wells is concerned, is the Gasconade formation. This formation consists mainly of cherty dolomite and varies in thickness within the area from 77 m (255 ft) to 87 m (290 ft).

Twelve wells bottom in the Gasconade formation within the Rolla area. Individual yields range from 8 to 34 gallons per minute. None of these wells is cased very deeply and the yields given above include water that comes from horizons above. Some of the wells originally obtained water from the Roubidoux formation until successive dry seasons made deepening necessary. They have been deepened to their present depth.

The static water levels in the Gasconade wells do not vary as greatly as the static water levels in the Roubidoux wells. They range from 250 m (834 ft) to 293 m (978 ft) above sea level. The static water is from 2 m (7 ft) to 58 m (192 ft) above the top of the formation, but the variation is due to the elevation differences of the static water level. No relationship is indicated between static water level and structure.

c. Potosi Formation. Six of the seven wells that supply the city of Rolla and one University of Missouri- Rolla well obtain ground water from the Potosi formation. This rock unit consists of cherty dolomite 69 m (230 ft) to 86 m (286 ft) in thickness. It is relatively flat lying with either local structure or a former erosional surface as indicated by elevation relief of the upper and lower contacts of the formation. Too few wells penetrate the Potosi formation for a strict interpretation of its structure. Fissures and caverns are not uncommon in this formation.

The Potosi wells yield water at the rate of 300 to 580 gallons per minute with 6 m (20 ft) to 39 m (130 ft) of drawdown. These wells are cased to points below the Roubidoux, so total yields noted are obtained from the Gasconade, Eminence and Potosi formations.

d. Other Aquifers. Minor water producing formations are the Jefferson City, Eminence, and LaMotte. Production from the dolomitic Jefferson City formation is weak and the formation is not important as a water producer in the Rolla area.

The Eminence formation consists of a cherty dolomite with sandstone lenses. This formation provides water for two wells that bottom in it and possibly for wells that pass through it into deeper formations. The Gunter sandstone, which is about 10 m (30 ft) thick and occurs at the top of the formation, provides water in other areas, but the Eminence wells in the area of this report bottom 21 m (70 ft) and 26 m (85 ft) below the Gunter. This indicates that water in the formation comes from the cherty and sandy dolomite rather than from the sandstone at the top.

The LaMotte formation throughout that area occurs at a depth of more than 480 m (1600 ft). Its thickness is unknown, but may range from 75 m (250 ft) to 150 m (500 ft) based on the data outside the area. It is considered a poor producer of water, but one known well yields about 250 gallons per minute from it in the Rolla area.

The Elvins group and the Bonneterre dolomite are non-producers of ground water in the area. The former, made up of the Derby-Doerun and Davis formations, consists of beds of shale, limestone, and non-cherty dolomite. The thickness of the Elvins group is about 78 m (260 ft). The Bonneterre dolomite is a non-cherty and is about 78 m (260 ft) thick.

e. Summary and Conclusions. Six aquifers are known beneath the Rolla area. These supply the city and immediate area with water. In decreasing importance, on the basis of the number of wells bottomed in them, the producing strata are: the sandstones of the Roubidoux formation, the fractured cherty dolomites of the Gasconade, Potosi and Eminence formations, the sandy and cherty dolomites of the Jefferson City formation and the sandstone of the LaMotte formation.

The Potosi wells supply the city of Rolla and the University of Missouri -Rolla at the rate of one-half to one and one-half million gallons of water per day. Figures for other formations are not available and part of the supply is from horizons above the Potosi formation, but Potosi production probably is greater than production from other aquifers.

The lens-like character of the sandstones and lateral change in lithology of the Roubidoux formation greatly influenced the yield and static water level in the Roubidoux wells.

2.4.2 Surface Water

Surface drainage from the reactor site is toward the east. Natural topography, modified by street fills and culverts conduct the runoff to Frisco Lake, a body of water about 3 acres in surface area. Frisco Lake, now a part of the Rolla Park System, was created by the damming of surface drainage by the Frisco railroad fill. Overflow from Frisco Lake drains eastward to the Little Dry Fork; then to the Dry Fork and Meramec Rivers. Route of surface drainage from the reactor site within a 40 km (25 miles) radius is shown in Figure 6.

Downstream from the reactor site the first known use of this drainage for human consumption is at the St. Louis suburbs of Valley Park and Kirkwood. Here wells are sunk into the Meramec River channel sands and gravels. Perforated horizonal radials from these wells pick up water which is probably largely seepage from the Meramec River.

Ninety air miles from the Rolla reactor site, Valley Park is probably at least 290 km (180 miles) away in terms of stream channel distance. In the unlikely event of a release of radioactivity from the reactor and subsequent escape of radioactive fluid from Frisco Lake, it appears that tremendous dilution would occur before any fluid from the reactor site would reach Valley Park or Kirkwood water systems.

The Meramec River enters the Mississippi River about 19 km (12 miles) south and downstream from St. Louis with an average discharge greater than 1,000,000 gals/min. At Eureka, records over a 10 year period indicate that the maximum flow was greater than 12,000,000 gals/min. and the minimum flow 115,000 gals/min. Downstream about 121 km (75 miles) from the Meramec-Mississippi confluence, Cape Girardeau, Missouri is the first town to use the river for domestic water supplies. Possibility of significant contamination of Cape Girardeau water supply from

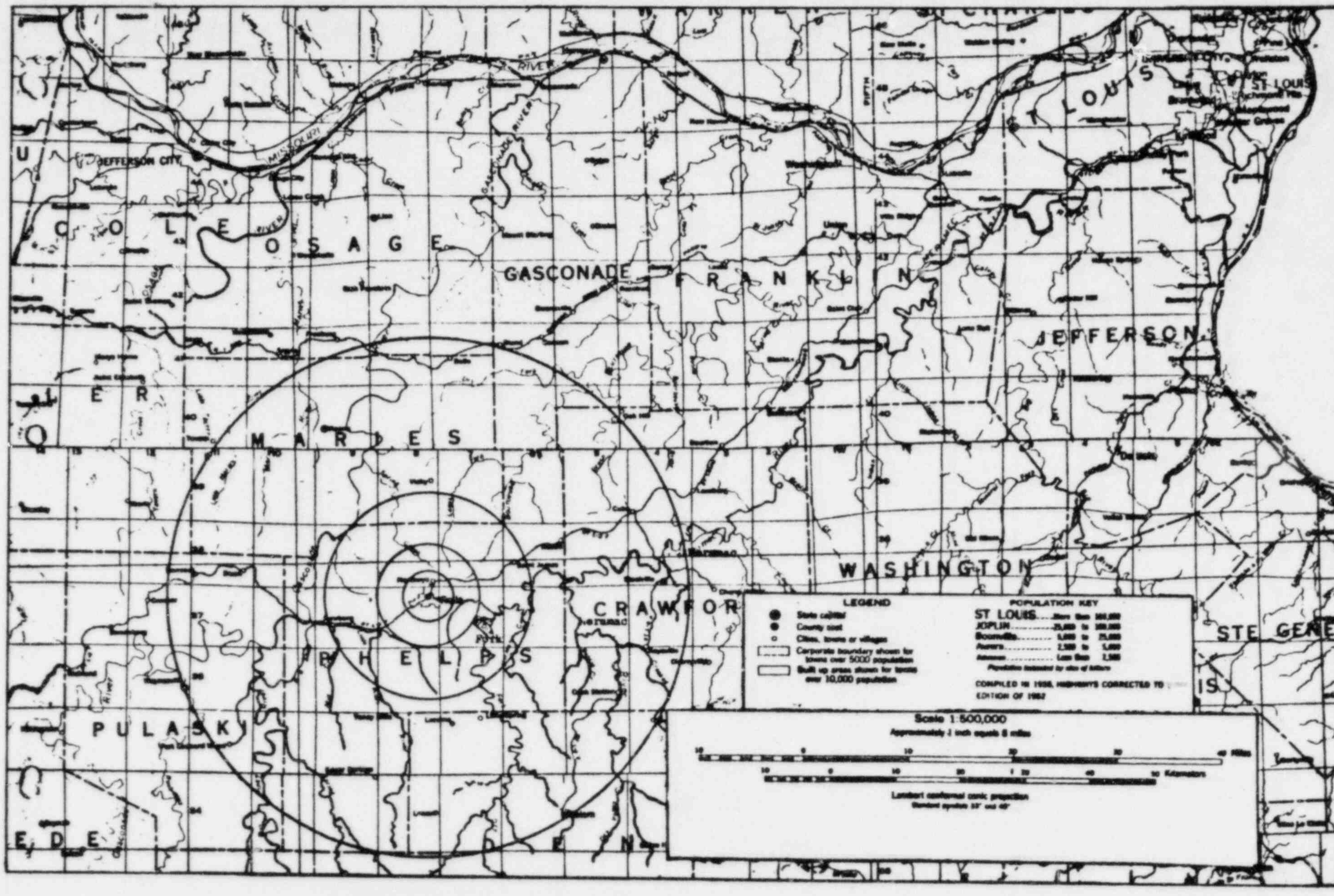


Figure 6. Route of surface drainage from reactor site.

the Rolla reactor site seems very remote.

2.5 Meteorology

Weather observations taken at the Missouri School of Mines, now the University of Missouri-Rolla, cover the period of October 1950 to June 1958 and temperature and rainfall data was extracted from these records. Direction and speed of winds was not available for the reactor site itself; however, complete records have been taken for a number of years at the CAA station at Vichy, Missouri which is 21 km (13 miles) north of the Rolla site. The topography at and surrounding Vichy is quite similar to the Rolla area. The Vichy elevation is 330 m (1100 ft) msl, same as that of Rolla. There seems no valid reason to assume that the data which has been collected at Vichy will not be adequate for the evaluation of the Rolla site.

The general climate of Missouri is a continental mid-western type. The area has generally adequate rainfall without extreme variations from year to year. Temperatures have, in general, a continental range with hot summers to generally mild winters ranging from over 38°C (100°F) to -20° C (-4° F). The prevailing wind across the area is South-Westerly. More specific analysis of the individual elements, particularly those affecting diffusion of material by the atmosphere, follows.

2.5.1 Wind Direction

Hourly wind observations for a 6 year period, 1948 to 1954, for the CAA Vichy Station were studied in detail. Table IV presents the percentage frequency of wind directions and average velocity for the period 1948 to 1954, inclusive. It is immediately evident that there is little variation of the most frequent winds from day to night, during periods of precipitation, and also when the visibility is low. These figures show that, on the average, the distribution of wind directions will be about the same regardless of the type of weather that is occurring. A detailed examination of the seasonal variations shows that this holds true for all seasons. The only major variation with seasons is that the west to northwest winds are more frequent during the winter as would be expected and that the highest wind velocities occur during the spring.

Figure 7 shows the remarkably constant prevailing wind directions with various wind conditions somewhat more graphically than does the table. Major flow is from the SSW quadrant regardless of the weather conditions occurring at the time. Highest wind speeds generally flow from the NW quadrant. The maximum wind speed observed for this period of record was 97 km/hr (60 mph). It is not improbable that rare wind gusts might reach as high as 137 km/hr (85 mph).

The data on winds occurring with precipitation was included in order that one might consider the effect of washout of

Tab[illegible]IV.

Annual Frequency of Wind Directions (Percent) and Average Speed

Direction Windspeed 6.4 km/hr (4 mph)	Daylight (07 - 1700 EST)	Night (18 - 0600 EST)	During Precipitation	During Low Visibility	Mean Wind Speed (km/hr)	(MPH)
N	3.8	3.3	7.3	6.7	13	8.1
NNE	2.3	2.6	3.2	4.4	13.8	8.6
NE	3.6	3.0	3.5	4.0	11.9	7.4
ENE	3.3	4.1	3.1	4.3	15.1	9.4
E	3.5	4.7	3.0	5.4	14.5	9.0
ESE	5.3	4.0	5.9	5.8	16.1	10.0
SE	5.8	6.6	6.4	8.3	14.8	9.2
SSE	6.2	8.7	9.3	7.7	17.5	10.9
S	11.6	12.6	9.8	7.8	18.4	11.4
SSW	8.4	11.2	5.0	7.6	17.9	11.1
SW	10.8	8.5	6.4	5.4	15.8	9.8
WSW	4.4	2.2	2.4	2.8	17.4	10.8
W	6.6	4.9	5.6	4.1	15.6	9.7
WNW	9.9	8.6	8.1	7.5	17.9	11.1
NW	6.7	5.4	8.8	5.8	14.2	8.8
NNW	4.0	3.7	9.0	6.0	15.6	9.7
3 mph and calm	3.6	4.5	3.4	6.1		

TOTAL MEAN WIND SPEED 15.9 km/hr (9.9 mph)

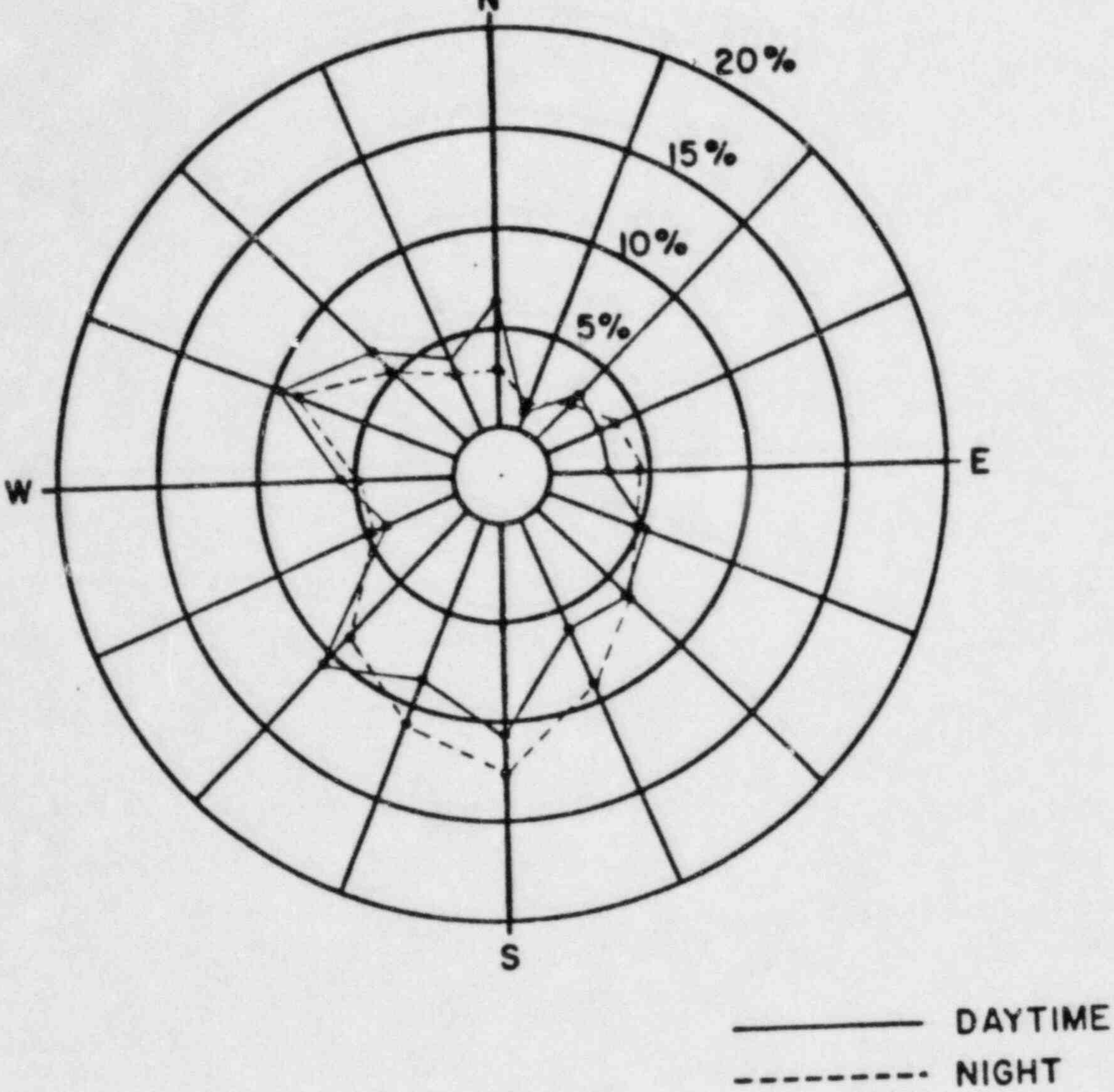


Figure 7. Annual frequency and wind direction.

potenti.. airborne contaminants. The wind frequency during periods of low visibility was included as a method of estimating the wind direction during periods of atmospheric stability. Since these do not differ markedly from the day or night wind frequencies, no special consideration of variation in weather conditions seems necessary in considering the transport of pollutants by the wind.

Another point of uniformity that can be noticed in the wind is the distribution of wind speeds with various weather conditions. Table V illustrates the annual frequency of various wind speed classes. It is noted that by far the largest proportion of the winds are between 6.5 km/hr (4 mph) and 19 km/hr (12 mph) averaging over 50% in all circumstances. The second largest occurrence is in the 21 km/hr 13 mph) to 39 km/hr (24 mph) category.

The Winds Aloft Summary for the St. Louis, Missouri area was examined. St. Louis is one of the nearest stations to Rolla which take upper wind observations. The general flow of air is from the west with most frequent flow from the west-northwest quadrant. Velocities increase steadily as the elevation above the surface increases.

Table V.

Annual Frequency of Wind Speeds (Percent)

km/hr (mph)	Calm	1.6-4.8 (1 - 3)	1.8 m/s 6.4-19.3 (4 - 12)	21-38.6 (13 - 24)	40¼-49.9 (25 - 31)	51.5-74.1 (32 - 46)	75.7 (47)
Daylight	3.0	1.0	68.2	27.7	.1	0.1	0.1
Night	3.6	0.94	82.9	12.3	0.3	0.1	0.1
During Precipitation	5.0	1.1	80.7	12.8	0.28	0.13	0.1
Low Visibility	3.1	0.69	73.3	20.3	2.4	0.1	0.17

2.5.2 Precipitation

Climatological observations for the University of Missouri-Rolla site were examined for the years 1951 to June 1958. Average annual precipitation for this period was 0.99 m (39 in) per year. The minimum annual precipitation was 0.57 m (22.5 in) period with most precipitation is generally April through August and the least amounts are recorded in December and January. The range of average precipitation is from about 3.8 cm (1 1/2 in) per month at minimum periods to around 11.5 cm (4 1/2 in) per month at the time of the rainy season. Table VI shows the average number of days with precipitation equal to or greater than certain specified amounts. From this table, it can be seen that precipitation amounts equal to or greater than a tenth of an inch will occur about 20% of the days in a year. Heavy amounts of 1 cm (1/2 inch) are less frequent. It should be noted, however, that precipitation is extremely variable. This is borne out by the range of precipitation occurrence which is presented in Part (b) of Table VI. The central Missouri area, including the town of Rolla, is subject to storms producing heavy precipitation. These storms may occur in any season of the year but high intensity-short duration rainfall can be expected with considerable frequency during the spring and summer months with the passage of thunder storms over the area. Table VII is a listing of the maximum precipitation recorded during the period of record between 1950 and 1958.

A small proportion of the wintertime precipitation will be

Table VI.

(a) Average Number of Days of Precipitation Per Year

(Inches) cm	Jan.	Feb.	March	April	May	June	July	Aug.	Sept.	Oct.	Nov.	Dec.
(.10) > 0.25	4.3	7.4	6.3	8.1	9.6	6.3	6.8	5.0	4.6	5.4	4.7	5.4
(.50) > 1.27	1.4	3.7	2.4	3.0	4.4	2.4	3.4	2.4	2.0	2.4	1.6	1.3
(1.00) > 2.54	*	*	0.6	0.6	2.1	2.0	1.0	1.3	0.7	0.8	0.6	*

(b) Range of Precipitation Occurrence

(.10) > 0.25 from 55/year to 107/year

(.50) > 1.27 from 21/year to 50/year

(1.00) > 2.54 from 4/year to 16/year

* Less than 1/2

Table VII.

Maximum Precipitation

Duration (Hours)	Amount (cm)	Amount (in)	Date
1/2	4.78	(1.88)	July 1957
1	3.58	(1.41)	July 1952
2	7.62	(3.00)	April 1953
3 1/2	7.85	(3.09)	August 1955
4	12.59	(4.96)	June 1958
6	8.00	(3.15)	December 1957
24	9.93	(3.91)	July 1951
48	12.80	(5.04)	June 1958

recorded as snow. It can be expected that only about 0.66 m (26 in) of snowfall will be recorded each winter and this can be expected to melt off and not accumulate. Heavy snowfalls are uncommon. The maximum snow fall recorded during any one 24 hour period was 0.5 m (19.5 in) on November 5, 1951.

3. REACTOR

The University of Missouri-Rolla Reactor (UMRR) is a thermal, heterogeneous, pool-type reactor. It is presently licensed for 200 kW of thermal power. The reactor core is fueled with enriched uranium-235 and is immersed in a pool filled with deionized water. The pool water serves as a moderating, reflecting, shielding, and heat removing medium. Some important design characteristics of the UMRR are given in Table VIII.

The reactor produces no steam. It is operated primarily for educational and research purposes. The facility is also made available for the training of personnel for industry and electric utilities.

3.1 Reactor Building

The Reactor Building (Figure 8) is constructed of insulated steel curtain walls. The doors and windows are weather stripped, the vents connected with the ventilation system are automatically closed when the system is shut down, and other points where air may leak out of the building are caulked.

The main floor contains a reactor bay, control room, counting room, and office space. At the beam port and thermal column end of the reactor bay, the floor is dropped to provide access to the beam tube and thermal column as they emerge from

Table VIII. Design and Operational Characteristics of the UMRR

General	
Type	Open pool
Core	Heterogeneous
Licensed Power	200 kW
Moderator	H_2O
Coolant	H_2O
Reflector	H_2O or H_2O and graphite
Shield	H_2O and concrete
Heat removal	Natural convection
Maximum inlet core temperature	(135°F) 57.2°C
Fuel	
Type	MTR
Geometry	Curved plates
Fuel	Uranium - 235
Enrichment	$\approx$ 90%
Active length	(24 in.) 0.61 m
Control rods	
Shim/Safety	3 rods
Material	Boron and Stainless Steel
Drive speed	(6 in/min) 15¼ cm/in
Regulating	1 rod
Material	Stainless Steel
Drive speed	(24 in/min) 0.61 m/min

Table VIII. (cont.)

Nuclear data

Average thermal neutron flux	1.6×10^{12} n/s-cm^2
Reactivity coefficients	
Temperature (Moderator)	$\approx -1.0 \times 10^{-4}$ $\Delta k/k/°F$
Void	$\approx -7.0 \times 10^{-7}$ $\Delta k/k/cm^3$
Prompt neutron lifetime	4.5×10^{-5} s
Effective delayed neutron fraction	7.5×10^{-3}

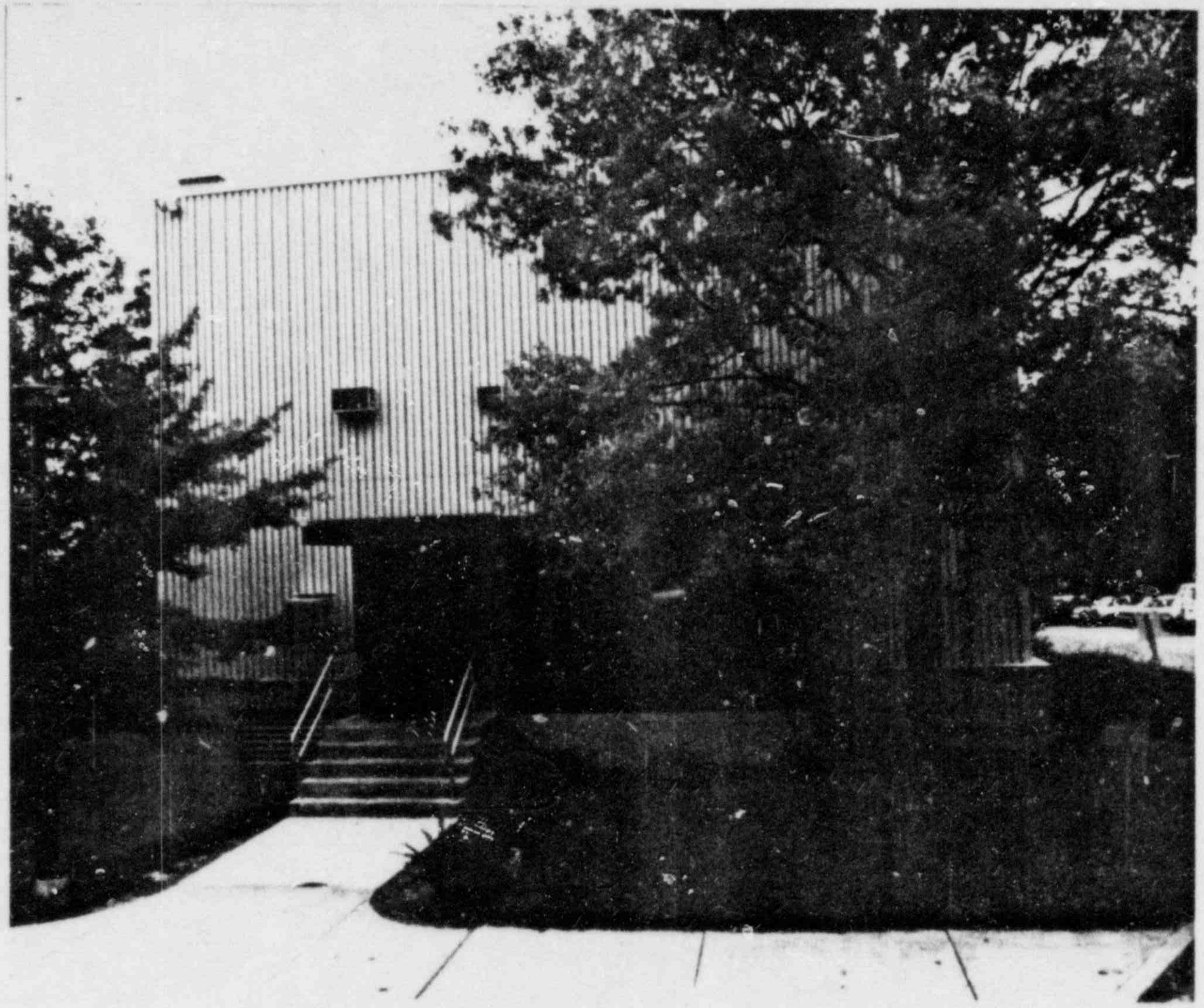


Figure 8. The UMR Reactor Building

the reactor pool. The facility layout is shown in detail in Figures 9 and 10. The volume of the Reactor Building is about $1.7 \times 10^3 m^3$ ($6.1 \times 10^4 ft^3$).

All a eas of the building are expected to remain free from radioactive contamination. If the reactor bay should become contaminated, it can be sealed off from all the other rooms on the main floor. The volume of the reactor bay is approximately $1.4 X 10^3 m^3$ ($5 X 10^4 ft^3$).

3.2 Reactor Core

The reactor core consists of MTR-type fuel elements and, if desired, one or two special elements. Four of the fuel elements contain movable neutron-absorbing rods which are used to control the reactor. The two special elements are used mainly for irradiation and isotope production purposes. Each element is positioned in a grid plate supported by an inverted tower suspended from the bridge spanning across the pool. In Figure 11 a reactor core configuration is partially shown.

3.2.1 Fuel Elements

A fuel element as designed by Curtiss-Wright Company is shown in Figure 12. It contains ten fuel bearing plates. Each plate is a sandwich consisting of a 0.051 cm (0.020 in) thick layer of aluminum-uranium oxide completely clad in a 0.051 cm (0.020 in) thick layer of aluminum. This thickness of aluminum

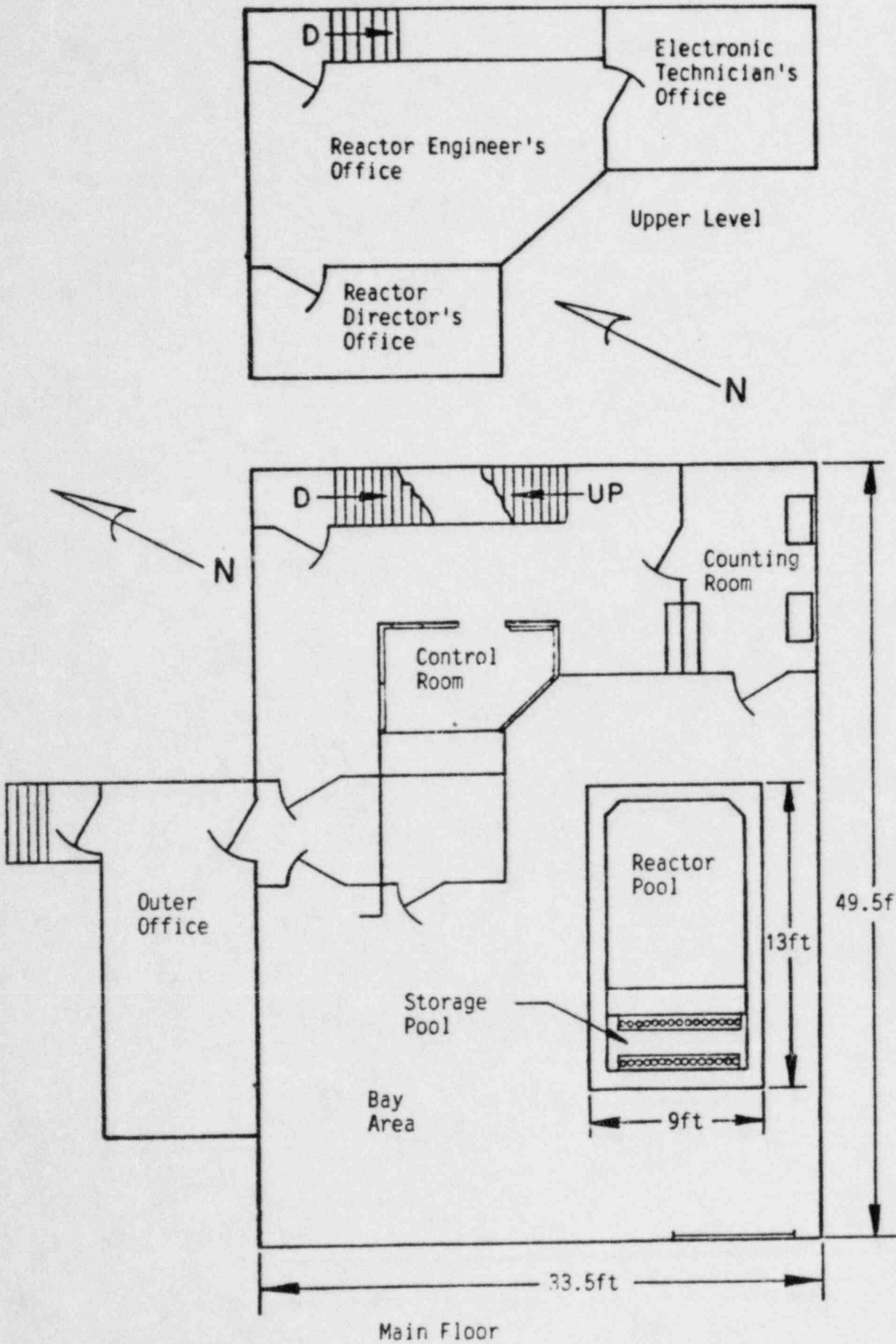


Figure 9. Facility layout.

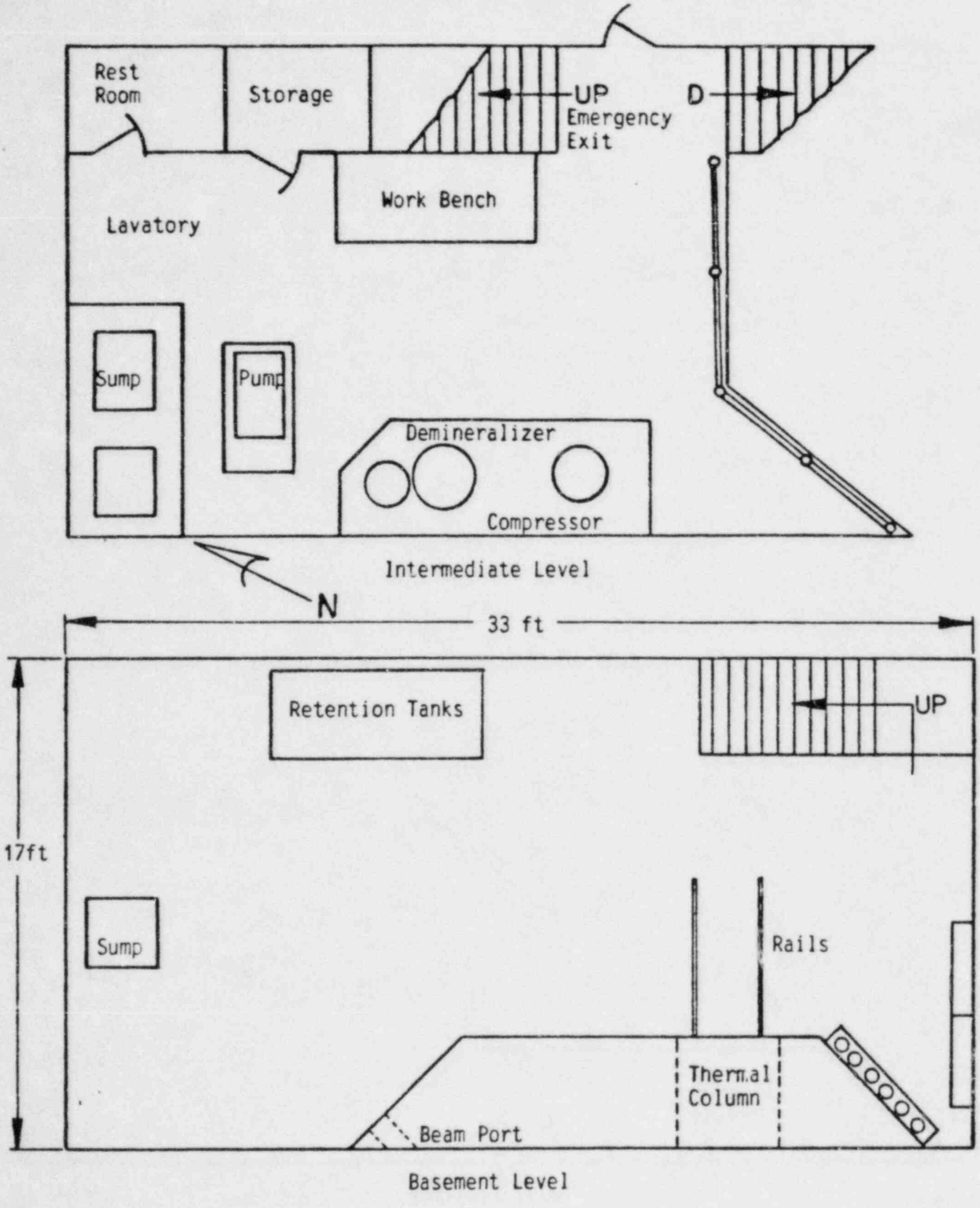


Figure 10. Facility layout.

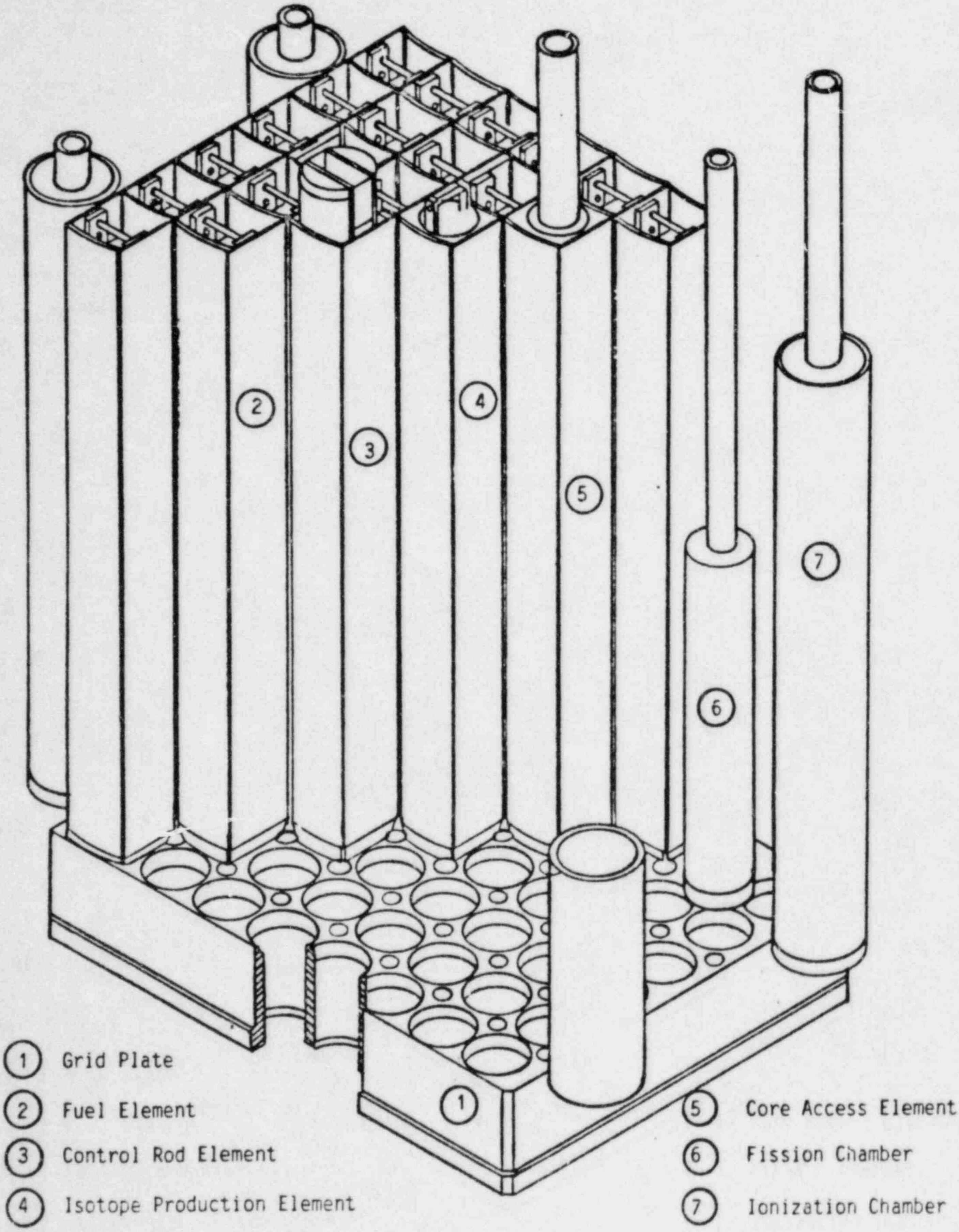


Figure 11. Reactor core.

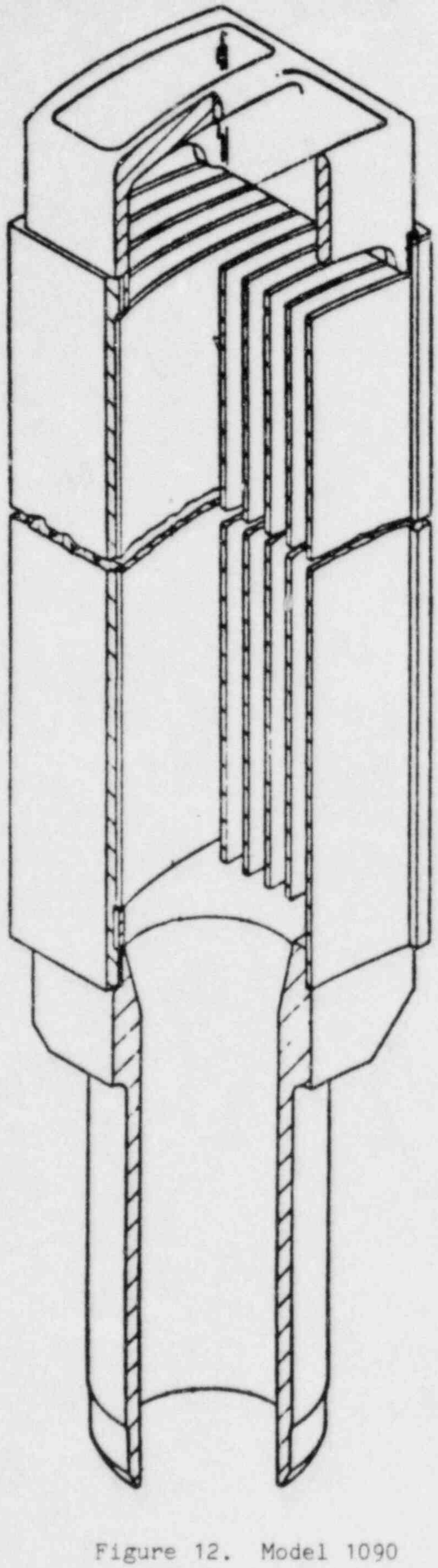


Figure 12. Model 1090
fuel element.

is sufficient to contain, under normal circumstances, all fission fragments. The uranium is enriched to 90% in the 235 isotope. The fuel layer is approximately 6.35 cm (2.5 in) wide and contains 17 grams of U-235. The finished plate is approximately 7.6 cm (3 in) wide, 60.9 cm (24 in) long and 0.152 cm (0.060 in) thick.

The fuel plates are fastened into groups of ten with aluminum side plates so that the finished element has an almost square cross section measuring 7.6 cm (3 in) by 7.6 cm (3 in). At one end a male guide section of circular cross section is attached and at the other end a handle, bringing the over-all length of an element to 0.91 m (3 ft). The guide piece is inserted into a tapered hole in the grid plate which supports the entire fuel element array or core. Two half elements, each consisting of five fueled and five dummy plates, are also available.

The elements and grid plate are designed so that the fuel bearing plates are spaced uniformly throughout the core. Both ends of the elements are open so that cooling water may flow between the fuel plates. The tolerances are set so that if all dimensions are off in the same direction there will be only a 20% reduction in coolant flow through any channel. The outer surfaces of the elements in the interior of the core are cooled by water which passes through a channel formed at the intersection of four elements and through an auxiliary coolant hole in the grid plate.

3.2.2 Control Rod Fuel Elements

A control rod fuel element has the central four plates removed to accomodate a guide tube for the control rod. The remainder of the fuel plates are spaced so that they have .9, .9, .6, ...6, .9, .9 times the coolant flow area of that in a normal element. The reduction to 0.6 is permitted since the control rod channel supports the removal of heat generated in both adjacent fuel plates. In Figure 13 a sketch of a control rod fuel element is shown. There are four control rod fuel elements in any core configuration.

3.2.3 Control Rods and Drive Mechanisms

Reactor power is regulated by using three shim/safety rods and one regulating rod.

All four control rod systems are equipped with console mounted electronic position indicators which measure the heights of withdrawal of each respective rod in inches. The remote position indication systems are accurate to within ± 0.25 cm (0.10 in), which translates to ≈ ± 0.01% delta k/k.

A shim/safety rod consists of a grooved, boron steel rod. The nominal dimensions are: 2.23 cm (7/8 in) thick, 5.7 cm (2 1/4 in) wide, and 61 cm (24 in) of active poison length. The borun content is about 1.5 to 1.7 percent natural boron.

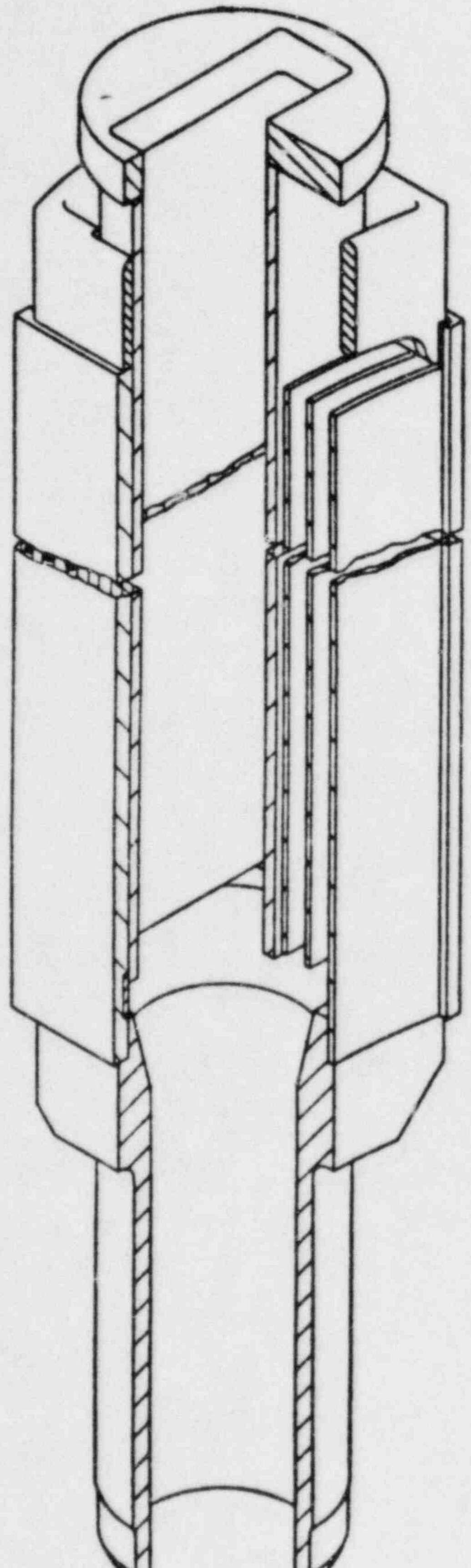


Figure 13. Model 690-C
control rod fuel element.

boron content is about 1.5 to 1.7 percent natural boron.

The shim/safety rods serve for both shim and rapid shutdown purposes. They are magnetically coupled to their rod drive extensions and in the event of power failure or receipt of a scram signal the current to the coupling magnets is interrupted and the rods fall freely into the core. The magnets do not drive-in automatically after a reactor scram. The normal magnet current is of such value as to limit the total weight lifted to only that required for satisfactory stable operation of the control system. A piston attached to the upper end of the safety rod enters a special damping cylinder as the safety rod approaches the full insert position. The water forced upwards around the piston provides a hydraulic snubbing action which permits the safety rod to come to rest without damage.

Each shim/safety rod provides approximately 3% delta k/k, the exact worth varying with different core loadings. The ganged worth of three safety rods is about 9% delta k/k.

The shim/safety rods have a maximum rate of withdrawal of 15.2 cm/min (6 in/min). At the most effective position, approximately 3.3 cm (13 in) withdrawn and designated as the shim range, this speed corresponds to a rate of change in reactivity for any one rod of about 0.02% delta k/k per second.

The regulating rod is used for fine control. It consists of a type 304 stainless steel tube having a wall thickness of

0.165 cm (0.065 in), a cross-sectional shape 2.23 cm (0.875 in) wide by 5.72 cm (2.25 in) long with an oval end, and an effective poison length of about 61 cm (24 in). The top tube end plug of the regulating rod contains a 0.953 cm (3/8 in) diameter hole to permit free circulation of water through the tube to eliminate the danger of trapping air in the rod and producing a variable void condition.

The regulating rod is limited to a total worth of 0.7% delta k/k and a maximum withdrawal rate of 61 cm/min (24 in/min). In its most effective position the maximum rate of change of reactivity of the regulating rod is about 0.02% delta k/k per second for a water reflector.

The regulating rod is bolted directly to the rod drive assembly instead of being connected through a magnetic coupling. It does not drive-in automatically upon the receipt of a scram signal.

The control rods are driven by an electro-mechanical linear actuator located at the bridge. The actuator is essentially a ball bearing type screw driven through a gear reduction unit by a low inertia servo motor. A variable loading ratchet type drive mechanism connects the screw to the gear reduction unit.

The following description of the mechanical arrangement of the shim/safety rod drive assembly used in this reactor (Figure 14) outlines the design safeguards incorporated in the control

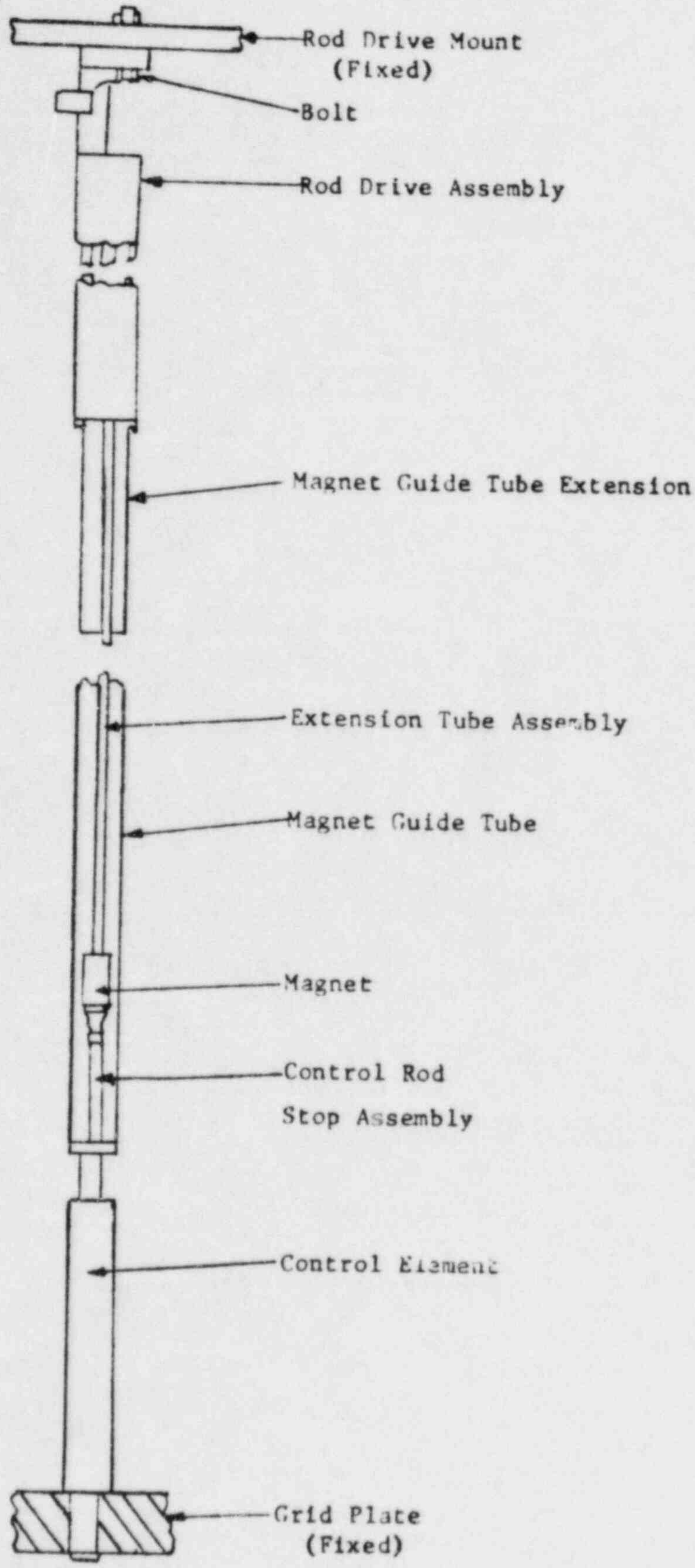


Figure 14. Control rod drive system.

rod drive system.

A control rod fuel element containing fueled plates and an axial hole for a control rod is inserted into the grid plate. Attached by bolts to a special flange of the control element is a stop assembly approximately 10.2 cm (4 in) in height. A guide tube assembly consisting of a magnet guide tube bolted to a magnet guide tube extension is placed over the stop assembly and rests on the control rod fuel element flange thus accomodating the top end of the control rod fuel element. The top end of the magnet guide tube extension is fastened to the rod drive assembly housing which is, in turn, bolted to the rod drive mount. This rod drive mount is bolted to the reactor bridge.

With this arrangement it can be seen that the accidental lifting of a control element out of the core by movement of a shim/safety rod is impossible without prior disassembly of the rod drive or deliberate omission of mechanical components. In addition, it is pointed out that a special adjustable slip clutch arrangement is incorporated between the drive motor and the linear actuator of the shim/safety rod drive to insure that excessive loading on the shim/safety rod drive will cause the clutch to slip, thereby preventing movement of the shim/safety rod. Furthermore, this special clutch is designed so that the force available to insert the shim/safety rod is always greater than that available for withdrawal, regardless of the clutch adjustment setting. The regulating rod drive assembly is identical to that of a shim/safety rod drive assembly.

3.2.4 Core Access and Isotope Production Elements

The core access element, shown in Figure 15 (a), is used to provide access to the inner part of the reactor core and is also a dry irradiation facility. The assembly is similar in shape to a fuel element and consists of a hollow aluminum can in the position usually occupied by the fuel plates. The top portion of the assembly will receive a sealing plug similar to that for the isotope production element except that it is tubular with an aluminum tube welded into its center. This aluminum tube projects upwards above pool water level and is curved under water to prevent neutron or gamma streaming out the upper portion of the pipe. Samples for irradiation are lowered down the pipe on the end of a leader.

The isotope production element (Figure 15 (b)) is filled with graphite with a hole passsing through it to permit a neutron start-up source or an irradiation sample to be inserted into the core. The graphite is entirely clad in aluminum, the inner cladding forming an aluminum tube. The element may be used as a dry irradiation facility. The top sealing plug contains a groove for an O-ring and a horizontal hole so that the plug may be secured to the complete assembly by means of aluminum pins.

3.2.5 Core Support Structure

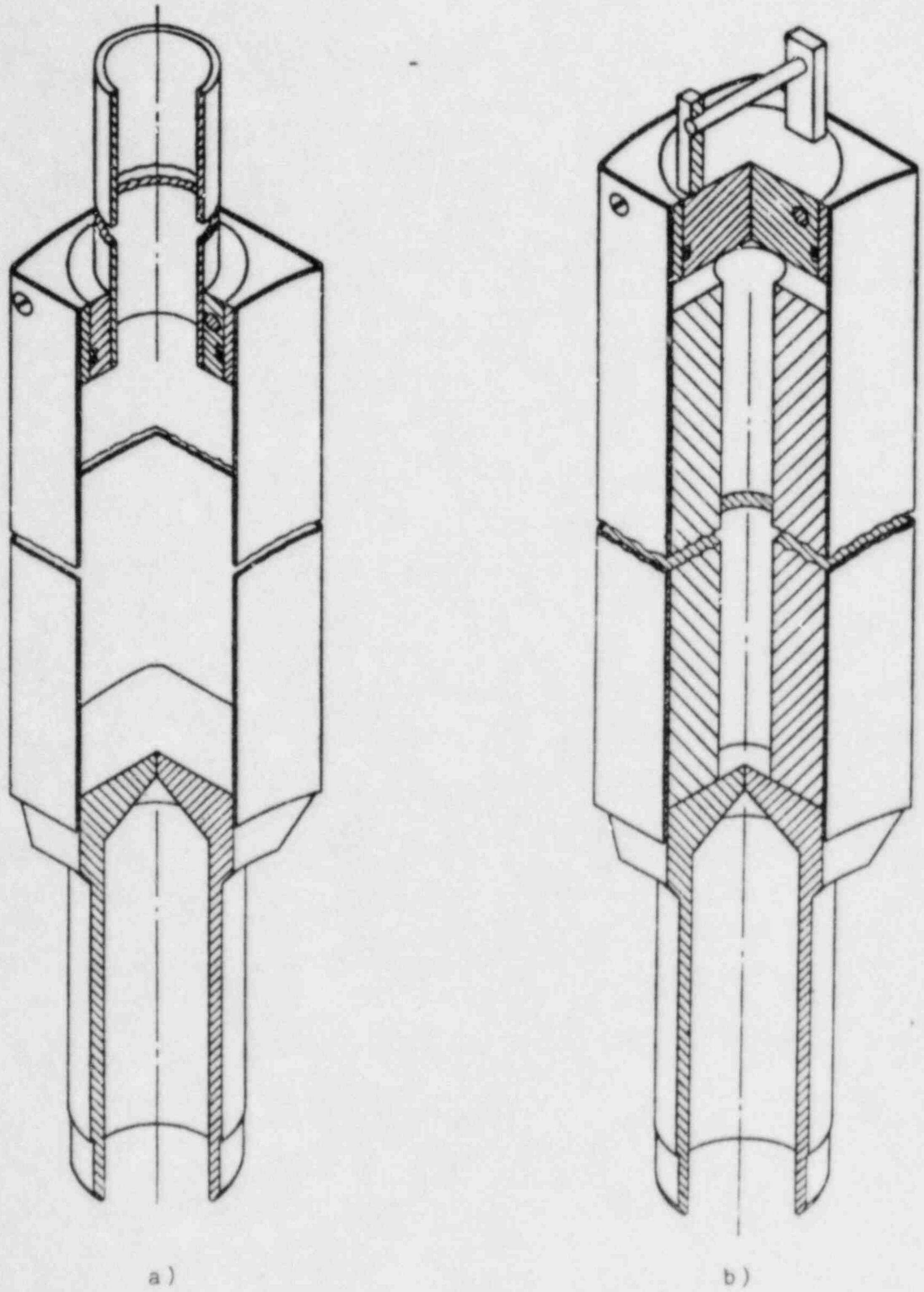


Figure 15. (a) Core access element

(b) Isotope production element.

The reactor core is supported by an inverted aluminum tower assembly suspended from the bridge which spans the pool as shown in Figures 16 and 17. The bridge is made of structural steel, approximately 3.3 m (11 ft) long and 1.35 m (4.5 ft) wide and is wheel mounted on tracks located parallel to the long axis of the reactor pool atop the pool walls. The bridge can be moved along its rails for a distance of approximately 1.8 m (6 ft) from its normal operating position, thus providing water shielding between the experimental facilities and the reactor core when required. Mechanical stops are provided on the bridge rails to limit bridge travel within the pool area. An inadvertent movement of the reactor bridge causes the reactor to be scrammed (see Section 3.5.8.)

The grid plate, shown in Figure 11, is made of 12.7 cm (5 in) thick special aluminum with 54 element positions arranged in a 6 x 9 array. The element holes pass through the grid plate to permit circulation of coolant through the core. The holes which do not hold an element are not plugged. Auxiliary coolant holes between the element holes are provided to permit coolant flow between outside plates of the fueled elements.

3.3 Reactor Pool

The reactor pool is a rectangle approximately 5.7 m (19 ft) long, 2.7 m (9 ft) wide and 8.1 m (27 ft) deep and houses the reactor, a beam port, and a thermal column (see Figures 16 and 17). It contains about 1.2×10^5 ℓ (32000 gal.) of highly

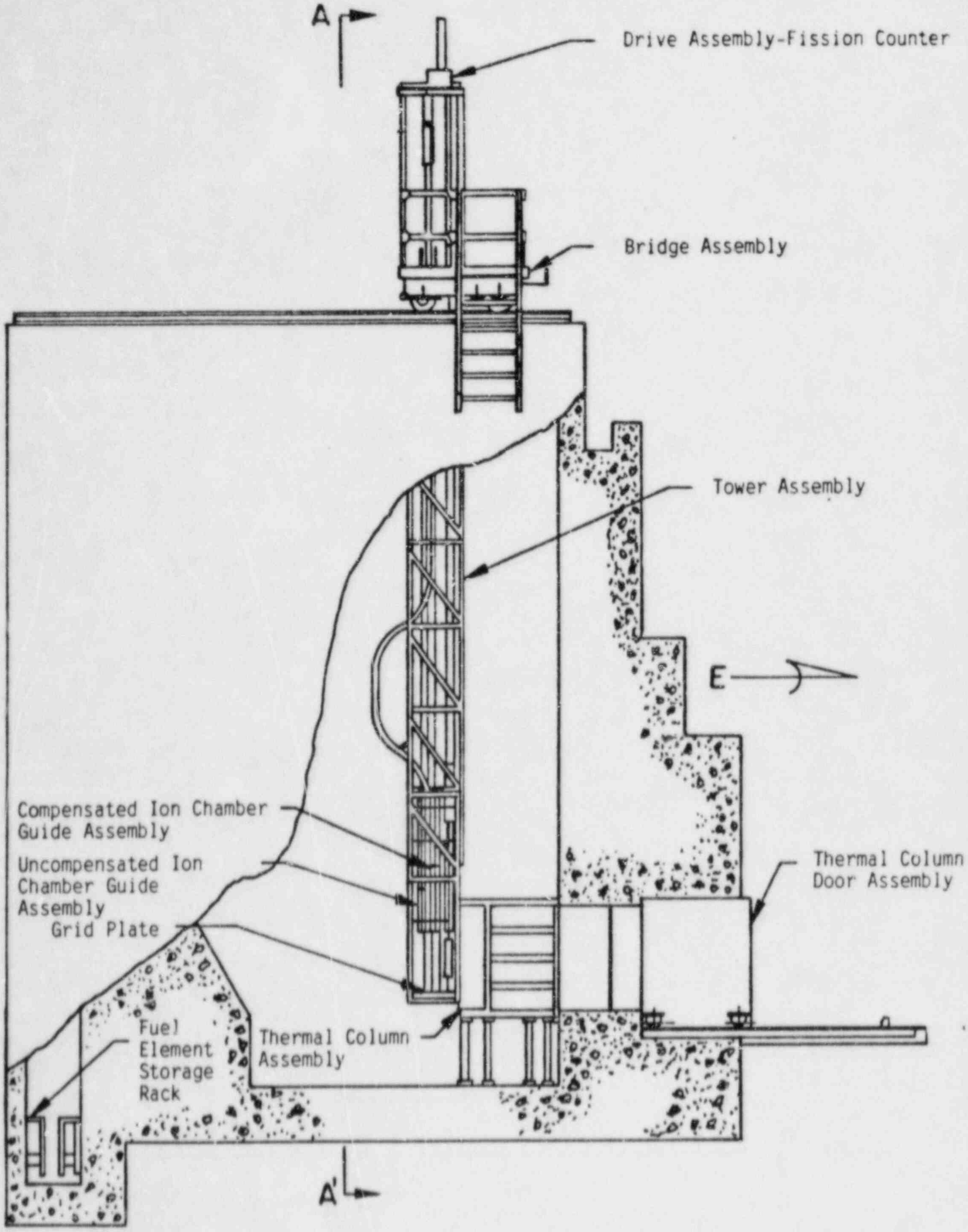


Figure 16. Cross section of the UMRR.

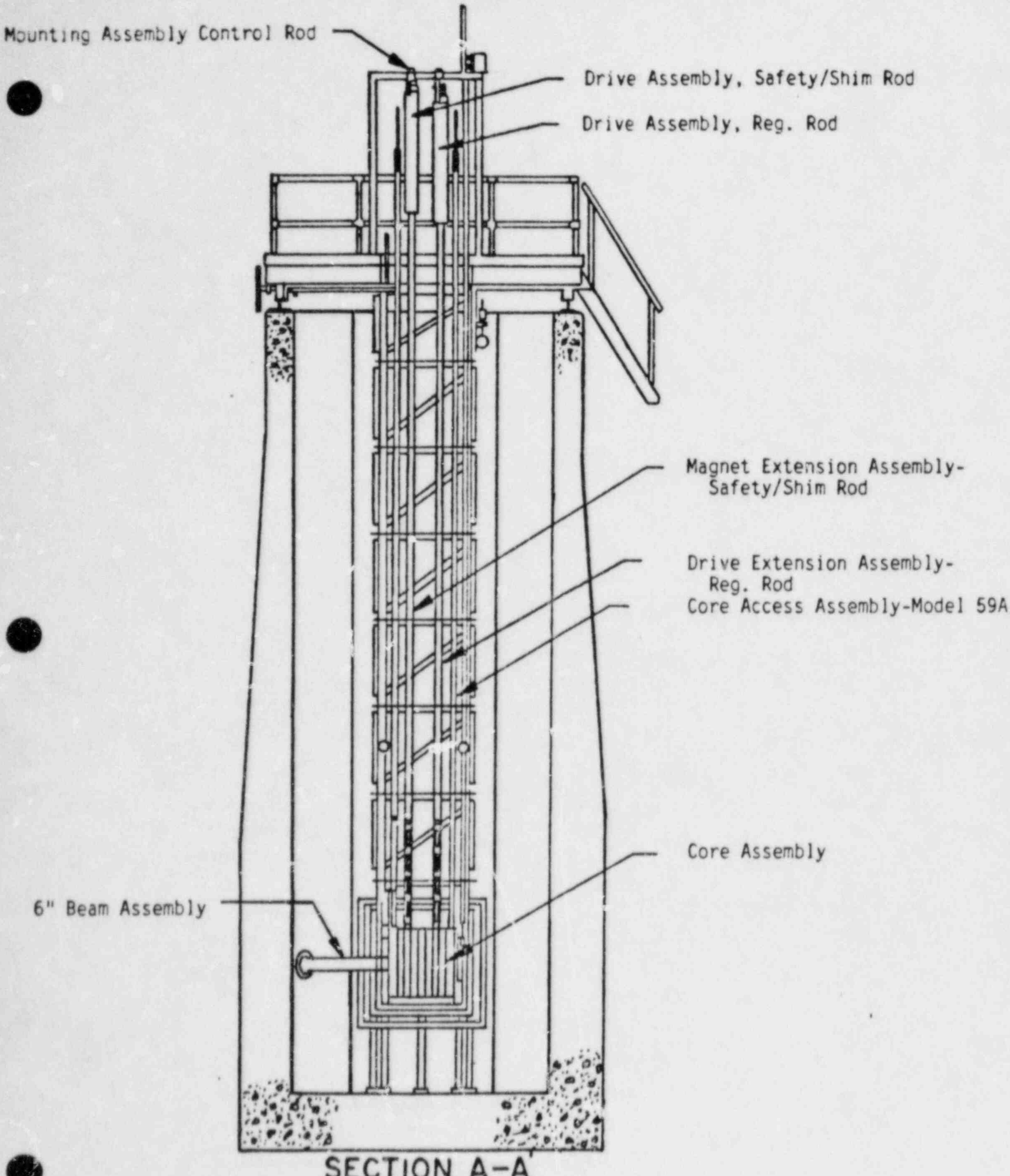


Figure 17. Cross section of the UMRR.

purified water. Pool walls are of ordinary reinforced concrete 45.7 cm (18 in) thick except at the beam hole and thermal column end where the thickness is increased to 1.98 m (78 in). The increase in wall thickness extends above the pool floor level in a stepped arrangement at the end of the pool (see Figure 16). The internal concrete sides and floor of the pool have several coats of protective vinyl paint to prevent excessive leaching of minerals from the concrete into the water.

At the opposite end of the pool from the thermal column is a fuel element storage space. This is formed by a reinforced concrete bulkhead extending 4.8 m (16 ft) above and 1.05 m (3.5 ft) below the pool floor. It is placed 0.6 m (2 ft) from the main pool wall. Fuel element storage racks are installed at the bottom of this section. If it becomes necessary to drain the reactor pool, fuel elements will be transferred to the storage rack prior to draining. The bulkhead will insure that at least 4.8 m (16 ft) of water covers the tops of the fuel elements at all times. A concrete insert between the bulkhead and the main pool floor insures adequate shielding to personnel working in the drained pool.

3.4 Reactivity Parameters and Heat Removal

The most important factors affecting criticality of the UMRR are discussed in this section. Furthermore, requirements on the excess reactivity of the reactor core are established. The heat removal is briefly described and the great potential of the reactor pool to dissipate generated heat is shown.

3.4.1 Moderator Temperature Coefficient

Many of the parameters which determine the multiplication factor depend on the reactor temperature. As a result, a change in the moderator temperature leads to a change in the multiplication factor, and hence alters the reactivity. This dependency is best expressed in terms of the moderator temperature coefficient of reactivity. It is defined as the ratio of the change in reactivity to the change in the moderator temperature.

It is desirable that the moderator temperature coefficient be negative since an increase in temperature will then lead to a decrease in the reactivity with a consequential reduction in the reactor power. Usually, the value of the moderator temperature coefficient is determined experimentally. The UMRR is designed such that the moderator temperature coefficient is no more positive than -4×10^{-5} delta k/k/°F. In practice, its magnitude is larger, with the current value being about -1.10^{-4} delta k/k/°F.

3.4.2 Void Coefficient

Another reactivity parameter encountered at the UMRR is the void coefficient. When the water is removed from the core or from its proximity changes occur in the moderation, leakage, and absorption of neutrons. These changes manifest themselves as reactivity changes. The void reactivity coefficient is defined as the ratio of the change in reactivity to the voided volume.

For the purpose of reactor safety and stability, it is desired that the void reactivity coefficient be negative. For the UMRR the design limit of the void reactivity coefficient is to be no more positive than -2×10^{-7} delta k/k/ cm^3 of void volume at the periphery of the core. The current value experimentally determined is about -1×10^{-6} delta k/k/cm^3.

3.4.3 Xenon-Poisoning

Many different fission products are created during the fission process. Because they absorb neutrons their buildup in the reactor reduces its multiplication factor. In thermal reactors such as the UMRR the most important fission product is xenon-135 because of its large thermal absorption cross section. The magnitude of xenon-poisoning is proportional to the neutron flux and operational history. In the UMRR, it is not the major reactivity effect. For example, the shutdown peak xenon-poisoning after 8 hours of full power operation was experimentally determined to be about 2×10^{-3} delta k/k.

3.4.4 Excess Reactivity

The excess reactivity is defined as that value of reactivity which would occur if all control rods were completely removed from the reactor core. It is measured for a given core loading starting from a clean, cold core. A designated core loading may include irradiation facilities such as pneumatic tubes, the isotope production element, the core access element or other facilities of such nature that they become a portion of the core when installed.

An excess reactivity must be built into the reactor core in order to compensate for a number of reactivity losses. (The most important ones for the UMRR have been discussed in previous sections.) Also, a sufficient reactivity must be available to allow for an adequate reactor period. Therefore, the minimum excess reactivity which is needed to allow for an extended operational flexibility (e.g. 24 hrs) consists of the following:

0.6% temperature effect

0.2% Xe-poisoning

+ 0.2% adequate reactor period

1.0% delta k/k total

However, this value of the excess reactivity does not account for any eventual negative reactivity effect due to irradiation experiments containing neutron-absorbing materials (e.g. cadmium). Therefore, depending on the experimental requirements more excess reactivity might be needed. In Section 9.4 an analysis of the upper reactivity limit requested for moveable experiments is presented.

3.4.5 Heat Removal

Heat generated in the reactor core is removed by natural convection and dissipated in the reactor pool water. Depending on the number of fuel elements in the core, the average heat flux at 200 kW is approximately 1.6 W/cm^2 (4.9×10^3 Btu/hrft^2). Using the measured peak factor for the neutron flux the corresponding maximum heat flux in the central fuel element is 3.2 W/cm^2 (10^4 Btu/hrft^2).

The coolant outlet temperature in one of the fuel elements was experimentally determined. Using this value, the coolant velocity calculated from a heat balance is about 0.1 m/s (0.3 ft/s).

This velocity is much too low to cause collapsing of parallel fuel plates due to a hydraulic pressure unbalance across the plates. Such failures have been observed in fuel-plate assemblies of some earlier reactors (12) at the flow velocities above 10 m/s (30 ft/s). Therefore, it is concluded that there is a substantial safety margin against fuel collapse in the UMRR.

The size of the reactor pool is such that the reactor, when started up at the usual operating pool water temperature of 20°C (68°F), could be operated for about 24 hrs at full power before the bulk temperature in the pool reaches its operational limit of 57°C (135°F) (see Sec. 5.2). The capability to remove decay heat after a full power run is best documented when considering the rate with which the reactor pool naturally cools off when heated up above the ambient temperature. This rate is about one order of magnitude larger than the rate with which decay heat heats up the pool water after a reactor shutdown from full power.

Heat transfer calculations have shown that the temperature drop across the "fuel meat" and the aluminum cladding is very small. For example, for the heat flux of 1.6 W/cm^2 it is about 4°C. On the contrary, the temperature drop between the cladding and the bulk coolant is relatively large. Therefore, it is this temperature drop which largely determines the central temperature of the fuel.

The pertinent literature was researched for heat transfer correlations available for natural convection flow. Two correlations, derived for vertical plates and tubes, were used in order to limit the range of uncertainties arising from a non-uniform heat and temperature distribution along the fuel element channel. Results of the calculations are summarized in Figure 18. Here, the wall temperature of the fuel cladding is shown as a function of the coolant bulk temperature, i.e. pool temperature, for different values of the cladding heat flux. At the maximum power, the peak heat flux, and within the operational limit of the UMRR the cladding temperature does not exceed 90°C (194°F). Consequently, this corresponds to the maximum fuel centerline temperature of about 94°C (201°F).

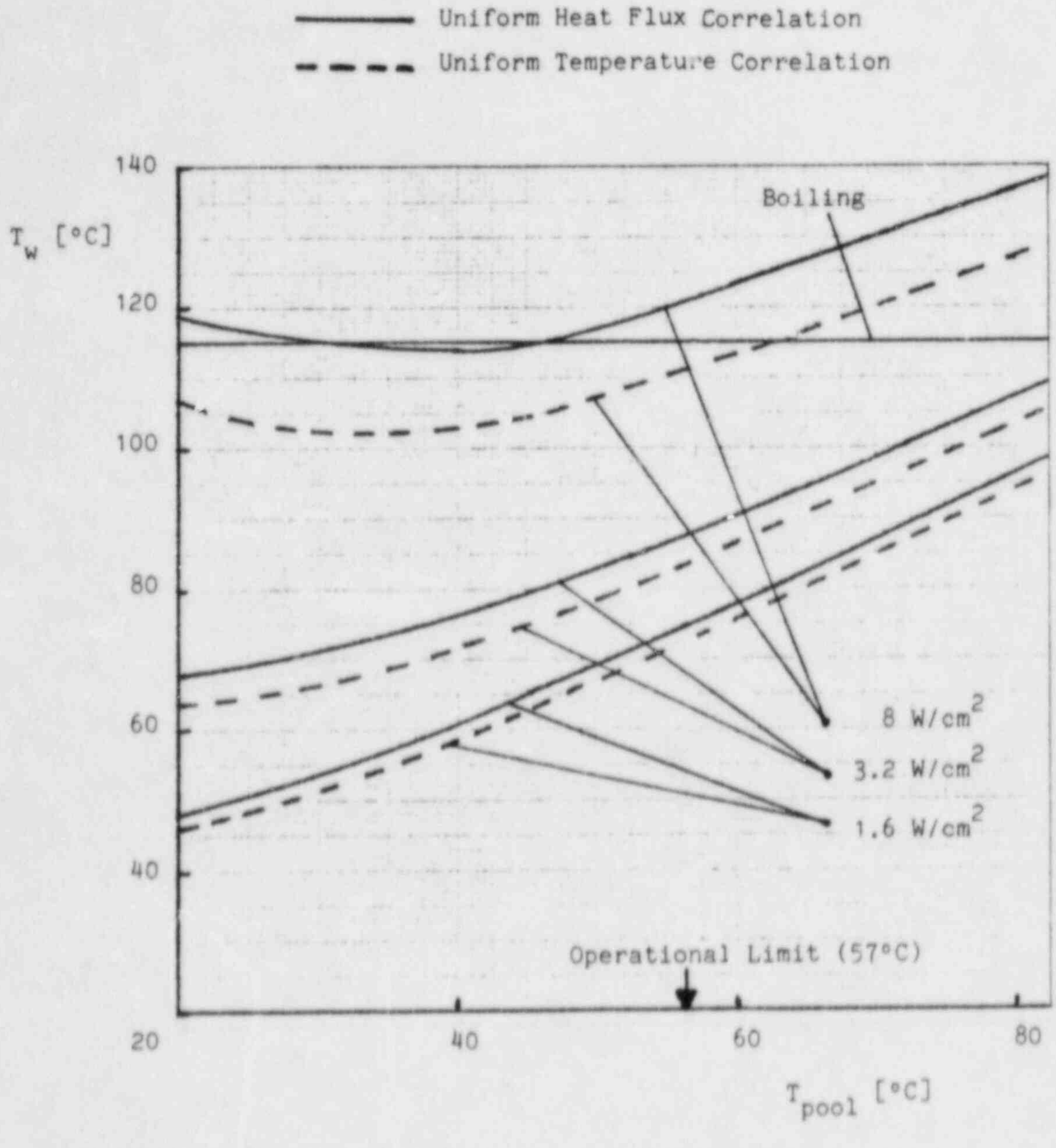


Figure 18. Calculated cladding wall temperature.

In the case of a failed movable experiment, as discussed in Section 9.4, the heat flux in the hot channel can temporarily reach the value of about 8 W/cm². Then, the corresponding cladding temperature might be equal and indeed might even exceed, the saturation temperature by a few degrees of C. This is the region of nucleate boiling heat transfer. It is known from the pool boiling experiments (5) that in this region, while the heat flux can be in the order of tens of W/cm², the cladding temperature is just about 20-30°C above the boiling temperature. For example, the peak heat flux of 50 W/cm², which would correspond to the reactor power of about 3 MW, would cause the cladding wall temperature to rise to about 140°C (284°F). The melting point of the aluminum cladding is, however, 660°C (1220° F). Therefore, the capability of heat removal in the region of nucleate boiling represents an additional safety factor in the thermal-hydraulics of the UMRR.

3.5 Reactor Instrumentation and Control

The function of the reactor instrumentation is to provide adequate information for the operator and to generate signals to control the reactor or to shut it down if necessary.

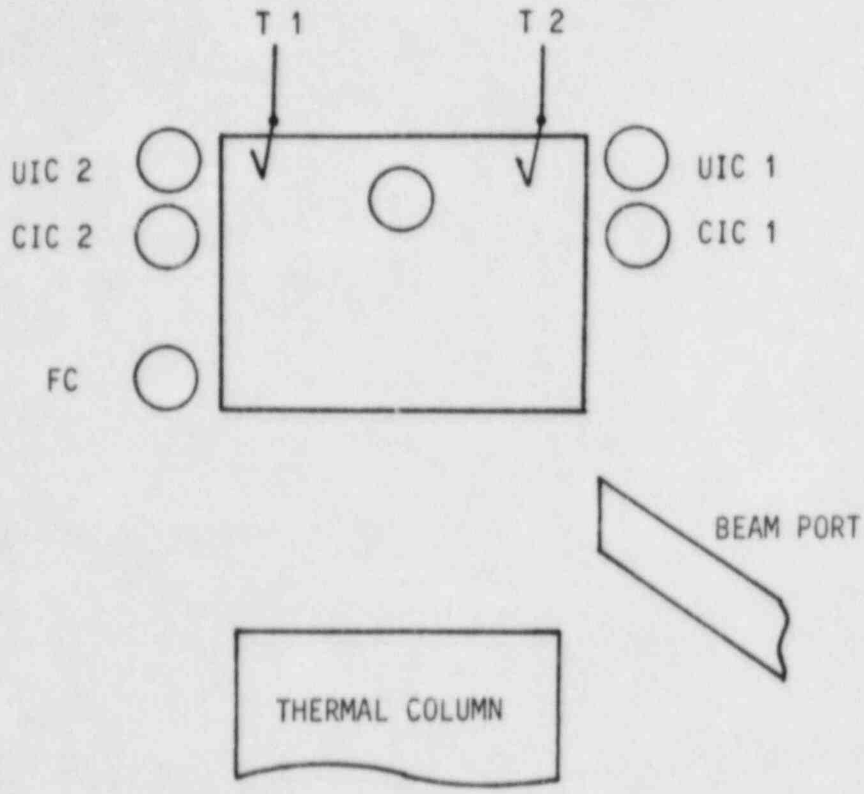
The nuclear instrumentation consists of a fission chamber, two compensated and two uncompensated ion chambers. All neutron detectors are arranged near the reactor core to facilitate repair, maintenance, and repositioning. They are encapsulated in water-tight aluminum tubes. The fission chamber is provided with a motor driven positioning mechanism and position indication system; the other detectors are provided with an adjustable screw mechanism to facilitate coarse and fine manual adjustment.

The non-nuclear reactor instrumentation consists of three thermocouples to measure the pool water temperature at the core inlet and outlet.

The spatial arrangement of the nuclear detectors and thermocouples is shown schematically in Figure 19.

3.5.1 Startup Channel

A fission chamber is used to monitor neutron flux and provide information to the console instruments during reactor startup and low-level reactor operation. The chamber,



FC = Fission Chamber
CIC = Compensated Ion Chamber
UIC = Uncompensated Ion Chamber
T = Thermocouple

Figure 19. Sketch of detector locations.

encapsulated in a water-tight aluminum tube assembly, is moved in and out of the core region by an electrical motor drive system. The drive system permits retracting the chamber into a boron shield assembly at high reactor power. The drive system is provided with a light indication system at the console to show "Insert Limit", "Top Half Travel", and "Withdraw Limit" positions of the chamber.

As the signal from the fission chamber approaches its limit on the recorder, the compensated ion chambers will have developed sufficient signals to control the reactor. At this point, the fission chamber can be retracted to minimize chamber burnout at high flux levels.

The pulses from the fission chamber are fed into a solid-state circuitry consisting of a preamplifier and a linear pulse amplifier. The amplified signal is fed to a counter scaler from which, if needed, a plot of reactor inverse multiplication can be determined. The pulse amplifier also feeds the log count rate meter, which is located on the front panel of the drawer, and the log count rate recorder. From the recorder two signals for the reactor interlock system are derived. The first signal prevents control rod movement until a minimum count rate (>2 counts/s) is obtained. This insures that the fission chamber is operating and that an adequate signal is available to begin a reactor startup. The second signal prevents control rod movement if the log count rate recorder is not turned on.

3.5.2 Linear Power Channel

The signal from a compensated ion chamber (CIC 1) is fed to a linear micro-microammeter which is essentially a vacuum tube electrometer designed and constructed especially for measuring small currents. The major panel controls are the range switch enabling switching ranges from 6×10^{-4} to 2×10^{6} W and the button to check the zero indication on the panel meter. An output is provided for driving a linear recorder-controller which, in conjunction with the servo-amplifier, provides automatic control of reactor power. If the reactor power, as indicated on the recorder, exceeds the set point of the controller by 20%, a signal is obtained from the recorder which actuates the reactor rundown system.

3.5.3 Log Power and Period Channels

The log power channel provides the operator with a continuous record from a power level of approximately 0.2W to 150% full power without switching interruptions. The signal from a compensated ion chamber (CIC 2) is fed into a solid-state log N amplifier. The output of the amplifier is used to drive a log power meter located on the front panel and an external log power recorder. From the log power recorder a signal for the reactor interlock system is obtained which prevents control rod movement if the recorder is not turned on. Another signal, which is obtained from this recorder if the reactor power reaches 120%

of full power, is fed into the control logic system which actuates an automatic control rod rundown.

Another output of the log power amplifier is used to feed a solid-state operational amplifier of the reactor period channel. The reactor period is defined as the time required for the neutron flux to change by a factor of e. For a visual indication, the period channel is equipped with a meter on the front panel of the drawer and with a period recorder. From this recorder three control signals are derived, two of which actuate the "Rod Withdrawal" prohibit system, and one which initiates the reactor rundown. In addition, the period channel initiates a reactor scram when the reactor period becomes shorter than 5 seconds. As the period decreases, the dc output of the period amplifier goes in a positive direction. When the level corresponding to a 5 second period is reached, the subsequent bistable circuit trips and de-energizes the scram relay, i.e. the scramming circuit of the safety amplifier (see Section 3.5.6).

3.5.4 Pool Water Temperature Channel

The pool water temperature channel consists of two (2) core inlet thermocouples placed just below the core, and one (1) outlet thermocouple placed about five (5) feet above the core. The thermocouples feed their signals to a multipoint recorder. The recorder provides a permanent record of core inlet and outlet temperatures while the reactor is operating.

Trip points are provided for the two (2) inlet temperatures and are set to trip at 57°C (135°F). This trip causes a rod withdrawal prohibit condition, preventing the rods from being withdrawn whenever the inlet temperature is 57 ° C (135 ° F) or greater (see Sec. 3.5.7).

3.5.5 Manual and Automatic Control

Three shim/safety rods and one regulating rod are used to control the reactor. Each shim/safety rod may be operated separately using an individual, spring loaded switch. The shim/safety rods may also be operated simultaneously, in a bank, by means of a joystick. There is an interlock system such that when the shim/safety rods are moved, power to the ac drive mechanism of the regulating rod is disconnected. Hence, this system makes it impossible to withdraw all four rods simultaneously. The position of each rod is continuously indicated to within $\pm$ 0.25 cm (0.1 in) at the reactor control console by an electrical transmitting system.

The shim/safety rods have console mounted "Insert Limit", "Shim Range" and "Withdraw Limit" lights which are actuated by micro-switches located on the rod drive mechanisms. Each shim/safety rod magnet contains a contact-actuated micro-switch which energizes a light on the console to inform the operator that the shim/safety rod is in contact with its magnet.

The regulating rod has "Insert Limit" and "Withdraw Limit" switches which energize console lights. In addition, signal lights are provided to indicate in which direction the regulating rod is being moved.

A servo-amplifier system is used to automatically control the reactor power at the desired set point on the linear recorder. When the reactor is at steady-state power, the servo system may be energized to automatically maintain power level. The servo system is interlocked so that the power level must be within ± 2% of the selected power level setting before the system may be engaged. Any time the power level exceeds the ± 2% variation limit, the control of the reactor reverts automatically to manual control.

3.5.6 Safety Channels

Two redundant safety channels are a part of the reactor protection system. They provide the mechanism for scramming the reactor when its power exceeds 150% of nominal power, i.e. 300 kW. Each safety channel consists of an uncompensated ion chamber and a sensing circuit within the safety amplifier. A current to operate the magnets which hold the shim/safety rods is supplied from the magnet amplifiers.

The sensing circuit contains two redundant circuits each capable of actuating a shut off of magnet amplifiers in two different ways. The first (master) circuit provides a fast scram which is obtained by applying cutoff bias directly to the grids of the magnet amplifiers. The second (slave) circuit provides a backup scram by cutting off the ac power supply to the magnet amplifiers. For all purposes, the time difference between a fast and a backup scram is negligible, being only a few milliseconds.

The high voltage power supply to the ion chambers is also contained in the safety amplifier. In the case of its failure, both scram circuits are actuated. In addition, the signal path between any ion chamber and the safety amplifier is monitored too. In the case when it is opened (e.g. disconnected), a fast scram will be initiated. The scram circuits are of a fail-safe design.

A relay connection is provided to couple the output of the period amplifier to the scram circuits. In this way, the reactor will be scrammed not only if the power level increases beyond a predetermined limit, but also if the reactor power level is increasing too rapidly. The period signal is coupled to the fast scram circuit. Lamps located on the front panel give an indication of some specific troubles developed within the system. For example, the availability of power supply to both safety and magnet amplifier is indicated by energizing respective control lights. Chamber power lights would become energized if power to either ion chamber were lost. In addition, a failure of -300 V power supply would cause the B- lamp to light. A connector is provided for an external alarm on the annunciator board which is energized if any of the panel lamp lights come on.

A selector switch and meter on the front panel monitor rod magnet currents and the + 300 and -300 Volt power supplies. A test switch is provided so that the scram function of both fast and backup scram circuits may be tested. A reset switch on the front panel resets the scram circuits after the system has been scrammed and the scram conditions are removed.

3.5.7 Alarms, Prohibits, Rundowns, and Scrams

There are a number of built-in engineered protective action levels derived from the UMRR instrumentation. According to the degree of their severity, some of them require only the attention of a reactor operator while the others initiate automatic actions. Table IX gives a summary of built-in protective actions used by the UMRR.

The lowest level of engineered protective actions consists of the audible and visual alarm annunciations only. Audible alarms are processed by a common alarm panel in the control room. In addition, a light on the alarm board indicates the alarming condition. The next level is designed in such a way as to prohibit any further rod withdrawal if any one of the conditions specified in Table IX occurs. As a result, the current to the rod drive motors is interrupted and control rods cannot be withdrawn while the abnormal condition persists.

The next two protection levels result in a reactor shutdown. Two types of reactor shutdown, rundown and scram, provide automatic protection against nuclear incidents. When a rundown is initiated all control rods are driven to their insertion limits at full speed. This rundown feature is designed to be the first line of defense against any incipiently dangerous condition such as the ones listed in Table IX. The ultimate level of engineered protection actions is the reactor scram which is reserved for only the most serious situations, such as the reactor power exceeding 150% of full power and reactor period less than 5 seconds. At this protective level the current through the shim/safety rod magnets is interrupted, causing the shim/safety rods to fall into the reactor under the effect of gravity. Other conditions which cause the reactor to

Table IX.

Protective Actions

Situation	Detector	Unit Initiating Action	Protective Action	Annunciation
Manual Scram	Operator	Scram Button	Scram	Yes
Period < 5 sec.	Compensated Ion Chamber	Log N-Period Amplifier	Scram	Yes
150% Full Power	Uncompensated Ion Chamber (2)	Safety Amplifier	Scram	Yes
Bridge Motion	Motion Switch	Motion Switch	Scram	Yes
Log N and Period Amp. Not Operative	Log N Period Amp.	Relay	Scram	Yes
120% Demand	Compensated Ion Chamber	Linear Recorder	Rundown	Yes
Period < 15 Seconds	Compensated Ion Chamber	Period Recorder	Rundown	Yes
Reg. Rod on Insert Limit in Auto-Control	Micro-Switch	Micro-Switch	Rundown	Yes
Low CIC Voltage	D.C. Relay	D.C. Relay	Rundown	Yes
120% Full Power	Compensated Ion Chamber	Log N Recorder	Rundown	Yes
*High Radiation	GM Tube (3)	RAM System	Rundown	Yes
*Period < 30 Seconds	Compensated Ion Chamber	Period Recorder	Rod Withdrawal Prohibit	Yes
Recorder Off	Relay	Relay	Rod Withdrawal Prohibit	Yes
*Log Count Rate < 2 CPS	Fission Chamber	Log Count Rate System	Rod Withdrawal Prohibit	Yes
Core Inlet Water Temp. 135°F	Thermocouple	Relay	Rod Withdrawal Prohibit	Yes
*Safety Rods Below Shim Range	Micro-Switch	Relay	Reg Rod Withdrawal Prohibit	No
*Safety Rods Below Shim Range and Reg Rod above Insert Limit	Micro-Switch	Relay	Safety Rod Withdrawal Prohibit	No

* Indicates that the situation may be key bypassed.

Table IX. (Cont.)

Protective Actions

Situation	Detector	Unit Initiating Action	Protective Action	Annunciation
Basement Sump level High	Micro-Switch	Micro-Switch	Operator Response	Yes
Interlock Bypassed	Key Switch	Key Switch	"	Yes
Effluent Pool Demineralizer Conductivity High	Conductivity Bridge	Relay	"	Yes
Beam Port or Thermal Column "Open"	Micro-Switch	Micro-Switch	"	Yes
High Neutron Flux in Beam Room	Neutron Detector	Relay	"	Yes
Manual Operation	Relay	Micro-Switch	"	Yes

be scrammed are summarized in Table IX.

3.5.8 Scram Logic

The scram logic circuitry contained in the power safety amplifier was discussed in Section 3.5.6. In this section the logic and operation of the circuit processing bridge motion, log power amplifier inoperative, and manual scram signals will be described.

The scram circuit consists of a set of open-on-failure relay contacts wired in series with a scram relay. Therefore, any scram signal or component failure will result in de-energizing the scram relay. This in turn opens the circuit of regulated power to the magnet power supply causing the current in the safety magnets to cease and to release the shim/safety rods. The scram relay can only be reset after the condition causing a scram has been removed and the reset relay energized by manually pushing the reset button.

The bridge motion scram is controlled by a micro-switch on the reactor bridge. As long as this switch is closed, a relay in the circuit is energized. A slight change in the position of the bridge, approximately 0.25 cm (0.1 in), will open the contact, de-energizing the motion relay which opens its contacts in the scram circuit.

If the log power amplifier were switched out of the operate position a monitor relay would become de-energized and its contacts would break the scram circuit de-energizing the scram relay.

When the manual scram button is pressed two contacts are mechanically opened: one of them causes the scram relay to de-energize and another one interrupts regulated power to the magnet power supply. Hence, the ac power circuit to the magnet power amplifier is opened in two different and independent ways.

In addition, the scram circuit also contains contacts of the relay which monitors the unregulated ac power. In the case when electrical power is lost the scram circuit opens and initiates a reactor scram.

3.6 Radiation Protection

Stationary radiation protection for the staff and the general public is accomplished by the biological shielding placed around the reactor. Various monitoring systems are used for an active radiation protection system to activate visual and audible alarm signals if an increased radiation level occurs in the Reactor Building.

3.6.1 Biological Shielding

The reactor core is shielded in all directions by water in the pool. The water level is maintained such that there is normally a water layer about 6 1/2 m (19.5 ft) thick between the top of the core and the water surface. The next shielding barrier is provided by concrete pool walls which are about 0.5 m (1.5 ft) thick, except for the east side of the pool where the wall thickness ranges from a maximum of about 2 m (6 ft) in the vicinity of the thermal column and the beam port to a minimum of 0.5 m (1.5 ft) at the top of the pool. Additional shielding by earth is provided on the other three sides since the lower part of the reactor pool is below ground level.

When the reactor is operated, N-16 is produced in the pool water by the reaction O-16 (n,p) N-16. Two water pumps are used to direct surface water downward to the top of the reactor core to delay the rise of radioactive nitrogen. This delay is sufficient enough to significantly reduce the radiation level due to N-16 at the pool water surface. The surveys at full power have shown that the maximum radiation level is about 3 mr/hr at one foot from the pool water surface.

3.6.2 Radiation Monitoring Systems

3.6.2.1 Radiation Area Monitors

The Radiation Area Monitoring (RAM) system consists of three Geiger-Mueller (GM) detectors for monitoring gamma radiation and one BF_3 detector to monitor neutron levels. Also included in the RAM system are associated alarms, indicators, and one automatic protective function.

The function of the RAM system is to monitor radiation levels in specific areas throughout the building. These areas are 1) on the reactor bridge to monitor radiation levels above the reactor pool, 2) near the demineralizer to check for a buildup of radioactive ions, and 3) at the area near the thermal column and beam port, where both a GM and a BF_3 detector are mounted to ensure that no increased radiation level is present before entry is made into the basement level.

In the vicinity of each GM detector there is a local alarm and meter. Remote indicators and alarms are located in the control room where the operator can also check the function of each GM monitor. This is done by actuating a solenoid that positions a small internal check source next to the detector.

This checks for proper operation and allows for checking setpoints.

When the radiation in any of the monitored areas exceeds the respective setpoint, a trip is actuated in the RAM control unit causing a reactor rundown, alarm annunciation, and providing local and control room indications of the radiation level associated with the RAM system. In addition, the radiation area monitor on the reactor bridge has a second, higher, setpoint which actuates the building evacuation horn.

The RAM system can be by-passed by using a special key to which only a Senior Reactor Operator has an access. Located on the control console, the by-pass switch overrides the high radiation area rundown, but does not impair the radiation alarm or the building evacuation alarm.

3.6.2.2 Continuous Air Monitor

The continuous air monitoring system consists of a Continuous Air Monitor (CAM), a recorder, and associated alarm and warning circuitry.

The function of the CAM is to measure the radioactivity of air-borne particulates by concentrating these solids on a filter paper. Air is drawn through a special filter paper at a controlled rate. The buildup of activity on the paper is detected by a Geiger-Mueller tube which gives a reading in counts per minute. The CAM is equipped with an alarm system to give audio and visual warning if the reading exceeds the alarm setpoint.

The CAM is located in the reactor bay area, where it takes in air, and sends a remote signal to the recorder in the auxiliary panel in the control room.

4. EXPERIMENTAL FACILITIES

The UMRR is used as a neutron source for research and irradiation purposes. In addition to irradiation capabilities in the pool, there are a number of facilities which can be used for sample irradiation and experiments. They are described in the following sections.

4.1 Thermal Column

The thermal column provides a readily accessible field of thermal neutrons for experimental purposes. The thermal column consists of a 1.1 m x 1.1 m x 1.5 m (3.5 ft x 3.5 ft x 5 ft) graphite pile extending from the reactor core to within about 1 m (3.3 ft) of the outer face of the beam room pool wall. The irradiation positions consist of one 51.6 square cm (8 square in) and four 2.6 square cm (4 square in) horizontal access ports, all of which are filled with graphite plugs when not in use. The thermal column door is constructed of steel and concrete and is 1.2 m x 1.2 m x 1.5 m (4 ft x 4 ft x 5 ft).

The reactor end of the thermal column is covered with a 10.2 cm (4 in) lead shield to reduce the gamma flux in the thermal column to a minimum. The front surface of the thermal column door, which is filled with heavy concrete, is lined with boral. Total shielding from the reactor core through the thermal column to the outer biological shield wall face is equivalent to that which would be provided by the intervening water and biological shield pool wall. A schematic drawing of the thermal column is shown in Figure 20.

4.2 Beam Tube

A stepped beam tube (see Figure 21) is provided primarily to obtain a beam of reactor neutrons which can be used for experimental purposes and to provide a wet or dry irradiation facility. The open end of the beam tube terminates at the beam room side of the pool wall and the operations required to remove or install equipment from the beam hole are performed from the beam room. The beam tube is constructed of aluminum and is closed at the reactor end.

A shutter assembly composed of two parts can be used to achieve a collimated beam of neutrons. It consists of a plug part containing a beam guide having a cross-section of 7.6 x 5.1 cm^2 (3x2 in^2) and a shutter part which provides an extension to the beam guide in the "Open" position and a radiation shield in the "Closed" position. Both positions are remotely controlled and their indication is displayed in the reactor control room.

The entire tube is lined with stainless steel. There is an additional lining of boral (aluminum-boron carbide-aluminum sandwich) to materially reduce activation of the stainless steel and concrete. Additionally, if the beam tube is not in use it can be filled with shielding plugs. The outer and inner concrete shielding plugs are contained in stainless steel. The

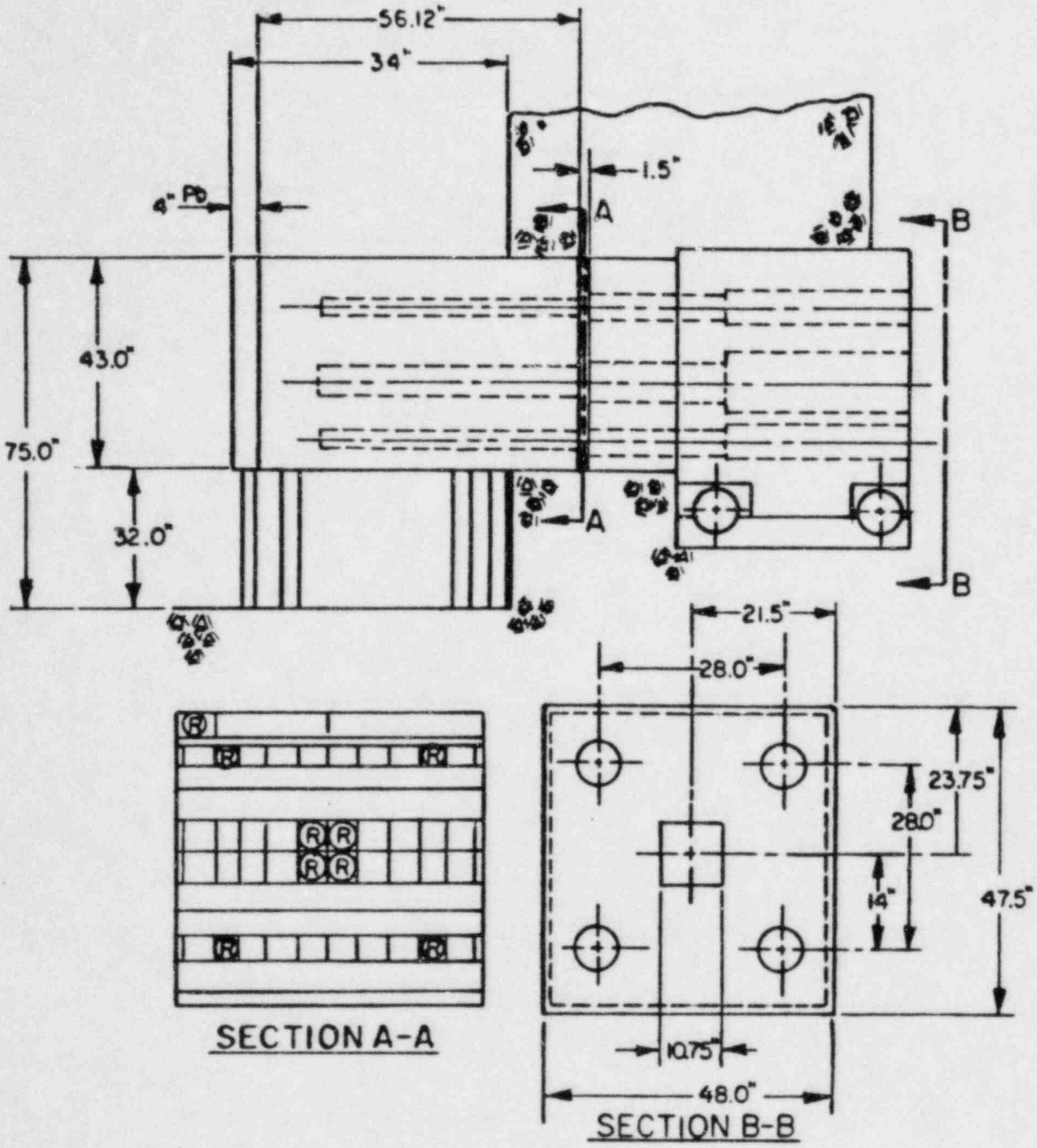


Figure 20. Thermal column.

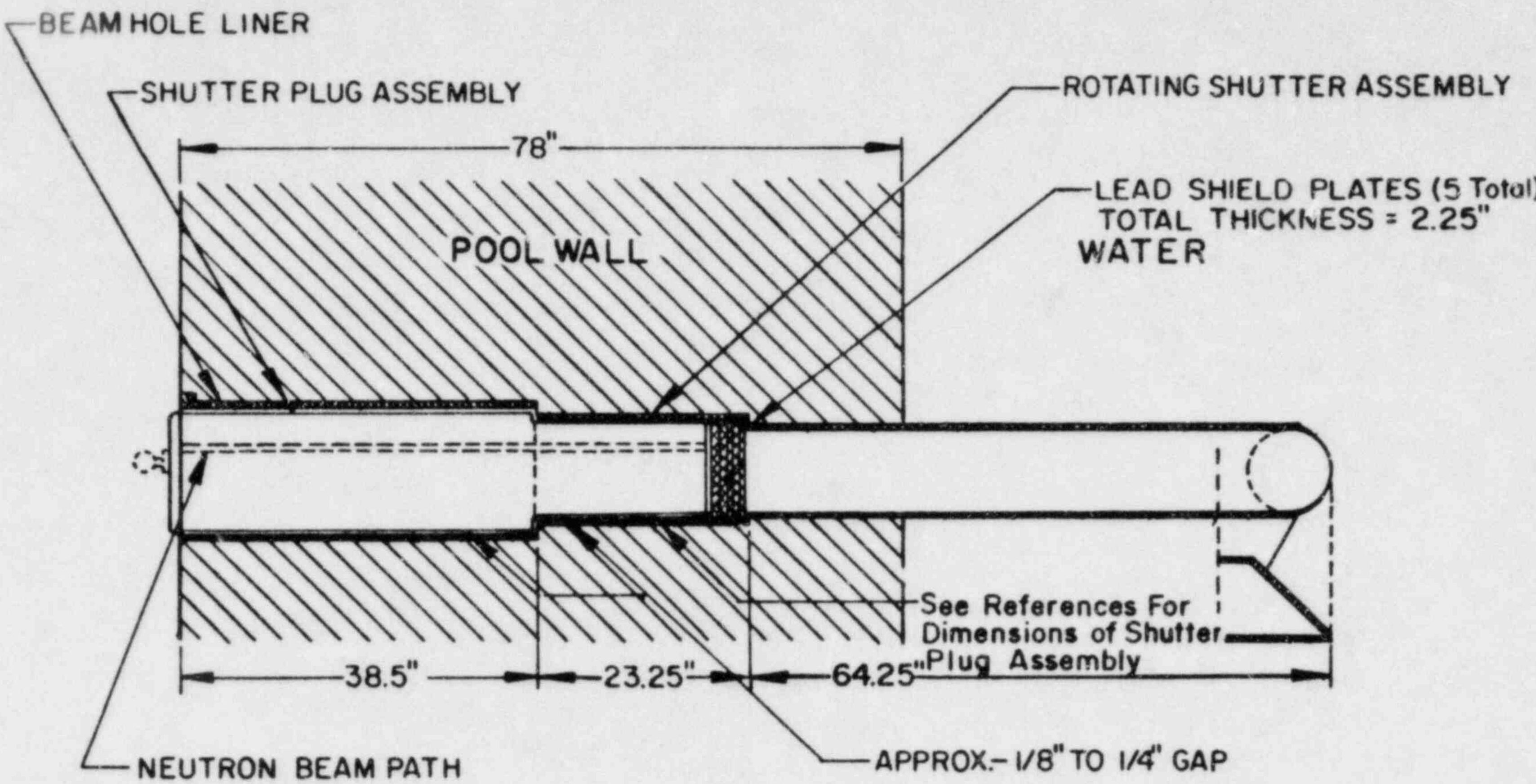


Figure 21. Beam tube with shutter assembly in place.

end of the plug nearest the reactor is covered with a boral sheet to reduce activation of the plug materials. The opposite end contains a lead plug for the attenuation of gamma rays.

4.3 Pneumatic Sample Transfer System

The rabbit tube system is used to rapidly transfer samples to and from the reactor core. It consists of two rabbit tubes, one of which is cadmium lined to prevent sample activation by thermal neutrons. The rabbit tube fits into the grid plate in a manner similar to a fuel element. The transfer tubing terminates in a glove box next to the reactor pool where samples are inserted or removed.

Each rabbit system consists of two stainless steel tubes with one tube being the sample tube and the other providing the pressure differential. The control system is semi-automatic, giving information and control relating to sample position, irradiation time and transport time.

Nitrogen gas is used as the transport medium in order to reduce the amount of gaseous Ar-41 activity produced. The rabbit system is vented through a high efficiency filter which exits just below a ventilation exhaust, thus reducing particulate activity.

The rabbit tubes are used in a core configuration that have at least one side open to a moderating medium.

4.4 Sample Rotor Assembly

The sample rotor is a device that rotates samples next to the core, thus enabling more uniform irradiation of samples which are irradiated simultaneously. Eight samples can be placed in the sample rotor at one time.

The sample rotor assembly is placed in the grid plate in a manner similar to a fuel element. It is positioned in an external core position and is rotated by a motor and gear arrangement that is mounted on the reactor bridge.

5. AUXILIARY FACILITIES

The auxiliary facilities described in this chapter are needed to support operations of the UMRR. Although they are not a part of the reactor, their function is such that it warrants their inclusion in this document.

5.1. Fuel Handling and Storage

Fuel handling at the UMRR is performed by manual handling tools. They are used to grasp, move, and position fuel elements either into the core grid plate or a storage rack.

Two storage racks are available in the fuel storage pit and each is capable of holding up to 15 fuel elements. The fuel elements are oriented in the storage racks in the same manner as in the core, i.e. standing. The geometry of the stored fuel is such that a criticality in the storage pit cannot be achieved.

5.2 Water Supply and Purification System

Supply water for the reactor pool is obtained from the University of Missouri-Rolla water system. An analysis of the impurities in the well dated September 12, 1958 is shown in Table X.

In order to reduce fuel element corrosion and prevent the

Table X.
Impurities in Supply Water

Substance	Parts/Million
Calcium	51.5
Magnesium	30.5
Sodium	10
Silicon Dioxide	21.7
Free Carbon Dioxide	16.7
Sulphate, SO_4	28.4
Chlorine Cl	3.6
Bicarbonate, HCO_3	1
Methyl Orange Alkalinity	252

build-up of impurities in the reactor pool water with consequential neutron-induced activity, the pool supply water is deionized prior to being supplied to the pool. This is accomplished by passing the water through a mixed resin bed ion exchanger (deionizer) to remove anions and cations. A limit of 57 C (135 F) is imposed on the pool water temperature. This limit is based upon the effective temperature range for the ion exchange resins. The effluent water contains fewer minerals than ordinary distilled water. A filter is installed at the ion exchanger input to remove undissolved solid particles. A schematic diagram of the water purification system is shown in Figure 22. The HCL and NaOH tanks are filled and used only during the resin regeneration. During this time the reactor pool is isolated from the water purification system. Neither the make-up water nor the pool water is directly connected to the raw water supply system. This prevents the possibility of any contamination of the raw water line.

The deionized water is circulated through the pool at a flow rate of 115 l/min (30 gal/min). The purity of the pool water is maintained at a specific resistance greater than 500 kOhm-cm. The corrosion of aluminum in the high purity water does not seem to be a problem. For example, in reference (17) it is reported that in most open literature publications the corrosion of aluminum in distilled water at temperatures less than 100° C is described as "ceasing" after an initial period of moderate reaction rate.

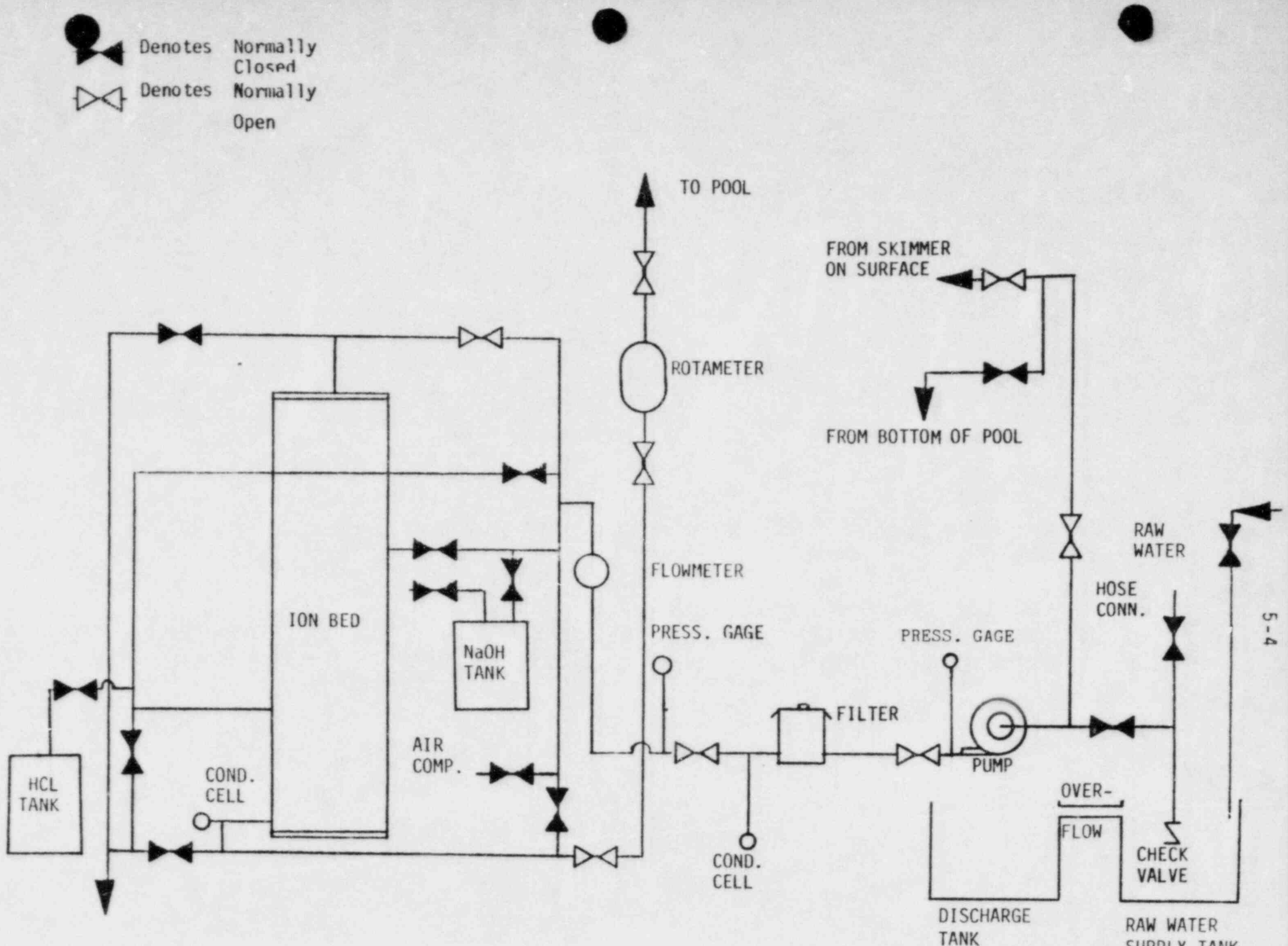


Figure 22. UMR Reactor pool water system.

5.3 Liquid Waste Holdup System

The liquid waste holdup system consists of two 1140 l (300 gallons) tanks, and associated pipes and valves. The function of the liquid waste holdup system is to facilitate the holding of ion exchanger regeneration liquids for sampling, decay and subsequent disposal after activity levels are below Maximum Permissible Concentration (MPC) limits specified in 10CFR20.

The tanks are large enough to hold the liquids from one complete regeneration, i.e. backwash, caustic flush and acid flush). Both tanks are vented through a high efficiency filter and drain into the lower level sump where the liquids are pumped to the middle level sump and out into the sanitary sewer. A schematic drawing of the liquid waste holdup system is shown in Figure 23.

5.4 Building Ventilation

Building ventilation is accomplished by a system of three fans which are mounted on the Reactor Building roof. Their combined flow rate is about 1000 m^3 /min (3.5×10^4 cubic feet/min). They exhaust air from the reactor bay and lower level. With all fans turned on, the air turnover rate in the reactor bay is about 1.4 min. Air enters the building through two intake systems which are equipped with fiberglass filters located on the lower level.

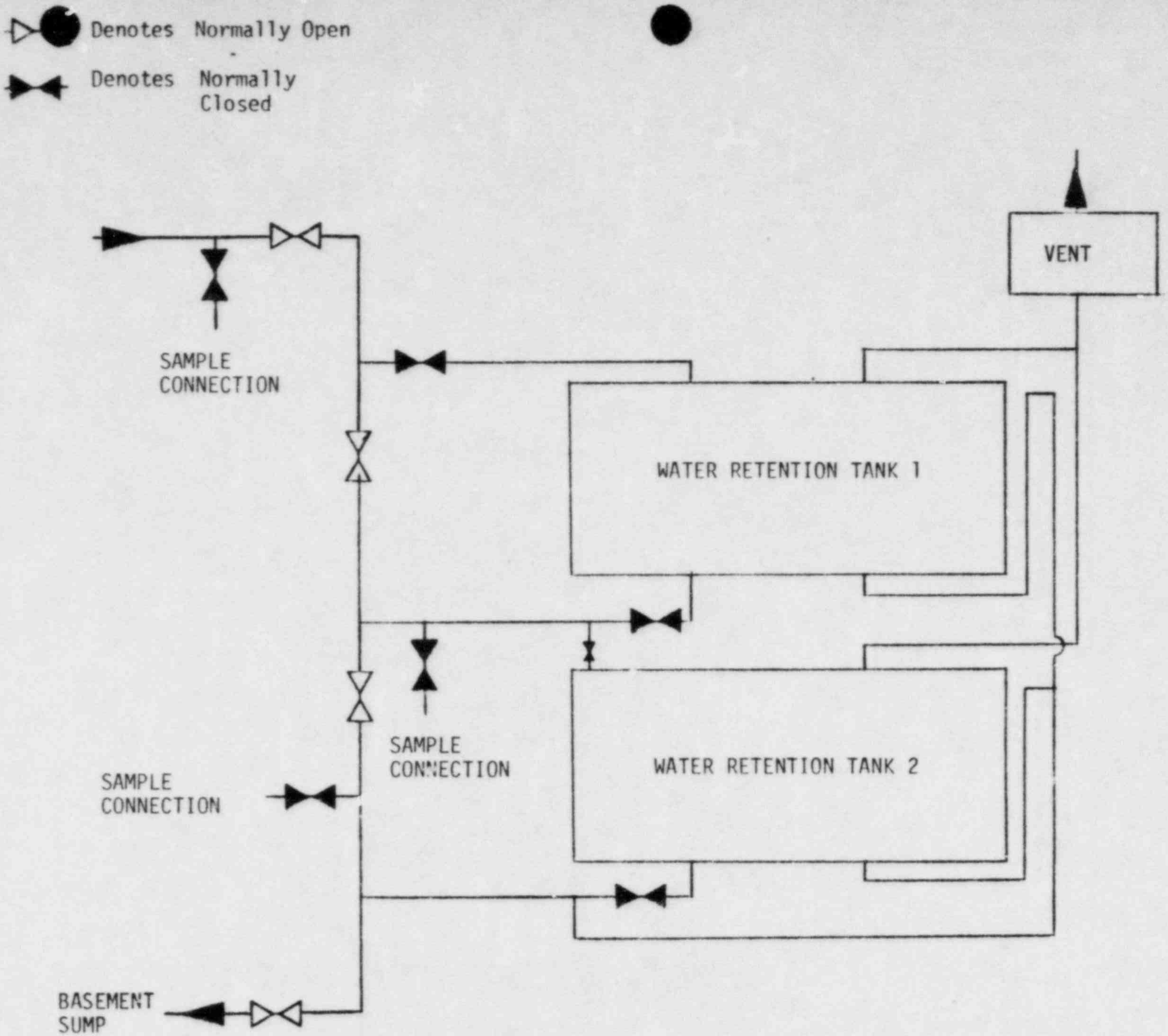


Figure 23. Liquid waste holdup system.

The exhaust ducts and intakes are equipped with louvers which close automatically when the fans are turned off. There are no filters in the exhaust ducts because during normal operation the UMRR does not produce any airborne particulate radioactivity. In the case of any abnormal situation arising in the Reactor Building, a pertinent emergency procedure will be followed. This procedure, included in the UMRR Standard Operating Procedures, specifies that all exhaust fans will be immediately turned off.

5.5 Fire Protection System

The function of the fire protection system is to give warning in the event of a fire or smoke development within the reactor building. If a smoke or fire situation arises, audible and visual alarms are actuated inside and outside the Reactor Building and a remote alarm is received at the campus police station.

The fire protection system consists of four heat sensors, two smoke detectors, two hand pull stations and an alarm and relay box. Two smoke detectors are located on the ceiling of the Reactor Building. Heat sensors are located at high points of the demineralizer level, the counting room, in the upstairs office space and in the electronics space behind the control room. The hand pull stations are located by the security door and by the emergency exit at the demineralizer level. There are two flashing lights, one of which is located on the south wall of the lower level and the other on the west wall in the bay area.

In the event that power is lost to the Reactor Building, there is a backup battery which will give an audible fire or smoke alarm to personnel in the Reactor Building and at the campus police station. There are also eight fire extinguishers located throughout the Reactor Building at strategically important locations.

6. RADIOACTIVE WASTE MANAGEMENT

Radioactive waste resulting from reactor operations is either discharged to the environment in gaseous form, released as liquid to the Rolla sanitary sewer system, or packaged as solids and transferred to Radiation Safety to be held for decay and then disposal, all in accordance with applicable regulations.

The University of Missouri - Rolla Radiation Safety Office has always operated with the As Low As Reasonably Achievable (ALARA) principle as a guideline, even before ALARA became a national standard. (See also section 7.1)

6.1 Solid Waste

Low-level solid waste generated as a result of reactor operations consists of ion exchange resins, filters, potentially contaminated paper and gloves, and occasional small, activated samples from laboratory experiments. All waste is packaged in accordance with applicable NRC and U.S. Department of Transportation regulations and transferred to the Radiation Safety Hazardous Waste Building to be held for decay and future disposal in accordance with applicable regulations. Major contributions to solid waste are listed in Table XI.

High-level solid radioactive material generated by routine reactor operations consists of spent fuel elements. Spent elements are stored in the reactor pool until the accumulation justifies shipment for delivery to the Department of Energy. To date, no spent fuel elements have ever been shipped from the UMRR.

Table XI.
Solid Waste

	Quantity/Time	Isotopes after 1 year decay	Activity after 1 year decay
Resin	12 cubic feet per 5 years	Gross Activity	2 microcuries per 12 cu.ft.
Pool water Filters	6 per change once a month	Gross Activity	0.001 microcuries per filter

Using a Ge(Li) detector and multichannel analyzer no peaks were identified. The above gross activity is calculated by subtracting the background from the total sample counts. An NBS traceable source is used to calibrate the detector prior to the analysis.

6.2 Liquid Waste

Several activities conducted within the reactor facility are capable of generating radioactive liquid waste. The largest volume of potentially contaminated water from the reactor is produced by the regeneration of the demineralizer and the lowering of the pool level for maintenance.

All potentially radioactive liquid waste is analyzed and checked for compliance with limits specified in 10CFR20 Appendix B, Table I , Column 2 prior to release to the Rolla sanitary sewer. Two 300 gallon fiberglass holding tanks located at the basement level of the reactor building are used to collect the liquids produced during resin regeneration. The resin regeneration consists of 3 steps: backwash, regeneration of anion resins, and regeneration of cation resins. The amount of regeneration liquid is about 325 gallons and the dilution amount is approximately 140 gallons of water for a total of 465 gallons. During each step samples are taken for isotopic analysis. Sampling is performed at the beginning of the step. In the middle, and near the end of the regeneration step. Hence, a total of three samples are taken during each step. The samples are mixed together to obtain a representative sample for analysis of each respective step. The analysis is used to determine the hold-up time (I′ required) before release to the Rolla sanitary sewer. Typically, only traces of sodium 24 or gross activ' y have been seen in the analysis. The isotopic analysis is performed on either a Ge(Li), a Na(I), or similar detector connected to a multichannel analyzer. A National Bureau of Standards, traceable standard is used to calibrate the detector and determine the efficiency.

6.3 Gaseous Waste

Possible airborne waste includes N-16, Ar-41, and neutron activated dust particulates. No fission products escape from the fuel cladding during normal operations as demonstrated by the monthly pool water analysis. The bulk of the radioactive airborne waste is due to Ar-41 which is produced mainly by the neutron irradiation of the air dissolved in the reactor pool water. Exposure from N-16 is reduced using the pool water diffusers which are described in Section 3.6.1 . The effectiveness of this system has been shown in radiation surveys performed at the pool surface.

Occupational exposure to personnel from airborne radioactivity is reduced by operating exhaust fans to sweep the air from the reactor bay and experimental area. In Sec 7.6.1 it is demonstrated that airborne radioactivity released to unrestricted areas does not exceed 10CFR20, Appendix B, Table II, column 1 limits.

7. RADIATION SAFETY PROGRAM

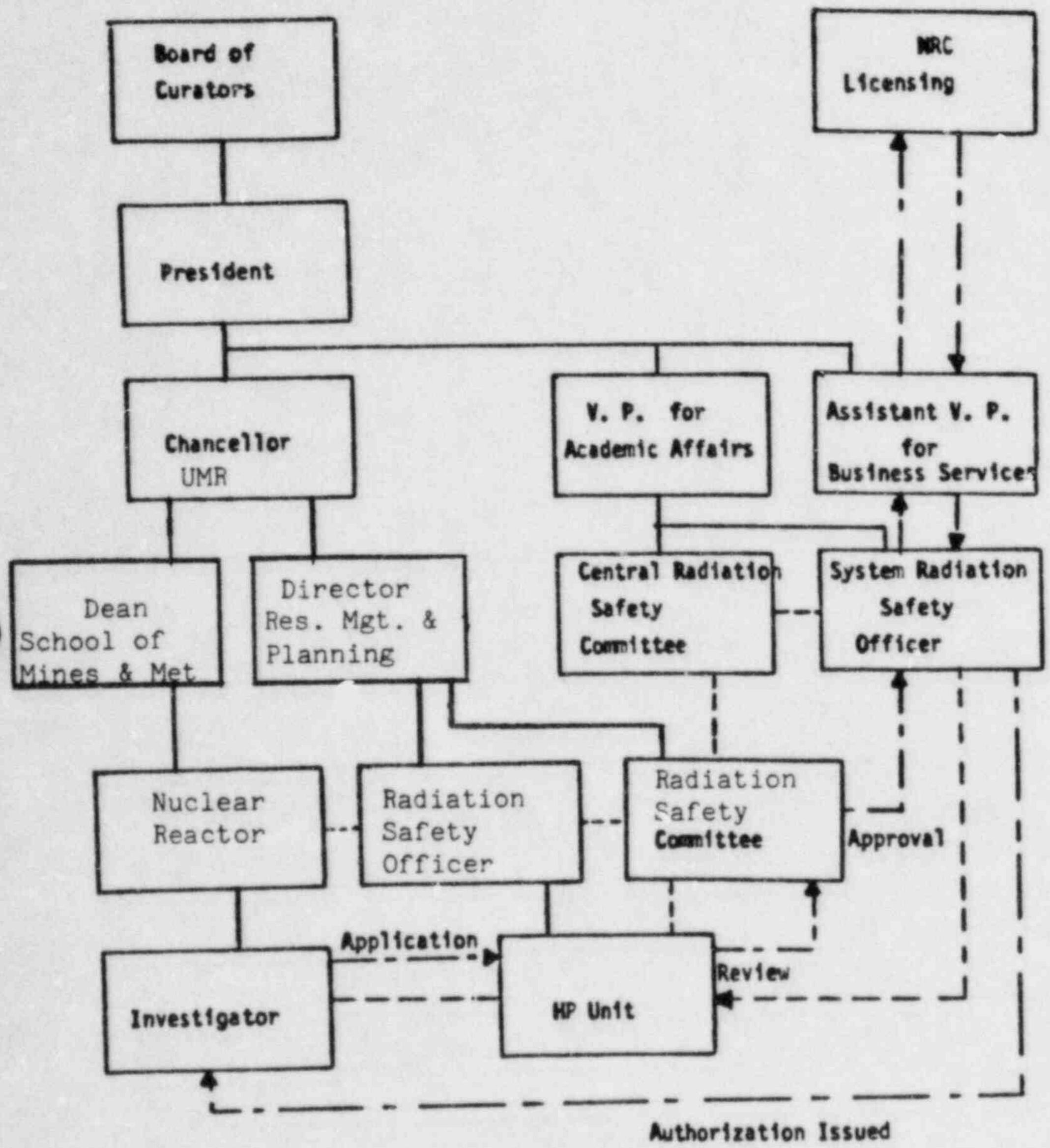
The Radiation Safety Office at UMR is a section of Administrative Services (see Fig. 24). Radiation safety at UMR is carried out by a part-time Radiation Safety Officer, a full time Health Physicist and a part-time technician. The reactor is provided with health physics coverage from the Radiation Safety Office. The Health Physicist or his designee monitors liquid effluents prior to release to comply with applicable regulations. Periodic grab samples are used to monitor for Ar-41 in the containment air. The Radiation Safety Office uses film badges as area monitors located in the Reactor Building to verify that radiation exposures in restricted areas and in the lobby within the facility are well within regulations specified in 10CFR20. Table XIV shows the results of area film badge monitors. Section 7.4.1 discusses radiation measurements taken at full power.

7.1 ALARA Commitment

The University of Missouri - Rolla Radiation Safety Office has always operated with the As Low As Reasonably Achievable (ALARA) principle as a guideline, even before ALARA became a national standard. A copy of the official policy is shown in Appendix B.

The following steps are used to implement the ALARA Principle.

1. <u>Film Badges</u>- Furnished to all students, faculty and staff as specified. Area radiation badges are also placed in the Reactor Building to monitor radiation levels.



——————— Indicates administrative responsibility including resource allocation.

— — — — — — Indicates technical support, coordination, surveillance, or audit as necessary.

—— - —— - — Indicates steps in the processing of an application to obtain an authorization to use a radiation source.

Figure 24.
Management structure for Radiation Safety Program at the University of Missouri.

2. Indoctrinations - Students, faculty, and staff must listen to the Indoctrination tape at the Reactor and receive a tour prior to working there. In addition, anyone who received a film badge must attend a Health Physics Indoctrination lecture.

3. Pocket Dosimeters - Everyone who enters the Reactor Building beyond the lobby must have either a self reading, pocket dosimeter or a film badge. In the case of tours usually 3 dosimeters are issued per group.

4. Levels of Action and Response - See section 7.2.2

5. Monthly Reactor (Health Physics) Audit - An audit of the following Health Physics activities is performed monthly to ensure compliance with applicable regulations and ALARA.

a) Sealed source leak tests,
b) Radiation area monitor calibration,
c) Health Physics instrument calibration,
d) Monthly contamination surveys,
e) Monthly air releases,
f) Waste water analysis,
g) Monthly area radiation surveys,
h) Monthly pool water analysis,
i) By-Product material releases,
j) Semi Annual Pool Water H-3 analysis.

6. Reactor Health Physics SOP's Have been written and implemented

for the activities specified in item 5 above.

7. <u>Campus Radiation Safety Committee</u> - New project requests are reviewed by the committee to ensure safety and consistency with the ALARA principle.

7.2 Health Physics Program

7.2.1 Health Physics Staff Qualifications

The current UMR Health Physics staff consists of two professionals plus one part-time technician. The qualifications for the Health Physicist are a bachelor's degree in health physics, or an equivalent combination of education and experience from which comparable knowledge and abilities can be acquired. The qualifications for the Health Physics Technician are an associate's degree in health physics, mathematics, physics, or an equivalent combination of education and experience from which comparable knowledge and abilities can be acquired.

The Radiation Safety Officer is a faculty member with an appropriate education and experience in the health physics area (e.g. HP degree, NE degree, etc.)

7.2.2 Health Physics Procedures and Responsibilities

Listed below are the responsibilities of the Radiation Safety Office for the Health Physics activities at the Reactor Facility:

a) Semi-annual sealed source leak tests,

b) Semi-annual radiation area monitor calibration,

c) Semi-annual Health Physics instrument calibration low--scale: yearly--high scale,

d) Monthly contamination surveys,

e) Monthly air release calculations,

f) Waste water analysis (as needed),

g) Monthly area radiation surveys,

h) Monthly pool water analysis,

i) By-Product material releases as required,

j) Semi-annual pool water H-3 analysis.

Health physics procedures have been prepared and placed in the reactor SOP Manual, that address the above-listed activities. However, the reactor staff participates in the calibration of radiation area monitors and portable health physics instruments. They also collect liquid samples, and monitor activated samples to ensure that they do not leave the reactor pool unless a sample is <100 millirem per hour on contact. Samples >100 millirems must be monitored by Health Physicist.

Administrative limits and action points are listed below:

a) External Exposures

Are monitored by film badges and thermoluminescent dosimeters (TLD)

Limit: 10CFR20 limits are used as guidelines but Health Physics will contact personnel who receive in excess of 50 millirems per month. Exposure summaries are provided to personnel annually.

b) Internal Exposures

Are monitored only if the quantity of material handled exceeds the amounts specified in section 2.2.4 of <u>the University of Missouri Handbook of Radiological Operations</u>.

<u>Tritium</u>

Limit: The maximum continuous body burden is 28 microcuries per liter.

Action: Detected by urinalysis. Health Physics will investigate any activities found over 0.28 microcuries per liter.

<u>Radioiodine</u>

Limit: The maximum continous body burden for Iodine-125 is 0.58 microcuries and 0.15 microcuries for Iodine-131.

Action: Detected by thyroid count. Health Physics will investigate any activities greater than 0.01 microcuries

Bioassays for other radioisotopes would be performed as needed.

c) Radiation Surveys

Data obtained with a G-M survey meter are reported in millirem per hour (mrem/hr). Exposure levels greater than (0.1 mrem/hr) are generally reported as to location. Based on a 40 hour work week and a 50 week year (0.1 mrem/hr) would equal 200 mrem/year.

d) Radiation Contamination Surveys

Data obtained from the swipe contamination surveys are reported in picocuries per 100 square centimeters (pCi/100cm^2). Activities below 100 pCi/100 cm^2 are reported as "no contamination evident".

e) Radiation Spills

In case of a spill of radioactive material, Health Physics must be contacted immediately to supervise the decontamination if more than microcurie activities are involved or if the contamination extends beyond the work area.

7.2.3 Instrumentation

The UMRR has a variety of detecting and measuring instruments available for monitoring potentially hazardous radiation. This includes the following instruments:

(1) Multichannel pulse height analyzer,

(2) Low background alpha-beta gas-proportional counter,

(3) Scintillation counters,

(4) Thin window G-M counters,

(5) TLD reader,

(6) Pocket dosimeter charger and reader,

(7) Low-through-high range portable beta-gamma dose rate meters capable of measuring from 0.1 mr/hr to 2.5 r/hr,

(8) portable neutron dose rate meter capable of measuring from 0.1 mr/hr to 700 mr/hr,

(9) GM "friskers",

(10) Fixed GM radiation area monitors,

(11) High-velocity portable air sampler.

The portable hand held beta-gamma instruments are calibrated to an NBS traceable source according to ANSI N323-1978. The portable hand held neutron instrument is calibrated with a PuBe source using calibration values traceable through Victoreen Corporation. The fixed GM radiation area monitors are calibrated according to SOP using an NBS traceable source.

7.2.4 Training

Health Physics training of the licensed operators is part of their requalification program. Lectures and indoctrinations are provided by the campus Health Physicist for the reactor non-licensed personnel. The minimum requirements used for training are 10CFR 19.12, the reactor indoctrination film and Regulatory Guides 8.29 and 8.13.

7.3 Radiation Sources

Sources of radiation directly related to reactor operations include radiation from the reactor core, ion exchange column, and radioactive gases.
An isotopic analysis of the Reactor Building air has shown that the primary contribution to gaseous radioactivity originates from Ar-41 (see also Sec.7.6.1).

Sources of radiation that may be considered as incidental to the normal reactor operation, but are associated with reactor use include activated foils, activated components of experiments, and activated samples or specimens. To minimize personnel exposure no activated material is removed from the pool unless its activity is less than 100 mrem/hr on contact or unless the Health Physicist is present to monitor material which reads greater than 100 mrem/hr.

7.4 Routine Monitoring

7.4.1 Radiation Surveys

Area radiation surveys are performed monthly, using portable, handheld, beta-gamma and neutron instruments according to written procedures. Survey results taken in July 1984 after the reactor had been operated at 200 kW for over four hours showed only three areas inside of the reactor building to be greater than 1 millirem per hour. One of the areas was directly over the core area of the pool and the other area was next to the demineralizer which read 48 millirem/hr on contact and 1 millirem/hr in the general area around the demineralizer. All other areas inside of the building showed less than 0.8

millirem/hr.

7.4.2 Pool Water Analysis

Once a month a one-liter sample of pool water is drawn and analyzed. The analysis is performed on either a sodium iodide or germanium detector connected to a multichannel analyzer. The purpose of the analysis is to look for fission products such as cesium 137 and cobalt 60 in the pool water. The analysis procedure involves drawing a 1 liter pool water sample and counting the sample and then counting an N.B.S. traceable 1 liter Co-60 standard to obtain the efficiency of the detector and thus activity of the pool water sample. The action level which would be used used to identify a leaking fuel element is if any of the fission products Co-60 or Cs-137 were identified in the pool water sample. The action to be taken would be as follows:

1. Identify the leaking element.

2. Remove the element from the core and store it in the fuel storage pool.

3. Using the sipping method, periodically sample the water in the vicinity of the leaking element. If the activity found is larger than the activity allowed in 10CFR20 Appendix B Table I, column 2 restricted area limits, then the element would be shipped.

7.4.3 Swipe Tests

Once a month Health Physics performs random swipe surveys in the Reactor

Building to check for possible contamination. Watman number 1 or equivalent filters are used to smear an area of approximately 100 cm^2. The filters are counted for alpha and beta contamination on either a gas-proportional counter connected to a single channel analyzer or an alpha meter and an end window Geiger-Mueller counter. (See Section 7.2.2 for limits and action points.) In the past no major contamination has been found at the reactor facility.

7.5 Occupational Dosimetry

7.5.1 Personnel Monitoring Program

The reactor personnel and radiation worker monitoring program is based upon 10CFR20.101 specified limits and ALARA. To summarize the program personnel exposures are measured by the use of film badges assigned to individuals who might be exposed to radiation. In addition, TLD's and self-reading pocket ion chambers are used, and instrument dose rate and time measurements are used to ensure that administrative exposure limits are not exceeded. Visitors and tour groups are monitored by pocket dosimeters and are limited to 10CFR20.104 limits to allow for minors.

7.5.2 Personnel Exposures

The UMRR reactor personnel annual exposure history for the last five years is given in Table XII.

7.6 Effluent Monitoring

7.6.1 Airborne Releases

Table XII.

Reactor Occupational Exposure Summary

	NUMBERS				
WHOLE BODY EXPOSURE(REMS)	1979	1980	1981	1982	1983
No measurable exposure	44	44	38	47	31
Less than 0.1	0	13	1	1	8
0.1 to 0.25	0	3	2	0	0
0.25 to 0.5	0	0	0	0	0
0.5 to 0.75	0	0	0	0	0
0.75 to 1.0	0	0	0	0	0

Note: This summary is reported as per 10 CFR 20.407(b).

Experience has shown that the average annual thermal output of the UMRR is about 10 megawatt-hours which is equivalent to 50 hours of operation at the full power of 200 kW. In reality, most of the time the UMRR is operated at low power levels, for example at 20 watts, in which case the production of airborne radioactivity is negligible.

A grab-sample system has been used with the reactor operating at full power to analyze Ar-41 in the reactor bay one foot over the fuel storage end of the pool. Concentration levels of Ar-41 were measured in consecutive time intervals of approximately 1.5 hours duration. During this experiment a ventilation fan with a flow capacity of 140 m^3/min (5 x 10^3 ft^3/min) was used.

The half-life of Ar-41 is 1.8 hours. Therefore, when argon-41 is produced it reaches its natural equilibrium after about 8 hours. (At that time 95% of Ar-41 is produced.) Measurement data show that at this time the concentration level of Ar-41 in the reactor bay is approximately 4.5 x 10^{-7} microcuries/ml. This value is well below the limit of 2 x 10^{-6} microcuries/ml, which is the limit established in 10CFR20 for the concentration of Ar-41 in restricted areas.

Since the exhaust fans are mounted at the building roof, the air containing Ar-41 is discharged from the Reactor Building at the roof level. The outside concentration, C_o of Ar-41 downwind from the point of discharge is given by

$$C_o = D \times \dot{v} \times C_B$$

where D = dilution factor (s/m^3)

$\dot{v}$ = fan flow rate (m^3/s)

C_B = Ar-41 concentration in the Reactor Building (microcurie/ml)

The dilution factor due to the wake of the Reactor Building is calculated using the relationship given by Lamarsh (15)

$$D = \frac{1}{c \times \bar{u} \times A}$$

where c = an empirical constant (0.5)

$\bar{u}$ = average wind velocity (m/s)

A = cross-sectional area of the building

The cross-sectional area of the Reactor Building is about 100 m^2. Using $u = 1$ m/s (see also Sec. 9.7) the building dilution factor is calculated to be 2×10^{-2} s/m^3. From the above relationship for C_o the concentration of Ar-41 near the Reactor Building is calculated to be

$$C_o = 2 \times 10^{-2} \times 2.33 \times 4.5 \times 10^{-7}$$
$$= 2.1 \times 10^{-8} \text{ microcuries/ml}$$

This value is below the limit of 4.0×10^{-8} microcuries per milliliter for Ar-41 discharged into an unrestricted area as stated in 10CFR20.

To summarize the results of the analysis of airborne radioactivity at the UMRR data demonstrates that the major gaseous radioactivity is due to argon-41. Furthermore, airborne radioactivity released to unrestricted areas does not exceed 10CFR20 guidelines. In addition, it should be kept in mind that the total ventilation capacity available at the UMRR is by a factor of 7 higher than the one used in the analysis. Therefore, a further dilution at the discharge point can easily be achieved.

7.6.2 Liquid Releases

Liquid releases are covered in detail in section 6.2 and Table XIII.

7.7 Environmental Monitoring

Environmental monitoring is accomplished from within the Reactor Building by film badges located in strategic areas. The results over the last 19 years are shown in Table XIV. During a 200 kW power run of over four hours duration in July 1984 the highest reading found at one spot outside of the building was 0.2 millirem/hr over eight feet above the ground level and on contact with the south Reactor Building wall adjacent to the reactor bridge. All other readings outside of the building were less than 0.18 millirem per hour on contact with the building. These measurements are within 10CFR20.105 limits.

Table XIII.

Water Release Summary

FOR FISCAL YEARS

1978 TO 1983

Year	Gallons	Activity
1978-1979	1,500	0.50 mCi
1979-1980	1,200	1.04 mCi
1980-1981	9,255	0.846 mCi
1981-1982	4,160	0.541 mCi
1982-1983	3,955	0.125 mCi

The above values were for gross activity only. There were no peaks identified in any of the samples. These values were obtained from Gamma-Ray spectroscopy.

Table XIV

History of Exposure In The Reactor Building

Exposure indicated by film badge located in control room and on bridge. (millirem)

year	control room	bridge	cumulative permanent
1965	----	0	0
1966	----	0	0
1967	----	560	560
1968	----	250	810
1969	----	110	920
1970	120	520	1440
1971	0	360	1800
1972	0	110	1910
1973	20	170	2080
1974	0	280	2360
1975	0	180	2540
1976	0	250	2790
1977	0	40	2830
1978	0	30	2860
1979	30	0	2860
1980	50	1570[1)]	4430
1981	0	230	4660
1982	0	0	4660
1983	0	80	4740
1984	0	80	4820

1) Caused by numerous high power runs and handling of samples near the film badge.

8. ADMINISTRATIVE CONTROLS

8.1 Organizational Structure

The organization of the University of Missouri-Rolla as related to ensuring the proper use of the nuclear reactor and radioactive materials is shown in Figure 25. This organization involves a single, major committee, the Radiation Safety Committee.

8.2 Radiation Safety Committee

The reactor is operated under NRC License R-79 granted in 1961. As required by the license, a reactor advisory committee was appointed at the time and, as time went by, it has been called by different names. Its present, official title is the UMR Radiation Safety Committee. The organization within the University is shown in Figure 25 and the organization of the Reactor Facility is shown in Figure 26.

The UMR Radiation Safety Committee has the dual responsibility of:

(1) Advising the administration regarding matters relating to custody and use of radioisotopes on campus.

(2) Reviewing and making recommendations concerning experimental and operational activities of the UMR Nuclear

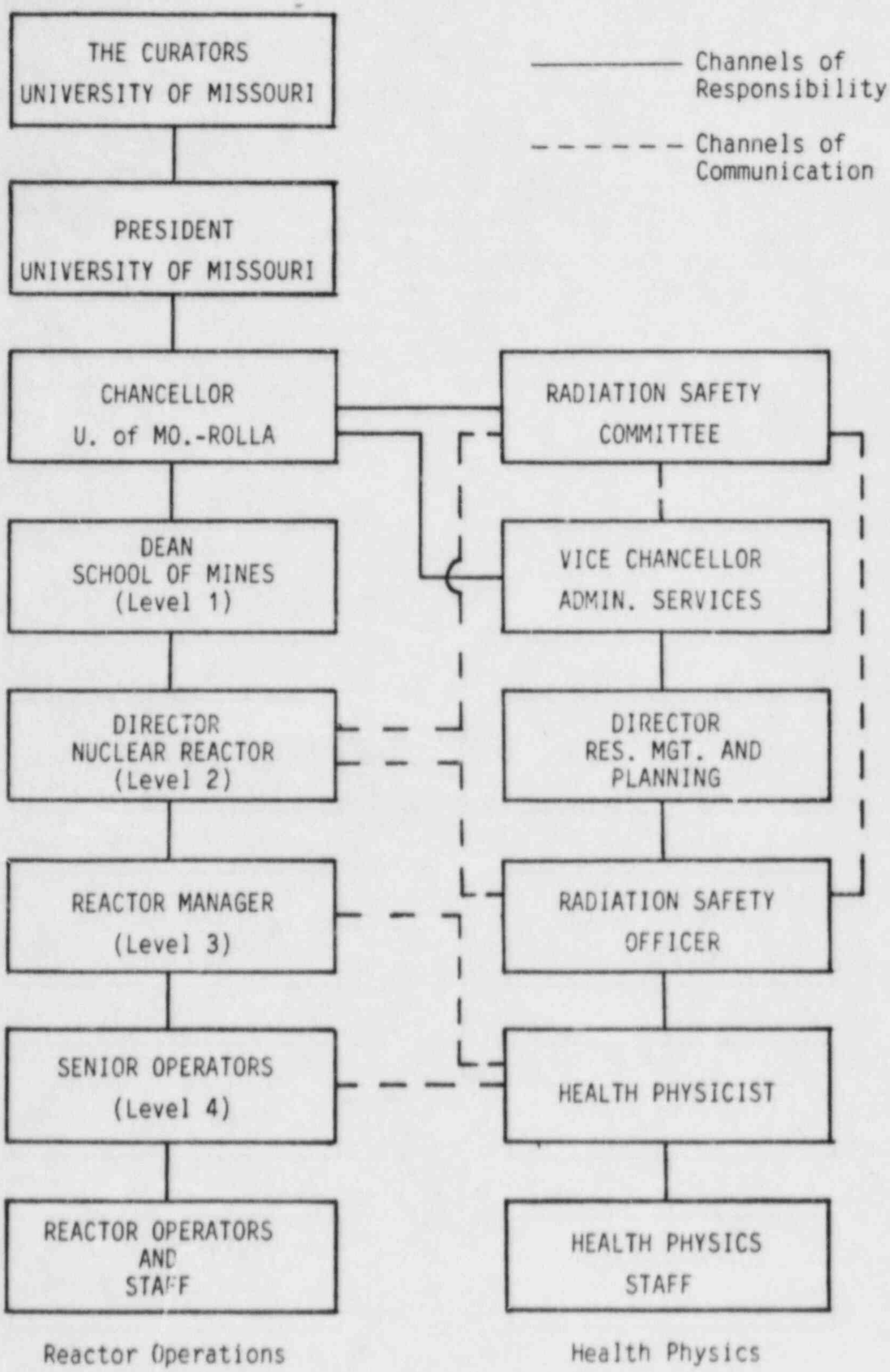


Figure 25. Organizational structure of the University of Missouri related to the UMR Nuclear Reactor Facility.

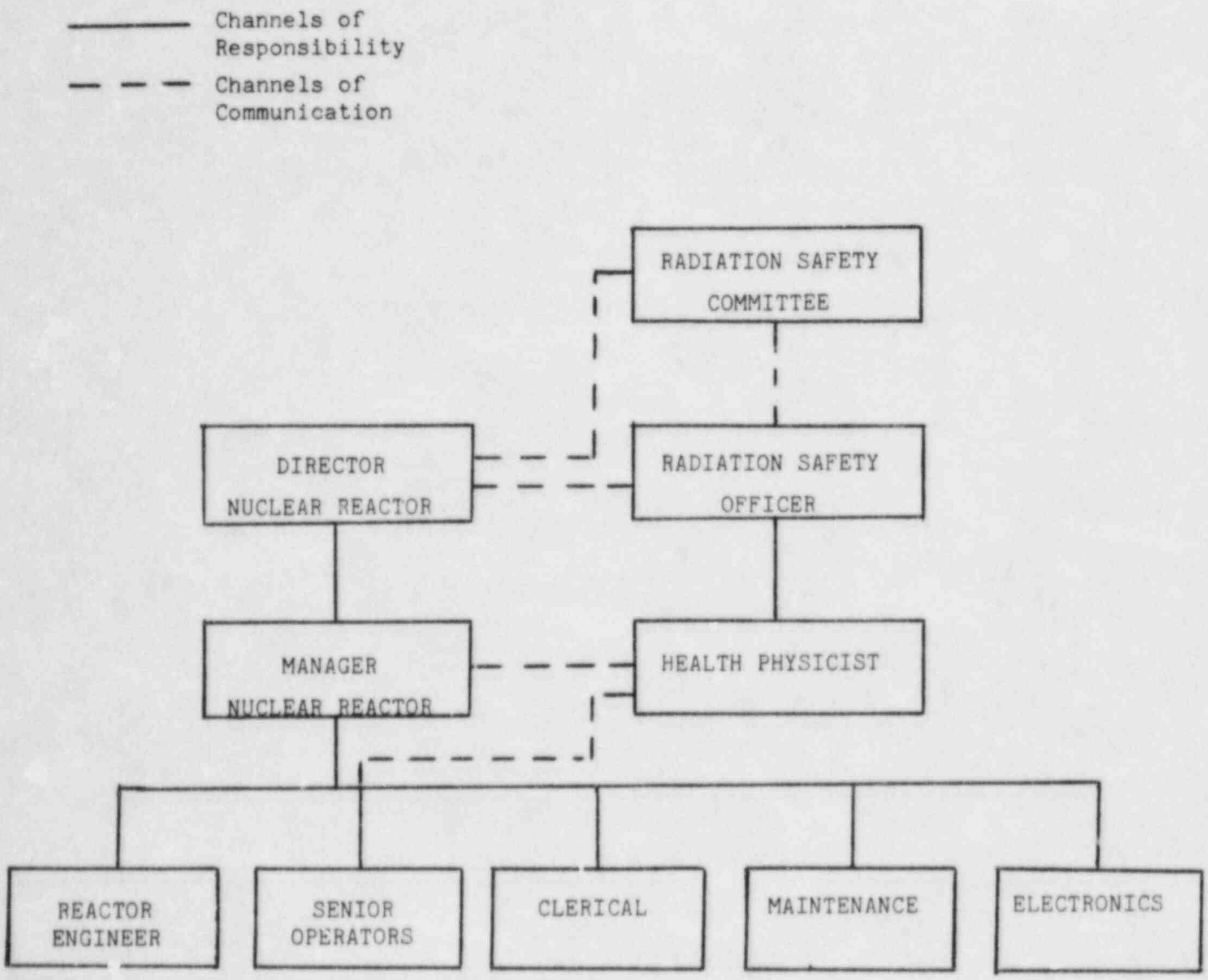


Figure 26. Organization of the UMR Reactor staff.

Reactor.

The Committee is appointed by the Chancellor to satisfy requirements imposed by the federal government. The Dean of the School of Mines and Metallurgy is the designated liaison officer through which the Committee reports to the administration.

The Committee is responsible for initial review of applications for use of radiation sources upon the campus. Applications are submitted to the Chairman through the Health Physicist and are reviewed in the same manner and using the same procedure as outlined for the University-Wide (U-W) Radiation Safety Committee in the Handbook of Radiological Operations. Safety, rather than feasibility, is the basis of criteria governing the Committee's evaluation of applications brought before it. After the review, all applications recommended for approval are forwarded to the Radiation Safety Officer. He submits the applications for review to the U-W Radiation Safety Committee. If the application is denied by the campus Committee, it is returned to the applicant with a statement of reasons for denial. The Committee is also responsible to insure that radiation safety is maintained on the campus.

In its role of reviewing the activities at the UMR Reactor, the Committee advises the Director of the Reactor Facility in matters pertaining to the safe operation of the reactor and with regard to planned research activities and use of the facility building and equipment. It will respond to matters brought before it by the Director, researchers, or other University administrative officials.

The responsibilities of the Committee are as follows:

(1) Review all requests for reactor time which are forwarded to it. This review shall encompass only matters concerning health and safety, and shall not touch upon the technical feasibility or advisability.

(2) Approve, provisionally approve with recommendations for change in the program, or disapprove all properly submitted requests, and advise the interested parties of the review.

(3) Review special reports issued by the Reactor Manager following any significant malfunctions, violations, or accidents. In addition to this review, the Committee shall either approve the corrective action already taken, or recommend further action.

The Committee shall meet at least quarterly. The Committee will maintain minutes of its meetings to include the items considered (particularly, the safety-related issues discussed), actions taken, and the recommendations made. Meetings are conducted in accordance with Robert's Rules of Order.

8.3 Independent Audit

Independent audits of the facility consist of reviewing the records, procedures, and operating procedures and they are performed at least annually. Such audits are done either by qualified faculty or staff, who are not associated with the reactor, or by reactor staff members from the University of Missouri Research Reactor which is located at Columbia.

Copies of the audits are enclosed in the Annual Progress Report submitted to the NRC in April each year.

8.4 Operating Procedures

The reactor is operated in accordance with written procedures established under the approval of the Reactor Director. These procedures include normal startup, operation and shutdown of the reactor as well as emergency procedures and special procedures for unusual operations. General procedures for the handling of experiments are promulgated but these are often supplemented by special procedures which apply only to the experiment under consideration.

All procedures concerning the modification of the reactor or its safety systems and associated reactor experiments must have the approval of the Reactor Director and Radiation Safety Committee and may be changed only by their authorization. However, in the final analysis, the safe operation of the reactor is dependent upon the reactor staff and their exercise of good judgement.

8.5 Staff Training Program

The UMR Reactor Facility has an NRC-approved operator requalification program that all licensed reactor operators and senior reactor operators must complete as a condition for renewal of their licenses. Persons who are preparing to take the NRC operators licensing examination participate in the sa e training program, as well as receive intensive "hands-o reactor operations training at the console. All licensed operators at UMRR participate in the program and must satisfactorily complete this program during each license renewal period. Each licensed operator or senior operator includes in his/her license renewal application a statement that he/she has satisfactorily completed the requirements of the requalification program.

The requalification program is divided into three major areas which are designed to provide assurance that all operators maintain competence in all aspects of licensed activities. The three areas are as follows:

(1) An annual written examination which is used to verify the operator's knowledge level. Special lectures are used to retrain those operators who demonstrate deficiencies in any part of the examination.

(2) On-the-job training which will ensure that the operator

maintains his/her competence in manipulating the controls and in operating all apparatus and mechanisms required by his/her license; that the operator is cognizant of all design, procedure and license changes implemented during the requalification period; and that he/she has a thorough understanding of all abnormal and emergency procedures.

(3) Periodic observation and evaluation which will be used to determine the performance of the operators to actual and simulated plant conditions.

8.6 Emergency Planning

The UMR Reactor Emergency Plan includes the guidelines, policy, and organization required to mitigate the consequences of an emergency. Specific implementation procedures are provided for each type of emergency in the Standard Operating Procedures for the UMR Reactor.

The principal objectives of the Emergency Plan are:

(1) to protect the health and safety of the general public beyond the site boundary,

(2) to establish the safety of reactor personnel and all persons within the site boundary,

(3) to establish controls and guidelines for those having

authority and responsibility for coping with the emergency situation to minimize any deleterious effect on the safety and welfare of all individuals involved,

(4) to provide division of responsibility and authority to facilitate and expedite remedial actions, and

(5) to provide for recovery and restoration of all affected zones.

The Emergency Plan was written in the fall of 1982 in accordance with NRC guidelines and submitted for their approval on October 25, 1982.

8.7 Physical Security Plan

There is a Physical Security Plan for the UMR Reactor Facility which describes the physical protection system and security organization which provides protection against radiological sabotage and detection of theft of special nuclear material from the facility.

The general performance objectives of the physical protection system and security orgnization described in the plan are as follows:

(1) to provide protection against acts of industrial sabotage,

(2) to minimize the possibilities of unauthorized removal of

special nuclear material consistent with the potential consequences of such actions, and

(3) to facilitate the location and recovery of missing special nuclear material.

In order to achieve these objectives the physical protection system provides the following:

(1) early detection and assessment of unauthorized access or activities by an external adversary within the vital areas and controlled access areas containing special nuclear material,

(2) early detection of removal of special nuclear material by an external adversary from controlled access areas,

(3) assures proper placement and transfer of custody of special nuclear material, and

(4) responds to indications of an unauthorized removal of special nuclear material and then notifies the appropriate response forces of its removal in order to facilitate its recovery.

The NRC-approved security measures went into effect on September 4, 1981.

9. ACCIDENT ANALYSES

In this chapter, details of the analyses and bases for the limiting safety system settings, established in the Technical Specifications for the UMRR, are given. Also, a whole spectrum of accidents, ranging from a credible accident up to the maximum hypothetical one, is discussed. The potential effects of the accidents on the health and safety of the public are analyzed.

9.1 Fuel Element Handling Accident

Fuel element maneuvers are always conducted in the reactor pool under a sufficient depth of water. They are removed from the core and moved into the storage space, one at a time, using a hand-held fuel handling tool. The procedure for unloading the core, always proceeds from outside to inside. During the core loading, the steps are performed in reverse order.

A fuel element weighs about 5.6 kg (12 lb) in air and only about 3.6 kg (8 lb) in water. Therefore, even if one fuel element should fall from the handling tool during its transfer it is not heavy enough to cause any considerable damage. The most severe consequence, likely to occur, would be some denting of the end fittings since the fuel element, being an elongated object, would tend to fall in water in a rather upright position.

The UMRR Standard Operating Procedures define administrative steps which are intended to prevent a fuel handling mishap. They are:

(1) All fuel handling is done in accordance with written procedures.

(2) Loading operations are done by qualified personnel under direct supervision of a Senior Operator.

(3) Fuel handling tools are kept locked with the keys secured to prevent unauthorized movement of fuel.

9.2 Flooding of an Irradiation Facility

A sudden replacement of a voided, i.e. air filled, space next to or within the core by water would cause a stepwise reactivity insertion. Its magnitude depends on the void volume being replaced and its position relative to the core. For example, experiments have shown (2) that flooding of the beam tube with water does not have any noticeable effect upon the reactivity of the core.

However, flooding of the isotope production element or core access element positioned in the central position of the core has been shown (2) to cause a reactivity change of about 0.7% delta k/k. If the special element is located at the core periphery its flooding would give rise to a reactivity of about 0.1% delta k/k. It is shown in Section 9.6 that a sudden reactivity insertion greater than 0.7% delta k/k into a critical core of the UMRR can be tolerated with a sufficient safety margin. Therefore, it is concluded that flooding of any irradiation facility would not endanger the reactor and would not pose any hazards to public health and safety.

9.3 Loss of Coolant Accident

The reactor pool is designed to prevent the possibility of an unintentional drainage. It is constructed of reinforced concrete and set in bed rock to resist the most severe earthquake. The pool has no drains. Therefore a sudden loss of coolant is considered to be extremely remote. But even if the pool drained instantaneously, while the reactor were operating, the loss of water (moderator) would shut the reactor down.

The most severe problem identified in this accident scenario is the removal of decay heat during and after loss of coolant. There is no danger of significant fuel overheating as long as the core stays immersed and heat can be removed by the water. If the core were to become uncovered, heat transfer would occur by natural convection of ambient air. For this case, steady-state heat transfer calculations show that the amount of heat removed is proportional to the cladding temperature (see Figure 27). Decay heat generation after reactor shutdown is shown in Figure 28. According to this Figure, and from operational experience, the decay power of the UMRR immediately after the shutdown from full power is about 14 kW. The corresponding cladding temperature to remove this power amounts to about 425° C. This is well below the melting temperature of 660° C for aluminum cladding. Moreover, the results of this analysis are conservative in that the amount of heat stored in the "fuel meat" and cladding during the heat-up period was not taken into account. The decay power rapidly decreases as indicated in Figure 28, being some 7 kW after 1 min. of cooling.

In addition to the inherent cooling mechanisms, discussed above, a fire hose is kept near the pool as an auxiliary measure in the case of the loss of pool water. It can be connected to a

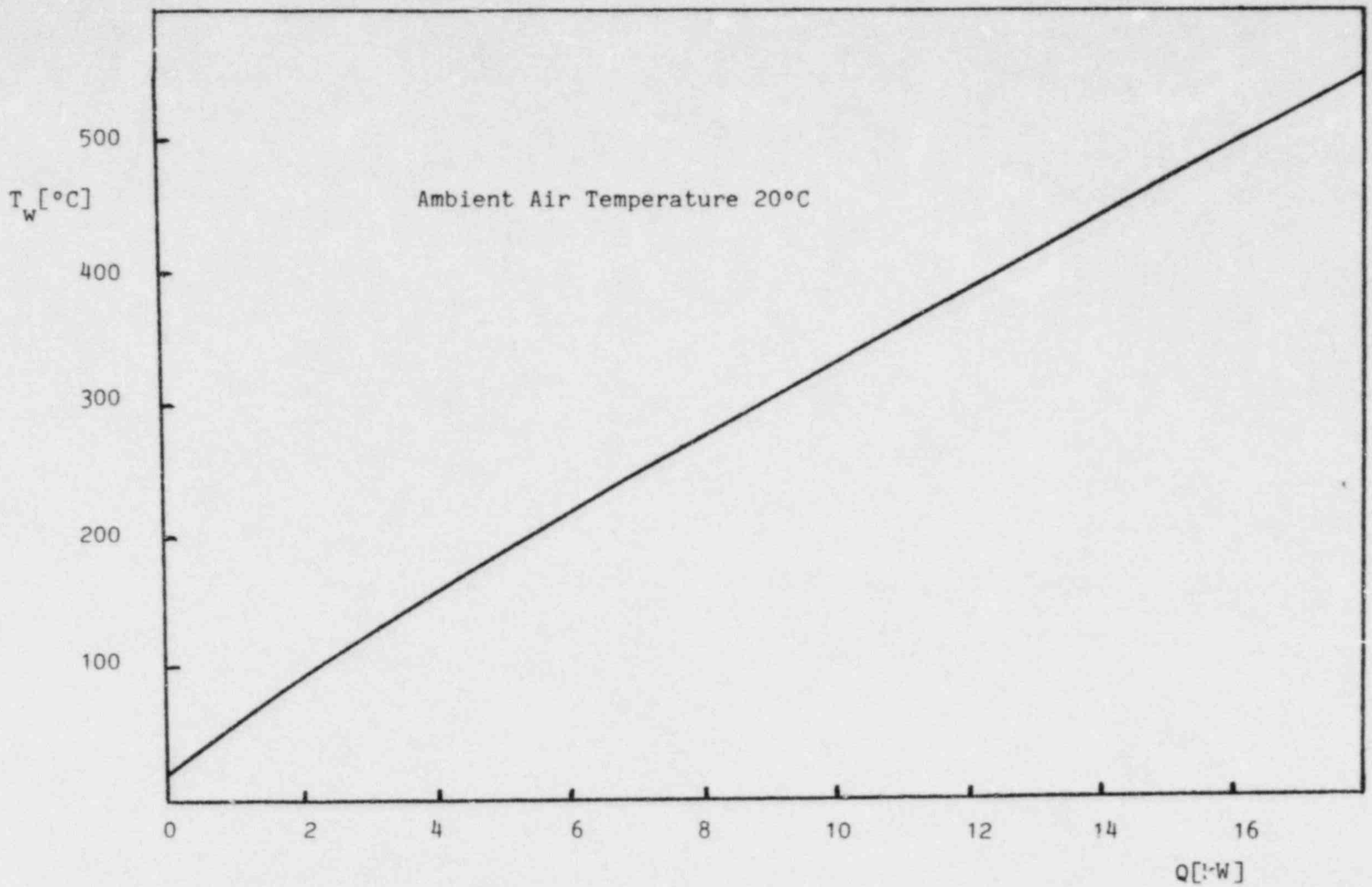


Figure 27. Cladding temperature versus core power dissipated by air convection.

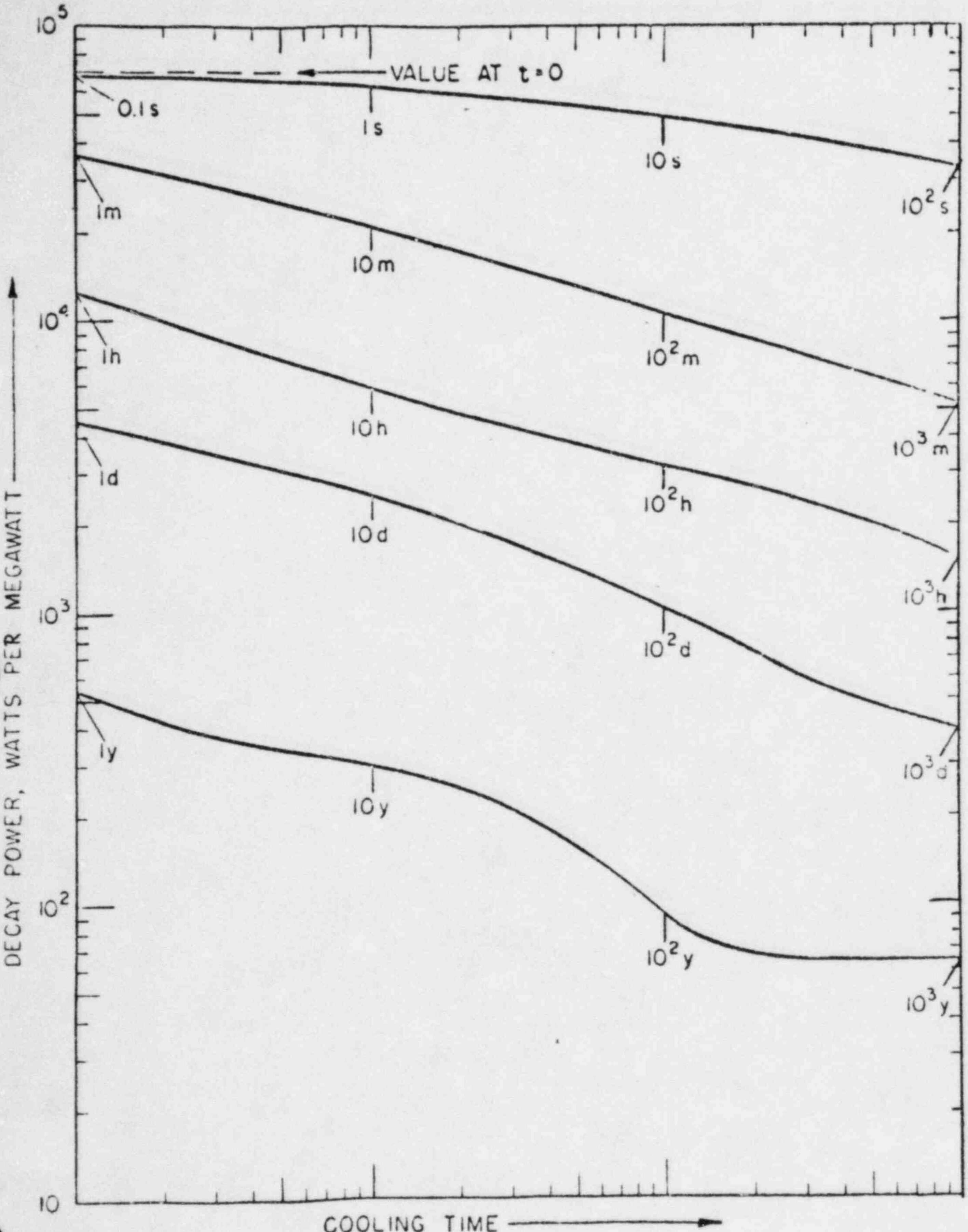


Figure 28. Decay power after infinite irradiation.

nearby fire hydrant and water added to the pool.

In any accident which is reasonably conceivable, the leakage of water from the reactor pool is expected to be rather slow. In such a case, two different automatic monitoring systems are available to signal a gradual loss of pool water. The radiation area monitor mounted on the reactor bridge, directly above the core, would detect any additional radiation coming from the core due to a decreasing pool water level. If the basement sump is being filled due to a pool leak (with the flow rate larger than 1 gal/min) a signal is sent to the annunciator board in the control room. Besides these automatic actions, the pool water level is checked during daily routine operations. It is concluded, that a slow leak of pool water would be discovered early and specific actions could be taken to mitigate its consequences.

9.4 Failure of a Movable Experiment

A sudden (stepwise) introduction of a positive reactivity into the critical reactor will cause a transient power increase. Its magnitude and course depend on the amount of the inserted reactivity. At the UMRR, the maximum reactivity worth of a movable experiment is limited by Technical Specifications to 0.4% delta k/k. In the following analysis an assumption is made, although it is highly unlikely under current operational practice, that an experiment with the maximum reactivity worth suddenly moves out of the core. This would result in a positive stepwise reactivity change of 0.4% delta k/k. A number of other conservative assumptions is made in that:

1) The reactor power is 200 kW.
2) All control rods are in a position with the least differential reactivity worth.
3) The most reactive control rod cannot be scrammed (stuck rod criterion).
4) The power excursion does not start to reverse until the reactor is brought back to critical.
5) No thermal feedback effects are taken into account.

Using the prompt jump approximation, the calculated power increase before the power excursion is reversed amounts to about 440 kW. (Details of the calculation are given in Appendix A.) The stable reactor period, corresponding to the reactivity insertion of 0.4% delta k/k, is 3 sec. Information in Table IX shows that this accident would be terminated by a number of protective action levels. Ultimately, two scram channels -- "Reactor Period < 5 sec" and "150% Full Power" -- would be activated too. Therefore, there are both redundancy and diversity available to terminate a mild power excursion such as described above.

The heat flux expected in the hot channel at the reactor power of 440 kW is less than 8 W/cm . In Section 3.4.5 it has been shown that such heat can be safely removed from the reactor core. It should be pointed out, that the results in Section 3.4.5 have been derived for steady-state heat transfer. However, in the accident discussed above, the reactor power of 440 kW is only an instantaneous power peak and immediately after the reactor has been scrammed it would decrease as shown in Figure 28. Therefore, as a result of the decreasing power, the cladding wall temperature in the hot channel during the power excursion would remain distinctly below 115 °C (240 ° F). Consequently, the safety margin available between the wall temperature and the melting temperature of cladding is larger than the steady-state heat transfer calculation used in this analysis indicates. It is concluded, therefore, that this reactivity insertion accident does not endanger the integrity of the reactor fuel.

9.5 Reactor Startup Accident

The reactor startup accident analyzed in this report is that condition during which reactivity is continually inserted into the UMRR at a given rate while the reactor is still subcritical or critical, but essentially at zero power. According to (14), both cases are quite similar. They exhibit similar power traces, the governing parameter being the reactivity insertion rate. The startup accident, which is postulated here, is assumed as being due to an uncontrollable simultaneous withdrawal of the regulating rod and all three shim/safety rods. (This is a highly unlikely situation since different control circuits, e.g. the gang switch for shim/safety rods and the interlock system between the reg rod and shim/safety rods, would both have to fail simultaneously.) The reactivity insertion rate of each control rod is given in Table XX.

Table XX. Maximum Reactivity Insertion Rate (% delta k/k per second)

Regulating rod	0.010
Shim/Safety rod No. 1	0.019
Shim/Safety rod No. 2	0.019
Shim/Safety rod No. 3	0.026

Total	0.074

It should also be pointed out that this reactivity insertion rate is a maximum which is available only along a short portion of the total distance each control rod can travel.

There is a number of protective action levels (as shown in Table IX) which would be consecutively activated to terminate a reactor startup accident. The very first protective action available is "120% Demand" which is automatically activated whenever the pointer on the linear power recorder exceeds the mark of 120. Next, as the amount of the inserted reactivity is continuously increased during the withdrawal of control rods, the reactor period becomes shorter, which causes the set points "Period < 30 s", "Period < 15 s", and "Period < 5 s" to be exceeded and corresponding protective actions to be automatically activated. Ultimately, if the reactor power exceeds 150% of full power, two independent power safety channels are activated scramming the reactor. Signals for each protective action discussed above are sensed by different sensors and are processed by different signal processing equipment. Hence this equipment diversity which is available at the UMRR provides a large safety margin which would not allow any startup failure to develop in a potentially serious accident.

A startup accident can be hypothesized even further by assuming that no protective action would be automatically initiated during a ramp reactivity insertion. This scenario was already analyzed for the UMRR in the Preliminary Hazards Evaluation (1). In that analysis, data obtained from the BORAX and SPERT experiments, in which self-shutdown behavior was investigated, were used. It was concluded that the UMRR could withstand a ramp addition of almost 2.5% delta k/k at a rate of 0.09% delta k/k per second without mechanically damaging the fuel elements or approaching the melting point of the elements. At this rate, it would take more than 27 seconds of continuous withdrawal of all rods to approach the point of danger. Therefore, even in this very unlikely scenario, there would be ample time for the reactor operator to take an appropriate corrective action.

This analysis has shown how unlikely it is that a startup failure would develop into a serious accident. Therefore, no adverse consequences are to be expected to the health and safety of the public nor to the reactor staff from this type of accident.

9.6 Maximum Reactivity Insertion

In this section mechanisms which could give rise to a large reactivity increase (such as $\rho > \beta$) are analyzed in order to identify the maximum reactivity insertion for which a safety analysis needs to be performed. The Technical Specifications specify the excess reactivity for the UMRR as follows:

> The fuel loading shall be such that the excess reactivity above the reference core condition will be no more than 1.5% delta k/k, except that the excess reactivity may be increased up to a maximum of 3.5% delta k/k for the purposes of control rod calibraton only. This increase in excess reactivity above 1.5% delta k/k will be permitted no more than twice a year and for no more than five consecutive working days each time. The reactor shall be operated only by a licensed Senior Operator when the excess reactivity is greater than 1.5% delta k/k.

In spite of extensive staff discussions and literature research no credible accident scenario has been found which could possibly lead to a sudden release of excess reactivity larger than 1.5% delta k/k. Therefore, an instantaneous insertion of the excess reactivity larger than 1.5% delta k/k has been excluded from further analysis.

Experiments at the Curtiss-Wright Research Reactor (11) have shown that the worth of a fuel element at the core periphery is less than 1.5% delta k/k. This is consistent with experience at the UMRR gained with different core configurations. Depending on its position at the core periphery a standard fuel element can be worth between 0.5% and 1.5% delta k/k. (The reactivity worth of a fuel element within the reactor core is not well known since the Technical Specifications preclude reactor operation with an empty internal lattice position.)

A hypothetical accident can be postulated assuming that a fuel element has been placed next to a barely subcritical core, thus resulting in a positive step reactivity insertion of 1.5% delta k/k. A sudden reactivity insertion of such a magnitude would cause the reactor to become prompt critical with a subsequent exponential power increase. The reactor period at the beginning of the prompt critical power excursion can be approximately calculated (14) from the expression

$$T \doteq \frac{\ell_p}{\rho_o - \beta}$$

where ℓ_p = prompt neutron lifetime

(for the UMRR it is 4.5×10^{-5}s)

β = delayed neutron fraction

Using $\beta = 0.0075$ the reactor period corresponding to the above postulated reactivity insertion is 6 ms.

In the analysis of short power excursions the total energy release and the resulting maximum fuel plate temperature are two of the most important physical parameters. In order to establish a relationship between those two parameters for the accident investigated in this work, a comparison with experimental data was sought. Fortunately, a large collection of data from the excursion experiments performed at the BORAX and SPERT facilities is available. Especially, some of the

SPERT-1 experiments using the DU-12/25 core are applicable to the analysis of the UMR Reactor since the fuel geometry and composition are very similar (9). A detailed comparison is given in Table XV.

A series of self-limiting power excursion tests was carried out in SPERT-1 using 5 core loadings. The input variable commonly referred to in these experiments was the reactor period induced by a stepwise reactivity insertion. At periods of the order of 6×10^{-3} to 9×10^{-3} sec the tests have shown some plate buckling and a ripple pattern due to thermal expansion stresses in the plate (10). During shorter period transients the plates appeared to have softened and remained in a plastic state for several days. From the tests it was concluded that the mechanism responsible for self-limiting the power excursion consists of fuel and moderator thermal expansion and boiling. (The latter being the dominant shutdown mechanism.)

There was one experiment in which the reactor period was 6×10^{-3} sec. The total energy released in the excursion was 13.2 MW-sec. Onset of the self-limiting mechanism occurred when about 7.2 MW-sec of the thermal energy was generated. No damage to the fuel cladding was observed. The maximum fuel surface temperature recorded was 560°C (1040°F) which is well below the aluminum cladding melting temperature of 660°C (1220°F).

Thus, from the results of experiments with various stepwise reactivity insertions it can be concluded that the

Table XV. Comparison of Important Fuel Data

Geometry	UMRR Plate	SPERT-1 Plate
Length [cm]	61	61
Width [cm]	7.6	7.6
Thickness [cm]	0.15	0.15
Water gap [cm]	0.63	0.45
Fuel		
Material	U_3O_8-Al	U-Al
Enrichment [%]	≈90	100
Weight fraction of U	0.36	0.24
Thickness [mm]	0.51	0.51
Cladding		
Material	Al	Al
Thickness [mm]	0.51	0.51

above-postulated accident would be safely terminated by this self-limiting shutdown mechanism. This is a rather surprising result. However, the short time constant of the thin fuel plates allows a large amount of energy to be transferred into the water channels even during very short reactor periods. Consequently, boiling becomes the rapid and dominating shutdown factor. Such an accident can, therefore, be terminated even if the safety instrumentation, e.g. both power safety channels, were inoperable.

In spite of this UMRR safety feature, administrative steps have been established in the Standard Operating Procedures which are designed to prevent a fuel handling accident:

(1) All fuel handling is done in accordance with written procedures.

(2) Loadings are planned to include the sequence of loading and positions of individual elements. Also a loading schedule is prepared prior to commencement of loading.

(3) Loading operations are done by qualified personnel under the direct supervision of a licensed Senior Operator.

(4) Fuel handling tools are kept locked with the keys secured to prevent unauthorized movement of fuel.

(5) Loading of the core is done from the inside to the

outside.

Finally, it should be pointed out that the assumptions leading to this accident are very unlikely, and therefore it is not believed that such an accident would ever happen. The analysis, however, has been quite useful in showing the inherent safety capacity of the UMRR. Therefore, no effects on the health and safety of the public nor on the reactor staff are to be expected from this type of accident.

9.7 Failure of a Fueled Experiment

In this section an analysis is performed to assess the hazard associated with the failure of an experiment in which fissile material has been irradiated in the reactor. In the scenario of this accident it is assumed that a capsule containing irradiated fissile material breaks and a portion of the fission product inventory becomes airborne. The consequences of the release are analyzed for both the reactor staff and general public. Since the potential impact of this postulated accident is greater than in any other accident analyzed, the failure of a fueled experiment is designated as the maximum hypothetical accident for the UMRR.

The limiting criterion used in the analysis of a fueled experiment is the power generated within the irradiated fissile material. In this analysis the consequences of a failed experiment generating 1 W and 100 W, respectively, were studied. The fission products expected to become airborne, in the case where the experiment capsule were to lose its integrity, are noble gases and elemental iodine. Other fission products and actinides are not volatile at the temperature (which is essentially at room temperature) at which the fueled experiment would be performed. The amount of noble gases and radioiodine is assumed to be that specified in (13), i.e.: 100% of the noble gases and 50% of the iodine inventory. If the experiment were to be run in the reactor pool a credit for the absorption of iodine in water can be taken (14). This partition factor amounts to 10, i.e. only about 5% of the total iodine inventory would reach the Reactor Building atmosphere in an accident.

A conservative assumption was made in the analysis in that the irradiation time was considered to be infinite. Therefore, the fission inventory used in the analysis represents for some long-lived radionuclides, e.g. Kr-85, most likely an overly conservative value. Furthermore, it was assumed that the fission products are instantaneously released and uniformly distributed in the Reactor Building air. The free volume of the Reactor Building is approximately 1.7×10^3 m^3.

The external dose rate (in mrem/hr) due to γ- and β-radiation was calculated using the relationships given in (15)

$$\dot{D}_\gamma = 9.43 \times 10^{11} \times X \times \bar{E}_\gamma$$

where X = radionuclide concentration (Ci/cm^3)

$\bar{E}_\gamma$ = average γ-energy per disintegration (MeV)

and

$$\dot{D}_\beta = 8.24 \times 10^{11} \times X \times \bar{E}_\beta$$

where $\bar{E}_\beta$ = average β-energy per disintegration (MeV). The dose rate to the thyroid (in rem/hr) due to the inhalation of radioiodines is given by

$$\dot{D}_T = DCF \times B \times X$$

where DCF = dose-conversion factor for the thyroid (rem/Ci)

B = breathing rate (cm^3/hr)

X = radioiodine concentration (Ci/cm^3)

The standard breathing rate recommended (14) is 1.25×10^6 cm^3/hr. The thyroid dose-conversion factors are given in Table XXI.

Table XXI. Iodine Dose-Conversion Factors for the Thyroid (14)

Isotope	DCF (rem/Ci)
I-131	1.0×10^{6}
I-132	6.6×10^{3}
I-133	1.8×10^{5}
I-134	1.1×10^{3}
I-135	4.4×10^{4}

The calculated saturation activity for each respective radioisotope and its concentration in the Reactor Building after experiment failure is shown in Table XVI for the experiment power of 1 W. Also shown in this table are the associated γ - and β - radiation energies together with calculated dose rates for the whole-body, skin, and the thyroid. With a γ -dose rate in the reactor building as high as 250 mrem/hr any one of radiation area monitors would cause an automatic reactor shutdown, audible and visual alarms in the control room, and in addition the reactor bridge monitor would activate the building evacuation alarm system. From the past experience, it is known that the reactor building can be evacuated within 3 minutes. For the purpose of this analysis it is assumed that the time elapsed between the release of radioactivity and the end of evacuation is 5 minutes. Therefore, it is assumed in the following calculation that the exposure time to the members of the reactor staff is 5 minutes. The resulting radiation doses are: whole-body dose 20.6 mrem, skin dose 11.21 mrem, and the thyroid dose 0.93 rem.

Table XVI.

Dose Rates in the Reactor Building from a Failed Fuel Experiment (Power = 1 W)

ISOTOPE	A_{sat} (Ci)	$\bar{E}_\gamma$ (MeV)	$\bar{E}_\beta$ (MeV)	X_β ($^{Ci}/cm^3$)	$\dot{D}_\gamma$ ($\frac{mrem}{hr}$)	$\dot{D}_\beta$ ($\frac{mrem}{hr}$)	$\dot{D}_T$ ($\frac{rem}{hr}$)
I-131	2.45 E-2	3.71 E-1	1.97 E-1	7.20 E-12	2.52 E 0	1.17 E 0	9.00 E 0
I-132	3.71 E-2	2.40 E 0	4.48 E-1	1.09 E-11	2.47 E 1	4.03 E 0	9.00 E-2
I-133	5.48 E-2	4.77 E-1	4.23 E-1	1.61 E-11	7.25 E 0	5.60 E 0	3.62 E 0
I-134	6.06 E-2	1.94 E 0	4.55 E-1	1.78 E-11	3.26 E 1	6.65 E 0	2.45 E-2
I-135	5.06 E-2	1.78 E 0	3.08 E-1	1.49 E-11	2.50 E 1	3.78 E 0	8.16 E-1
Kr-83m	5.90 E-3	2.60 E-3	1.03 E-2	3.47 E-12	8.51 E-3	2.95 E-2	
Kr-85m	1.27 E-2	1.51 E-1	2.23 E-1	7.47 E-12	1.06 E 0	1.37 E 0	
Kr-85	2.53 E-3	2.11 E-3	2.23 E-1	1.49 E-12	2.12 E-1	2.74 E-1	
Kr-87	2.00 E-2	1.37 E 0	1.05 E 0	1.18 E-11	1.52 E 1	1.02 E 1	
Kr-88	3.12 E-2	1.74 E 0	3.41 E-1	1.84 E-11	3.02 E 1	5.17 E 0	
Kr-89	3.96 E-2	1.60 E 0	1.33 E 0	2.33 E-11	3.52 E 1	2.56 E 1	
Xe-131m	2.53 E-4	2.00 E-2	1.40 E-1	1.49 E-13	2.81 E-3	1.72 E-2	
Xe-133m	1.35 E-3	3.26 E-1	1.55 E-1	7.94 E-13	2.44 E-1	1.01 E-1	
Xe-133	5.48 E-2	3.00 E-2	1.46 E-1	3.22 E-11	9.11 E-1	3.87 E 0	
Xe-135m	1.77 E-2	4.22 E-1	9.74 E-2	1.04 E-11	4.14 E 0	8.35 E-1	
Xe-135	5.23 E-2	2.46 E-1	3.22 E-1	3.08 E-11	2.90 E 1	8.17 E 0	
Xe-137	5.31 E-2	1.50 E-1	1.37 E 0	3.12 E-11	4.41 E 0	3.52 E 1	
Xe-138	5.57 E-2	1.10 E 0	8.00 E-1	3.28 E-11	3.40 E 1	2.16 E 1	
				Total	2.47 E 2	1.34 E 2	1.36 E 1

In Table XVII dose rates in the reactor building from a failed fuel experiment generating 100 W are shown. It was assumed that this experiment would be run at the reactor core within the water pool. Therefore, as discussed previously, in the calculation of the iodine concentration in the reactor building air a retention factor of 10 was assumed for the reactor pool. Assuming the same evacuation time as above, the respective radiation doses to the staff members are calculated: whole-body 1.37 rem, skin 0.96 rem, and to the thyroid 11.3 rem.

For the radiation calculations outside of the reactor building it was assumed that all fission products released in the reactor building would leak out within 24 hours. Since the reactor building does not have any windows and has only a few openings such as the ones for fans, air conditioners, etc., which could be readily sealed from outside in the case of an emergency, the assumption about the leak rate is considered to be conservative. Another conservative assumption was made in that no radioactive decay and hence no decrease in the source strength was taken into account while calculating the dose rates outside the reactor building. The radionuclide concentration at the nearest boundary of the exclusion zone was calculated using the atmospheric dispersion factor recommended in (16)

$$\frac{X}{Q^1} = \frac{1}{\pi \bar{u} \sigma_Y \sigma_Z}$$

where X/Q^1 = atmospheric dispersion factor (s/m^3)

Q^1 = source rate (Ci/s)

Table XVII.

Dose Rates in the Reactor Building

From a Failed Fuel Experiment (Power = 100 W)

ISOTOPE	X_B (Ci/cm^3)	$\dot{D}_\gamma$ ($\frac{mrem}{hr}$)	$\dot{D}_\beta$ ($\frac{mrem}{hr}$)	$\dot{D}_T$ ($\frac{rem}{hr}$)
I-131	7.20 E-11	2.52 E 1	1.17 E 1	9.00 E 1
I-132	1.09 E-10	2.47 E 2	4.03 E 1	9.00 E-1
I-133	1.61 E-10	7.25 E 1	5.60 E 1	3.62 E 1
I-134	1.78 E-10	3.26 E 2	6.65 E 1	2.45 E-1
I-135	1.49 E-10	2.50 E 2	3.78 E 1	8.16 E 0
Kr-83m	3.47 E-10	8.51 E-1	2.95 E 0	
Kr-85M	7.47 E-10	1.06 E 2	1.37 E 2	
Kr-85	1.49 E-10	2.12 E 1	2.74 E 1	
Kr-87	1.18 E-9	1.52 E 3	1.02 E 3	
Kr-88	1.84 E-9	3.02 E 3	5.17 E 2	
Kr-89	2.33 E-9	3.52 E 3	2.56 E 3	
Xe-131m	1.49 E-11	2.81 E-1	1.72 E 0	
Xe-133m	7.94 E-11	2.44 E 1	1.01 E 1	
Xe-133	3.22 E-9	9.11 E 1	3.87 E 2	
Xe-135m	1.04 E-9	4.14 E 2	8.35 E 1	
Xe-135	3.08 E-9	2.90 E 3	8.17 E 2	
Xe-137	3.12 E-9	4.41 E 2	3.52 E 3	
Xe-138	3.28 E-9	3.40 E 3	2.16 E 3	
	Total	1.64 E 4	1.15 E 4	1.36 E 2

$\bar{u}$ = average windspeed (m/s)

σ_Y = lateral plume spread (m)

σ_Z = vertical plume spread (m)

In the above expression for the atmospheric dispersion factor no credit was taken for so-called building wake effects and horizontal plume meandering both of which help in spreading the radioactive plume. Using the average windspeed of 1 m/s and Pasquill type F atmospheric conditions the dispersion factor at the boundary of the exclusion zone (~100 m) is calculated to be 4×10^{-2} s/m^3.

(Actually, the average windspeed in about 4.4 m/s as shown in Table IV. Therefore, the results of this analysis are conservative at least by a factor of 4.)

Calculated dose rates at the exclusion area boundary for experiment powers of 1 W and 100 W are shown in Table XVIII and Table XIX, respectively. If it is assumed, in accordance with 10 CFR 100, that an individual is located at the exclusion area boundary for 2 hrs following the fission product release from a postulated experiment failure at 100 W then his/her resulting radiation dose to the whole body would be 22.6 mrem and to the thyroid 2.1 rem. These doses are, however, only fractions (about 1%) of those which are referred to in 10 CFR 100 in conjunction with the determination of an exclusion area. The respective doses to the public at the exclusion area boundary caused by the failure of an experiment operating at 1 W are even smaller, e.g. for the thyroid by a factor of 10.

Table XVIII.

Dose Rates at the Exclusion Area Boundary

From a Failed Fuel Experiment (Power = 1 W)

ISOTOPE	X_{EZ} ($^{Ci}/cm^3$)	$\dot{D}_{\gamma}$ ($\frac{mrem}{hr}$)	$\dot{D}_{\beta}$ ($\frac{mrem}{hr}$)	$\dot{D}_{T}$ ($\frac{rem}{hr}$)
I-131	1.50 E-15	5.25 E-4	2.43 E-4	4.50 E-2
I-132	8.80 E-15	2.00 E-2	3.26 E-3	1.74 E-3
I-133	7.30 E-15	3.29 E-3	2.55 E-3	3.94 E-2
I-134	1.44 E-14	2.64 E-2	5.40 E-3	4.75 E-4
I-135	1.21 E-14	2.02 E-2	3.06 E-3	1.60 E-2
Kr-83m	2.80 E-14	6.88 E-5	2.38 E-4	
Kr-85m	6.03 E-16	8.58 E-5	1.11 E-4	
Kr-85	1.20 E-15	2.39 E-6	2.21 E-4	
Kr-87	9.14 E-15	1.18 E-2	7.91 E-3	
Kr-88	1.48 E-14	2.44 E-2	4.17 E-3	
Kr-89	1.88 E-14	2.84 E-2	2.06 E-2	
Xe-131m	1.20 E-16	2.27 E-6	1.39 E-5	
Xe-133m	6.44 E-16	1.98 E-4	8.22 E-5	
Xe-133	2.61 E-14	7.38 E-4	3.14 E-3	
Xe-135m	8.41 E-15	3.34 E-3	6.75 E-4	
Xe-135	2.49 E-14	5.77 E-3	6.60 E-3	
Xe-137	2.53 E-14	3.57 E-3	2.85 E-2	
Xe-138	2.65 E-14	2.75 E-2	1.75 E-2	
	Total	1.76 E-1	1.04 E-1	1.03 E-1

Table XIX.

Dose Rates at the Exclusion Area Boundary

From a Failed Fuel Experiment (Power = 100 W)

ISOTOPE	X_{EZ} ($^{Ci}/cm^3$)	$\dot{D}_\gamma$ ($\frac{mrem}{hr}$)	$\dot{D}_\beta$ ($\frac{mrem}{hr}$)	$\dot{D}_T$ ($\frac{rem}{hr}$)
I-131	1.50 E-14	5.25 E-3	2.43 E-3	4.50 E-1
I-132	8.80 E-14	2.00 E-1	3.26 E-2	1.74 E-2
I-133	7.30 E-14	3.29 E-2	2.55 E-2	3.94 E-1
I-134	1.44 E-13	2.64 E-1	5.40 E-2	4.75 E-3
I-135	1.21 E-13	2.02 E-1	3.06 E-2	1.60 E-1
Kr-83m	2.80 E-12	6.88 E-3	2.38 E-2	
Kr-85m	6.03 E-14	8.58 E-3	1.11 E-2	
Kr-85	1.20 E-13	2.39 E-4	2.21 E-2	
Kr-87	9.14 E-13	1.18 E 0	7.91 E-1	
Kr-88	1.48 E-12	2.44 E 0	4.17 E-1	
Kr-89	1.88 E-12	2.84 E 0	2.06 E 0	
Xe-131m	1.20 E-14	2.27 E-4	1.39 E-3	
Xe-133m	6.44 E-14	1.98 E-2	8.22 E-3	
Xe-133	2.61 E-12	7.38 E-2	3.14 E-1	
Xe-135m	8.41 E-13	3.34 E-1	6.75 E-2	
Xe-135	2.49 E-12	5.77 E-1	6.60 E-1	
Xe-137	2.53 E-12	3.57 E-1	2.85 E 0	
Xe-138	2.65 E-12	2.75 E 0	1.75 E 0	
	Total	11.3 E 0	9.12 E 0	1.03 E 0

It is concluded that experiments using fissile material can be irradiated at the UMRR within the power limits analyzed in this section. There is no undue hazard to the general public nor to the reactor staff in the very hypothetical case of a failed experiment as postulated above.

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Appendix A

Appendix A

Assume a stepwise reactivity insertion of ρ_o =0.4% delta k/k. Then using the "prompt jump" approximation the reactor power vs. time elapsed is given by

$$P = \frac{\beta}{\beta - \rho_o} P_o e^{t/T} \qquad (1)$$

where β = fraction of delayed neutrons

P_o = initial power

T = reactor period

t = time

The reactor period corresponding to the reactivity insertion of 0.4% delta k/k is found to be 3 sec [8]. Therefore, the UMRR would be scrammed due to exceeding the protection limit, "Period < 5 sec."

<u>Assumptions</u>:

1) β = 0.75%
2) P_o = 200 kW
3) the most reactive rod, i.e. rod No. 3, gets "stuck"
4) the other rods are in a position with the least differential reactivity worth, i.e. $d\rho/dx \approx 0$
5) the power does not start to reverse until the reactor is subcritical

To get the UMRR back to critical, i.e. $\rho = 0$, rods No. 1 and No. 2 must be inserted (during the scram) at least 4 inches.

From the periodic measurements of rod drop times it is known that the average value of

1) the rod separation time is 25 msec.

2) the falling speed of a scramming rod is 1 inch in 12 msec.

Therefore, the time elapsed from the reactivity insertion ($\rho = \rho_0$) until the reactor is "scrammed" to critical ($\rho = 0$) is

$$t = 25 \text{ msec} + 4 \text{ in} \times 12 \frac{\text{msec}}{\text{in}} = 73 \text{ msec.}$$

$$\approx 8 \times 10^{-2} \text{ sec}$$

From eq(1) the power increase until the time that the reactor is critical again is calculated to be

$$P = \frac{0.75}{0.75-0.4} \quad 200 \text{ kW} \times e^{8\times10^{-2}/3}$$

$$= 440 \text{ kW}$$

Note:

Because of such a prompt power increase the UMRR would also be scrammed by a "150% Full Power" protection level. This scram signal would occur almost simultaneously with the period signal, "Period < 5 sec."

Appendix B

UNIVERSITY OF MISSOURI-ROLLA

Office of the Vice Chancellor

Administrative Services

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JULY 2, 1984

TO: UMR Radiation Safety Committee

FROM: Neil K. Smith

RE: Commitment to the ALARA Principle

The University of Missouri - Rolla Radiation Safety Office has always operated with the As Low As Reasonably Achievable (ALARA) principle as a guideline, even before ALARA became a national standard. At this time, I would like to inform the committee that the ALARA principle has been, is, and will continue to be the official guide with respect to radiation exposure of students, faculty, staff and the public. Since this statement constitutes the official position of the UMR administration, it is expected that the UMR Radiation Safety Committee will be taking actions which are always consistent with the ALARA concept.

mm

cc: Chancellor Marchello
Dean Warner
Members of Radiation Safety Committee
Dr. Al Bolon
Ray Bono