UNIVERSITY OF MISSOURI-ROLLA

TRIGA REACTOR

LICENSE NO. R-79

DOCKET NO. 50-123

SAFETY ANALYSIS REPORT

REDACTED VERSION

SECURITY-RELATED INFORMATION REMOVED

Redacted text and figures blacked out

or denoted by brackets

SAFETY ANALYSIS REPORT
FOR THE
UNIVERSITY OF
MISSOURI-ROLLA REACTOR

Revision 1
August 13, 2004

SAFETY ANALYSIS REPORT

FOR

THE UNIVERSITY OF MISSOURI-ROLLA REACTOR

WITH LOW ENRICHED URANIUM FUEL

License Number R-79
Docket Number 50-123

Prepared by:

William E. Bonzer
Christopher M. Carroll

With contributions by:
David W. Freeman,
Reactor Staff,
And
Health Physics

University of Missouri-Rolla
Revision 1, August 13, 2004

Table of Contents

List of Figures

Figure | Page

List of Tables

Table Page

Abbreviations

1. THE FACILITY

1.1 Introduction

The purpose of this Safety Analysis Report (SAR) is to summarize the results of the safety review of the University of Missouri-Rolla Reactor (UMRR) and to delineate the scope of the technical details considered in evaluating the radiological safety aspects of continued operation. This SAR will serve as the basis of renewal of the license for operation of the UMRR facility at power levels up to and including 200 kW_t. The facility was reviewed against Federal regulations (10 CFR 20, 30, 50, 51, 55, 70 and 73), applicable regulatory guides (principally Division 2, Research and Test Reactors), and appropriate accepted industry standards (American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series).

This revised SAR was written as a part of the documentation required for the conversion of the UMRR to low enriched uranium (LEU) fuel. Preceding this report, three documents, [1-1] [1-2] and [1-3] were submitted to the regulatory agencies at various stages of the facility development and its license renewal. These previous documents analyzed the UMR reactor with the core consisting of high enriched uranium (HEU) fuel.

The UMRR began to operate in December 1961 as an open-pool-type reactor, with fuel of the Materials Testing Reactor (MTR) type. At that time it was licensed for the power level of 10 kW_t. In 1967, an amendment was granted to increase the maximum power to 200 kW_t. In 1992, the fuel was converted from HEU to LEU. The average yearly thermal output is about 10 MW_t–hrs. The reactor is operated by a professional staff within the School of Materials, Energy & Resources 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 UMR faculty, UMR graduate students, and UMRR staff. The UMRR is made available to users from outside the University under suitable contract arrangements, e.g. to the electric utilities for their 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 (DOE).

1.2 Summary and Conclusions on Principal Safety Considerations

The design, testing, and performance of the reactor structures, systems, and components important to safety during normal operation are inherently safe and safe operation can reasonably be expected to continue.

The expected consequences of a broad spectrum of postulated credible accidents have been considered, emphasizing those that could lead to a loss of integrity of fuel-element cladding. Conservative analyses of the most serious credible accidents have been performed and determined that the calculated potential radiation doses outside of the reactor room would not exceed 10 CFR 20 guidelines for personal in unrestricted areas.

The management organization, conduct of training and research activities, and its security measures are adequate to ensure safe operation of the facility and protection of its special nuclear material. The systems provided for the control of radiological effluents can be operated to ensure that releases of radioactive wastes from the facility are within the limits of the Commission's regulations and are as low as reasonably achievable (ALARA). The Technical Specifications, which provide limits controlling operation of the facility, are such that there is a high degree of assurance that the facility will be operated safely and reliably. The financial data provided are such that the licensee has sufficient revenues to cover operating costs and eventually to decommission the reactor facility.

1.3 General Description of the Facility

The UMRR is a heterogeneous, swimming-pool-type non-power reactor. The core is cooled by natural convection of light water, moderated by water, and reflected by water and graphite. The reactor core is located near the bottom of a water-filled pool formed by a reinforced concrete shielding structure. The core and control systems are suspended from a bridge that rides on rails above the reactor pool; this arrangement permits controlled movement of the reactor system to provide radiation fields in various locations within the pool.

The reactor core is composed of approximately 20 fuel elements positioned in holes in an aluminum grid plate. The grid plate is suspended from the movable bridge by an aluminum framework. The grid plate contains a 6 by 9 array of holes to allow changing fuel element locations. Each fuel element consists of several thin metal plates assembled into a unit about 7.6 cm x 7.6 cm (3 in x 3 in) with an active fuel length of ~0.61 m (~2 ft). Fuel elements of this general configuration were first designed for and used in the Materials Testing Reactor and thus are referred to as MTR-type fuel elements. Four of the fuel elements were fabricated with the middle plates missing, providing space for the positioning and movement of the reactor control rods.

Reactivity of the reactor core is changed by the operator moving the control rods that are suspended from fail-safe electromagnets located on the support bridge. The ionization chambers used for sensing the neutron and gamma-ray flux are suspended near the core. The control console, from which the operator can observe the reactor bay and the top structures of the reactor through a large window, is located in a small room adjacent to the reactor bay. The control console consists of typical read-out and control instrumentation.

The UMRR is housed in a small building designed and dedicated for that purpose on the east side of the campus of the University of Missouri in the city of Rolla. The nearest large cities are St. Louis and Kansas City, Missouri, at distances from the site of 161 km (100 miles) and 290 km (180 miles), respectively.

1.3.1 Reactor Building

The Reactor Building (Figure 1.1) 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 the reactor pool. The facility layout is shown in detail in Figures 1.2 and 1.3. The volume of the Reactor Building is about 1.7E+3 m^3 (6.1E+4 ft^3).

All areas 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 1.4E+3 m^3 (5E+4 ft^3).

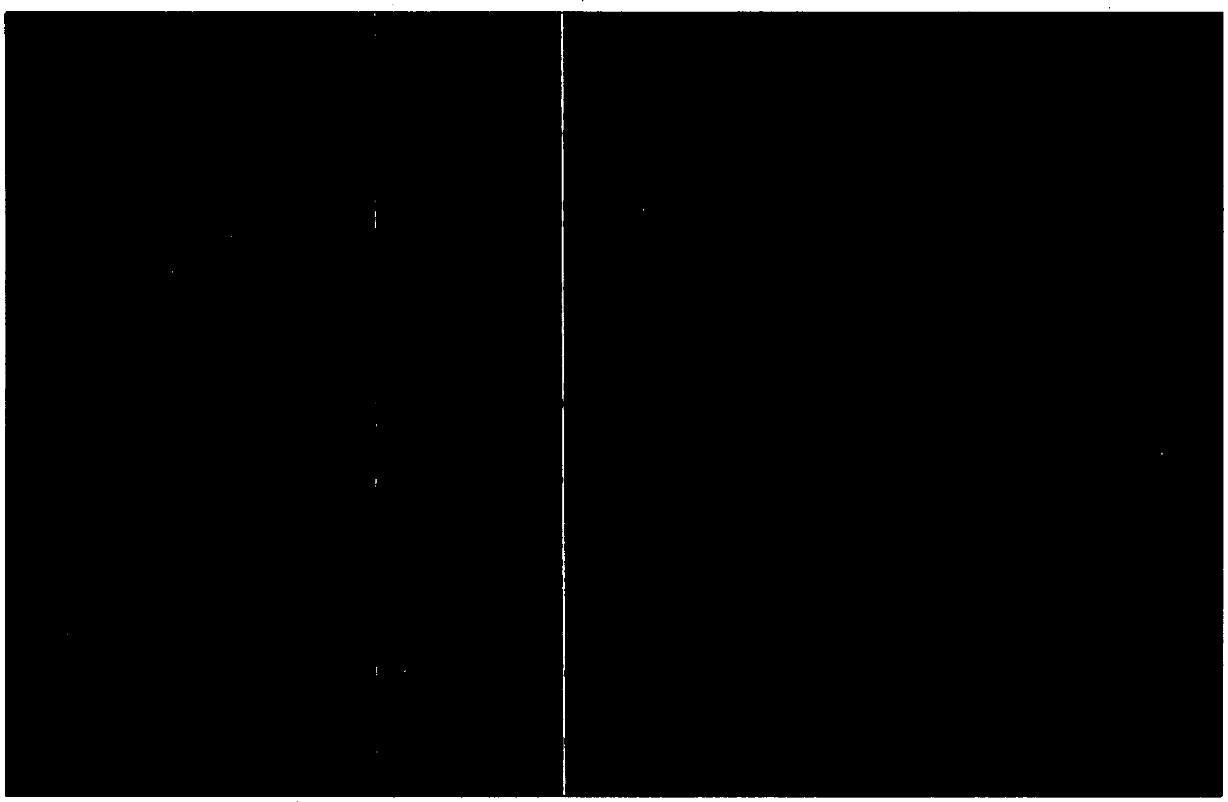


Figure 1.1-The UMR Reactor Building.

[redacted]

Figure 1.2-UMRR Upper Level and Main Floor Layouts.

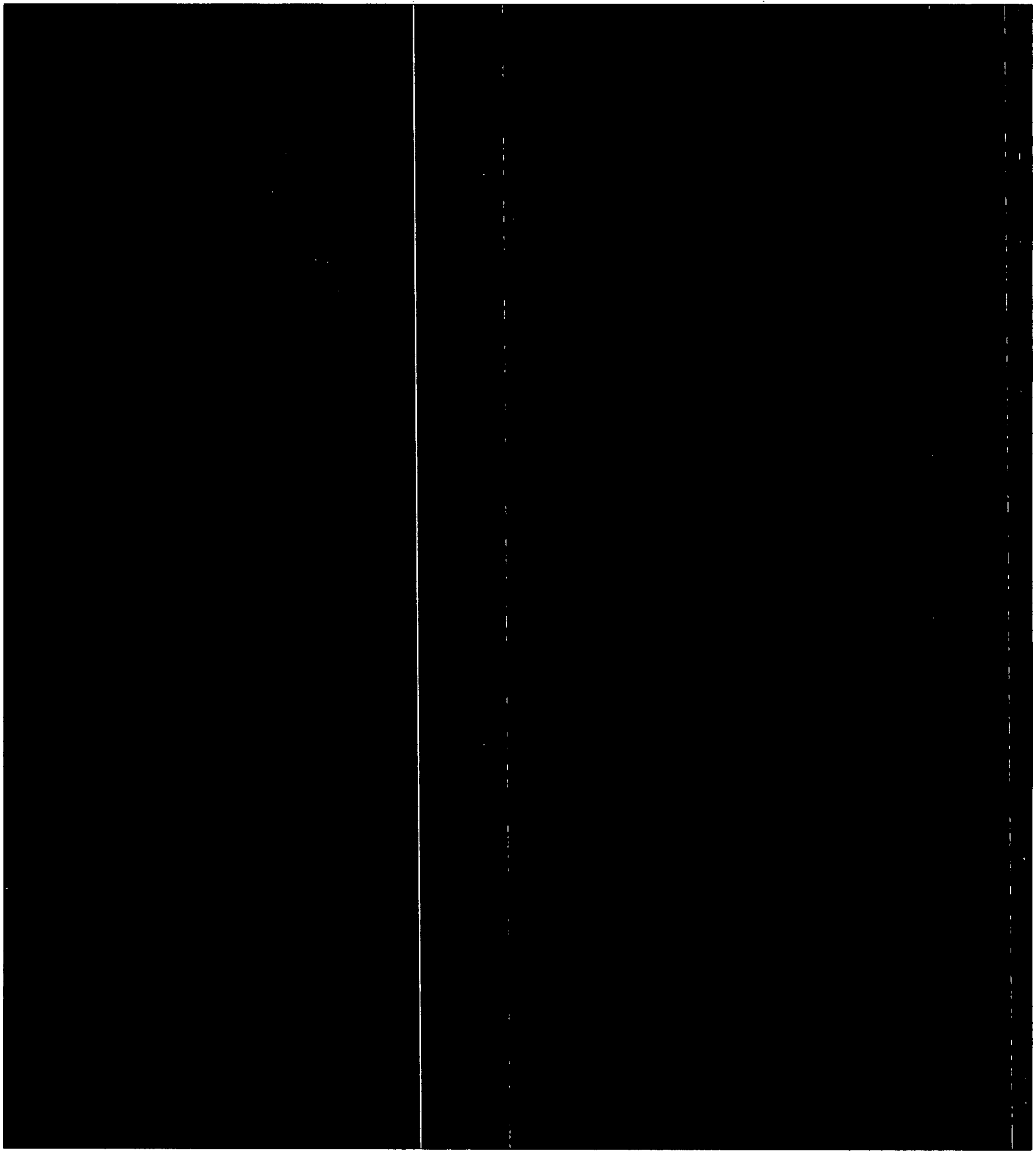


Figure 1.3-UMRR Intermediate Level and Basement Level Layouts.

1.4 Shared Facilities and Equipment and Special Location Features

The reactor building is separate from other buildings on campus, but obtains utility services such as water, electricity, and sanitary sewage from the main campus systems. There are no special features about the facility location.

1.5 Comparison with Similar Facilities

The fuel used in the UMRR is based on the MTR design and is very similar to the fuel used in other research reactors operating in the United State and in foreign countries. The control and instrumentation systems, while different in detail, are based on the same operating principles used for other research or test reactors.

1.6 Summary of Operations

The UMR Reactor Facility supports several UMR courses, provides training to students of other schools in the Midwest region of the United States, presents tours to the public and is used for research with several UMR departments. The UMRR has operated safely and efficiently over the years with no significant safety-related incidents or personnel exposures.

1.7 Compliance with the Nuclear Waste Policy Act of 1982

Section 302(b) (1) (B) of the Nuclear Waste Policy Act of 1982 provides that the NRC may require, as a precondition to issuing or renewing an operating license for a research or test reactor, that the applicant shall have entered into an agreement with the DOE for the disposal of high-level radioactive waste and spent nuclear fuel. DOE (R.L. Morgan) has informed the NRC (H. Denton) by letter dated May 3, 1983, that it has determined that university and other government agencies operating non-power reactors have entered into contracts with DOE that provide that DOE retain title to the fuel and is obligated to take the spent fuel and/or high-level waste for storage or reprocessing.

Because the University of Missouri-Rolla has entered into such a contract with DOE, the applicable requirements of the Waste Policy Act of 1982 have been satisfied.

1.8 Facility Modifications and History

In December 1961, the UMRR began to operate with a licensed power of 10 kW_t. In 1967, an amendment was granted to increase the maximum power to 200 kW_t. In 1992, there was a conversion from HEU to LEU.

UMRR has not undergone significant or safety-related physical or operational modifications since the last renewal was issued.

1.9 References

[1-1] Preliminary Hazards Evaluation, The Curators of the University of Missouri, School of Mines and Metallurgy, 10 kW Training Reactor, Rolla, Missouri, December 26, 1958.

[1-2] Hazards Summary Report for the University of Missouri at Rolla Nuclear Reactor, November 1, 1965.

[1-3] Safety Analysis Report for the University of Missouri-Rolla Reactor. Rolla, Missouri. September 27, 1984.

2. Characteristics

Demographic and natural factors pertaining to the site of the University of Missouri – Rolla Reactor (UMRR) are discussed in this chapter including those characteristics used in design and analysis discussions presented in other sections. Much of the information is based on the Preliminary Hazards Evaluation [2-1] and Hazards Summary Report [2-2].

2.1 Geography and Demography

2.1.1 Site Location and Description

The reactor is located on the east side of the campus of the University of Missouri-Rolla (UMR) at a latitude of [redacted] and longitude of [redacted] [2-3]. The UMR campus is located in Rolla, Missouri, which is located in Phelps County. Figure 2.1 presents a general Missouri State map. Rolla is located about 161 km (100 mi) southwest of St. Louis, Missouri, and about 290 km (180 mi) southeast of Kansas City, Missouri. Figure 2.2 shows the region within an approximate 8 km (5 mi) radius of the reactor with concentric circles centered on the reactor having radii: 1 km (0.62 mi), 2 km (1.24 mi), 4 km (2.48 mi), 6 km (3.73 mi), 8 km (5 mi), and 40 km (25 mi). Figure 2.3 shows the location of the Reactor Facility on a more detailed map of the area within about a one-mile radius of the facility. Figure 2.3 has concentric circles centered on the reactor having radii: 100 m (328 ft), 1 km (0.62 mi), and 2 km (1.24 mi).

The general terrain 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 Rolla, especially near the town of Rosati. The land surface is too rocky and sloped in most areas for intensive agricultural practice.

2.1.2 Population Distribution

Rolla has a population of about 14,090 [2-4]. Typical enrollment at UMR is about 5,000 students. The university employs about 2,000 Faculty and Staff members. The university personnel, including students and staff, total about 7,000.

Fort Leonard Wood, located about 40 km (25 miles) southwest of Rolla, has 15,863, primarily military personnel. The town near Fort Leonard Wood, Waynesville, has a population of about 3,207. Table 2.1 presents population data from Phelps County and for surrounding counties. Figure 2.4 presents a county map of Missouri. Population centers within 40 km (25 miles) of Rolla, with distance and direction from Rolla, are tabulated in Table 2.2.

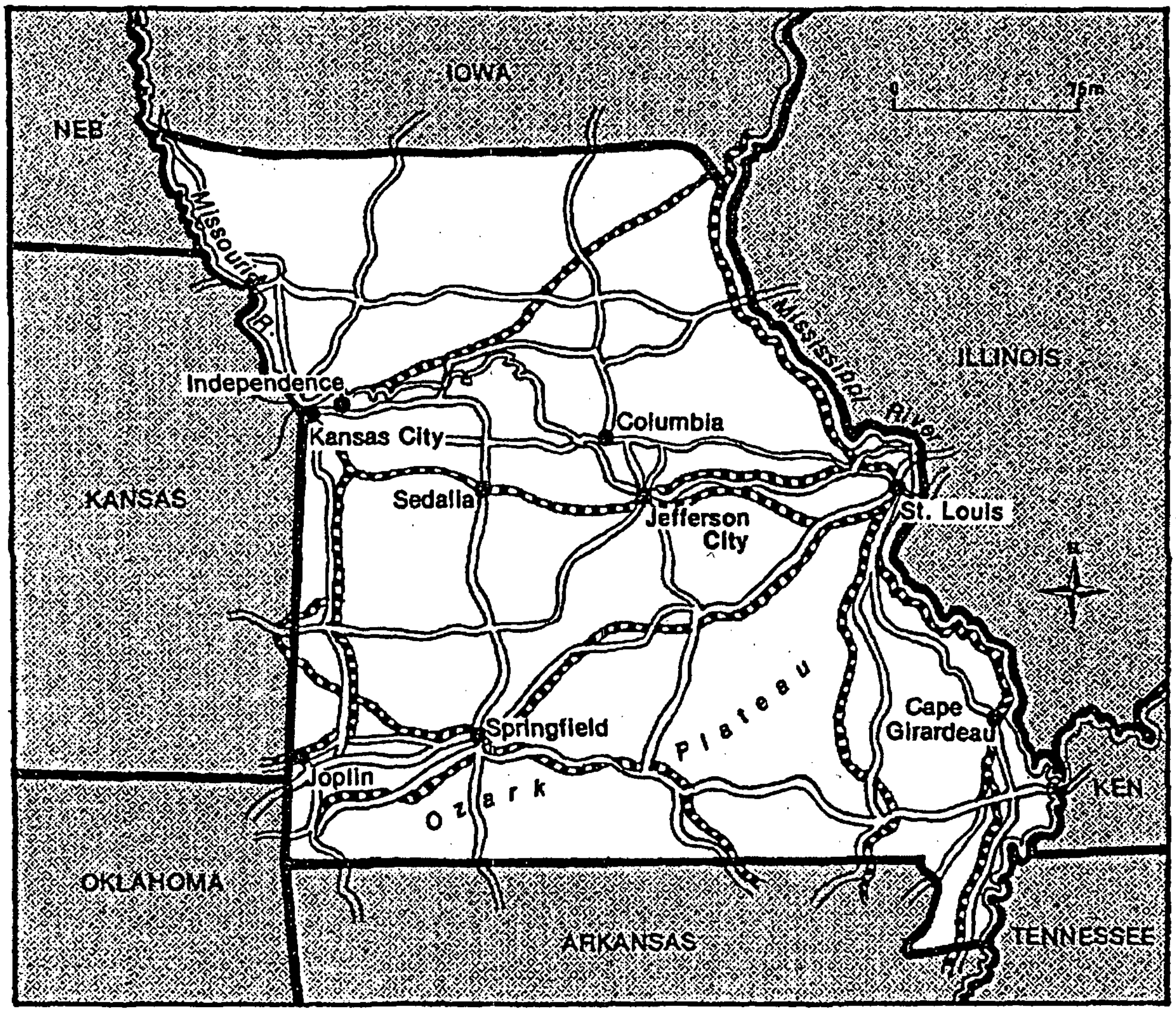


Figure 2.1-Map of the State of Missouri.

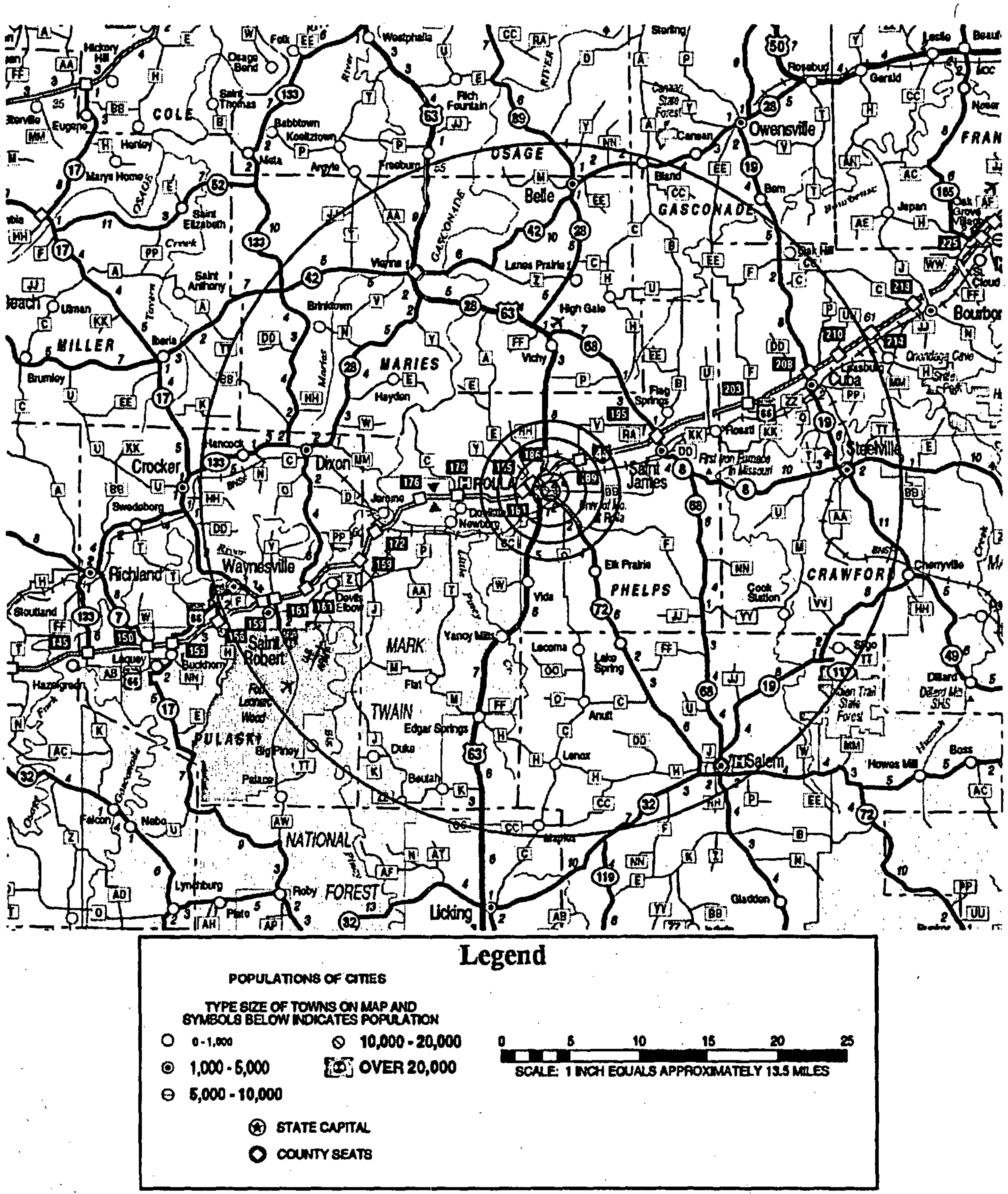


Figure 2.2-Rolla Area within 8 km of Facility.

Figure 2.3-Detailed Map of Rolla, Missouri.

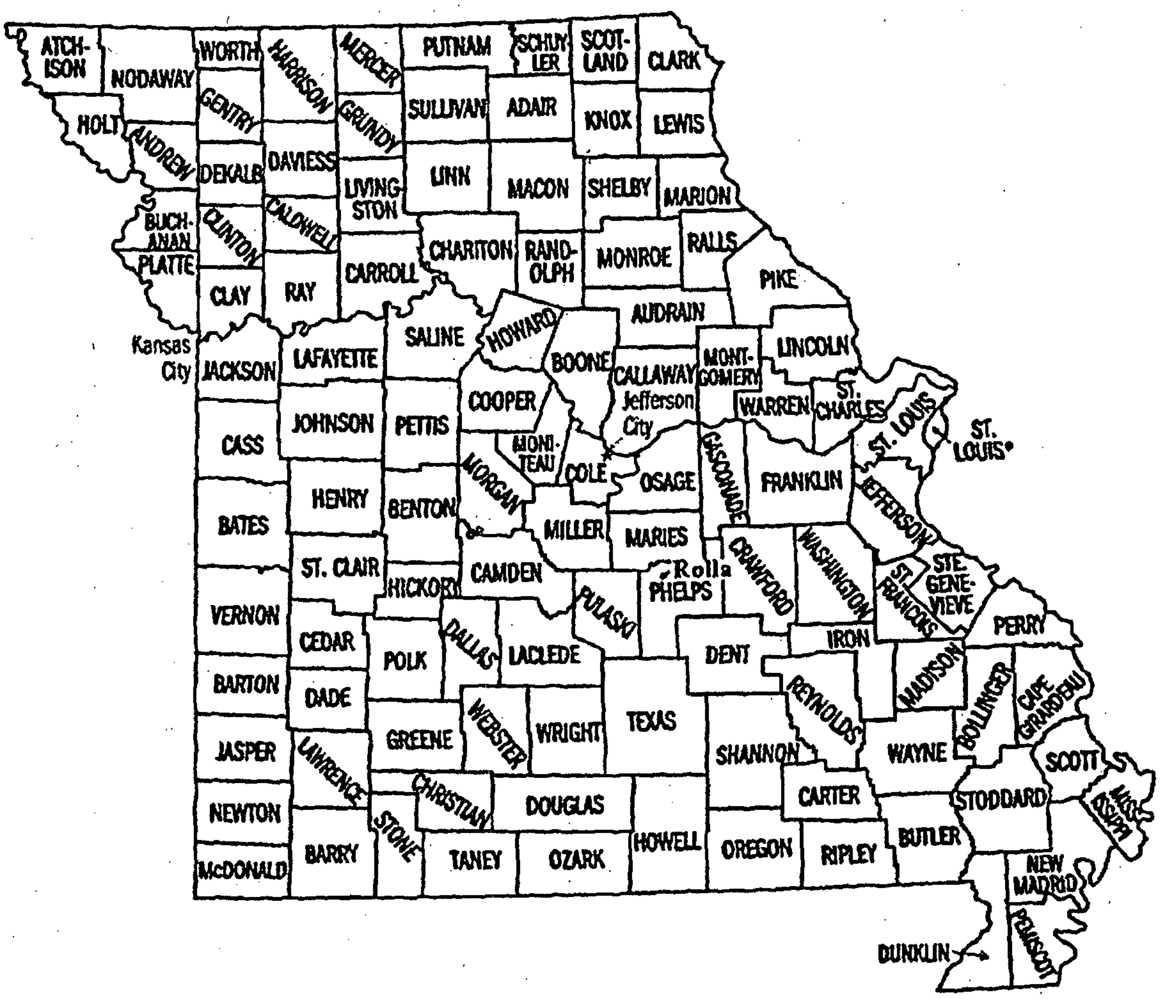


Figure 2.4-County Map of Missouri.

Table 2.1-Populations for Phelps and Surrounding Counties.

County	Population [4]	Population Density (per mi^2)
Crawford	19,845	27
Phelps	36,123	54
Dent	13,640	18
Texas	21,714	18
Pulaski	43,693	80
Maries	8,153	15
Gasconade	14,144	27

Table 2.2-Population Centers within 40 km (25 mi) of Rolla.

Town	Population [4]	Distance from Rolla		Direction from Rolla
		(km)	(mi)	
Fort Leonard Wood	15,863	40	25	Southwest
Salem	4,486	40	25	Southeast
St. James	3,256	16	10	East Northeast
Waynesville	3,207	40	25	Southwest
Cuba	2,537	32	20	East Northeast
Steelville	<2,500	35	21	East
Newburg	<2,500	16	10	Southwest
Dixon	<2,500	32	20	West
Belle	<2,500	37	23	North
Bland	<2,500	40	25	North
Vienna	<2,500	29	18	Northwest
Vichy	<2,500	22	13	North

Table 2.3-Population Distribution Estimates within 8 km (5 mi) of Rolla.	
Location	**Population**
Within 1 km (0.62 mi) radius	10,568
Between 1 km (0.62 mi) radius and 2 km (1.24 mi) radius	3,634
Between 2 km (1.24 mi) radius and 4 km (2.48 mi) radius	594
Between 4 km (2.48 mi) radius and 6 km (3.73 mi) radius	989
Between 6 km (3.73 mi) radius and 8 km (5.0 mi) radius	1,385
Total population within 8 km (5.0 mi) radius	17,170

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.1 Locations and Routes

There are three major manufacturers with 8 km (5 mi) of the facility. Table 2.4 shows the company name, a brief description of what each company produces, its distance from the reactor, and which direction it is.

Table 2.4-Major Manufacturers.				
Company	**Output**	**Distance from Reactor**		**Direction**
		(km)	**(mi)**	
Briggs & Stratton Corp.	Motors	5.3	3.4	Northeast
Can Tex Inc.	PVC Pipe	3.4	2.2	Northeast
Pet Products Plus Inc.	Pet Foods	1.9	1.2	Southwest

There are also several rock quarries within 8 km (5 mi) of the facility. Table 2.5 shows the distance and direction to the quarries in the area.

There are no major water routes within 8 km (5 mi) of the facility, but there are three land transportation routes and one airport. Interstate 44, at closest approach, is 0.6 km (0.37 miles) to the Northwest of the facility. US-63, at closest approach, is 0.4 km (0.25 miles) to the northwest. The Burlington Northern Railroad runs 0.4 km (0.25 miles) to the east.

Table 2-5. Rock Quarries within 8 km (5 mi) of Facility.		
Distance from Reactor (km)	(mi)	Direction
4.2	2.6	East Northeast
4.3	2.7	West Northwest
4.8	3	Southwest
5.6	3.5	South
5.8	3.7	North Northwest
6	3.8	West Northwest
6.2	3.9	North

2.2.2 Air Traffic

Rolla National Airport in Vichy is located approximately 22 km (13 miles) north of the city of Rolla. It has the ability to land planes as large as DC-9 jets, and can have as many as 13 or 14 large planes in the ground. It has mostly single and twin-engine planes lading and taking off. Most planes fly over in the commercial zone. Air traffic is heaviest around holidays, when people are more prone to fly their private planes.

The airport is Class E airspace. There is no control tower, but pilots call in when they are about 30 minutes away to announce their distance, direction, speed, and altitude.

None of the industries, transportation routes, or other facilities described above poses a threat to UMRR.

2.3 Meteorology

Weather observations taken in Vienna, MO cover the period of 1961 to 1990. Temperature, rainfall, and snowfall data were extracted from these records. Direction and wind speeds were not available for the reactor site itself; however, records were obtained from the Civil Aeronautics Administration (CAA) station at Vichy, MO, 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 (100 ft), same as that of Rolla. The data collected at Vichy appears to be adequate for use at the Rolla site.

2.3.1 General and Local Climate

The general climate of Missouri is a continental Midwestern type not influenced by any local mountains or large bodies of water. 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 a high of 43.3°C (110°F) in 1980 to a low of -32.8°C (-27°F) in 1977 with a mean annual temperature of 12.5°C (54.5°F).

Climatologically observations for Vienna, Mo. were examined for the years 1961 to 1990. Average annual precipitation for this period was 105.1 cm (41.4 in) per year. The period with the most precipitation is generally April through August, and the least amounts are recorded in January and February. Average monthly precipitation ranges from 4.24 cm (1.67 in) at minimum periods to around 12.70 cm (5.00 in). Table 2.6 shows the average number of days with precipitation equal to or greater than certain specified amounts.

Table 2.6-Average Number of Days of Precipitation per Year.

Inches	Jan	Feb	Mar	Apr	May	June	July	Aug	Sept	Oct	Nov	Dec
0.01	7.1	6.8	9.4	9.7	10.4	9.3	7.8	7.6	7.8	7.6	8.2	7.6
0.5	1.1	1.4	2.4	2.7	3.3	3.1	2.5	2.8	2.9	2.7	2.7	1.7
1	0.5	0.4	0.9	1.1	1.3	1.4	1.2	1.3	1.3	1.1	1.1	0.9

From the table above, it can be seen that precipitation amounts equal to or greater than one one-hundredth of an inch will occur about 27% of the days in a year. Heavy amounts exceeding 1.27 cm (0.5 in) are less frequent. It should be noted, however, that precipitation is extremely variable. The central Missouri area including the town of Rolla can be subjected 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 thunderstorms over the area.

The annual average snowfall is about 46.5 cm (18.3 in). Heavy snowfalls are uncommon. The maximum snowfall recorded during any month was 63.75 cm (25.1 in) in January 1979. Table 2.7 shows monthly and annual low, mean, and high precipitations and snowfalls for the period of record.

Table 2.7-Low, Mean, and High Precipitation and Snowfall, in Inches.													
Type	Jan	Feb	Mar	Apr	May	June	July	Aug	Sep	Oct	Nov	Dec	Annual
Low Prec.	0	0.2	1.44	1.17	1.86	0.03	0.28	1.03	0.76	1	0.18	0.35	28.03
Mean Prec.	1.67	2.08	3.73	3.79	5	4.18	3.17	3.62	3.87	3.78	3.51	2.97	41.37
High Prec.	6.5	4.7	8.14	13.9	10.4	11.3	10.7	9.36	12.1	10.6	9.72	11.1	63.06
Mean Snow	5.6	4.5	3.2	0.4	0	0	0	0	0	0	1.5	3.8	18.3
High Snow	25.1	11.5	12.5	5	0	0	0	0	0	0	8.5	19.5	38.1

2.3.2 Site Meteorology

Climatic wind data for the United States was summarized for the period of 1930 to 1996 by the National Climatic Data Center. Mean wind speeds, prevailing wind directions, and peak gusts are shown in Table 2.8 for four Missouri cities, amongst which Rolla is centrally located.

Table 2.8-Wind Direction, Speed, and Peak Gust for Missouri.			
City	Prevailing Wind Direction	Mean Wind Speed (m.p.h.)	Peak Wind Gust (m.p.h.)
Columbia	S	10	95
Kansas City	S	11	75
St. Louis	WNW	10	72
Springfield	SSE	11	72

Hourly wind observations for a 6 year period, 1948 to 1954, for the CAA Vichy Station were studied in detail. Table 2.9 presents the percentage frequency of wind directions and average velocity for the period 1948 to 1954, inclusive. It is evident that there is little variation of the most frequent winds from day to night, during periods of precipitation, and also then 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 variation shows that his 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.

Table 2.9-Annual Frequency of Wind Directions (Percent) and Mean Speed.						
Direction Wind Speed 6.4 km/hr (4 mph)	Daylight (0700-1700 EST)	Night (1800-0600 EST)	During Precipitation	During Low Visibility	Mean Wind Speed (km/hr)	(mph)
N	3.8	3.3	7.3	6.7	13.0	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
Calm	3.6	4.5	3.4	6.1	4.8	3.0
Total Mean Wind Speed					15.9	9.9

Figure 2.5 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 potential 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 when 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 2.10 illustrates the annual percentage frequency of various wind

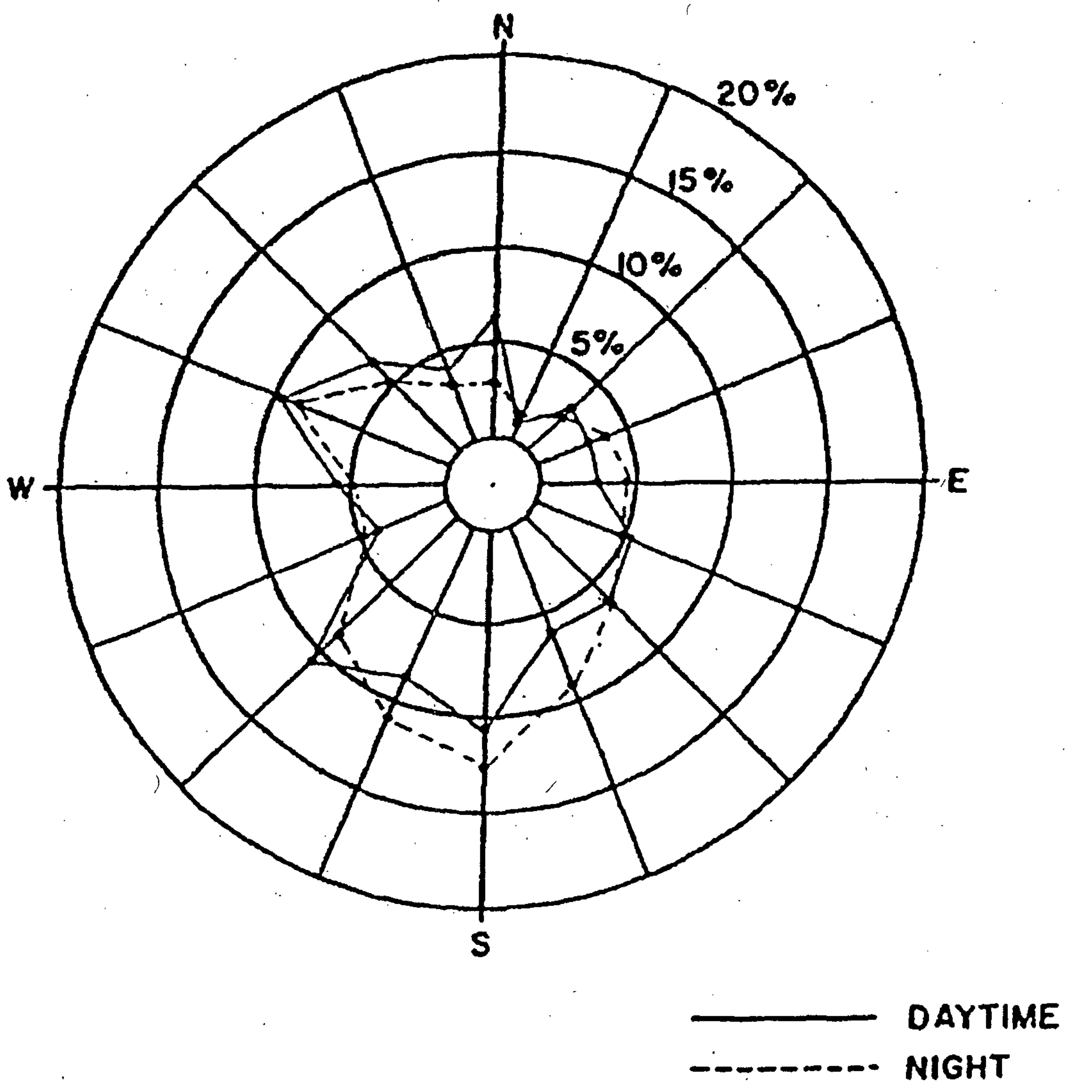


Figure 2.5-Annual Frequency and Wind Direction.

speed classes for three Missouri cities, amongst which Rolla is centrally located, for the period of 1951 to 1960

Table 2.10-Annual Frequency of Wind Speeds (Percent).

City	0-3 mph	4-7 mph	8-12 mph	13-18 mph	19-24 mph	25-31 mph	32-38 mph	39-46 mph	> 47 mph
Kansas City	9	29	35	23	5	1	< 0.5	< 0.5	< 0.5
St. Louis	10	29	36	21	3	< 0.5	< 0.5	< 0.5	< 0.5
Springfield	4	13	34	32	13	1	< 0.5	< 0.5	< 0.5

It is noted that by far the largest proportion of the winds are between 8 mph and 12 mph, averaging over 34% in all cities. Other frequent occurrences are in the 4-7 mph group and the 13-18 mph group.

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.

Tropical disturbances generally do not influence the weather in Rolla, and even though tornados occur frequently in some areas of the Midwestern, their frequency and intensity in the Rolla area is not high [2-5].

2.4 Hydrology

2.4.1 Ground Water

Wells furnishing water for the city of Rolla are cased for varying depths from the surface. Ground water is restricted to aquifers. In order of decreasing importance with respect to wells bottomed in them, the aquifers are the Roubidoux, Gasconade, Potosi, Jefferson City, Eminence, and Lamotte formations illustrated in Figure 2.6.

The Roubidoux sandstones and the Gasconade Formation outcrop along courses which drain the reactor site toward the east. 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.

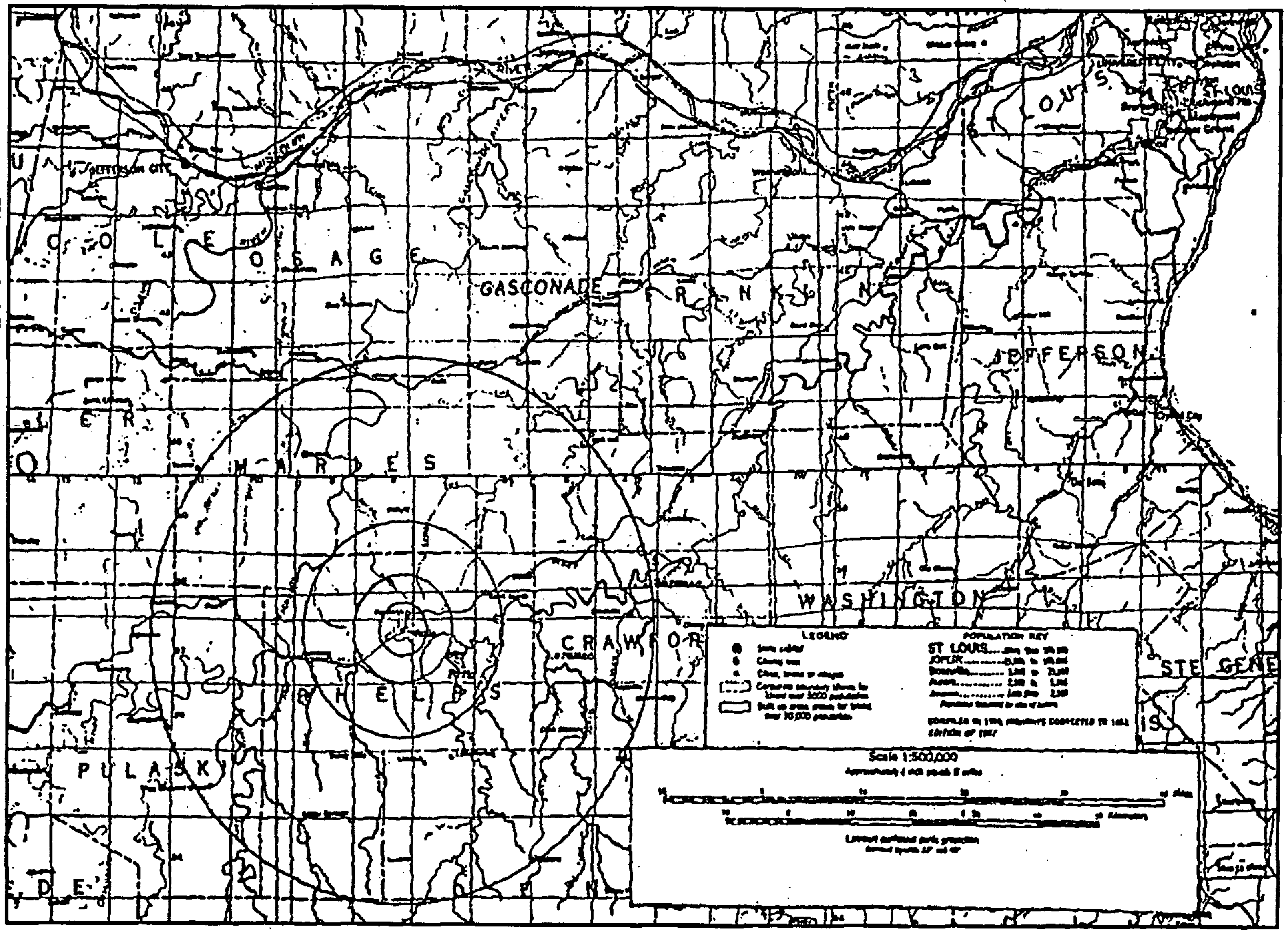


Figure 2.6-Route of Surface Drainage from Reactor Site.

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 dolomites. 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 gpm. 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 42 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. 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.

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 from 78 m (255 ft) to 88 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.

The static water levels in the Gasconade wells do not vary as greatly as the static water levels in the Roubidoux wells. Water levels range from 254 m (834 ft) to 298 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.

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 70 m (230 ft) to 87 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 gpm with 6 m (20 ft) to 40 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.

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. Water production figures for other formations are not available and part of this supply is from horizons above the Potosi Formation, but Potosi production probably is greater than production from other aquifers.

Minor water producing formations are the Jefferson City, Eminence, and Lamotte. Production from the Jefferson City Dolomite 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 provided 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 9 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 76 m (250 ft) to 152 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 gpm from it in the Rolla area.

The Elvins group and the Bonne Terre 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 79 m (260 ft). The Banterer dolomite is a non-cherty and is about 79 m (260 ft) thick.

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 2.6.

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 horizontal 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 gpm. At Eureka, records over a 10 year period indicate that the maximum flow of the Meramec River was greater than 12,000,000 gpm and the minimum flow 115,000 gpm. 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. The possibility of significant contamination of Cape Girardeau water supply from the Rolla reactor site seems very remote.

2.5 Geology, Seismology, and Geotechnical Engineering

2.5.1 Site Geology

Rolla is located toward the northern edge of the Ozark uplift. A generalized section of the rocks on the surface and subsurface is shown in Figure 2.7. The sedimentary rock section in the Rolla area averages about 518 m (1700 ft.) in total thickness. This section consists largely of Paleozoic dolomites with some sandstone and shale intervals. The Cambrian Lamotte Formation, the basal sandstone of the sedimentary rock section encountered in deep wells. The Lamotte uncomfortably overlies Precambrian metamorphic and igneous rocks. The surficial materials in the area consist primarily of residuum from cherty sandstone and dolomite [2-6].

The geographic center of the Ozark uplift lies to the southeast of Rolla. Consequently, the regional dip of the rocks in the Rolla area is toward the northwest, with a very gentle gradient of less than 1°. The regional structure, however, is not uniform as exemplified by the structure contour map of the Roubidoux - Jefferson City contact in Figure 2.8. Locally sink structures developed in the Gasconade, Roubidoux, and Jefferson City formations causing high local dips and even faulting.

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

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

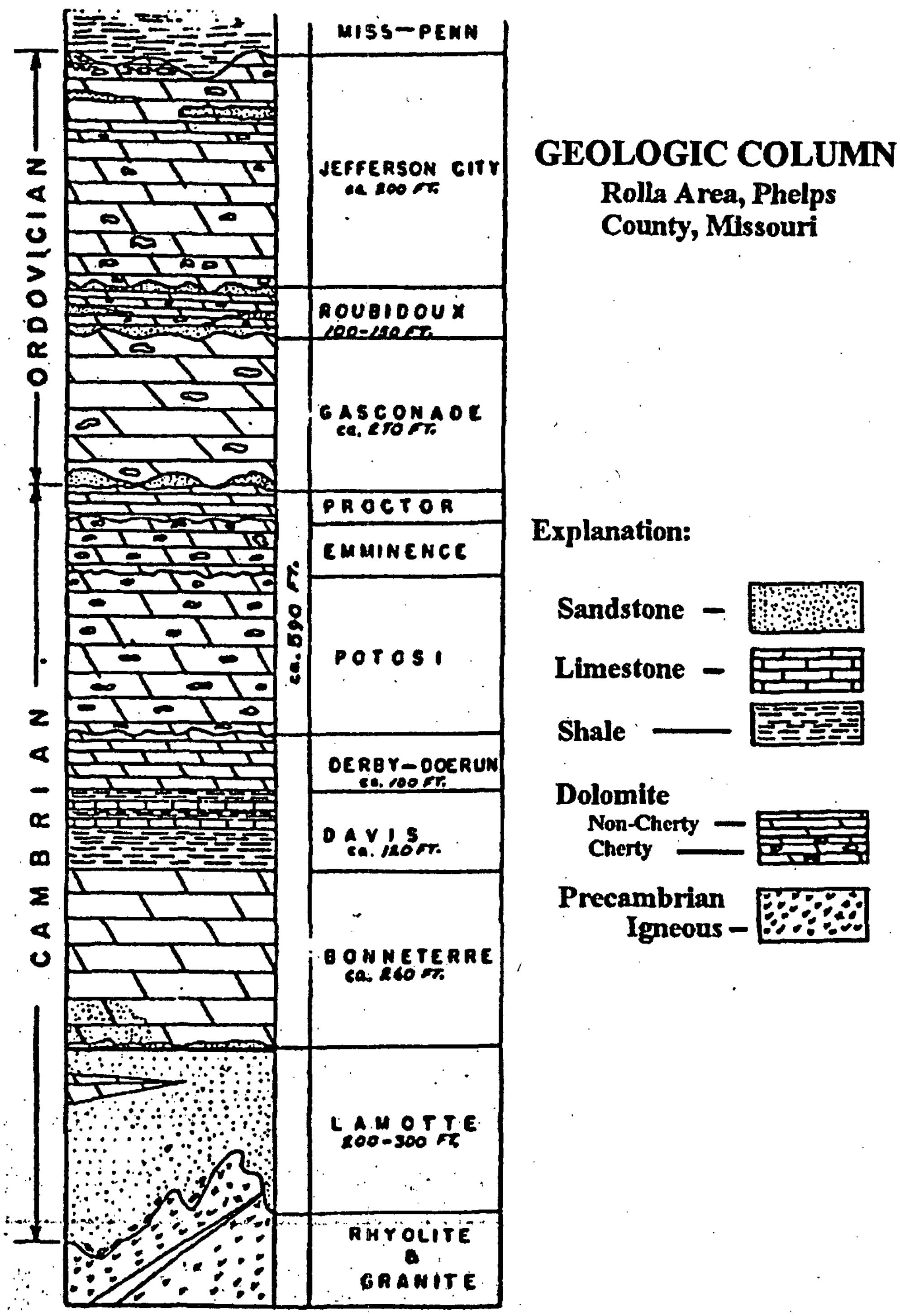


Figure 2.7-Geologic Column, Rolla Area, Phelps County, Missouri.

Roubidoux-Jefferson City Contact

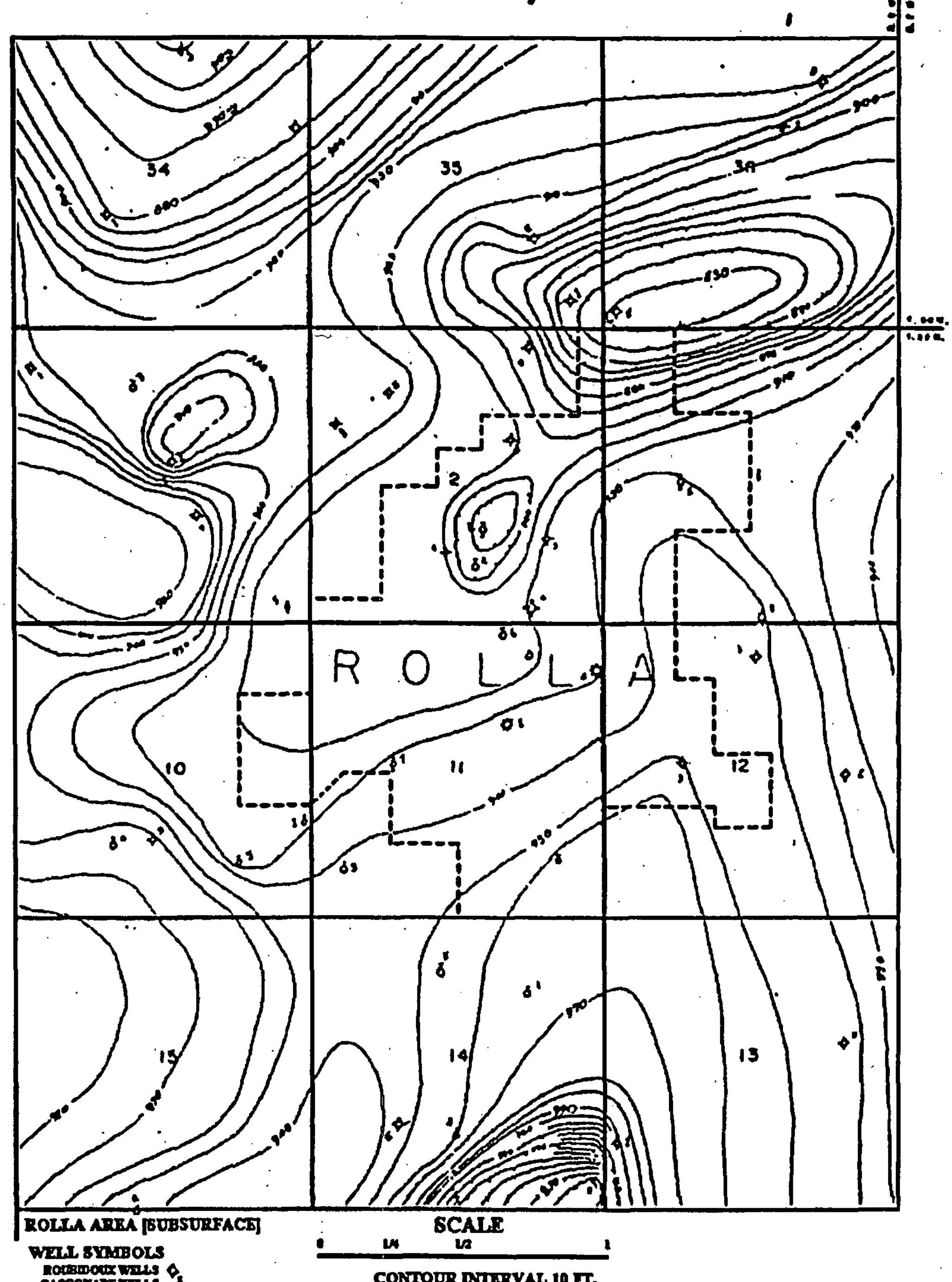


Figure 2.8-Subsurface Contour Map.

2.5.2 Seismology

Examination of 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 classes 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) that occurred over a period of more than one year with some 1,874 individual shocks being reported. The affected area included southeast Missouri, northeast Arkansas, western Kentucky and Tennessee. Visible surface effects covered an area of 50,000 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 Table 2.11 (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 in the period from 1811 to 1954, 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 have ranged from a minimum of 1 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 earthquakes activity in Missouri since 1811 are shown in Table 2.11.

From the above consideration, 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 and the intensity of any earthquakes 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.

Table 2.11-Occurrence and Intensity of Earthquake Activity in Missouri Since New Madrid Shocks of 1811-1812.			
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 MO 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 08, 1872	Western Missouri	III	
Nov 08, 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	VII	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 05, 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, SC
Oct 18, 1895	New Madrid	II	
* Oct 31, 1895	Charleston	VII-IX	Felt as far as New Mexico
Dec 02, 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	

(* violent enough to cause damage)

Table 2.11-Continued...				
	Date	Place	Intensity	Remarks
*	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, 1903	MO, IN, KY, TN	VI	Considerable damage in St. Louis
	Feb 23, 1906	Anabel	II	
	Mar 6, 1906	Hannibal	IV	
	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

(* violent enough to cause damage)

Table 2.11-Continued...			
Date	**Place**	**Intensity**	**Remarks**
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	
* 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	

(* violent enough to cause damage)

Table 2.11-Continued...			
Date	**Place**	**Intensity**	**Remarks**
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. MO

(* violent enough to cause damage)

2.6 References

[2-1] Preliminary Hazards Evaluation, The Curators of the University of Missouri, School of Mines and Metallurgy, 10 kW Training Reactor, Rolla, Missouri, December 26, 1958.

[2-2] Hazards Summary Report for the University of Missouri at Rolla Nuclear Reactor, November 1, 1965

[2-3] State of Missouri Department of Natural Resources, Topographic Map, County

[2-4] U. S. Bureau of the Censes, County and City Data Book 1994, Washington D. C. U. S. Government Printing Office, 1994

[2-5] Safety Evaluation Report related to the renewal of the operating license for the research reactor at the University of Missouri-Rolla, Docket No. 50-123, U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, December 1984.

[2-6] Missouri Department of Natural Resources Division of Geology and Land Survey, "Surficial Material Map of Missouri", PO Box 250, Rolla, Missouri, 1990.

3. DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Reactor Building

The reactor is housed in sturdy steel framed, double walled building designed to restrict leakage. Weather stripping of doors and windows and caulking of potential air leakage points limits the out-leakage of air typical in this type of construction. In addition, vents in the ventilation system automatically close in the event of shutdown of the ventilation system, providing confinement of the building air. While the ventilation system is operating normally, a negative pressure is maintained within the reactor building.

The building is essentially a rectangular structure, approximately 15 m x 10 m x 10 m high. An office/reception/entrance area was added to the building in 1980. The main floor contains the reactor bay, control room, counting rooms, and office space. The reactor building free volume is approximately 1700 cubic meters. There is a normally locked truck door at one end of the reactor bay. The basic building layout is shown in Figures 1.2 and 1.3.

A structural assessment was made of the UMR Reactor Building as a senior class term project in 1999 [3-1]. Analyses were performed in accordance with standard design specifications and building codes [3-2], [3-3], [3-4]. The assessment evaluated installation of a 5-ton overhead crane and considered live loads (wind, roof, rain, snow, and earthquake from sections 1606 through 1610 of the Building Officials and Code Administrators, BOCA, code [3-4]) and dead loads. The conclusion of the assessment was that the building is far over designed in terms of structural strength and has more than adequate excess capacity to handle a standard 5-ton crane.

3.2 Meteorological Damage

The Rolla, Missouri area experiences few extreme wind conditions such as tornadoes or inland hurricanes. Further, the reactor building is constructed of a reinforced steel frame and poured concrete floor, and the reactor pool is formed by a poured concrete biological shield that is reinforced. On this basis, wind or storm damage to the reactor facility is very unlikely. The structural assessment [3-1] evaluated live wind loads up to 49.21 lbs/ft^2 as recommended by the BOCA code [3-1] and showed that the facility can easily withstand such winds.

3.3 Water Damage

The reactor site is situated on gently sloping terrain, but well above any flood plain. Significant damage to the reactor because of flooding is not likely.

3.4 Seismic Damage

The information on past seismic activity in the area of Rolla, Missouri, indicates that the UMRR is located in a region of low probability of severe seismic activity. The structural assessment [3-1] evaluated live earthquake overturning and foundation overturning loads of 16.7 kip-ft (as recommended by the BOCA code [3-4]) and showed that the facility can easily handle such loads. In the event of an earthquake causing catastrophic damage to the reactor building and/or the reactor pool, water might be released. However, Section 13 shows that loss of coolant in the UMRR does not lead to core damage, and mechanical damage to fuel cladding would release only a small fraction of the fission product inventory. The risk of radiological hazard resulting from seismic damage to the reactor facility is not significant.

3.5 Systems and Components

The mechanical systems important to safety are the neutron-absorbing control rods suspended from the reactor superstructure. The motors, gear boxes, switches, and wiring are above the level of the water and readily accessible for visual inspection, testing, and maintenance. A preventive maintenance program has been in effect for many years at the UMRR facility to ensure that operability of the reactor systems is in conformance with the performance requirements of the Technical Specifications.

3.6 References

[3-1] "Crane Feasibility and Design Report for the University of Missouri-Rolla Nuclear Reactor Building Senior Class Term Project for Civil Engineering Course CE326, Advanced Design of Steel Structure, May, 1999.

[3-2] American Institute of Steel Construction "Manual of Steel Construction: Load and Resistance Factor Designing" Second Edition, Volume I & II, Chicago, Illinois, 1998.

[3-3] Segui, William, T. "LRFD Steel Design", Second Edition. Brooks/Cole Publishing Company, California.

[3-4] Building Officials and Code Administrators International Inc., "The BOCA National Building Code/1996", Thirteenth Edition.

4. REACTOR DESCRIPTION

4.1 Summary Description

The University of Missouri-Rolla Reactor (UMRR) is a thermal, heterogeneous, pool-type reactor licensed for a power of 200 kW_t. The reactor core is made up of MTR plate-type fuel and is fueled with low-enriched U-235. The core is cooled by natural convection of the pool water. The pool water serves as a moderating, reflecting, shielding, and heat removing medium. Some important reactor parameters of the UMRR are presented in Table 4.1.

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.

4.2 Reactor Core

The reactor core consists of fuel elements, four control rods, and if desired, in-core experimental facilities. Each core component is positioned in the grid-plate, which is supported by an inverted tower suspended from the bridge and is shown in Figure 4.1.

The reactor core is supported by an inverted aluminum tower assembly suspended from the bridge which spans the pool as shown in Figure 4.2. The bridge is made of structural steel, approximately 3.33 m (10'11") long and 1.37 m (4'6") 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 to within the pool area. An inadvertent movement of the reactor bridge causes the reactor to be scrammed.

The grid plate, depicted in Figure 4.3, is made of 12.7 cm (5 in) thick aluminum with 54 element positions arranged in a 6 x 9 array. The element holes, which have a 6.91 cm (2.42 in) diameter, pass through the grid plate to permit circulation of coolant through the core. The holes which do not hold an element are not plugged. Smaller auxiliary coolant holes, which have a 2.22 cm (0.875 in) diameter, are provided between the larger element holes to permit coolant flow between outside plates of the fueled elements in the interior of the core.

Table 4.1-General Design Characteristics of the UMRR.	
Type	Open Pool, light water
Licensed Power	200 KW
Heat Removal	Natural convection
Fuel Type	MTR - Curved Plates
Fuel Meat	
Composition	[redacted]l
Enrichment	[redacted]%
Thickness	[redacted]h
Width	[redacted]h
Height	[redacted]h
Water Gap	[redacted]h
Shim/Safety Rods	
Material	Boron Stainless Steel
Drive Speed	6 in/min
Regulating Rod	
Material	Stainless Steel (hollow)
Drive Speed	24 in/min
Reactivity Coefficients	
Temperature (Moderator)	$\approx -1.3\times10^{-4}$ Δk/k/°C
Temperature (Fuel)	$\approx -1.3\times10^{-5}$ Δk/k/°C
Void (Periphery)	$\approx -9.0\times10^{-7}$ Δk/k/°C
Prompt Neutron Lifetime	5.0×10^{-5} seconds
Effective Delayed Neutron Fraction	0.0079

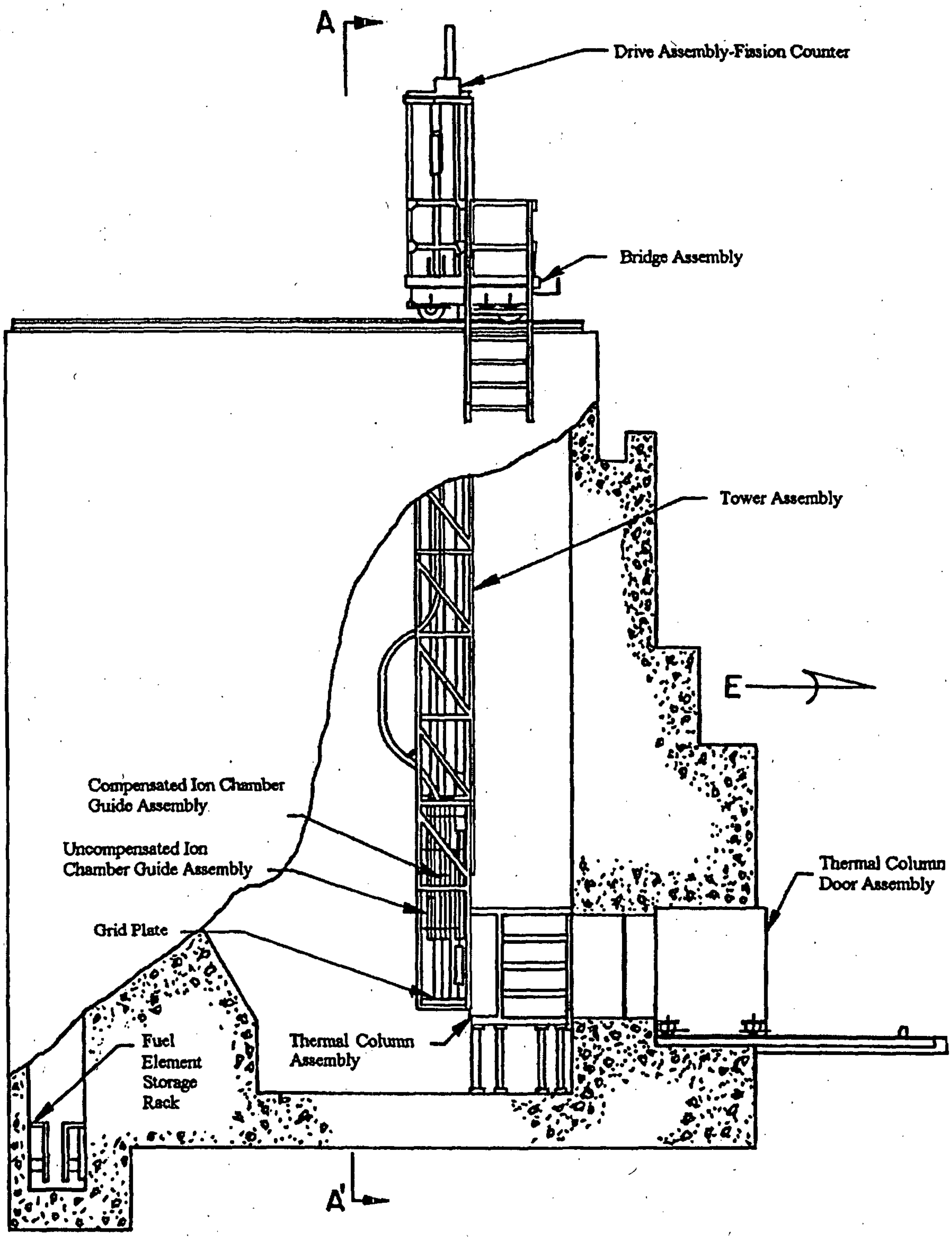


Figure 4.1-Side View of Reactor Pool Cross Section.

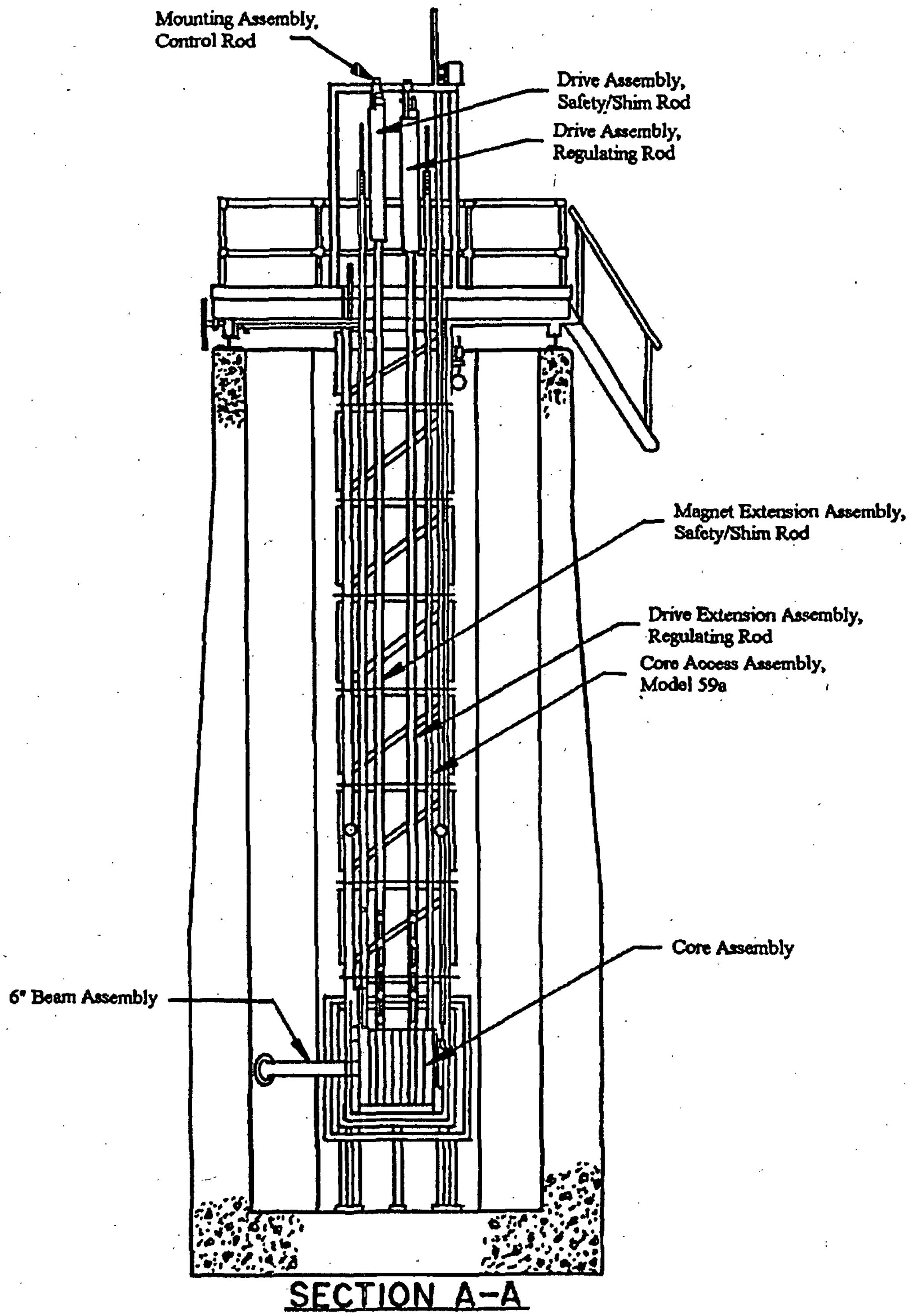


Figure 4.2-Front View of Reactor Pool Cross Section.

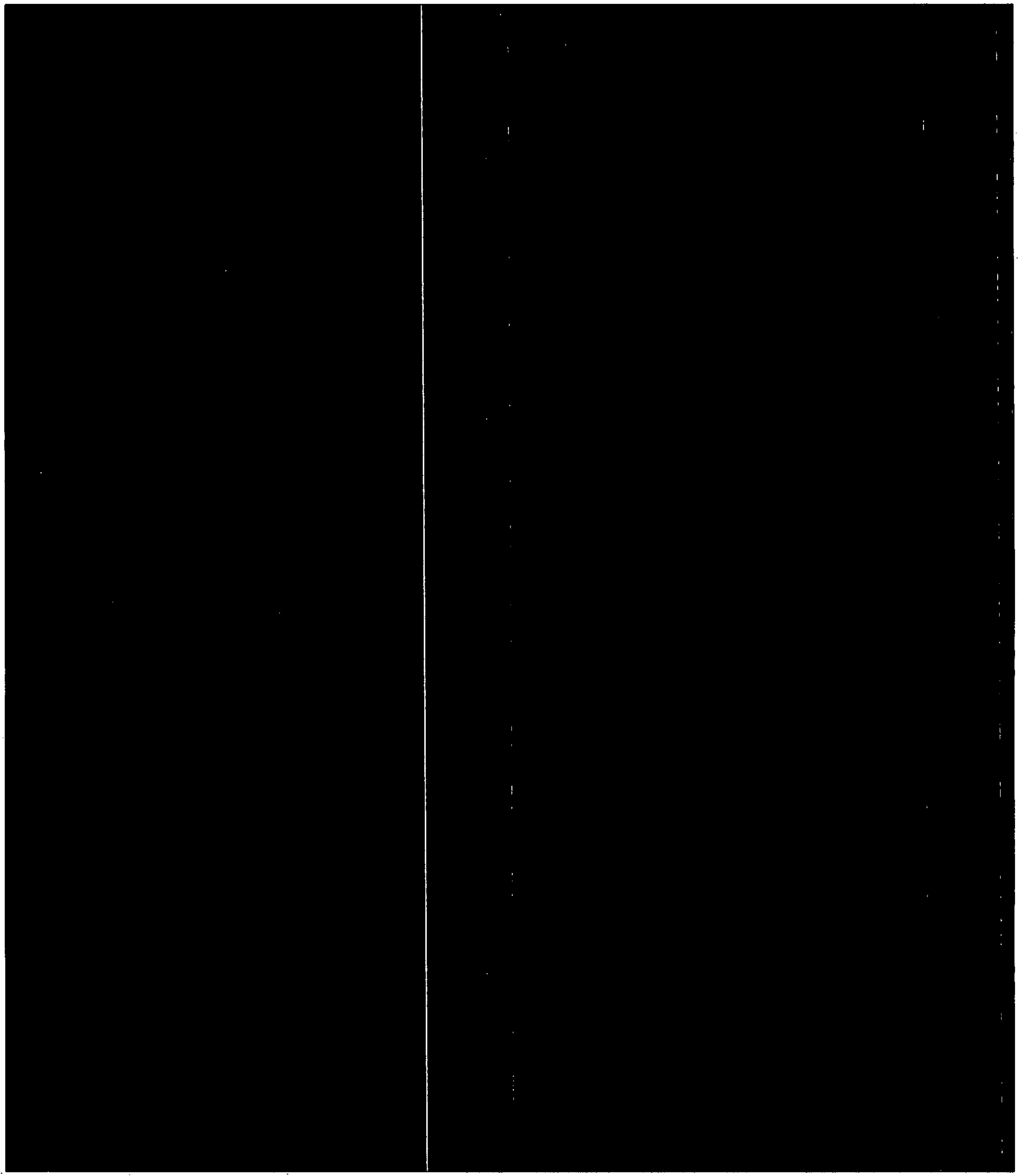


Figure 4.3-The Grid Plate.

4.2.1 Reactor Fuel Elements

4.2.1.1 Standard Fuel Elements

A standard fuel element (Figure 4.4) contains [redacted]. The [redacted]. The standard element is [redacted] tall and has an almost square cross sectional area of about [redacted] The [redacted] has a circular cross section which can be [redacted].

Fuel plates consist of [redacted]. The fuel meat dimensions are approximately [redacted] The cladding is a layer of [redacted]. The overall plate thickness is about 0[redacted]. The u[redacted]nium fuel is [redacted]. Each plate contain[redacted].

The elements and grid plate are designed so that the fuel bearing plates are basically spaced uniformly throughout the core. The water gap thickness between plates is approximately 0.315cm (0.124 inch). Both ends of the elements are open so that water may flow between the fuel plates. The outer surfaces of the elements in the core interior 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.

4.2.1.2 Half Fuel Elements

Half elements are identical to standard fuel elements except that only nine plates contain fuel. The other nine plates contain only aluminum and are referred to as "dummy" plates. Depending on the element, either the front or rear nine plates are fueled.

4.2.1.3 Control Rod Fuel Elements

A control rod fuel element (Figure 4.5) has the central eight plates removed to accommodate a guide tube for the control rod. The guide tube prevents the control rod from coming into contact with fuel plates. There are four control rod fuel elements in any core configuration.

4.2.1.4 Irradiation Fuel Element

The irradiation fuel element is used for irradiations within the reactor core. It is identical to a standard fuel element, except that the fuel plates in positions 11 through 16 have been removed and the fuel plates in the positions 10 and 17 have been replaced with aluminum dummy plates. Thus, the irradiation fuel element contains only nine fuel plates. The irradiation fuel element has a space about [redacted] to accommodate various samples for irradiation.

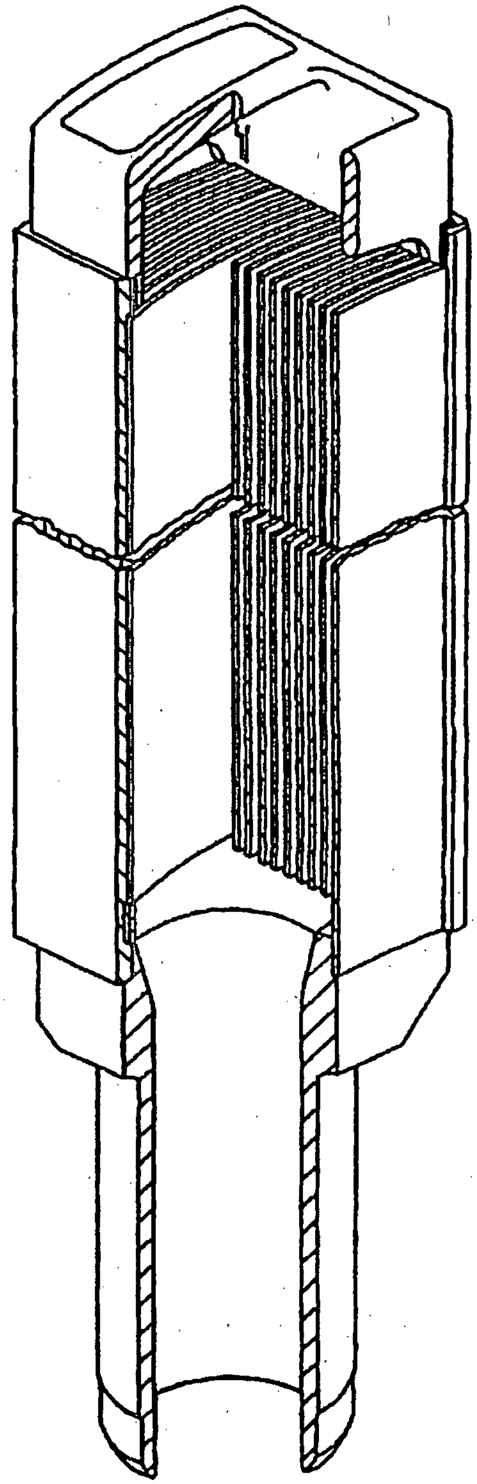


Figure 4.4-Standard Fuel Element.

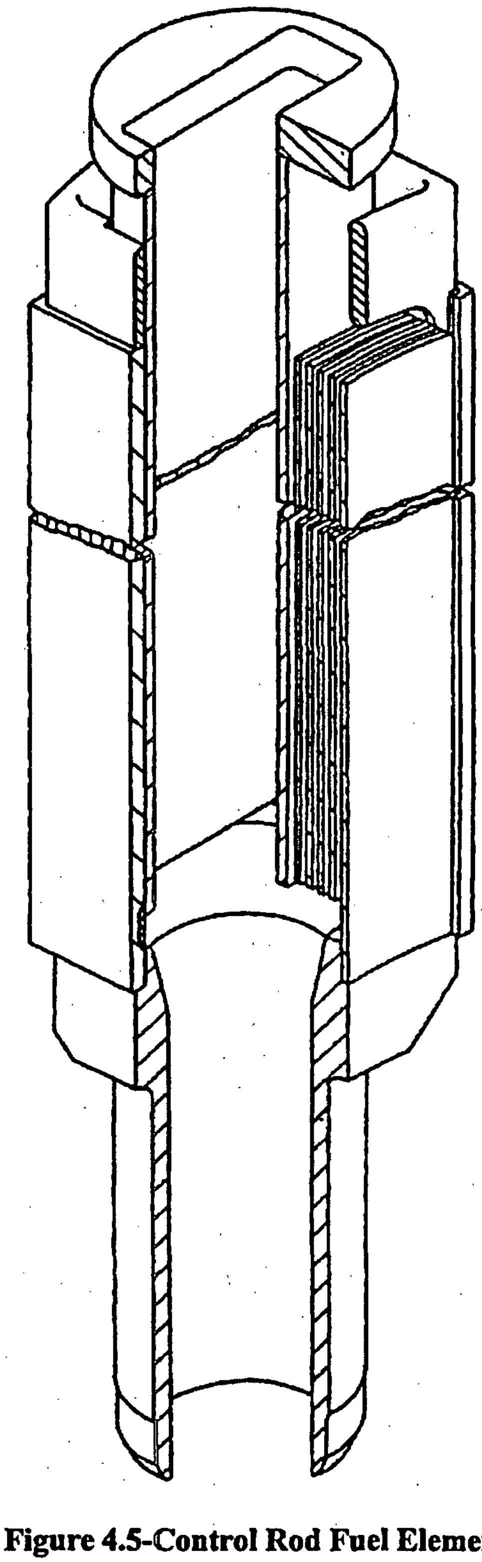


Figure 4.5-Control Rod Fuel Element.

4.2.2 Control Rods

There are four control rods loaded in any particular core configuration. Three of the control rods are shim/safety rods and the fourth is referred to as the regulating rod. Each control rod fits into a control rod fuel element specially designed to accommodate the control 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 about ± 0.25 cm (0.10 in).

Each shim/safety rod consists of a grooved, boron stainless steel rod. The boron content is about 1.5 % natural boron. The nominal dimensions are 2.23 cm (7/8 in) thick, 5.7 cm (2.25 in) wide, and 61 cm (24 in) of active poison length.

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 by gravity. 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 mounted on top of the control rod element 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.

The regulating rod is used for fine control. It consists of a Type 304 stainless steel tube with a wall thickness of 0.165cm (0.065 in), a cross-sectional shape 2.23cm (0.875 in) wide by 5.72 cm (2.25 in) long with an oval end, and an effective 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 bolted directly to the rod drive assembly. 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 mechanical arrangement of the shim/safety rod drive assembly is shown in Figure 4.6. A control rod [redacted] containing an axial hole for a control rod is inserted into the grid plate. A stop assembly approximately 4 inch in height is attached by bolts to a special flange at the top of the control element. The 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 [redacted] flange thus accommodating the top end of the control rod [redacted]. 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, accidental lifting of a control element out of the core by movement of a shim/safety rod is impossible [redacted]. In addition, a special adjustable slip clutch arrangement is incorporated between the drive motor and the linear actuator of the shim/safety rod drive to ensure 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.

4.2.3 Neutron Moderator and Reflector

Reactor pool water serves as the moderator for the reactor. Pool water also serves as the primary reflector. The reactor can be run with two different reflector modes; the "W-mode" and the "T-mode". In the "W-mode" the reactor, which is suspended from a moveable bridge, is rolled away from the thermal column. In this mode, the reactor is water reflected on all sides. In the "T-mode" the reactor bridge is rolled back such that the rear face of the core is essentially touching the thermal column. In the T-mode, the rear face of the core is reflected by the graphite thermal column. The thermal column is a superior reflector. The core excess reactivity in the T-mode is on the order of 0.4% $\Delta k/k$ higher than in the W-mode, depending on the particular core configuration.

The bridge is secured in position by two bridge clamps which may be manually tightened and loosened A bridge motion detection switch is installed which provides a reactor scram if the bridge is moved. The bridge motion scram is required by technical specifications and protects against possible reactivity swings that would be associated with changing modes during while operating.

4.2.4 Neutron Start-Up Source

The reactor used a PuBe startup source. The source is sufficiently strong such that when coupled with the shutdown core, it provides a minimum count rate of at least 2 cps on the startup channel. Technical Specifications require a startup channel count rate of at least 2 cps in order to start the reactor.

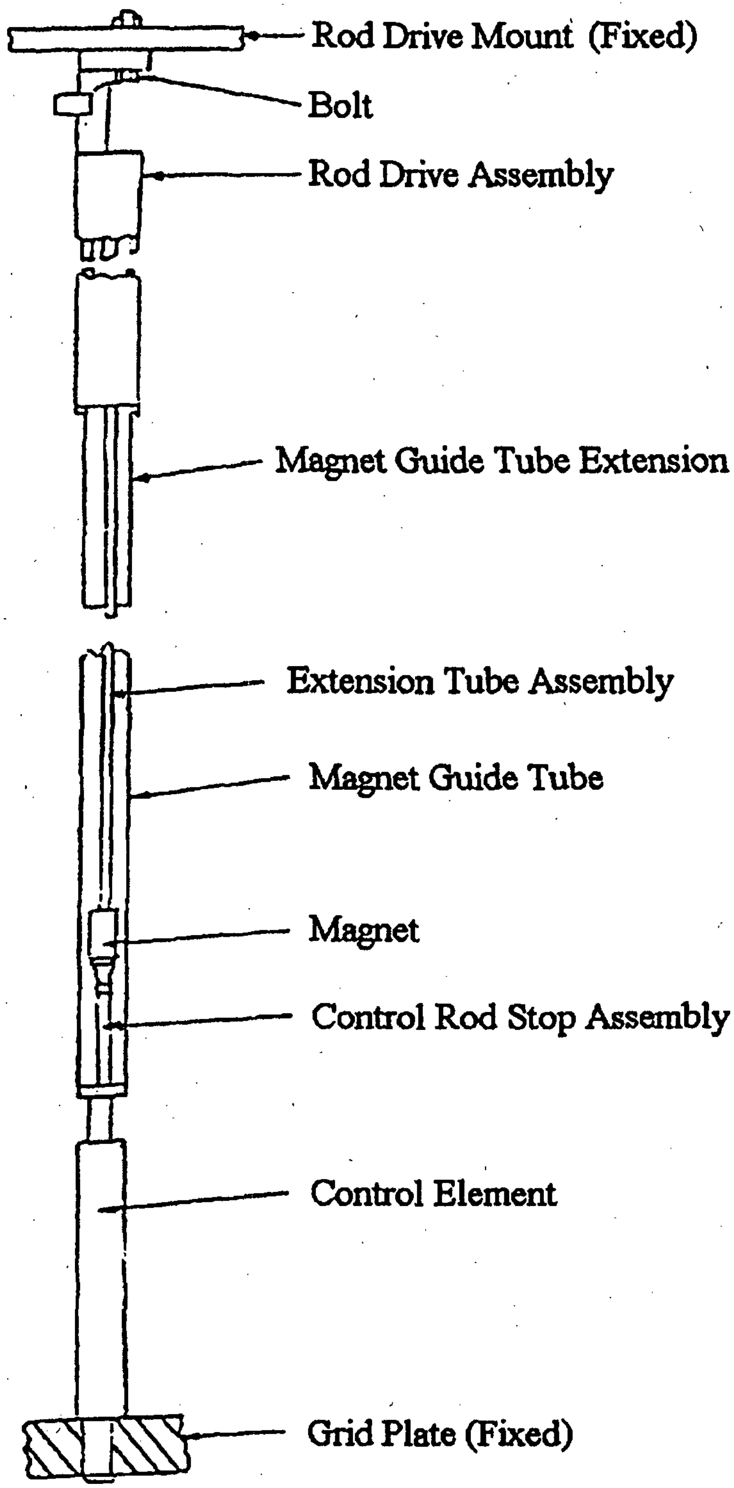


Figure 4.6-Control Rod Drive System.

4.3 Reactor Pool

The reactor pool is a rectangular approximately 5.79 m (19 ft) long, 2.74 m (9 ft) wide and 8.23 m (27 ft) deep and houses the reactor, a beam port, and a thermal column (see Figures 4.1 and 4.2). It contains about 113.56 kiloliters (30,000 gallons) of highly purified water. Pool walls are made of ordinary reinforced concrete and are 30.5 cm (12 in) at the top of the pool and taper up to 55.9 cm (22 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 4.1). The reactor pool is set in bedrock, which is resistant to earthquakes. 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.

The pool has no drains. Water is pumped to the demineralizer system through piping that is at least 4.88 m (16 ft) above the core. The demineralizer system has a siphon break, consisting of a hole drilled in the in-pool piping, which precludes the possibility of pumping or siphoning pool water below approximately 4.88 m (16 ft) above the top of the core.

At the opposite end of the pool from the thermal column is a fuel element storage pit. This is formed by a reinforced concrete bulkhead extending 4.88 m (16 ft) above and 1.07 m (3.5 ft) below the pool floor. It is placed 0.61 m (2 ft) from the main pool wall. Two fuel element storage racks are installed at the bottom of this section each capable of holding 15 fuel elements. The two racks can hold the entire inventory of LEU fuel in a criticality safe geometry with adequate heat dissipation. The effective multiplication factor was measured to be less than 0.6 for the racks when loaded with HEU fuel. The reactivity of the LEU elements has been shown to be virtually identical to the HEU elements.

If it becomes necessary to drain the reactor pool, fuel elements could be transferred to the storage rack prior to draining. The bulkhead will ensure that at least 4.88 m (16 ft) of water covers the top of the fuel elements at all times. A concrete insert between the bulkhead and the main pool floor ensures adequate shielding to personnel working in the drained pool. The fuel storage pit contains no drainage pipes.

4.4 Biological Shield

The reactor core is shielded by water in the pool. The water level is maintained such that there is normally a water layer about 6.10 m (20 ft) between the top of the core and the water surface. The next shielding barrier is provided by concrete pool walls which are about 0.56 m (22 in) thick at the base of the pool, except for the east side of the pool where the wall thickness ranges from a maximum of about 1.98 m (78 in) in the vicinity of the thermal column and the beam port to a minimum of 0.30 m (1.0 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.

A detailed radiation survey of the reactor shield wall was performed in April of 1999, the reactor at high power. Measured dose rates on the mid-level basement shield wall ranged from [redacted] to [redacted]. All measurements were taken at a distance of [redacted]) from the wall.

Measured dose rates on the lower-level basement shield wall ranged from [redacted] to [redacted]. Slightly higher dose rates were measured at the thermal column and beam port facilities, which protrude through the lower-level basement shield wall. The dose rate at the closed beam port face was [redacted]. Some minor streaming occurs around the periphery of the thermal column with gamma dose rates up to about [redacted]

A detailed radiation survey was also conducted over the reactor pool during high power operations. All measurements were taken at a height of approximately one foot above the pool surface. Measured dose rates ranged from [redacted] to [redacted].

4.5 Nuclear Design

The reactor is provided with redundant and diverse rapid-response controls and nuclear instrumentation to attain versatile and safe operation. The reactor core system is designed to have negative moderator temperature and void coefficients of reactivity. The ultimate void (total loss of coolant) removes the principal neutron moderator and shuts down the reactor. The most important factors affecting reactivity are discussed in this section. Furthermore, requirements on the excess reactivity of the reactor core are established. Heat removal is also addressed.

4.5.1 Normal Operating Conditions

The grid plate provides a six-by-nine array of locations in which fuel elements (standard, half, control, and irradiation elements), control rods, experimental facilities, and experiments may be loaded. Limitations on excess reactivity and shutdown margin serve as the primary constraints on core configuration. An additional constraint that no core may be taken critical with an open (unfilled) internal lattice position is required by Technical Specifications to prevent the possibility of an accidental insertion of a fuel element. No additional restrictions are placed on core configurations. Limitations on experiments and experimental facilities are discussed in Chapter 10.

A wide variety of core configurations are possible, and many have been used in the past. Because of the relatively low licensed power level of the UMRR, fuel burnup is not a major factor. The average annual burnup at the UMRR over the past 20 years (1984 to 2004) is less than 10 MW_t-hr per year.

Analyses presented in Section 4.6 show that the highest power density is obtained when the irradiation fuel element is placed in a central location within the core. The associated peak cladding temperature at full power has been shown below 90°C (194°F). The wide safety margin between

90°C° and the Safety Limit of 580°C° provides assurance that no other restrictions on core configurations are required, beyond the Technical Specification limits on excess reactivity and shutdown margin.

Approximate reactivity worths, dependent upon core configuration, of various components are listed below:

- Standard fuel element at core periphery (0.5% Δk/k to 1.5% Δk/k)
- Half fuel element at core periphery (0.5% Δk/k)
- Thermal Column (0.4% Δk/k)
- Bare rabbit and cadmium rabbit combined worth (0.85% Δk/k)
- Shim/safety (3% Δk/k each)
- Regulating rod (0.5% Δk/k)

As discussed previously, the reactor is controlled by three shim/safety rods and one regulating rod. Each shim/safety rod provides approximately 3% Δk/k, the exact worth varying with different core loadings. The ganged worth of three safety rods is about 9% Δk/k. The shim/safety rods have a maximum rate of withdrawal of about 15.2 cm/min (6 in/min). At the highest differential worth position, this speed corresponds to a rate of change in reactivity for any one rod of about 0.02% Δk/k per second.

The regulating rod is typically worth about 0.5% Δk/k and has a maximum withdrawal rate of about 61 cm/min (24 in/min). In its highest differential worth position, the maximum rate of change of reactivity of the regulating rod is about 0.014% Δk/k-sec.

4.5.2 Reactor Core Physics Parameters

4.5.2.1 Temperature Coefficients

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. For the UMRR, the moderator temperature coefficient has been calculated to be about -1.3 x 10-4 Δk/k °C [4-1]. Measurements performed over the temperature range of 23.9°C (75°F) to 31.1°C (88°F) confirm this value. The zero (e.g. 10 W_t) to full power (200 kW_t) reactivity swing has been measured to be about 0.15% Δk/k. This effect is due primarily to heating of the moderator as it travels through the core.

In the UMRR, with a low enriched uranium core, the overwhelming contribution to the fuel temperature coefficient is due to the Doppler broadening of the resonance capture cross sections of Uranium-238. Its value has been calculated to be -1.1E-5 Δk/k/°C [4-1]. This value is comparable with the data for other similar reactors of the pool type [4-2].

4.5.2.2 Void Coefficient

Another reactivity parameter encountered at the UMRR is the void coefficient. When the water is removed from the core or from its periphery 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 calculation shows [4-1] that the void reactivity coefficient at the reactor periphery is about 9E+7 Δk/k/ cm3. Measurements performed with the LEU core have repeatedly yielded a similar value.

4.5.2.3 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 Xe-135 because of its large thermal absorption cross section. The magnitude of xenon poisoning is proportional to the neutron flux and operational history. Because of the relatively low licensed power level, xenon poisoning is not a major reactivity effect. For example, the shutdown peak xenon poisoning after 8 hours of full power operation (as experimentally determined for the previous HEU core) was only about –2E-3 Δk/k.

4.5.2.4 Kinetics Parameters

The effective delayed neutron fraction (k_{eff}) for the UMRR has been calculated to be $k_{eff} = 0.0079$. The prompt neutron lifetime (ℓp) has been calculated to be 50 microseconds.

4.5.2.5 Operating Limits - Excess Reactivity and Shutdown Margin

Technical Specifications specify limits on excess reactivity and shutdown margin for all core configurations. A designated core configuration may include irradiation facilities such as the pneumatic rabbit facilities, the isotope production element, the core access element, and secured experiments. In such instances, these items are considered part of the core with regard to excess reactivity and shutdown margin requirements.

Excess reactivity must be built into the reactor core in order to compensate for a number of reactivity losses. Additionally, a sufficient reactivity must be available to allow for an adequate reactor period. The minimum excess reactivity, which is required to allow for an extended operational flexibility (e.g. 24 hrs), consists of the following:

0.6% Δk/k	temperature effect
0.2% Δk/k	Xe-poisoning
0.4% Δk/k	Moveable experiments
0.3% Δk/k	adequate reactor period and operational flexibility
1.5% Δk/k	Total

The maximum excess reactivity in the UMRR core is limited to 1.5% Δk/k by the Technical Specifications under normal operating conditions. To provide sufficient excess reactivity for accurate control rod total and differential worth measurements, the Technical Specifications permit an excess reactivity of 3.5% Δk/k no more than twice a year for periods not to exceed 5 working days each time. The additional reactivity is obtained by loading fuel elements at the periphery of the core. The worth of one fuel element in such a position is less than 1.5% Δk/k. The analyses of rapid reactivity insertions (Chapter 13) indicate that a step insertion of 1.5% Δk/k will not result in fuel or core damage.

The Technical Specifications prescribe a minimum reactivity shutdown margin of 1.0% Δk/k in a cold, xenon-free core with the highest worth control rod and the regulating rod fully withdrawn. The shutdown margin limitation provides adequate flexibility to load sufficient excess reactivity into the core to compensate for the effects of experiments, temperature coefficients of reactivity, and fission product poisoning, while still ensuring that the reactor can be controlled under any condition of operation even if the most reactive safety rod and regulating rod were to fail to insert.

Technical Specifications allow the excess reactivity and shutdown margin requirements to be exceeded temporarily solely for the purpose of performing excess reactivity and shutdown margin measurements. In such instances, the reactor power is limited to $\leq 2\ kW_t$. In the event that the measurements indicate either excess reactivity or shutdown margin requirements are exceeded, the reactor will be shutdown and core modified as appropriate.

4.6 Thermal - Hydraulic Design

Heat generated in the reactor core is removed by natural convection and is dissipated in the reactor pool water. The major mechanism for heat removal from the pool is through evaporation.

The coolant outlet temperature in one of the fuel elements was experimentally determined for the HEU core. Using this value, the coolant velocity calculated from a heat balance was about 0.1 m/s (0.33 fps). Since the LEU core is very similar to the HEU core and the licensed power is the same, the coolant velocity will not appreciably change in the LEU core. 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 at the flow velocities above 10 m/s (32.8 fps) [4-3]. Additionally, the LEU fuel has been used since 1992 with no problems. 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 about 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 570°C (1350°F). 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.

A computational analysis [4-1] performed with a 2-D diffusion code [4-4] investigated different core configurations and identified a maximum total power peaking factor of 2.4 for a core that has the irradiation fuel element located in a central region of the core, adjacent to two control rod elements. This value is only slightly higher than the power peaking factor of 2.2 calculated for a control rod fuel element in a core without the irradiation fuel element. Depending on the number of fuel elements in the core, the average heat flux at 200 kW_t is approximately 0.9 W/cm^2. Using the maximum power peaking factor of 2.4 the corresponding maximum heat flux in the irradiation fuel element is 2.2 W/cm^2. (For the previous HEU core, the measured peaking factor in a central fuel element was about 2).

Heat transfer calculations have shown that the temperature drop across the fuel meat and the aluminum cladding is negligible. On the contrary, the temperature drop between the cladding and the bulk coolant is relatively large. It is this temperature drop which largely determines the central temperature of the fuel.

The pertinent literature was researched for heat transfer correlations 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 4.7 for the more conservative Uniform Heat Flux correlation. Here, the wall temperature of the fuel cladding is shown as a function of the bulk pool temperature, for different values of the cladding heat flux. The peak heat flux of 2.2 W/cm^2 has an associated peak cladding temperature below 90°C (194°F) (assuming a pool temperature of 57°C (134.6°F)). The peak cladding temperature associated with the Limiting Safety System Setting of 300 kW_t would be below 100°C (212°F). The margin of safety between the peak cladding temperature and the Safety Limit of 580°C is dramatic.

The results of the analysis also temperature below 90°C (194°F) (assuming a pool temperature of 57°C (134.6°F)). The peak cladding temperature associated with the Limiting Safety System Setting of 300 kW_t would be low 100°C (212°F). The margin of safety between the peak cladding temperature and the Safety Limit of 580°C (1076°F) is dramatic. The results of the analysis also demonstrate that no additional restrictions, beyond excess reactivity and shutdown margin requirements, are needed for core configurations within the grid-plate.

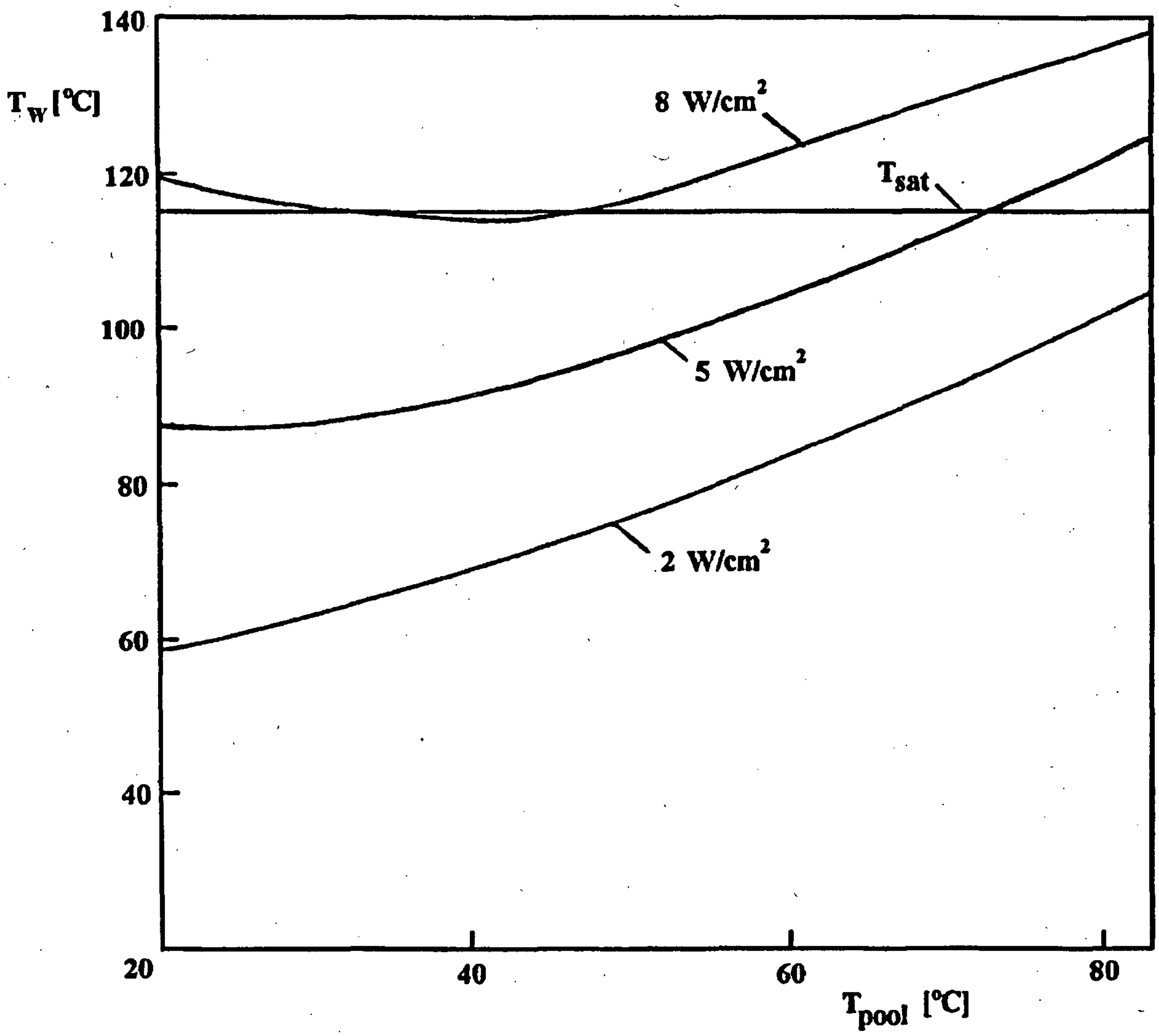


Figure 4.7-Calculated Cladding Wall Temperature using a Uniform Heat Flux Correlation.

In the case of a failed movable experiment, as discussed in Chapter 13, the heat flux in the hot channel could briefly reach the value of about 5 W/cm^2. Then, the corresponding cladding temperature could reach about 103°C (217.4°F), which is well below the saturation temperature (T_{sat}) of 115°C (239°F). The region above the saturation temperature is known as the region of nucleate boiling heat transfer. It has been shown in the pool boiling experiments [4-5] that in this region, while the heat flux can be in the order of tens of W/cm^2, the cladding temperature remains just about 20-30 K above the boiling temperature. For example, the peak heat flux of 50 W/cm^2, which would correspond to the reactor power of about 4.5 MW$_t$, would cause the cladding wall temperature to rise to about 140°C (284°F) which is well below the melting point of the aluminum cladding (580°C (1076°F)). Therefore, the capability of heat removal in the region of nucleate boiling represents an additional safety factor in the thermal-hydraulics of the UMR Reactor.

4.7 References

[4-1] Covington, L., "A Neutronic and Thermal-hydraulic Study of the Conversion of the University of Missouri–Rolla Reactor to Low Enriched Uranium Fuel," MS Thesis, University of Missouri-Rolla, December 1988.

[4-2] Liverhant, S. E., Introduction to Nuclear Reactor Physics, J. Wiley, 1960.

[4-3] Gambill, W. R, "Collapse of Parallel-Plate Fuel Assemblies, "Nuclear Safety, Vol. 1, No. 1.

[4-4] Little, W. W. Jr. and Harde, R. W., "2DB User's Manual–Rev. 1", BNWL-831, Rev. 1, 1969.

[4-5] Stelzer, F., Warmeuebertragung and Stroemung, K. Thiemig, Muenchen, 1971.

5. REACTOR COOLANT SYSTEMS

5.1 Primary Coolant System

The UMRR core is submerged in an open pool containing 113.56 kiloliters (30,000 gallons) of demineralized light water. The core is cooled by natural convection. Heat from the pool water is dissipated primarily by evaporation into the reactor bay and can be discharged to the environment with the ventilation system. At full power, the pool temperature rises by only a few degrees Fahrenheit every hour. The pool water serves as a moderator, reflector, coolant, radiation shield, and heat sink. A description of the reactor pool is provided in Section 4.3 of Chapter 4.

Technical Specifications require the resistivity of pool water to be greater than 0.2 MΩ-cm as long as fuel elements are in the pool. This water quality requirement assures corrosion of pool components will be minimized. Components in the pool are made with corrosion resistant materials, primarily aluminum or stainless steel. The corrosion of aluminum in high purity water does not seem to be a problem. For example, in Reference [5-1] it is reported that in most open literature publications the corrosion of aluminum in distilled water at temperatures less than 100°C (212°F) is described as "ceasing" after an initial period of a moderate reaction rate. Furthermore, no significant fuel plate corrosion has ever been observed during over 43 years of operating experience at the UMRR. Provisions are made in Technical Specifications that allow for this water quality requirement to be temporarily exceeded for a period of up to one month every three years as a matter of operational convenience. In the unlikely event that the demineralizer system was to become inoperable, the one-month time period should be sufficient to repair the system. No significant corrosion is expected to be associated with a one-month period of slightly degraded water quality.

Technical Specifications require pool water resistivity to be measured at least once every 2 weeks during periods when the reactor is operating. If the reactor is not operated, resistivity is to be measured monthly. Experience has shown that pool water quality normally does not change rapidly but rather changes gradually. The measurement intervals specified assure that the need to regenerate or replace resins in order to maintain water quality will be identified in a timely fashion.

Technical Specifications require at least 4.88 m (16 ft) of water above the top of the core for reactor operations. This provision is primarily to assure sufficient water depth for shielding but also provides assurance that a natural convection flow path will be available. Normally, the pool is filled to within about 1 to 2 feet of the top of the pool walls. This provides about 6.10 m (20 ft) of water shielding between the top of the core and the pool surface.

There is a low pool level float switch that actuates an alarm if the pool level drops significantly. In addition, there is a siphon break located in the demineralizer inlet piping, approximately 4.88 m (16 ft) above the core that prevents the possibility of pumping the pool down below that level.

5.2 Pool Water Cleanup and Makeup System

The pool water is circulated through the demineralizer system at a flow rate of about 115 liter/min (30 gpm). Water flowing to the demineralizer system must first flow through a particulate filter and then through the mixed bed demineralizer column. A conductivity cell located in the outlet piping of the demineralizer column provides an alarm in the control room if the water quality deteriorates beyond the set point. Another conductivity cell is located between the particulate filter and the demineralizer column. This cell provides information on inlet water quality in order to assess the performance of the demineralizer column. The HCL and NaOH tanks are filled and used only during 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.

Technical Specification's limit the temperature of the pool water by requiring a rod withdrawal prohibit to occur if the pool temperature at the core inlet reaches 57.2°C (135°F). This limit is based upon the manufacturer's recommended maximum temperature of 60°C (140°F) for the ion exchange resins.

Supply water for the reactor pool is obtained from the University of Missouri-Rolla water system. The supply water is relatively hard as shown by the analysis of impurities shown in Table 5.1. Table 5.1 presents average values of samples taken from several Rolla public water supply wells by the State of Missouri Department of Natural Resources in 1997. The supply water is purified prior to being supplied to the pool. This is accomplished by passing the water through a charcoal filter bed and mixed resin bed ion exchanger.

Makeup water is normally provided with a dedicated system connected to the UMR water supply so that the makeup water system components (e.g. tanks, resin, hoses, etc.) are not contaminated. Makeup water can be provided by passing raw water through the pool demineralizer system; however, this practice is discouraged because the raw water significantly depletes the pool resins which have to be handled as potential radioactive waste. A siphon break is installed between the pool and the makeup system to prevent the possibility of siphoning contaminated water into the potable water supply.

Table 5.1-Average Results of Samples from Several Rolla Public Water Supply Wells.

Analysis Performed	Results	MCL[1]	SS[2]	Units
Total dissolved solids	270	0	500	mg/L
Hardness as CaCO3	272	0	0	mg/L
Fluoride	<0.804	4	2	mg/L
Sulfate	29.1	0	250	mg/L
Chloride	2.268	0	250	mg/L
Cyanide	<0.09	0.2	0	mg/L
Silver, Dissolved	<5	0	100	g/L
Aluminum, Dissolved	<10	0	200	µg/L
Arsenic, Dissolved	<2	50	0	µg/L
Barium, Dissolved	158	2000	0	µg/L
Beryllium, Dissolved	<1	4	0	µg/L
Calcium, Dissolved	58.04	0	0	mg/L
Cadmium, Dissolved	<1	5	0	µg/L
Chromium, Dissolved	<2.282	100	0	µg/L
Copper, Dissolved	76.92	1300	1000	µg/L
Iron, Dissolved	<96.02	0	300	µg/L
Mercury, Dissolved	<0.2	2	0	µg/L
Potassium, Dissolved	0.98	0	0	mg/L
Magnesium, Dissolved	30.86	0	0	mg/L
Manganese, Dissolved	<2.108	0	50	µg/L
Sodium, Dissolved	2.95	0	0	mg/L
Nickel, Dissolved	<3	100	0	µg/L
Lead, Dissolved	<4	15	0	µg/L
Antimony, Dissolved	<3	6	0	µg/L
Selenium, Dissolved	<2	50	0	µg/L
Thallium, Dissolved	<1	2	0	µg/L
Zinc, Dissolved	31.018	0	5000	µg/L

[1] MCL: Maximum Contaminant Level, regulated maximum contaminant level.
[2] SS: Secondary Standards (SMCL), recommended maximum contaminant level.

5.3 Nitrogen - 16 Control System

When the reactor is operated, N-16 is produced in the water passing through the core by the O-16 (n,p) N–16 reaction. Because the reactor is cooled by natural convection, a "thermal plume" of warm water naturally tends to rise to the surface during high power operations. N-16 traveling with the thermal plume can cause elevated surface dose rates. Two water pumps, referred to as "diffusers", are installed to direct surface water downward above the reactor core to delay the rise of N-16 as needed. Because the half-life of N-16 is only about 7 seconds, this delay is sufficient to significantly reduce the radiation level at the pool water surface. Operational experience at the UMRR has shown that N-16 does not produce significant radiation levels at reactor powers less than 20 kW_t even without the use of the diffusers. Full power surveys have shown that a maximum radiation level is about 3 mrem/hr at one foot from the pool water surface with a single diffuser turned on.

5.4 References

[5-1] Draley, F. E. and Ruther, W. E., "Aqueous Corrosion of Aluminum," Part 1, Corrosion, Vol. 12, September 1956.

6. ENGINEERED SAFETY FEATURES

Engineered safety features (ESF) are systems provided by a reactor facility to mitigate the radiological consequences of accidents. At the University of Missouri – Rolla Reactor (UMRR), ESF are not needed due to the low operating power, conservative design, and the fact that the accident analysis, as discussed in Chapter 13, in the past and present have shown they are not necessary to mitigate any accident postulated for the UMRR. The Maximum Hypothetical Accident for the UMRR, Failure of a Fueled Experiment as discussed in Chapter 13.2.1, has shown that for the reactor staff, the maximum expected whole-body dose and thyroid dose are respectively 1.37 rem and 11.3 rem. The whole-body dose to an individual located directly outside the reactor facility was determined to be 242 mrem. These values, which were evaluated without taking into consideration the use of any ESF, are well below the 10 CFR Part 20 dose limits. The accident analysis discussed in Chapter 13 demonstrates that for the UMRR facility, no ESF are necessary to maintain safe operations or to protect the health and safety of the public and reactor staff.

7. INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Summary Description

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.

7.2 Reactor Control and Protection System

7.2.1 Control Rods and Drive Mechanisms

There are four control rods loaded in any particular core configuration. Three of the control rods are shim/safety rods and the fourth is referred to as the regulating rod. Each control rod fits into a control rod fuel element specifically designed to accommodate the control rod. A detailed physical description of control rods, control rod fuel elements and control rod drive mechanisms is presented in Chapter 4.

7.2.2 Reactor Instrumentation

The nuclear instrumentation system is comprised of five separate channels:

1. Start-Up Channel
2. Linear Channel
3. Log and Linear Channel
4. Safety Channel #1
5. Safety Channel #2

Table 7.1 presents the type of detector, monitored parameters, approximate ranges, outputs, trips, and setpoints for each channel. Figure 7.1 illustrates a block diagram of the system layout. The trips actuate automatic safety or control functions for the reactor as required in the facility's Technical Specifications.

All neutron detectors are arranged near the reactor core and are encapsulated in water-tight aluminum housings. The fission chamber is attached to a motor driven positioning mechanism; the other detectors are manually adjusted.

Four types of automatic engineered protective actions are provided with the UMRR instrumentation:
1) Informational with audio/visual (WHITE) alarms
2) Rod Withdrawal Prohibit (RWP) with audio/visual (YELLOW) alarms

3) Reactor Rundown with audio/visual (BLUE) alarms
4) Reactor Scram with audio/visual (RED) alarms.

The non-nuclear reactor instrumentation consists of three thermocouples that measure pool water temperature near the core inlet and outlet. The spatial arrangement of the nuclear detectors and thermocouples is shown schematically in Figure 7.2.

Table 7.1-Nuclear Instrumentation Channel Description.

Channel	Type of Detector	Monitored Parameters	Range	Output Displays	Trip Function (Set Point)
1 Startup	Fission Chamber	Log CR	1 to 10E4 CPS (Moveable)	2. Recorder	RWP (Recorder off)
2 Log and Linear	CIC	Log N	10^{-6}% to 140%	1. Meter	Scram (Not Operate Mode)
				2. Recorder	Rundown (P > 120%)
					Rundown (Low HV 80%)
					RWP (Recorder Off)
		Period	-30 to +3 Seconds	1. Meter	RWP (Recorder Off)
				2. Recorder	RWP (Period< 30 Sec)*
					Rundown (Period < 15 Sec)
					Scram (Period < 5 Sec)
		Power Range	0 to 125%	1. Meter	None
3 Linear	CIC	Linear Power	0 to 150% Scale w/ Selectable Scales (About 0.2 W to 300 kW)	1. Meter	RWP (Recorder Off)
				2. Recorder	Rundown (P > 120% Scale)
					Rundown (Low HV 80%)
4 Safety #1	UIC	Power Range	0 to 150%	Meter	Scram (P > 150%)
5 Safety #2	UIC	Power Range	0 to 150%	Meter	Scram (P > 150%)

RWP = Rod Withdrawal Prohibit
CIC = Compensated Ion Chamber
UIC = Uncompensated Ion Chamber
CR = Count Rate
CPS=Counts per Second
HV = High Voltage
P = Power
* = May be key bypassed

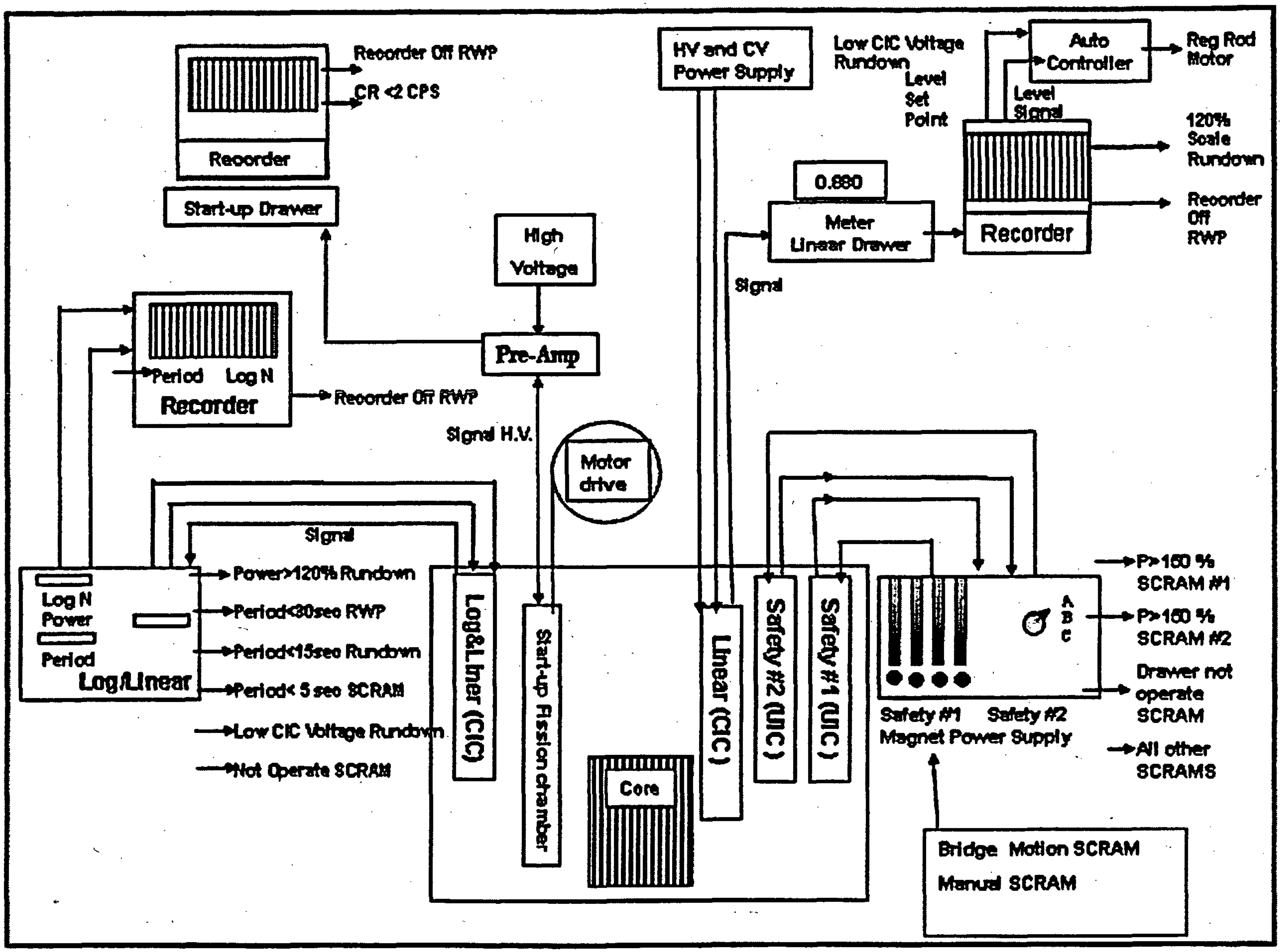
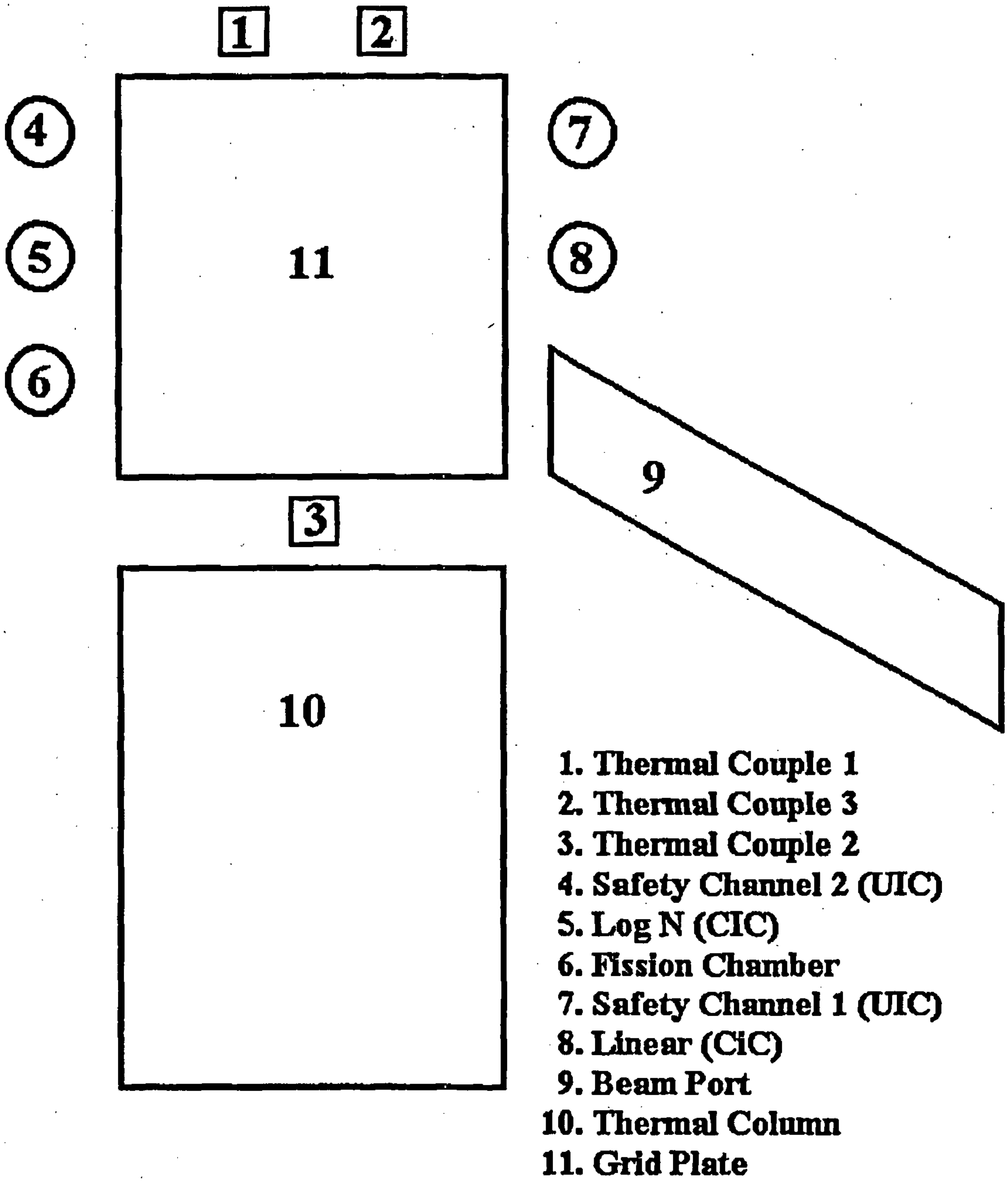


Figure 7.1-Block Diagram of Reactor Nuclear Instrumentation.



Note: Thermal Couple 1 and 3 are located beneath the gridplate, Thermal Couple 2 is located above the core.

Figure 7.2-Sketch of Detector Locations (Top View).

7.2.2.1 Startup Channel

The startup channel uses a fission chamber detector to monitor neutron flux during reactor startup and low power reactor operation. The channel has a limited range of four decades. The detector is physically moved in and out of the core region into a boron shield assembly by an electric drive system to maintain an on-scale indication. 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.

During a startup, the fission chamber is normally fully inserted near the core. In this position, the startup channel is sensitive enough to monitor reactor neutrons from shutdown to a power of about 1 W. The CIC based Linear Channel begins to respond at a power level of about 0.2 W. The CIC based Log and Linear Channel begins to respond at slightly lower power levels. Thus, the startup channel is used to monitor reactor power from shutdown power up to a power level sufficient to drive the two CIC channels. Once positive indications have been registered on the CIC channels, the fission chamber may be retracted away from the core. Because the fission chamber must be moved, it can only be used to provide "relative" power trends and is not used to indicate absolute power.

The pulses from the fission chamber are fed into a solid-state circuitry consisting of a preamplifier and a linear pulse amplifier. The pulse amplifier feeds the log count rate meter, which is located on the front panel of the drawer, and the log count rate recorder.

The recorder provides two signals to the rod withdrawal prohibit (RWP) system. The first signal, which originates from a micro switch on the recorder cam, prevents control rod movement if the count rate (CR) is less than 2 counts per second (cps). This interlock 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 (RWP) if the Startup Channel rate recorder is off.

The CR < 2 cps rod withdrawal trip may be key bypassed at the reactor console by the Senior Reactor Operator (SRO) on Duty as provided for in the Standard Operating Procedures (SOP). The CR < 2 cps bypass is used to raise control rods during such evolutions as core loading.

7.2.2.2 Linear Power Channel

The Linear Channel consists of a CIC detector coupled to the linear picoampmeter and strip-chart recorder. The Linear picoampmeter panel control consists of range switches enabling switching ranges from 0.2 W to 300 kW and a zero check switch. The output drives the Linear Recorder-controller, which in conjunction with the servo-amplifier provides automatic control of reactor power. If the reactor power exceeds 120% of the selected range (demand), a micro-switch in the recorder actuates a reactor Rundown. The Linear Recorder is equipped with a RWP, which actuates if the recorder is off.

The CIC detector high voltage (HV) and compensating voltage (CV) power supplies are housed in a separate instrument drawer. The CIC HV is monitored and a reactor Rundown is actuated if the HV drops to 80% of its rated value.

7.2.2.3 Log and Linear Power Channel

The Log and Linear Power Channel monitors three parameters; 1) log power, 2) period, and 3) power range. The system is comprised of the Log and Linear drawer, CIC detector, and a two pen strip-chart recorder.

The CIC detector is powered by HV and CV power supplies located inside of the Log and Linear signal processing drawer. The signal from the detector is fed to the log amplifier portion of the processing drawer.

The drawer monitors log percent power from 10^{-6} to 140% full power without switching interruptions. Log percent power is displayed on a digital meter (range of 10^{-6} to 140%) and LCD bar graph, (range of 10^{-6} to about 108%) which is located on the front panel of the signal processor drawer. The drawer provides a log percent power output signal for the strip-chart recorder and also provides a relay trip output, which actuates a reactor rundown if reactor log power reaches 120% of full power.

The signal processing drawer monitors reactor period over a range of -30 seconds to +3 seconds. The log amplifier feeds a signal to the period amplifier. The period signal is displayed on a digital meter and an LCD bar graph located on the front panel of the drawer. The drawer also provides a period output signal for the strip-chart recorder. The drawer provides the following three relay trip outputs for reactor period:

1. Period < 30 second rod withdrawal prohibit
2. Period < 15 second rundown
3. Period < 5 second scram

The Period < [redacted].

The drawer also monitors power range from 0 to 125%. The power range signal is displayed on a digital meter and an LCD bar graph display on the front panel of the drawer.

Detector HV is monitored by a Non-Operate circuit. If HV drops to less than 80% of nominal operating voltage, the Non-Operate circuit trips a relay which initiates both a reactor scram and reactor rundown.

The recorder is a two pen analog strip-chart recorder. One pen records the log percent power signal while the other records the period signal. The recorder provides a relay trip to actuate a rod withdraw prohibit trip when the recorder is not turned on.

7.2.2.4 Safety Channels

Two redundant safety channels are a part of the reactor protection system. They provide the mechanism for scramming the reactor when power exceeds 150% of licensed full power. Each safety channel consists of an UIC 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 power supply. The sensing circuit of either safety amplifier is capable of actuating a shut off of magnet current.

An indicator lamp located on the front of each safety amplifier becomes energized if reactor power should reach a predetermined limit. The safety amplifier scram circuit activates a shutting off of the magnet current. An additional safety amplifier relay activates the 150% full power annunciator light and audible alarm. Magnet current will remain shut off until reactor power is below the predetermined set point and the safety amplifier reset switch is depressed.

The safety amplifier scram circuit will activate a shutting off of the magnet current. The safety amplifier scram circuit consists of relay connections providing a current from the negative to positive input of the magnet power supply scram input. If a relay is actuated, the magnet current will shut off. The safety amplifier scram circuit consist of relays from two safety amplifiers, two HV power supplies, a five second period trip, and the scram logic series containing bridge motion, Log and Linear non-operative, and manual scram circuitry described in Section 7.2.2.5. 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. A test switch is mounted on the front of each safety amplifier to provide testing of the scram circuitry. The scram circuits are of a fail-safe design.

Safety amplifiers are contained in separate NIMs (Nuclear Instrumentation Module). HV power supplies for the ion chambers are contained in a single NIM. In the case of failure of either HV power supply, the scram circuit is actuated. An indicator light will illuminate on the HV NIM upon failure of the HV power supply. The magnet power supply is contained in a NIM. A SCRAM indicator lamp will illuminate on the magnet power supply when the safety channel scram circuit is initiated. The four NIMs are located in a NIM Bin power supply, which provides power to each NIM.

7.2.2.5 Scram Logic

The scram logic circuitry contained in the Safety Drawer was discussed in the previous section. This section describes logic and operation of the circuit processing the Bridge Motion Scram, Log and Linear Not Operative Scram, and Manual Scram.

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 interrupting the current in the safety magnets thus releasing the shim/safety rods. The scram relay [redacted]

[redacted].

The Bridge Motion Scram is controlled by a micro-switch located 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 will trip the micro-switch thus de-energizing the motion relay, which opens the contacts in the scram circuit.

When the Log and Linear drawer goes into Not Operate mode, a relay de-energizes causing contacts to break thus de-energizing the scram relay.

Pushing the manual scram button mechanically opens two contacts: one causes the scram relay to de-energize and the other interrupts regulated power to the magnet power supply, thus AC power 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 line conditioned AC power. In the case when electrical power is lost the scram circuit opens and initiates a reactor scram.

7.2.2.6 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 withdrawn in a bank, power to the AC drive mechanism of the regulating rod is disconnected. The position of each rod is continuously indicated, to within about ± 0.1 inch (0.25 cm) 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. 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 rod withdrawal prohibit trip will occur when withdrawing control rods that are below shim range while the regulating rod is not fully inserted. The rod withdrawal prohibit with a control rod in shim range can be key bypassed at the reactor console by the Senior Operator on Duty as provided for in the Standard Operating Procedures. The shim range bypass is used during calibration of a control rod while the control rod is below the shim range level.

A servo-amplifier system is used to automatically control the reactor power. The Linear Recorder has an adjustable set point control for the servo system. When the reactor is at steady-state power, the servo system may be energized to automatically maintain power level at the set point. The servo system is interlocked so that the power level must be within about ± 2 % of the set point before the system may be engaged. If the power level deviates outside of the ± 2 % limit, control of the reactor reverts to manual control and visual and audible alarms are actuated.

7.2.2.7 Pool Water Temperature Channel

The pool water temperature channel consists of two core inlet thermocouples placed just below the grid plate, and one outlet thermocouple placed several feet above the core (see Figure 7.2). The thermocouples feed their signals to a recorder located in the control room.

The two inlet thermocouples and are set to trip at T ≤135°F (57°C). This trip causes a rod withdrawal prohibit condition, preventing the rods from being withdrawn. The purpose of the trip is to protect the demineralizer system resins, which are rated to 140°F. An additional rod withdrawal prohibit condition occurs when the recorder is not turned on.

7.3 Control Console and Display Instruments

The main reactor console contains the Startup, Linear and Log N drawers and recorders in a console area that is about 4½ feet tall and 5½ feet wide. This area also includes the manual scram button, control rod indicators and controls, annunciator controls, magnet power key switch, fission chamber controls and bypass switches.

The Startup, Linear and Log N channels have instrument drawers with analog or digital displays mounted about at eye level when the operator is sitting. Recorders are mounted above their respective drawers about at eye level when the operator is standing. Both meter displays and recorders can easily be seen when sitting or standing. The annunciator panel is mounted above the recorders. Controls and indicators for the control rods, fission chamber, magnet power, manual scram button, and bypass switches are mounted below the drawers and are easily accessible to the operator.

To the right of the console is a rack that is 19 inches wide and 6 foot tall. The rack contains the RAM system, magnet power supply, safety channels, temperature and CAM recorders, pool light controls and linear power supply. The reactor operator can easily see this instrumentation while sitting or standing.

7.4 Radiation Monitoring System

Three systems, the Gamma Radiation Area Monitoring (RAM) system, Basement Neutron Monitor, and Constant Air Monitor (CAM) are used for radiation protection while operating the reactor. The Gamma RAM system consists of three separate gamma monitoring channels. The gamma monitors are located at the following strategic locations: 1) in the reactor bay above the reactor pool (Reactor Bridge Monitor), 2) in the mid-level basement on the wall behind the demineralizer system (Demin Monitor), and 3) on the wall in the sub-basement experimental area (Experiment Room Monitor).

Each gamma monitor consists of a Geiger-Muller (G-M) detector, a remote readout unit with an audible and visual alarm, and a local readout in the control room. The output meters of the gamma channels have a range of 0.1 mrem/hr to 10,000 mrem/hr. The control room read-out modules are interfaced into the reactor console to provide "High Area Radiation" reactor rundown and audible and visual alarms. The "High Area Radiation" rundown is required by Technical Specifications to be set to actuate at or below radiation levels of 20 mrem/hr.

The Reactor Bridge Monitor has a second alarm set point that initiates the building evacuation alarm. The building evacuation alarm is required by Technical Specifications to be set at or below 50 mrem/hr.

The Basement Neutron Monitor consists of a BF-3 detector mounted on the wall in the basement experimental area, adjacent to the beam port and thermal column. The beam port and thermal column facilities provide access to neutrons leaking from the reactor and create significant neutron fields when opened during reactor operations. The Basement Neutron Monitor output is displayed on an analog meter that is located in the control room. The meter output ranges from 0.1 mrem/hr to 10,000 mrem/hr. If neutron radiation exceeds the predetermined set point, audio and visual alarms are actuated on the control console.

The RAM system rundown can be key bypassed by the Senior Reactor Operator on Duty in accordance with SOPs. [redacted], but does not impair the radiation alarm or the building evacuation alarm. The high area radiation bypass may be used when experiments that have higher radiation levels are being performed near a RAM detector.

The continuous air monitoring (CAM) system consists of a monitor, recorder, and associated alarm and warning circuitry. The function of the CAM is to measure the radioactivity of airborne particulates in reactor bay air by concentrating 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. The CAM is equipped with an alarm system to give audio and visual warning if the reading exceeds the alarm set point. The CAM system is a stand-alone unit and is not interfaced with the control console.

7.5 Other Protective Actions

A conductivity probe located at the outlet side of the demineralizer system is monitored by a conductivity meter that provides an audible and visual alarm on the control console if the predetermined set point is exceeded. Technical Specifications require that the pool water resistively be greater than 0.2 meg-ohm centimeters as long as fuel elements are in the pool.

A water level sensor is mounted in the basement sump, which provides an audible and visual alarm to the control room in the event that the water level becomes too deep in the sump. This allows early indication of failure of the sump pump(s) and provides some protection against water damage of experimental equipment.

Float switches monitor the level of the pool water with in a 12 to 16 inch range to activate an alarm if the level of the pool water is too low or high. This provides an early detection of water leakage, make-up water needing added to the pool or over filling the pool. An additional float switch is located in a holding tank used for containing water leakage from the demineralizer system. A rise of the water level within the holding tank will activate the pool water level alarm. The alarm activated by the float switches is continuously monitored by the campus police.

7.6 Summary of Reactor Instrumentation Protective Actions

Table 7.2 provides a summary of reactor instrumentation protective actions that are interfaced with the control console.

Table 7.2-Reactor Instrumentation Protective Actions.		
Situation	**Unit Initiating Action**	**Protective Action**
Manual Scram	Scram Button	Scram
Period < 5 Seconds	Log & Linear Drawer	Scram
150% Full Power	Safety Amplifier	Scram
Bridge Motion	Motion Switch	Scram
Log & Linear Drawer Non-Operate (Includes 80% HV Trip)	Log & Linear Drawer	Scram & Rundown
120% Demand	Linear Recorder	Rundown
Period <15 seconds	Log & Linear Drawer	Rundown
Reg. Rod on Insert Limit in Auto-Control	Micro-Switch	Rundown
120% Full Power	Log & Linear Drawer	Rundown
High Radiation[1]	RAM System	Rundown
Period <30 Sec[1]	Log & Linear Drawer	RWP
Recorder Off	Relay	RWP
Log Count Rate <2 CPS[1]	Log Count Rate System	RWP
Core Inlet Water Temp. 135°F	Relay	RWP
Safety Rods Below Shim Range[1]	Relay	Regulating Rod RWP
Basement Sump Level High	Micro-Switch	Operator Response
Interlock Bypassed	Key Switch	Operator Response
Effluent Pool Demineralizer Conductivity High	Conductivity Bridge	Operator Response
Beam Port or Thermal Column "Open"	Micro-Switch	Operator Response
High Neutron Flux in Beam Room	Neutron Detector	Operator Response
Manual Operation	Micro-Switch	Operator Response

[1] Indicates that the situation may be key bypassed.

8. ELECTRICAL POWER SYSTEM

The electrical power system at the UMRR facility is a standard and well-accepted electrical supply system, designed and constructed to specifications similar to those at other research reactor facilities.

8.1 Normal Electrical Power Systems

A 110/208 V distribution panel in the reactor building is fed from a campus substation and meets the 1996 National Electric Code (NEC).

The reactor console instrumentation is powered by line conditioned 120VAC. The AC Line Conditioner has battery back-up potential that is not presently used. In the future it may be used as a convenience feature to avoid dropping control rods during a power loss of a few seconds or less.

8.2 Emergency Electrical Power Systems

No emergency electrical power is required for the UMRR operation. Because the reactor will scram in case of a power interruption and the decay heat generated in the core after scram will not cause fuel heating above acceptable levels (see Chapter 13, Section 13.1.2), no emergency power is required. Battery-operated emergency lighting for personnel movement during a power outage is provided throughout building

The security and fire alarm systems have individual battery back-up systems. Both of the systems are operational during a power outage to the Reactor Facility.

9. AUXILIARY FACILITIES

9.1 Heating, Ventilation, and Air Conditioning

A recirculating air conditioner, located in the reactor bay, regulates building air for human comfort and assists in dissipating reactor pool heat. Several room air conditioners are located throughout the facility for regulating room temperature for comfort of the staff. Room air conditioners are mounted in the exteriors walls of the building. The facility is heated with steam that is provided by the campus.
A system of three exhaust fans has been mounted on the reactor building roof and provides ventilation for the building. Fans #1 and #2 each have a rated discharge of about 425 m^3/min (15,000 cfm) and Fan #3 has a rated discharge of about 142 m^3/min (5,000 cfm). Fans #1 and #2 have intakes equipped with filters located on the lower level. Each fan has a separate controller and may be run in any combination. The on/off switches are mounted adjacent to the control room within view of the reactor operator, and each vent fan controller has a light that shows if the fan is either on or off.

The UMRR is a 200-kW pool-type reactor whose operation between 1984 and 2003 has averaged the equivalent of about 50 full-power hours per year. Therefore, the fission product inventory is low. In the case of any abnormal situation involving a significant airborne release, an emergency procedure will be followed, which specifies that all exhaust fans must be immediately turned off. The exhaust fans and intakes are equipped with louvers that close automatically when the fans are turned off or power is lost. Other building openings are not sealed; thus, some air movement caused by atmospheric pressure changes and temperature differentials are expected to occur.

Technical Specifications require that a ventilation fan with a rated capacity of at least 127 m^3/min (4,500 cfm) be turned on within 10 minutes of reaching full power operations. Experience has shown that this is sufficient to maintain Ar-41 concentrations at acceptable levels in the bay. Technical Specifications also state that the bay door, ventilation inlet and exhaust duct louvers, and the personnel security door shall be visually checked for proper operation quarterly.

9.2 Handling and Storage of Reactor Fuel

The fuel storage pit is located at [redacted]. The fuel storage pit contains two storage racks; each is capable of holding up to 15 fuel elements. The fuel elements are oriented in the storage racks in an upright position. The geometry of the stored fuel is such that a criticality in the storage pit cannot be achieved.

Technical specifications require that the fuel storage pit be capable of storing the complete inventory (28 elements) of Low Enriched Fuel (LEU), but the fuel storage pit is actually capable of storing up to 30 elements.

Technical specifications also require $k_{eff} < 0.9$ for the fully loaded storage pit. Measurements with the previous High Enriched Fuel (HEU) showed $k_{eff} < 0.6$ for the loaded fuel pit. The reactivity of the LEU fuel elements have been shown to be very similar to that of the previously used HEU elements.

[redacted]. Technical specifications require that a licensed SRO supervise fuel movement and associated Health Physics monitoring. SOPs are in place that assures this requirement will be met.

9.3 Fire Protection System

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

The fire protection system consists of thirteen smoke detectors, two hand pull stations, and an alarm and relay box. Smoke detectors are located throughout the reactor building on each floor. The hand pull stations are located by the front entry door and by the emergency exit at the demineralizer level. There are four flashing warning 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. The control room and front office contain the remaining flashing warning lights. Eight fire extinguishers are also located throughout the Reactor Building at strategically important locations. The Rolla Fire Department, located less than 1.6 km (1.0 miles) from the reactor facility, responds to all calls.

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. Emergency lighting is located throughout the reactor building at strategic locations and is activated when power is lost.

9.4 Communication Systems

The control room is located adjacent to the reactor bay and has large windows that allow for visual communication between the reactor operator and the reactor bridge and pool area. An intercom system provides audible communication between the reactor operator and personnel throughout the building. In addition to the intercom system, there are telephones located in the reactor control room, Reactor Manager's office, in the front office, and various other locations throughout the facility.

The building evacuation alarm, which sounds throughout the building, may be activated manually from the control room to communicate the need for a building evacuation.

9.5 Possession and Use of By-Product, Source, and Special Nuclear Material

The reactor facility is licensed to receive, possess and use the following materials:

1. Up to [redacted] enriched to less than 20% in the form of reactor fuel;
2. Up to [redacted] of any enrichment in the form of fission chambers and flux foils used in connection with operation of the reactor;
3. Up to [redacted] in the form of sealed Pu-Be neutron sources to be used in connection with operation of the reactor; and
4. By product material as may be produced by operation of the reactor.

All of reactor related materials are possessed, used, and stored within the confines of the reactor facility building. By product, source, or special nuclear material to be moved out of the reactor facility building is transferred to another valid license prior to leaving the facility.

The facility also possesses check sources and some calibration sources that are licensed under the campus Materials License (License # 24-00513-40).

10. EXPERIMENTAL FACILITIES

10.1 Summary Description

The UMRR is used primarily as a teaching facility and as such experiments are conducted on a regular basis. In addition to irradiation capabilities in the pool, there are a number of facilities which can be used for sample irradiation and experiments. These include the thermal column, beam port, pneumatic sample transfer system, sample rotor assembly, core access and isotope production elements, and void tubes and are described in the following sections. In addition to those facilities which are described below, the irradiation fuel element (see Chapter 4, Section 4.2.1.4) may also be used to accommodate experiments. All experiments are subject to the requirements set forth in Section 10.3 and approval as discussed in Section 10.4.

10.2 Experimental Facilities

10.2.1 Thermal Column

The thermal column protrudes through the pool wall and sits behind the reactor core. The thermal column provides a readily accessible field of thermal neutrons for experimental purposes. The thermal column assembly consists of a door that opens into the basement experimental level, a graphite assembly, and a shield (Figure 10.1). Total shielding from the reactor core through the thermal column to the outer biological wall face is roughly equivalent to that which would be provided by the intervening water and biological pool wall.

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. The graphite assembly measures 1.1 m x 1.1 m x 1.75 m (3.5 ft x 3.5 ft x 5.75 ft) and extends from the pool wall by 1 m (3.3 ft). The irradiation positions within the thermal column consist of one 412.9 cm^2 (64 in^2) and four 103.2 cm^2 (16 in^2) horizontal access ports, all of which are filled with graphite stringers (plugs) when not in use (Figure 10.1; Section A-A).

The Thermal column door is filled with concrete and has a front plate made of boral (35% B_4C). The door measures 1.2 m x 1.2 m x 1.5 m (4 ft x 4 ft x 5 ft). Access to the graphite may be obtained by rolling the door out by its rails or by removing the 103.2 cm^2 (16 in^2) round plugs or the 412.9 cm^2 (64 in^2) square plug to gain access to the irradiation positions (Figure 10.1; Section B-B).

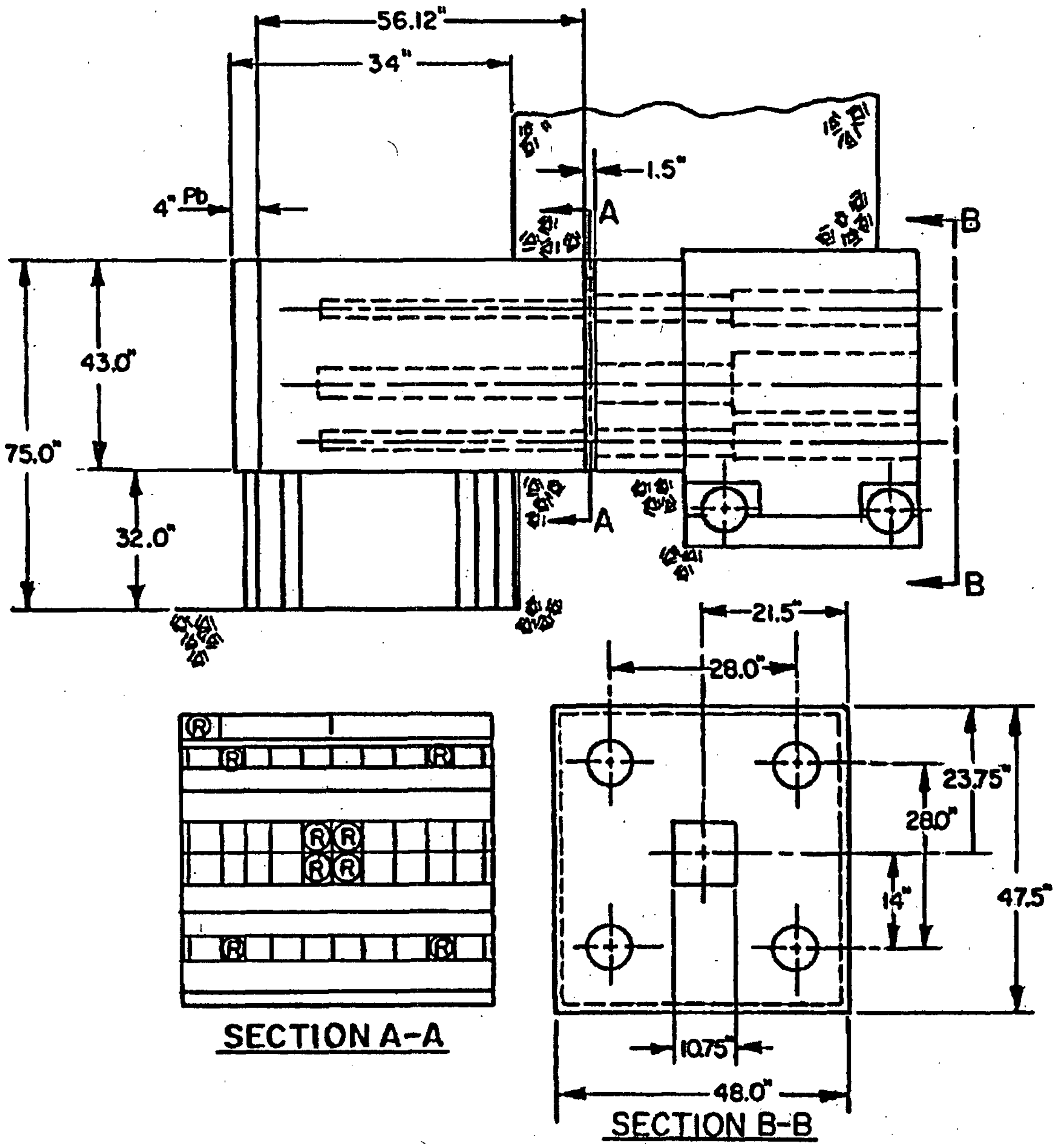


Figure 10.1-Thermal Column Assembly.

10.2.2 Beam Port

A stepped beam port (Figure 10.2) provides a beam of reactor neutrons for experimental purposes. The beam port protrudes through the pool wall and extends into the reactor pool near the core. The open end of the beam tube terminates in the basement experimental area. Operations required to remove or install equipment from the beam port are performed from the basement experimental area.

The beam tube is constructed of aluminum and is closed at the reactor end.

A shutter assembly composed of two parts may be used to achieve a collimated beam of neutrons. It consists of a plug containing a beam guide having a cross-section of 7.0 cm^2 x 4.4 cm^2 (2.75 in^2 x 1.75 in^2) and a shutter 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 from the control room. An open/closed 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. The beam tube may be filled with shielding plugs if desired. The outer and inner concrete shielding plugs are contained in stainless steel. The 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. The beam port can be sealed with a blank cap if desired.

Experiments with the HEU core have shown [10-1] that flooding of the beam tube with water has no noticeable effect upon core reactivity.

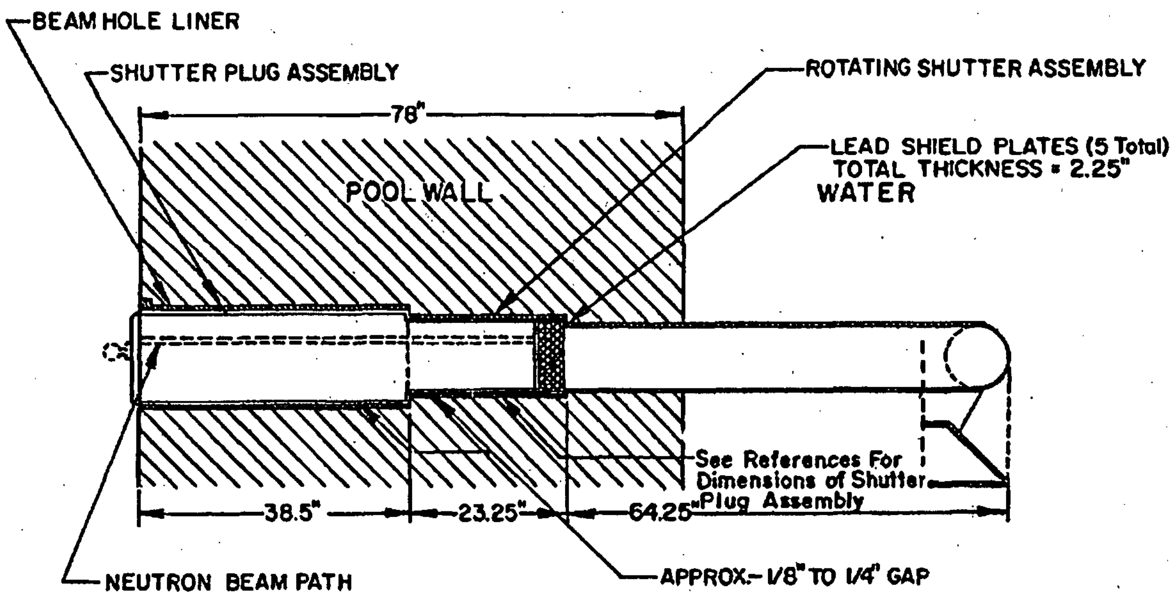


Figure 10.2-Beam Port.

10.2.3 Pneumatic Sample Transfer (Rabbit) System

The pneumatic transfer (rabbit) 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 reduce sample activation by thermal neutrons. The rabbit tubes fit into the grid plate in a manner similar to that of a fuel element and are positioned on the core periphery. The transfer tubing terminates in a glove box next to the reactor pool where samples are loaded and removed from the system.

Each rabbit system consists of two stainless steel tubes with one tube being the sample tube and the other providing a pressure differential. The stainless steel tubes terminate just above the pool water surface. Flexible tubing is used to connect the stainless steel tubes to the glove box and gas system. Nitrogen gas is used as a transport medium in order to reduce Ar-41 activation. The rabbit system is vented through a high efficiency filter thus reducing particulate activity.

The system may be controlled manually or by a computer system located in the reactor control room. A switch is available to bypass the computer controller and to return the system to manual control if desired.

10.2.4 Sample Rotor Assembly

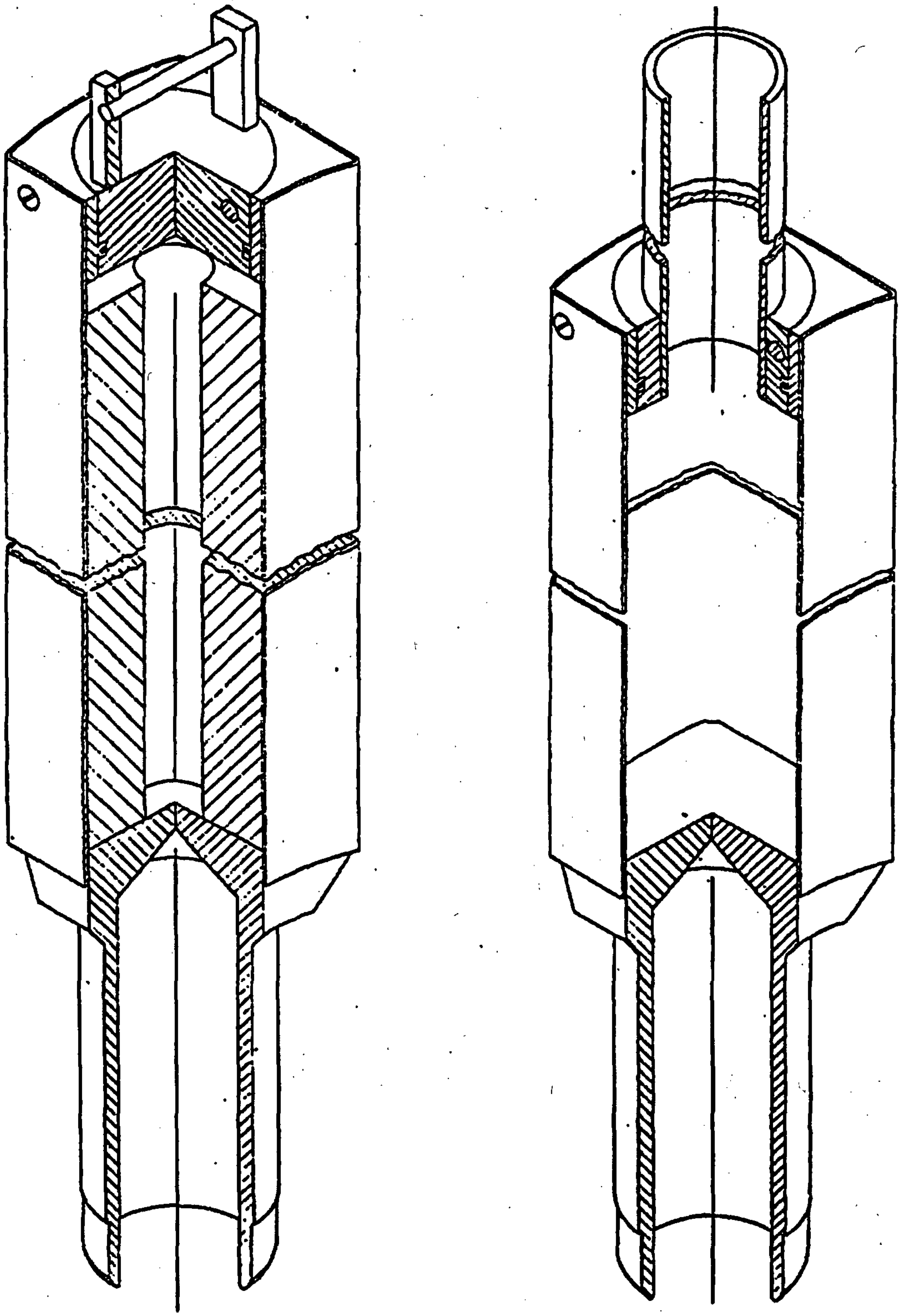
The sample rotor is a device that rotates samples next to the core, thus enabling a more uniform irradiation of samples which are irradiated simultaneously. Eight samples can be placed in the sample rotor at one time. The sample rotor assembly is placed in the grid plate in a manner similar to that of 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. Experiments irradiated in the sample rotor assembly must conform to the reactivity and materials requirements specified in the Technical Specifications.

10.2.5 Core Access and Isotope Production Elements

The core access element and isotope production elements are unfueled and are similar in shape to the fuel elements. These elements serve as experimental facilities and may be placed in various grid plate locations.

The core access element, shown in Figure 10.3 (a), is used to provide access to the inner and outer parts of the reactor core and is a dry irradiation facility. The assembly consists of a hollow piece of graphite clad entirely in aluminum. The top portion of the assembly accommodates an aluminum tube which projects upwards above the pool water level and is curved underwater to prevent neutron and gamma streaming. Samples for irradiation are lowered down the pipe on the end of a leader.

The isotope production element, shown in Figure 10.3 (b), is also used to provide access to the inner and outer parts of the core. It is filled with graphite and has a hole passing through it to permit a neutron start-up source or an experiment 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 wet or dry irradiation facility. A top sealing plug may be used which contains a groove for an O-ring and a horizontal hole so that the plug may be secured to the assembly by means of aluminum pins.



a) Isotope Production Element b) Core Access Element

Figure 10.3-Core Access and Isotope Production Elements.

10.2.6 Void Tube

The void tube is a cylindrical, aluminum facility that is equipped with a nozzle that fits into the core grid-plate. The void tube is equipped with a seal and may be used as an air-filled assembly, water filled assembly, or a partial mix of air and water. The void tube may be placed at various positions within the grid-plate and constitutes a moveable experiment.

10.2.7 Other Experimental Facilities

Moveable experiments may be inserted or removed from the core using "stringers" which simply consist of a long string. In such instances, samples are affixed to the stringer and weighted to overcome buoyancy if necessary. Stringer samples may be manually lowered into or removed from the core region in accordance with applicable procedures.

Wires may be inserted between fuel plates on stringers. The arrangement may be used to measure the axial flux profile of the core.

Experiments approved by the reactor staff may be positioned next to the shutdown reactor for various purposes; including gamma irradiation using the shutdown reactor as a gamma source. Such an experiment does not constitute a "reactor experiment" as long as the reactor remains shutdown the entire time the experiment is positioned near the core.

Various other experiments approved by the reactor staff may be positioned in the reactor pool. Such experiments are not considered "reactor experiments" as long as they are positioned outside of the neutron field produced by the reactor.

10.3 Restrictions on Experiments

A wide variety of experiments may be run in conjunction with reactor operations. Moveable experiments may be inserted and removed from the reactor while operating and are limited by Technical Specifications to an absolute reactivity worth of $\leq 0.4\ \%\ \Delta k/k$. Failure of a moveable experiment is analyzed in Chapter 13, Section 13.1.6.2.

An experiment with an absolute worth greater than $0.4\ \%\ \Delta k/k$ must be run as a secured experiment. A secured experiment is held in a stationary position relative to the reactor by mechanical means. This requirement minimizes the possibility that such an experiment could fall away from the core causing an undesired step insertion of reactivity. Experiments with an absolute worth greater than $0.4\ \%\ \Delta k/k$ must be inserted and removed from the reactor with the reactor shutdown using a procedure approved by the Radiation Safety Committee (RSC).

The sum of the absolute values of all experiments is limited by Technical Specifications to 1.2 %

$\Delta k/k$. This does not include experimental facilities. This places an acceptable upper limit on the worth of all experiments. This limit is lower than that assumed in the reactivity insertion accident analyzed in Chapter 13, Section 13.1.2 which showed a step insertion of 1.5 % $\Delta k/k$ and did not lead to significant consequences.

Experiments with moving parts shall not have a continuous reactivity insertion rate greater than +0.05 % $\Delta k/k$ per second. This requirement provides a restriction on certain reactor kinetics experiments. The value of 0.05 % $\Delta k/k$ per second is well below the insertion rate of 0.074 % $\Delta k/k$ per second analyzed in Chapter 13, Section 13.1.9, which showed no significant consequence.

Experiments to be irradiated in the reactor are to be either made of corrosion resistant materials or encapsulated within corrosion resistant containers. This requirement provides assurance that irradiation samples will not contaminate the pool water.

Explosive materials are not allowed in or near the reactor unless specifically approved by the RSC. Experiments reviewed by the RSC in which the material is potentially explosive, either while contained or if it leaked from its container, shall be designed to prevent damage to the reactor core or to the control rods or instrumentation, and to prevent any changes in reactivity. Known explosives in the amount of greater than 25 milligrams shall not be irradiated in or near the reactor core. In addition the pressure must be calculated or experimentally determined and shown to be below that required to cause the sample container to fail. Special case-by-case precautions would be taken before the unlikely irradiation of explosive materials would be allowed. The quantities would be restricted to very small masses and most likely such irradiations would be done at the far end of the beam tube or of the thermal column. In which case, the potential for core damage or reactivity change would be very small.

Fueled experiments are experiments containing more than trace quantities of [redacted]-[redacted]. [redacted]. Accident analysis presented in Chapter 13, Section 13.1.1 show that the potential fission product release associated with a [redacted] fueled experiment is within acceptable limits. Fueled experiments which generate more than [redacted] power shall be irradiated in the reactor pool at least 16 ft (4.88 m) deep under the pool water surface. Pool water reduces the amount of fission product released to the bay in the unlikely event of an experiment failure. Analysis presented in Chapter 13, Section 13.1.1 show that the potential release associated with a [redacted] experiment are such that it does not need to be run underwater. Therefore, fueled experiments which generate [redacted] or less may be irradiated in the beam port or thermal column. Special case-by-case precautions would be taken before irradiation of a fueled experiment including RSC review.

Cooling shall be provided to prevent the surface temperature of an experiment being irradiated from exceeding the boiling point of the reactor pool water. Samples or containers irradiated in the pool are in contact with a large heat sink. However, in order to assure that departure from nucleate boiling does not occur, adequate heat removal must be provided.

Experimental apparatus, material, or equipment to be inserted in the reactor are not to be positioned so as to cause continuous shadowing of the nuclear instrumentation, interference with the control rods, or other perturbations that may interfere with the safe operation of the rector. An experiment which shadows any of the nuclear instrumentation could possibly cause said instrumentation to give erroneous information and thus degrade its performance. Experiments which could adversely affect proper operation of the control rods must be avoided.

10.4 Experiment Review

Technical Specifications requires the reactor staff to perform a thorough review of all proposed experiments in order to assure they meet applicable Technical Specification requirements, to determine if any safety issues are involved, and to determine if the proposed experiment constitutes a "new" experiment not described in the Safety Analysis Report. New experiments must be evaluated under the provision of 10 CFR Part 50.59. Standard Operating procedures are in place covering the installation, removal, and movement of experiments and experimental facilities in the core region. Following the reactor staff review, the proposed experiment may be approved at the site level if appropriate. Otherwise, the proposed experiment will be forwarded to the RSC for review.

Experiment reviews are based on American National Standards Institute (ANSI) and American Nuclear Society (ANS) standard "Review of Experiments for Research Reactors" (ANSI N401-1974/ANS 15.6). Changes that do not alter the original intent of an experiment procedure can be approved by the Reactor Manager. Such changes are subject to RSC approval.

10.5 References

[10-1] Hazards Summary Report for the University of Missouri at Rolla Nuclear Reactor, November 1, 1965.

11. RAD. PROTECTION PROGRAM AND WASTE MANAGEMENT

11.1 Radiation Protection

Radiation protection at the UMRR facility has long followed the principals of the ALARA program. This ensures that the health and safety of staff, students, and the general public will always be of the foremost concern and that unnecessary exposures do not occur. Stationary radiation protection for the staff and the general public from the reactor core is accomplished by the biological shielding provided by the reactor pool. Various monitoring systems are utilized for an active radiation protection system to activate visual and audible alarm signals if an increased radiation level occurs in the reactor building. In addition, monthly audits are conducted via the health physics staff to verify that all radiation or contamination levels are within regulation values and that they show compliance to the principals of the ALARA program.

11.1.1 Radiation Sources

11.1.1.1 Airborne Radiation Sources

The principal potential airborne radiation source is composed of Ar-41, N-16 and neutron-activated dust particulates. These are produced by the irradiation of dissolved air in pool water, O-16, and airborne particulates in the thermal column and other experimental facilities. Another activation product that can become airborne is N-16, produced within the coolant passing through the core of the reactor. To decrease the N-16 gas that becomes airborne, water above the core may be over stirred using the available diffuser pumps. This increases the transport time of the short-lived ($t_{1/2}$ = 7.1 s) N-16 from the core to the surface of the pool and allows additional decay time. As a result of this practice, the potential exposure rate from airborne N-16 is well below the limits prescribed by 10 CFR 20.

The UMRR staff has estimated the release of airborne radioactivity Ar-41 to typically be about 100 mCi/yr, see Table 11.1. Analyses show that this amount of release results in radiation exposures in unrestricted areas that are well within the limits specified in 10 CFR 20.

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 primarily produced by the neutron irradiation of dissolved air in the reactor pool water. Exposure from N-16 is reduced using the pool water diffusers. 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 Section 11.2.3.1 it is demonstrated that airborne radioactivity released to unrestricted areas does not exceed 10CFR20.

11.1.1.2 Liquid Radioactive Sources

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, lowering of the pool level for maintenance, and draining of the demineralizer column in order to replace resins.

Potentially radioactive liquid waste is analyzed and checked for compliance with limits specified in 10CFR20.2003 prior to release to the sanitary sewer. Liquid radioactive waste may also be dewatered through evaporation, in which case the remaining residue would be handled as solid radioactive waste. The liquid waste holdup system consists of retention tank(s), and associated plumbing. The function of the liquid waste holdup system is to facilitate the holding of liquids for sampling, decay and subsequent disposal after activity levels are below limits specified 10CFR20. Normally, a single tank with a minimum capacity of 100 gallons provides sufficient holdup capacity for the liquid waste generated at the facility.

In the event that a regeneration is to be performed, additional holdup capacity is required. In such an instance, tank(s) large enough to hold the liquids from one complete regeneration (approximately 465 gallons) will be installed. A schematic drawing of the liquid waste holdup system required for regeneration is shown in Figure 11.1.

Table 11.1-Ar-41 Release.

Year	Activity (μCi)	Year	Activity (μCi)
1984	8,671.85	1994	65,830.75
1985	46,378.08	1995	55,564.34
1986	150,117.14	1996	25,470.70
1987	445,240.22	1997	100,822.82
1988	71,145.21	1998	91,560.61
1989	104,487.56	1999	60,788.05
1990	181,546.35	2000	45,241.78
1991	115,656.38	2001	43,628.97
1992	46,405.76	2002	48,718.75
1993	64,992.67	2003	74,447.60

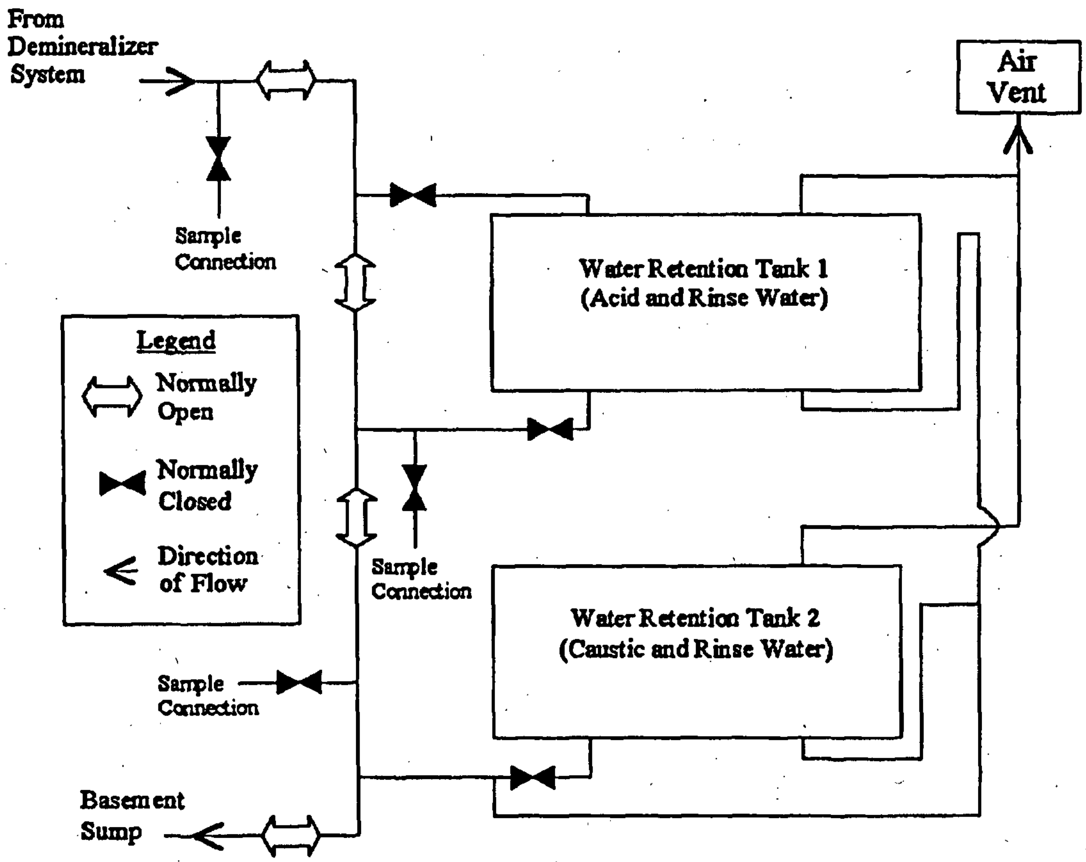


Figure 11.1-Liquid Waste Holdup System Required for Regeneration.

Holdup tank(s) located in the basement level of the reactor building are used to collect potentially contaminated liquid. Prior to initiating a regeneration, the holdup tank capacity will be checked and additional tanks installed as necessary to insure sufficient holdup capacity is available. The resin regeneration consists of 3 steps: backwash, regeneration of anion resins, and regeneration of cation resins. The amount of regeneration liquid is normally about 325 gallons followed by approximately 140 gallons of dilution water for a total of about 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 holdup time (if required) before release to the Rolla sanitary sewer. Typically, only traces of Na-24 or gross activity 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 Institute of Standards and Technology (NIST), traceable standard is used to calibrate the detector and determine the efficiency.

Normal reactor operations produce very little radioactive liquid waste. However, some of the research activities conducted within the facility are capable of generating such waste. Liquid waste drains from the reactor room and equipment areas into the lower level (basement) sump. The contents of the tank(s) are eventually released to the lower level sump, pumped to the middle level sump, and released into the sanitary sewer system if analysis shows that the limits of 10CFR20 will not be exceeded. See Table 11.2 for water release to the sanitary sewer.

Table 11.2-Water Release Summary.

Year	Gallons	Activity (μCi)
1985	3,255	48.22
1986	3,355	76.44
1987	6,975	162.58
1988	3,985	525.48
1989	1,430	26.59
1990	26,850	1194.66
1991	12,743	166.5
1992	13,777	449.58
1993	214	4.12
1994	0	0
1995	55	0.058
1996	0	0
1997	0	0
1998	0	0
1999	0	0
2000	0	0
2001	0	0
2002	0	0
2003	0	0

11.1.1.3 Solid Radioactive Sources

Sources of radiation directly related to reactor operations include radiation from the reactor core, ion exchange column, and radioactive gases. Solid radioactive waste is discussed in Section 11.2.

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.

Low-level solid waste generated as a result of reactor operations typically consists of ion exchange resins, filters, potentially contaminated paper and gloves, and small, activated samples from laboratory experiments. Waste is packaged in accordance with applicable NRC and U.S. Department of Transportation regulations, transferred to the campus Materials License, and then moved 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 11.3.

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. Irradiated reactor fuel is not considered "waste". The reactor fuel is owned by the Department of Energy. Irradiated and unirradiated fuel will ultimately be returned to possession of the Department of Energy in accordance with applicable regulations at that time.

Table 11.3-Solid Waste. (Activity in μCi)			
Year	Filters	Resins	Solid (excluding Filters and Resins)
1984	0	0	0
1985	0	0	0
1986	0	0	0
1987	0	0	0
1988	0	0	66
1989	0	0	0
1990	500	500	0
1991	0	0	0
1992	0	0	0
1993	0.134	4.468	11.98
1994	0	0	0
1995	0.008	4.86	0
1996	1.21	0.084	1.54
1997	0	0	0
1998	11.65	0	0
1999	0	0	0
2000	0.001	118	0
2001	0	0	0
2002	0	0	0
2003	0.112	0	0

11.1.2 Radiation Protection Program

The reactor is provided with health physics coverage from the Radiation Safety Office. The Radiation Safety Office at UMR is a section of Administrative Services (see Figure 11.2). Radiation safety at UMR is carried out by a part-time (0.1 FTE) Radiation Safety Officer, a Health Physicist and part-time Health Physics technician. 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 11.4 shows the results of area film badge monitors.

Table 11.4-History of Exposure in the Reactor Building. (mrem)				
Year	Control Room	Bridge	Office	Demineralization Area
1984	0	120	0	10
1985	0	120	0	0
1986	17	50	0	30
1987	0	110	10	30
1988	10	110	10	0
1989	0	140	0	0
1990	0	90	0	0
1991	10	20	20	40
1992	0	0	0	0
1993	0	0	0	40
1994	0	0	0	30
1995	0	0	0	0
1996	0	10	0	0
1997	0	0	0	0
1998	0	0	0	10
1999	0	0	0	10
2000	0	0	0	0
2001	0	0	0	0
2002	1	13	2	119
2003	3	43	2	163

11.1.2.1 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.

11.1.2.2 Health Physics Staff Qualifications

The qualifications for the Health Physicist and Radiation Safety Officer are a bachelor's degree in Chemistry, Health Physics, related area, or an equivalent combination of education and experience from which comparable knowledge and abilities can be acquired.

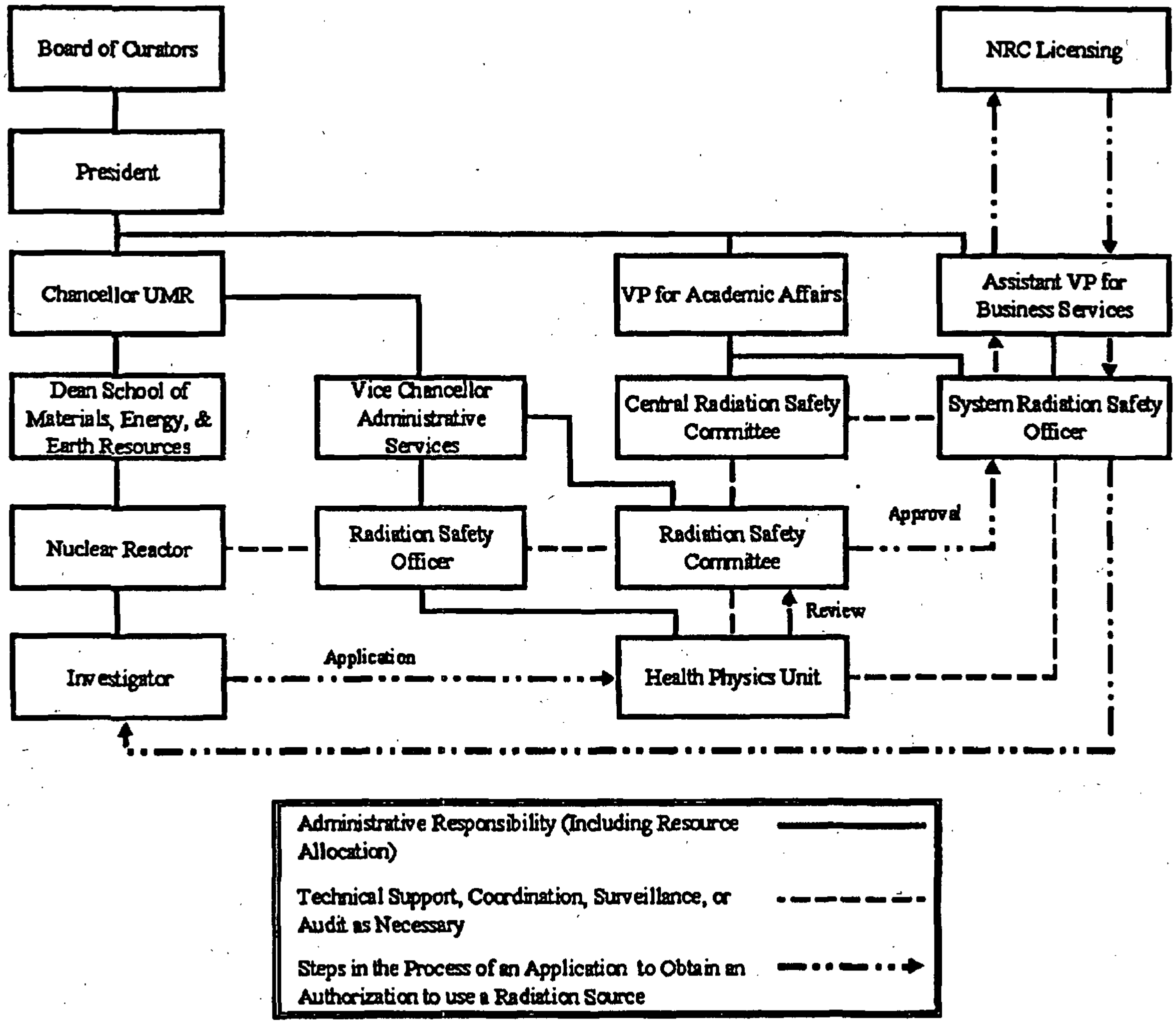


Figure 11.2-Management Structure for Radiation Safety Program at the University of Missouri.

11.1.2.3 Health Physics Procedures and Responsibilities

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

a) Monthly area radiation surveys,
b) Monthly contamination surveys,
c) Monthly air release calculations,
d) Monthly pool water analysis,
e) Semi-annual pool water H-3 analysis.
f) Semi-annual sealed source leak tests,
g) Annual Health Physics Instrument calibration of portable survey instruments,

h) By-Product material releases as required,
i) Health Physics coverage for various tasks as needed,
j) Waste water analysis (as needed).

Health physics procedures have been prepared and placed in the reactor SOP Manual that addresses the above-listed activities. Activated samples are monitored to ensure that they do not leave the reactor pool unless a sample is <100 millirem per hour at a distance of one foot. Samples reading greater than 100 millirem/hr at a distance of one foot will be monitored by the Health Physicist.

Administrative limits and action points are listed below:

a) External exposures are monitored by film badges and/or thermo-luminescent dosimeters (TLDs). 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 the University of Missouri Handbook of Radiological Operations. For tritium the maximum continuous body burden is 28 microcuries per liter. Tritium is detected by urinalysis. Health Physics will investigate any activities found over 0.28 microcuries per liter. For radioiodine the maximum continuous body burden for Iodine-125 is 0.58 microcuries and 0.15 microcuries for Iodine-131. Iodine is 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 that is not readily cleaned up, Health Physics must be contacted immediately to supervise the decontamination.

11.1.3 ALARA Program

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. UMR understand the ALARA principle to mean the following:

1. Merely restricting the dose to individuals or groups of individuals to below the maximum limit is not enough. Actions should be taken to decrease the dose to As Low As Reasonably Achievable.
2. Merely controlling the maximum dose to the individual is not sufficient; the collective dose to the group (measured in person-rems) also must be kept As Low As Reasonably Achievable.
3. "Reasonably achievable" is judged by considering the state of technology and the economics of improvement in relation to all of the benefits from these improvements.
4. Under the linear, nonthreshold concept, restricting the doses to individuals at a fraction of the applicable limit would be inappropriate if such action would result in the exposure of more persons to radiation and would increase the total person-rem dose [11-1].

The following steps are used to implement the ALARA Program:

1. Film Badges - furnished to students, faculty and staff working with or around radioactive material. Area radiation badges are also placed in the Reactor Building to monitor radiation levels.

2. Indoctrinations - Students, faculty, and staff receive indoctrination training at the Reactor and receive a tour prior to working there. In addition, anyone who receives 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 11.1.2

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

11.1.4 Radiation Monitoring and Surveying

11.1.4.1 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.

11.1.4.2 Pool Water Analysis

Pool water is drawn and analyzed on a monthly basis. The analysis is performed on either a sodium iodide or germanium detector connected to a multi-channel analyzer. The purpose of the analysis is to look for fission products such as Cs-137 and Co-60 in the pool water. The analysis procedure involves drawing a 1 liter pool water sample and counting the sample and then counting an NIST traceable standard to obtain the efficiency of the detector. The action level used to identify a leaking fuel element if any of the fission products Co-60 or Cs-137 are identified in the pool water sample is as follows:

1. Identify the leaking element.
2. Remove the element from the core.
3. Store the element in the fuel storage pool.

11.1.4.3 Swipe Tests

Health Physics performs swipe surveys in the Reactor Building to check for possible contamination on a monthly basis. Watman number 1 or equivalent filters are used to smear an area of approximately 100 cm^2. The swipes are counted for alpha and beta contamination on either a gas-proportional counter connected to a single channel analyzer or both an alpha meter and an end window Geiger-Mueller counter. In the past no major contamination has been found at the reactor facility.

11.1.4.4 Health Physics Instrumentation

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

1 Multi-channel pulse height analyzer,
2 Low background alpha-beta gas-proportional counter,
1. Scintillation counter,
2. Thin window G-M counters,
3. Pocket dosimeters,
4. Low and high range portable beta-gamma meters,
5. Portable neutron survey meter,
6. GM "friskers",
7. Fixed GM radiation area monitors
8. High-velocity portable air sampler.

The portable hand held beta-gamma instruments may be calibrated to an NIST traceable source according to ANSI N323-1978 [11-2] or may be sent out to a certified commercial vendor. The portable hand held neutron instrument may be calibrated with a PuBe source or may be sent out to a certified commercial vendor. The fixed G-M radiation area monitors are calibrated according to SOP using an NIST traceable source.

11.1.5 Radiation Exposure Control and Dosimetry

The reactor personnel and radiation worker monitoring program is based upon 10CFR20.1101 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 10CFR20.1301 limits to allow for minors. The UMR Reactor personnel annual exposure history for the last twenty years is given, in Table 11.5.

11.1.6 Contamination Control

UMRR has three instruments containing Geiger-Muller detectors for contamination control:

1) hand and foot monitor
2) frisker meter
3) portal monitor

Items 1and 2 are located in the reactor bay for personnel to used after completing labs where radioactive materials were handled. Personnel exiting the reactor bay pause in the portal monitor prior to leaving the Reactor Building. If radioactive contamination has been identified personnel are instructed to immediately contact reactor staff, which will start a decontamination process. If decontamination can not be completed adequately the reactor staff will request assistance from the Environmental Health and Safety Department.

Health Physics performs swipe surveys in the Reactor Building to check for possible contamination on a monthly basis. In the past no major contamination has been found at the reactor facility, see Section 11.1.4.3.

11.1.7 Environmental Monitoring

Environmental monitoring is accomplished from within the Reactor Building by film badges located in strategic areas. The results over the last 20 years are shown in Table 11.4. 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 10CFR20.1301 limits.

11.2 Radioactive Waste Management

Radioactive waste management at the UMRR facility has always operated within the realm of the ALARA program and below the limits presented in 10CFR20. The facility will continue to handle radioactive wastes in such a manner that the health and safety of staff, students, and the general public is of utmost concern.

Table 11.5-History of Occupational Personal Exposure Summary.				
Whole Body Exposure (mrem)				
Year	No Measurable Exposure	Less than 100	100-250	Greater than 250
1984	41	1	0	0
1985	34	1	0	0
1986	35	14	0	0
1987	61	8	0	0
1988	37	6	0	0
1989	45	2	0	0
1990	49	2	0	0
1991	45	5	0	0
1992	54	0	1	0
1993	51	0	0	0
1994	57	0	0	0
1995	64	1	0	0
1996	49	0	0	0
1997	48	0	0	0
1998	45	1	0	0
1999	49	1	0	0
2000	44	0	0	0
2001	58	15	0	0
2002	75	17	0	0
2003	24	66	0	0

11.2.1 Radioactive Waste Management Program

Because Ar-41 is the only airborne radionuclide released by the reactor to the environment during normal operations, the staff has reviewed the history, current practices, and future expectations of operations as regards to this radionuclide. The potential doses in unrestricted areas, as a result of actual releases of Ar-41, have never exceeded or even approached the limits specified in 10CFR20. Furthermore, computations of the dose beyond the limits of the reactor facility give reasonable assurance that the potential doses to the public as a result of Ar-41 release would not be significant even if there were major changes in the operating schedule of the UMRR.

The major radioactive waste generated by the reactor operations is activated gases, principally Ar-41. A limited volume of radioactive solid waste, principally spent ion exchange resins, is generated by reactor operations, and some additional solid waste is produced by the research programs involving the use of reactor facilities. Liquid radioactive waste is produced by regeneration of the resin bed in the water demineralizer system.

11.2.2 Radioactive Waste Controls

The waste management activities at the UMRR facility have been conducted and are expected to continue to be conducted in a manner consistent with 10CFR20 and with the ALARA principle. Among other guidance, the staff review has followed the methods of ANSI/ANS 15.11, "Radiological Control at Research Reactor Facilities" [11-3].

11.2.3 Release of Radioactive Waste

Radioactive waste resulting from reactor operations is 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.

11.2.3.1 Airborne Releases

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 Ar-41 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 Ar-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_0) of Ar-41 downwind from the point of discharge is given by:

$$C_0 = D \cdot v \cdot C_B$$

Where:

D = dilution factor (s/m^3)
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 [11-4].

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

Where:

c = an empirical constant (0.5)
$\bar{u}$ = average wind velocity (m/s)
A = cross-sectional area of the building (m^2)

The cross-sectional area of the Reactor Building is about 100 m^2. Using u = 1 m/s the building dilution factor is calculated to be 2 x 10^{-2} s/m^3. From the above relationship for C_0 the concentration of Ar-41 near the Reactor Building is calculated to be

$$\begin{aligned} C_0 &= 2 \times 10^{-2} \cdot 2.33 \cdot 4.5 \times 10^{-7} \\ &= 2.1 \times 10^{-8} \text{ microcuries/ml} \end{aligned}$$

This value is below the limit of 4.0 x 10^{-8} microcuries per milliliter for Ar-41 discharged into an unrestricted area as stated in 10 CFR 20.

To summarize the results of the analysis of airborne radioactivity at the UMRR data demonstrates that the major gaseous radioactivity is due to Ar-41. Furthermore, airborne radioactivity released to unrestricted areas does not exceed 10 CFR 20 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.

11.2.3.2 Solid Waste

The only solid waste generated as a result of reactor operations consists primarily of ion exchange resins and filters, potentially contaminated paper and gloves, and occasional small activated components. Some of the reactor-based research also results in the generation of solid low-level radioactive wastes in the form of contaminated paper, gloves, and glassware. This solid waste generation typically contains a few millicuries of radionuclides per year.

The solid waste is collected in specially marked containers. The solid waste is picked up by the health physics staff and held temporarily before being packaged and shipped to an approved disposal site in accordance with applicable regulations.

11.2.3.3 Liquid Waste

Liquid radioactive waste is produced by the regeneration of the demineralizer system, lowering of the pool level for maintenance, and draining of the demineralizer column in order to replace resins. The general philosophy of the facility administration has been to minimize liquid waste discharge. In recent years, the preference has been to replace resins as they become depleted rather than regenerating in order to minimize liquid waste. Liquid waste are analyzed to assure compliance with regulatory requirements and then released to the sanitary sewer system. Future liquid releases are expected to be minimal and are not expected to exceed the historical trends.

11.3 References

[11-1] UMR Handbook of Radiological Operations, 2nd ed. University of Missouri-Rolla, May 2003.

[11-2] ANSI N323-1978 "American National Standard Radiation Protection Instrumentation Test and Calibration" , published by The Institute of Electrical and Electronics Engineer, Inc, 345 East 47th Street, New York, NY 10017.

[11-3] ANSI/ANS 15.11, "Radiological Control at Research Reactor Facilities", published by the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, IL 60525 USA.

[11-4] Lamarsh, J.R., Introduction to Nuclear Engineering, Addison-Wesley pub., 1997.

12. CONDUCT OF OPERATIONS

12.1 Organization

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

12.1.1 Structure

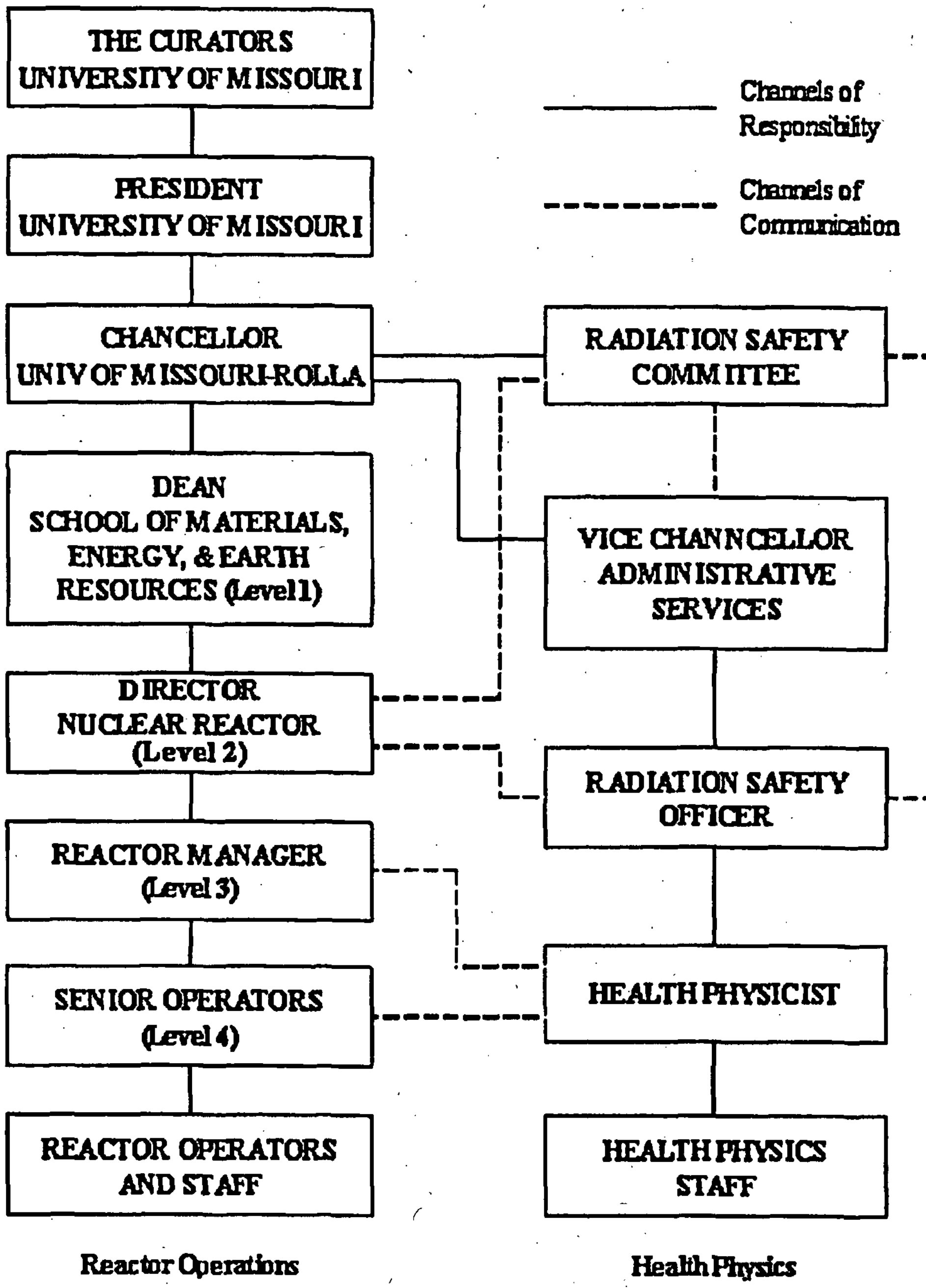
The Nuclear Reactor Facility is a part of the School of Materials, Energy & Earth Resources of the University of Missouri-Rolla. The organizational structure is shown in Figure 12.1.

12.1.2 Responsibilities

The Dean of the School of Materials, Energy & Earth Resources is the individual responsible for the reactor facility's licenses (Level 1). The Dean reports to the UMR Chancellor who is ultimately responsible for all activities at the UMR campus. The Chancellor reports to the President of the University of Missouri system, which include four separate campuses. The President reports to the Board of Curators of the University of Missouri system.

The Director of the Nuclear Reactor Facility is the contact person for the NRC and has overall responsibility for management of the facility (Level 2). The Reactor Manager (Level 2) shall be responsible for the day-to-day operation and for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license and the provisions of the Radiation Safety Committee.

The responsibilities of the Radiation Safety Committee are discussed in Section 12.2. The Radiation Safety Officer is responsible for the radiation safety of the entire campus, including the reactor facility. A Health Physicist, who is organizationally independent of the Reactor Facility operations group, as shown in Figure 12.1, is responsible for radiological safety at the facility. The Vice Chancellor for Administrative Affairs has administrative responsibility for the office of Occupational Health and Safety Services, which implements Health Physics coverage.



igure 12.1- Organizational Structure of the UM System Related to the UMRR Facility.

12.1.3 Staffing

When the reactor is operating the following staffing conditions shall be met:

a) At least two persons (one of who is a licensed Senior Reactor Operator) shall be present in the Reactor Building.

b) A licensed Reactor Operator or Senior Reactor Operator shall be present in the control room.

All rearrangements of the core, fuel movement, and associated Health Physics monitoring, shall be either performed or supervised by a licensed Senior Reactor Operator.

12.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall meet or exceed the requirements of American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4(5).

12.2 Review and Audit - The Radiation Safety Committee

The reactor is operated under NRC License R-79 initially granted in 1961. As required by the license, a reactor advisory committee was appointed at the time and, as time went by, has been called different names. Its present title is the UMR Radiation Safety Committee.

The UMR Radiation Safety Committee has the dual responsibility of:

(1) Reviewing and making recommendations concerning experimental and operational activities of the UMR Nuclear Reactor.

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

The Committee is appointed by the Vice Chancellor of Administrative Affairs. 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 reactor-related issues, which are forwarded to it. This review shall encompass matters concerning health and safety.

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

3) Review special reports issued by the Reactor Staff following any significant malfunctions, violations, or accidents.

The Committee shall meet at least once each calendar year. 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.

12.2.1 Composition and Qualifications

The Committee shall be composed of at least three members, one of whom shall be the Radiation Safety Officer of the campus. No more than two members will be from the organization responsible for reactor operations. At least three members of the committee shall collectively represent a broad spectrum of expertise in areas relating to reactor safety and research use of radioisotopes. Qualified alternates, approved by the Committee Chairman, may serve in the absence of regular members.

12.2.2 Charter and Rules

(1) A quorum of the Committee shall consist of at least one half of the voting members where the operating staff does not constitute a majority.

(2) The Committee shall meet at least once each calendar year. Minutes of all meetings shall be disseminated to Committee members and to other responsible personnel as designated by the Committee Chairman.

(3) The Committee shall have a written statement, or charter, defining such matters as the authority of the Committee, the subjects within its purview, and other such administrative provisions as are required for effective functioning of the Committee.

12.2.3 Review Function

As a minimum, the responsibilities of the Radiation Safety Committee include:

(1) Review in accordance with 10CFR50.59 of untried experiments and tests that are significantly different from those previously used or tested in the reactor, as determined by the Facility Director.

(2) Review in accordance with 10CFR50.59 of changes to the reactor core, reactor systems or design features that may affect the safety of the reactor.

(3) Review of all proposed amendments to the facility license and Technical Specifications.

(4) Review reportable occurrences and the actions taken to identify and correct the cause of the occurrences.

(5) Review significant operating abnormalities or deviations from normal performance of facility equipment that affect reactor safety.

This same Committee may have other responsibilities, for example oversight of the campus materials license. The Committee may assign sub-committees to act on its behalf.

12.2.4 Audit Function

The Radiation Safety Committee will arrange for a knowledgeable and impartial individual (or individuals) to review reactor operations and audit the operational records for compliance with reactor procedures, Technical Specifications, and license provisions. An impartial individual is one who is not directly affected by the findings or recommendations of the audit and has no reason to be biased concerning the review. These audits shall be performed annually.

12.3 Procedures

The reactor is operated in accordance with written procedures established under the approval of the Reactor Director. These procedures cover the following items:

1) Startup, operation and shutdown of the reactor.

2) Installation, removal and movement of fuel elements, control rods, experiments, and experimental facilities in the core region.

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

4) Periodic surveillance required by Technical Specifications for reactor instrumentation and safety systems.

5) Radiation control procedures.

6) Administrative control for experiments that could affect reactor safety or core reactivity.

7) Implementation of emergency plan.

Substantive changes to approved procedures having safety significance shall be made only with the approval of the Radiation Safety Committee. The Radiation Safety Committee should evaluate the change under the provisions of 10CFR 50.59 to determine if an unreviewed safety question is involved. Changes that do not change the original intent of the procedures may be made with the approval of the Director.

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 review of the Radiation Safety Committee. However, in the final analysis, the safe operation of the reactor is dependent upon the reactor staff and their exercise of good judgment.

12.4 Required Actions

12.4.1 Action to be taken in the Case of Safety Limit Violation

In the event that the safety limit is exceeded, the following actions will be taken:

1) The reactor shall be shut down, and reactor operations shall not be resumed until authorized by the NRC.

2) The safety limit violation shall be promptly reported to the Director of the Reactor Facility.

3) The safety limit violation shall be reported to the NRC (per Chapter 14, Section 14.7.2).

4) A safety limit violation report shall be prepared. The report shall describe the following:
   a) Applicable circumstances leading to the violation including, when known, the cause and contributing factors.
   b) Effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public.
   c) Corrective action to be taken to prevent recurrence.

5) The report shall be reviewed by the Radiation Safety Committee and any follow-up report shall be submitted to the NRC when authorization is sought to resume operation of the reactor.

12.4.2 Actions to be taken in Response to Certain Other Occurrences

The following actions shall be taken if an event of the type identified in Section 12.5.2(1)b or 12.5.2(1)c occurs:

1) The reactor shall be shut down.

2) The occurrence shall be reported to the Director and to the NRC (see Chapter 14, Section 14.7.2).

3) Operations shall not be resumed until authorized by the Reactor Director.

4) The Radiation Safety Committee at their next scheduled meeting shall review the occurrence.

12.5 Reports

12.5.1 Operating Reports

An annual progress report will be made by May 30 of each year to the NRC Document Control Desk with a copy to the Regional Administrator, which provides the following information:

1) A narrative summary of reactor operating experience including the energy produced by the reactor or the hours the reactor was critical, or both.

2) The unscheduled shutdowns including, where applicable, corrective action taken to preclude recurrence.

3) Tabulation of major preventive and corrective maintenance operations having safety significance.

4) A summary of changes to the facility or procedures, which affect reactor safety, and performance of tests or experiments carried out under the provisions of 10CFR 50.59.

5) A summary of the nature and amount of radioactive effluents released or discharged from the facility. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25 percent of the concentration allowed, a statement to this effect is sufficient.

6) A summary of exposures received by facility personnel and visitors where such exposures are greater than 25 percent of that allowed.

12.5.2 Special Reports

(1) There shall be a report not later than the following working day by telephone to the NRC Project Manager and the regional NRC office, to be followed by a written report that describes the circumstances of the event within 14 days of any of the following (see Chapter 14, Section 14.6.7.2):

a) Violation of safety limits
b) Release of radioactivity from the site above allowed limits
c) Any of the following:
   i) Operation with actual safety-system settings for required systems less conservative than the limiting safety-system settings.
   ii) Operation in violation of limiting conditions for operation (see Section 3) unless prompt remedial action is taken.
   iii) A reactor safety system component malfunctions which renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns.

NOTE: Where components or systems are provided in addition to those required by the technical specifications, the failure of the extra components or systems is not considered reportable provided that the minimum number of components or systems, specified or required, perform their intended reactor safety function.

iv) An unanticipated or uncontrolled change in reactivity greater than one dollar. Reactor trips resulting from a known cause are excluded
v) Abnormal and significant degradation in reactor fuel or cladding, coolant boundary, which could result in exceeding prescribed radiation exposure limits of personnel or environment.
vi) An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

(2) A written report shall be submitted within 30 days to the NRC Document Control Desk, with a copy to the Regional Administrator of the following:
a) Significant changes in the transient or accident analyses as described in the SAR.
b) Permanent changes in facility organization involving Level 1 or 2 personnel.

12.6 Records

Records may be in the form of logs, data sheets, or other suitable forms, including digital computer files. The required information may be contained in single or multiple records, or a combination thereof.

The following records are to be retained for the life of the facility:

1) gaseous and liquid radioactive effluents released to the environment,

2) radiation exposures for all personnel monitored,

3) Updated, corrected, and as-built drawings of the facility.

The following records are to be retained for at least five years:

1) Normal reactor facility operation (but not including supporting documents such as checklists, log sheets, etc., which shall be maintained for a period of at least one year)

2) Principal maintenance operations

3) Reportable occurrences

4) Surveillance activities required by the Technical Specifications

5) Reactor facility radiation and contamination surveys where required by applicable regulations

6) Experiments performed with the reactor

7) Fuel inventories, receipts, and shipments

8) Approved changes in operating procedures

9) Records of meeting minutes and audit reports of the Radiation Safety Committee

Records of retraining and requalification of licensed operators are maintained at all times the individual is employed or until their license is renewed. Following a license renewal, records subsequent to the renewal will be maintained.

12.7 Emergency Planning

The UMR Reactor facility has an NRC approved Emergency Plan. The 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.

12.8 Security Planning

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 organization 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.

12.9 Operator Training and Requalification

The UMR Reactor Facility has an NRC approved Operator Requalification Program that all licensed reactor operators and senior reactor operators participate in. Persons who are preparing to take the NRC operators licensing examination participate in the same training program, as well as receive intensive "hands-on" reactor operations training at the console.

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) A biennial written examination 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 ensures that the operators maintain his/her competency in manipulating the controls and in operating all apparatus and mechanisms required by license; that the operators are cognizant of all design, procedure and license changes implemented

during the Requalification period; and that the operators have a thorough understanding of all abnormal and emergency procedures.

3) Periodic observation and evaluation is used to assess the performance of the operators.

12.10 Environmental Reports

The UMRR is licensed to operate at 200 kW and operates approximately 10-11 MW-hours annually performing instruction, training and research for the UMR Campus and local community. The average annual waste releases from the UMRR are 1) Ar-41, less than 100 mCi, 2) liquids less than 0.2 mCi, and 3) solids less than 60 μCi. The release of these levels of activity causes no significant impact to the environment.

13. ACCIDENT ANALYSIS

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

13.1 Accident Scenarios and Analysis

In the following subsections various accident scenarios have been categorized according to their corresponding accident type as defined by the Nuclear Regulatory Commission (NRC) in NUREG 1537 Part 1. In each case the most limiting accident scenario for each accident type has been analyzed for the potential hazards posed to the health and safety of the public and reactor staff.

13.1.1 Maximum Hypothetical Accident - [redacted]

In this subsection an analysis is performed to [redacted].

The limiting criterion used in the analysis of [redacted].

As a conservative assumption, the irradiation time was considered to be infinite. Therefore, the fission product 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 x 10^3 m^3 (6.0 x 10^4 ft^3).

Table 13.1 on the following page presents relevant data for the primary fission product gases of interest [13-3] including half-life, saturation activities following an infinite irradiation and initial building concentrations (χ_o^{Bld}). From this data the external dose rate (mrem/hr) due to γ and β radiation was calculated using these relationships [13-4]:

$$\dot{D}_\gamma = 9.43 \times 10^{11} \cdot \chi \cdot \overline{E}_\gamma$$

Where:

$\dot{D}_\gamma$ = External dose rate due to gamma radiation (mrem/hr)
χ = Radionuclide concentrations (Ci/cm^3)
$\overline{E}_\gamma$ = Average gamma energy per disintegration (MeV)

$$\dot{D}_\beta = 8.24 \times 10^{11} \cdot \chi \cdot \overline{E}_\beta$$

Where:

$\dot{D}_\beta$ = External dose rate due to beta radiation (mrem/hr)
χ = Radionuclide concentrations (Ci/cm^3)
$\overline{E}_\beta$ = Average beta energy per disintegration (MeV)

The dose rate to the thyroid (rem/hr) due to the inhalation of radioiodines is given by:

$$\dot{D}_T = DCF \cdot B \cdot \chi$$

Where:

$\dot{D}_T$ = Dose rate to the thyroid (rem/hr)
DCF= Dose-conversion factor for the thyroid (rem/Ci)
B= Breathing rate (cm^3/hr)
χ= Radioiodine concentration (Ci/cm^3)

Table 13.1 - [redacted]

Experiment.

Isotope	Half-Life (min)	[redacted]		[redacted]	
		Asat (Ci)	χ_o^{Bld} (Ci/cm3)	Asat (Ci)	χ_o^{Bld} (Ci/cm3)
I-131	11577.60	2.450E-02	7.200E-12	2.450E+00	7.206E-11
I-132	83.40	3.710E-02	1.090E-11	3.710E+00	1.091E-10
I-133	1248.00	5.480E-02	1.610E-11	5.480E+00	1.612E-10
I-134	52.60	6.060E-02	1.780E-11	6.060E+00	1.782E-10
I-135	394.20	5.060E-02	1.490E-11	5.060E+00	1.488E-10
Kr-83m	111.60	5.900E-03	3.471E-12	5.900E-01	3.471E-10
Kr-85m	268.80	1.270E-02	7.470E-12	1.270E+00	7.471E-10
Kr-85	5643550.80	2.530E-03	1.490E-12	2.530E-01	1.488E-10
Kr-87	76.20	2.000E-02	1.180E-11	2.000E+00	1.176E-09
Kr-88	170.40	3.120E-02	1.840E-11	3.120E+00	1.835E-09
Kr-89	3.15	3.960E-02	2.329E-11	3.960E+00	2.329E-09
Xe-131m	17136.00	2.530E-04	1.488E-13	2.530E-02	1.488E-11
Xe-133m	3153.60	1.350E-03	7.940E-13	1.350E-01	7.941E-11
Xe-133	7549.92	5.480E-02	3.220E-11	5.480E+00	3.224E-09
Xe-135m	15.30	1.770E-02	1.040E-11	1.770E+00	1.041E-09
Xe-135	546.00	5.230E-02	3.080E-11	5.230E+00	3.076E-09
Xe-137	3.82	5.310E-02	3.124E-11	5.310E+00	3.124E-09
Xe-138	14.10	5.570E-02	3.276E-11	5.570E+00	3.276E-09

The standard breathing rate recommended [13-2] is 1.25×10^{6} cm^3/hr. The Thyroid dose-conversion factors are given in Table 13.2 below.

Table 13.2 - Iodine Dose-Conversion Factors for the Thyroid. [13-2]

Isotope	DCF (rem/Ci)
I-131	1.00E+05
I-132	6.60E+03
I-133	1.80E+05
I-134	1.10E+03
I-135	4.40E+06

The calculated dose rates for both cases, 1 W and 100W, are presented in Table 13.3. Also included in the table are the associated γ and β radiation energies and the total dose rate for each of the determined dose rates.

Table 13.3 - Dose Rates in the Reactor Building Following a [redacted].

Isotope	E-gam (MeV)	E-bet (MeV)	[redacted]			[redacted]		
			D-gam (mrem/hr)	D-Bet (mrem/hr)	D-Thy (rem/hr)	D-gam (mrem/hr)	D-Bet (mrem/hr)	D-Thy (rem/hr)
I-131	3.71E-01	1.97E-01	2.52E+00	1.17E+00	9.00E+00	2.52E+01	1.17E+01	9.00E+01
I-132	2.40E+00	4.48E-01	2.47E+01	4.03E+00	9.00E-02	2.47E+02	4.03E+01	9.00E-01
I-133	4.77E-01	4.23E-01	7.25E+00	5.60E+00	3.62E+00	7.25E+01	5.60E+01	3.62E+01
I-134	1.94E+00	4.55E-01	3.26E+01	6.65E+00	2.45E-02	3.26E+02	6.65E+01	2.45E-01
I-135	1.78E+00	3.08E-01	2.50E+01	3.78E+00	8.16E-01	2.50E+02	3.78E+01	8.16E+00
Kr-83m	2.60E-03	1.03E-02	8.51E-03	2.95E-02		8.51E-01	2.95E+00	
Kr-85m	1.51E-01	2.23E-01	1.06E+00	1.37E+00		1.06E+02	1.37E+02	
Kr-85	2.11E-03	2.23E-01	2.12E-01	2.74E-01		2.12E+01	2.74E+01	
Kr-87	1.37E+00	1.05E+00	1.52E+01	1.02E+01		1.52E+03	1.02E+03	
Kr-88	1.74E+00	3.41E-01	3.02E+01	5.17E+00		3.02E+03	5.17E+02	
Kr-89	1.60E+00	1.33E+00	3.52E+01	2.56E+01		3.52E+03	2.56E+03	
Xe-131m	2.00E-02	1.44E-01	2.81E-03	1.72E-02		2.81E-01	1.72E+00	
Xe-133m	3.26E-01	1.55E-01	2.44E-01	1.01E-01		2.44E+01	1.01E+01	
Xe-133	3.00E-02	1.46E-01	9.11E-01	3.87E+00		9.11E+01	3.87E+02	
Xe-135m	4.22E-01	9.74E-02	4.14E+00	8.35E-01		4.14E+02	8.35E+01	
Xe-135	2.46E-01	3.22E-01	2.90E+01	8.17E+00		2.90E+03	8.17E+02	
Xe-137	1.50E-01	1.37E+00	4.41E+00	3.52E+01		4.41E+02	3.52E+03	
Xe-138	1.10E+00	8.00E-01	3.40E+01	2.16E+01		3.40E+03	2.16E+03	
Total:			2.47E+02	1.34E+02	1.36E+01	1.64E+04	1.15E+04	1.36E+02

With γ-dose rates as high as [redacted] case and significantly higher for the [redacted] case, any one of the 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 past trials, it is known that the reactor building can be evacuated within three minutes. For the purpose of this analysis it is assumed that the time elapsed between the release of radioactivity and the end of the evacuation is five minutes. Therefore it is assumed that the exposure time to members of the reactor staff is five minutes. For the [redacted] case it is assumed that the experiment would be run at the reactor core within the pool water. As previously mentioned, in the calculation of the iodine concentration in the reactor building air a retention factor of 10 was assumed for the reactor pool. In Table 13.4 the radiation doses to reactor staff is presented assuming the five minute total exposure time for both the [redacted] cases.

Table 13.4-Radiation Doses to Reactor Staff for a [redacted].			
Case	Whole-Body Dose (rem)	Skin Dose (rem)	Thyroid Dose (rem)
[redacted]	2.06E-02	1.12E-02	0.93
[redacted]	1.37E+00	9.60E-01	11.3

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 twenty-four hours. Since the reactor building does not have any windows and has only a few openings (for fans, air conditioning units, etc.), this assumption is considered to be conservative. All openings for the reactor building could easily be sealed from the outside in the case of an emergency. Another conservative assumption was made in that no radioactive decay, hence no decrease in the source strength, was taken into account while calculating the dose rates outside of the reactor building. The radionuclide concentration just outside of the reactor building was calculated using the building wake dispersion factor of 2.0×10^{-2} sec/m^3 (5.66×10^{-4} s/ft^3).

The whole body dose to an individual located just outside the reactor building was calculated using the aforementioned methodology. The resulting whole-body dose associated with the [redacted]; the whole-body dose was lower by about two magnitudes for the [redacted] case. This dose is a factor of two lower than the 10CFR Part 20 annual limit of 500 mrem exposure to individuals in unrestricted areas.

It can be concluded from the analysis presented here that experiments using fissile materials can be irradiated at the UMRR within the power limits analyzed in this subsection. The analysis has shown that there is no undue hazard to the general public or to the reactor staff in the hypothetical case of a failed experiment as postulated and analyzed.

13.1.2 Insertion of Excess Reactivity

The [redacted] reactivity insertion scenario identified involves the [redacted] into a critical or barely subcritical core. Technical specifications limit excess reactivity during routine operations to 1.5% Δk/k. Experiments at the Curtiss-Wright Research Reactor [13-5] have shown that the worth of a fuel element at the core periphery is less than 1.5% Δk/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 [redacted].

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% Δk/k. Therefore, an instantaneous insertion of the excess reactivity larger than 1.5% Δk/k has been excluded from further analysis. Technical Specifications state that the reactor can only be operated when all lattice positions internal to the fuel boundary are occupied; assuring that an element can only be added to the core periphery. Technical Specifications also require a licensed operator to visually confirm that there are no unoccupied internal lattice positions prior to taking a new core configuration critical.

A hypothetical accident is postulated assuming that a fuel element is placed next to the reactor core, resulting in a positive step reactivity insertion of 1.5% Δk/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 approximated from the expression [13-6]:

$$\tau \cong \frac{l_p}{\rho_0 - \beta}$$

Where:

τ = reactor period
l_p = prompt neutron lifetime (5×10^{-5} sec)
ρ_0 = initial reactivity
β = delayed neutron fraction (0.0065).

The corresponding reactor period is about 6 msec.

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. The transient code PARET [13-7] has been used to analyze this accident. The reactor was assumed to be critical at low power (1W). All of the reactor protection systems were assumed to be inoperable. At t=0, a step insertion of 1.5% Δ k/k occurs, after which the power increases steeply until a peak power of [redacted] is reached at about t=0.144 seconds. At this time, energy released during the transient amounts to about [redacted]. A strong negative feedback caused by moderator voiding (about -0.70% Δk/k) and the Doppler Effect (about -0.16% Δk/k) reverses the sign of the period and brings the reactor to delayed supercritical. From this time on, the reactor power rapidly decreases and the transient quickly dies away. The maximum fuel centerline temperature is about [redacted] which is still distinctly below the melting temperature of the cladding. The maximum fuel temperature occurs at about 0.16 seconds. In the reactor average channel, a fuel temperature of about 245°C (473°F) is reached. At this point, both temperatures start to decrease with time and with the reactor now being subcritical at k_{eff} = 0.9981.

The results of the theoretical analysis are supported by a large collection of data from excursion experiments performed at the BORAX and SPERT facilities [13-8, 13-9, and 13-10]. In particular, SPERT-1 experiments using the DU-12/25 core are applicable to the UMR Reactor since the fuel geometry and composition are very similar [13-10]; comparison is given in Table 13.5.

In the series of SPERT experiments, there was an experiment in which the induced reactor period was 6 milliseconds. The total energy released in the excursion was 13.2 MW-s. Onset of the self-limiting mechanisms occurred when about 7.2 MW-s of the thermal energy was generated. No damage to the fuel cladding was observed [13-5]. It was concluded that the mechanisms responsible for self-limiting the power excursion were fuel and moderator thermal expansion and boiling (the latter being the dominant shutdown mechanism). This finding is consistent with the results of our theoretical analysis.

Table 13.5-Comparison of UMRR and SPERT-1 Fuel Geometry and Composition.

Description	UMRR LEU Plate	SPERT-1 Plate
Active Plate Length (cm)	[redacted]	61.00
Plate Width (cm)	[redacted]	7.60
Plate Thickness (cm)	[redacted]	0.15
Water gap (cm)	0.31	0.45
Fuel Composition	[redacted]	U-Al
Enrichment (%U-235)	[redacted]	100.00
Weight fraction of U	[redacted]	0.24
Thickness (mm)	0.51	0.51
Cladding Composition	6061Al	Al
Cladding Thickness (mm)	0.38	0.51

Both the theoretical and experimental analyses have shown that the postulated reactivity insertion accident can safely be terminated by a self-limiting shutdown mechanism. 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 (together with Doppler feedback) becomes the rapid and dominating shutdown factor. Therefore, such an accident can be terminated even if the safety instrumentation is inoperable.

In addition to this inherent safety feature, the following administrative steps designed to prevent such an accident have been established in the Standard Operating Procedures:

a) Fuel handling is done in accordance with written procedures.
b) Loadings are planned to include the sequence of loading and positions of individual elements. A loading schedule is prepared prior to commencement of loading.
c) Loading operations are performed under the direct supervision of a licensed Senior Operator.
d) Fuel handling tools are kept locked with the keys secured to prevent unauthorized movement of fuel.

13.1.3 Loss of Coolant

The reactor pool is constructed of reinforced concrete set in bed rock to resist the most severe earthquake. The pool is designed to prevent unintentional drainage and has no drains. The possibility of a sudden loss of coolant is considered to be extremely remote. None-the-less, the accident scenario considered here assumes that somehow all of the pool water drains from the pool instantaneously while the reactor is operating at full power. In such a case, the reactor would shut down immediately due to loss of moderator. Of primary concern is the removal of decay heat during and after loss of coolant. 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 as shown in Figure 13.1.

Decay heat generation after reactor shutdown is a function of operating history and shutdown time. The most conservative approach is to assume that the reactor has been operating at full power for an infinite time prior to the accident. Decay heat generation can be approximated using the Way-Wigne equation [13-11] for an infinite run:

$$P = 0.0622 \cdot t^{-.02} \cdot P_0$$

Where:

P = reactor power after time t
t = time prior to shutdown
P_0 = initial reactor power

If the pool water is assumed to "free fall" out of the pool by gravity, it would take approximately 1 second. Substituting t=1 second into the decay heat equation yields a decay heat of 12.4 kW.

Decay heat diminishes with time as shown in Figure 13.2. Assuming decay power remains constant at 12.4 kW, the corresponding cladding temperature would be about 410°C (770°F) according to Figure 13.1. In reality, the decay power rapidly decreases as shown in Figure 13.2, so the actual maximum clad temperature would be lower. After 1 minute of cooling the power falls to about 5.5 kW and has an associated clad temperature of about 200°C (392°F). This is well below the melting temperature for the aluminum cladding. 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.

This analysis demonstrates that no fuel damage would result from an instantaneous loss of pool water. In any accident which is reasonably conceivable, the leakage of water from the reactor pool is expected to be rather slow. Pool water make-up systems could be used to replenish leaking pool water.

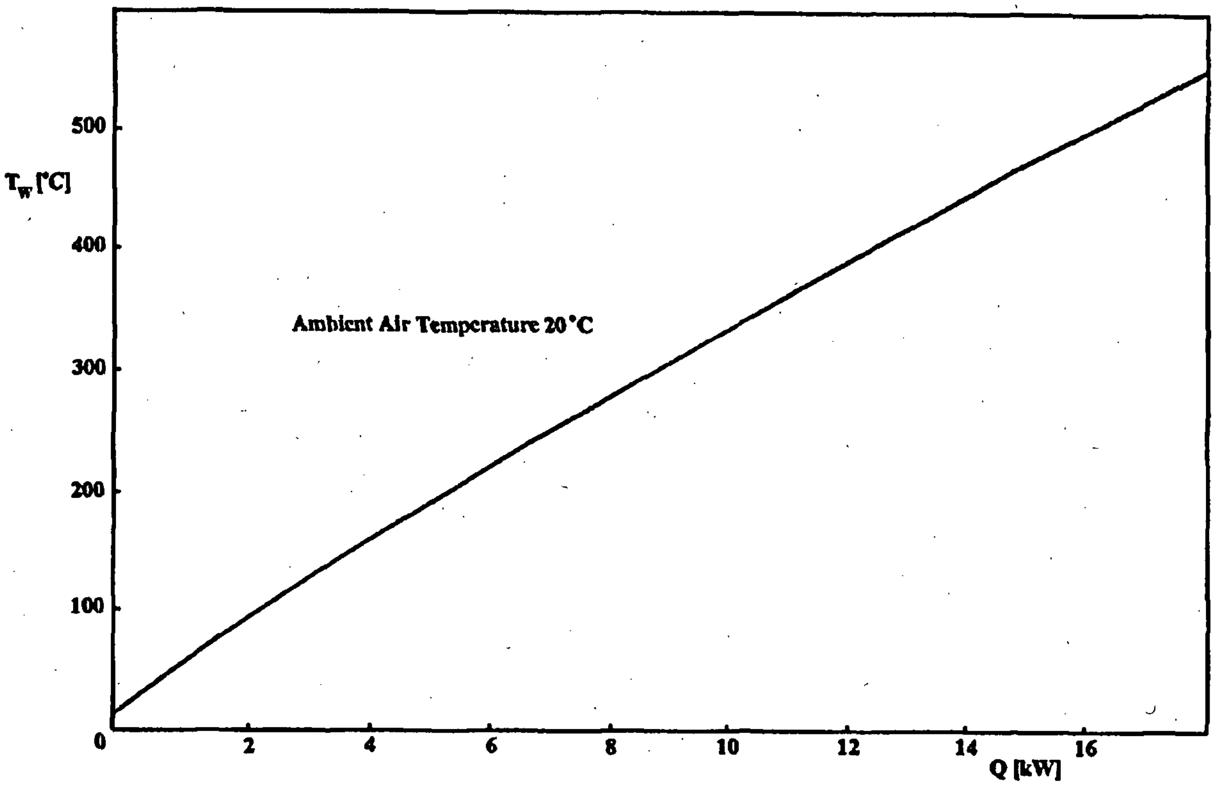


Figure 13.1-UMRR Cladding Temperature vs. Core Power Dissipated by Air Convection.

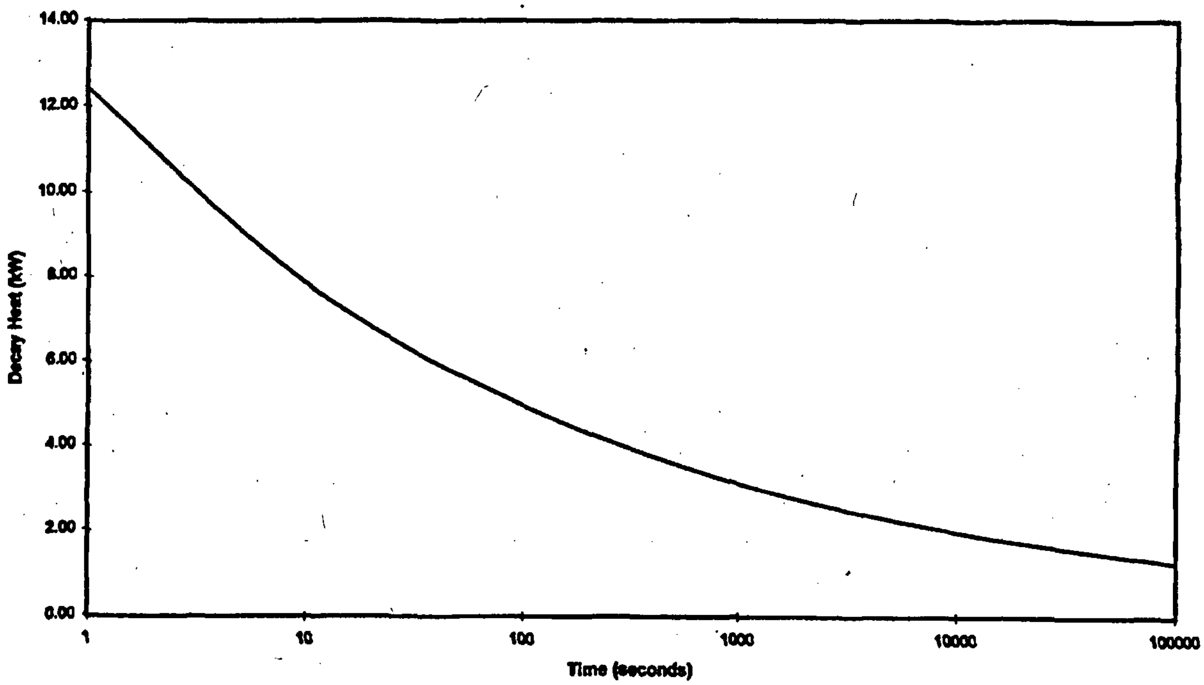


Figure 13.2-Decay Power after Infinite Irradiation.

13.1.4 Loss of Coolant Flow

No credible accident scenario could be envisioned in which the coolant channels of the reactor core could become blocked. This is due in part to the low rates of flow associated with natural convection, a cross flow provided by the demineralizer system, and the generally small size of the coolant channels. Standard operating procedures require the reactor core be inspected multiple times prior to operation. These inspections would prevent a large obstruction to the coolant channels while the reactor is operating.

13.1.5 Mishandling or Malfunction of Fuel

Fuel element maneuvers are conducted in the reactor pool under a sufficient depth of water for radiation shielding purposes. The elements are removed from the core and moved into the storage pit, one at a time, using a hand-held fuel handling tool. A fuel element weighs about [redacted] in air and only about [redacted] in water. Even if a 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 through water in an upright position.

The UMRR Standard Operating Procedures defines the following administrative steps intended to prevent a fuel handling mishap:

(1) Fuel handling is done in accordance with written procedures.
(2) Fuel handling is performed under the direct supervision of a Senior Operator.
(3) Fuel handling tools are kept locked to prevent unauthorized movement of fuel.

13.1.6 Experiment Malfunction

13.1.6.1 Flooding of an Irradiation Facility

Two special elements, the isotope production element and the core access element, can be used as air-filled irradiation facilities (see descriptions in Section 4). Both elements have outer dimensions similar to a fuel element and are made with graphite clad in aluminum with a hollowed center cavity. An accident scenario can be postulated where the facility suddenly develops a leak and is instantaneously filled with water. The sudden replacement of the voided space would cause a stepwise reactivity insertion. Its magnitude and sign depends on the void volume being replaced and its position relative to the core.

Flooding of the isotope production or core access element positioned in a central position of the HEU core has been shown [13-12] to cause a reactivity change of about -0.1% $\Delta k/k$. If the element were located at the core periphery, its flooding has been shown [13-12] to cause a reactivity change of about +0.2% $\Delta k/k$. This is consistent with the results of void experiments routinely performed in the past.

A code, based on neutron diffusion theory [13-13], has been used to calculate the reactivity worth of a voided volume at the HEU core periphery. The results compare within 0.05% Δk/k with the experimental data [13-14]. The same code has been used to calculate the reactivity change for flooding of a special element in the LEU core [13-15]. Flooding of an element in a central position of the LEU core was determined to cause a reactivity change of about -0.5% Δk/k. At the core periphery, the reactivity change was calculated to be about -0.1% Δk/k. In either case, if the element were flooded core reactivity would decrease.

13.1.6.2 Failure of a Moveable Experiment

The maximum reactivity worth of a movable experiment is limited by Technical Specifications to 0.4% Δk/k. In the following analysis it is assumed that an experiment worth -0.4% Δk/k suddenly falls away from the core causing a sudden stepwise reactivity insertion of +0.4% Δk/k. It should be noted that such a scenario is highly unlikely under current operational practice. Other assumptions made in the analysis are as follows:

1) The reactor is operating at a power is 200 kW.
2) The most reactive control rod cannot be scrammed (stuck rod).
3) All other control rods are fully withdrawn (low differential reactivity worth).
4) The power excursion does not start to reverse until the reactor is brought back to critical.
5) No mitigation by thermal feedback effects are taken into account.

Such a reactivity insertion would cause a prompt jump in power that would trip the 5 second period SCRAM. The subsequent period would be about 5.6 seconds [13-4].

In order to return the reactor to critical the two scramming control rods would need to insert approximately 4 inches (based on rod worths for a typical core). The time required for the rods to drop 5 inches has been estimated at 325 msec assuming a magnet release time of 25 msec (an average value based on periodic rod drop measurements) and a free fall time of 300 msec. The free fall time has been calculated assuming Newtonian Free fall with a drag force proportional to velocity. Technical Specifications require the rod drop time to be less than or equal to 1 second. The drag force was calculated to produce a total rod drop time of 1 second.

The power excursion can be estimated using the "prompt jump" approximation [13-16] given by:

$$P = \frac{\beta}{\beta - \rho} P_0 e^{t/\tau}$$

Where:

- β = effective delayed neutron fraction (0.0079)
- P_o = initial power
- τ = reactor period
- t = time
- ρ = core reactivity

At t = 0.325 seconds, reactor power is calculated to be [redacted].

The immediate prompt jump takes reactor power to about [redacted] while the additional power increase associated with the rod insertion time only accounts for an additional 45 kW. At times greater than 325 msec, the reactor power would decrease rapidly due to continued rod insertion. In reality, the actual peak power reached would be below the calculated value if thermal feed back mechanisms were taken into account.

In addition to the "Period < 5 sec" scram, the "150% Full Power" scram would activate. Technical Specifications require the "150% Full Power" scram as a Limiting Condition of Operation. Therefore, both redundancy and diversity are available to terminate the power excursion.

The heat flux expected in the hot channel at a reactor power of 450 kW is less than 5 W/cm^2. In Section 4.6 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 4.6 were derived for steady-state heat transfer. However, in the accident discussed above, the reactor power of 450 kW is only an instantaneous power peak. Immediately after the reactor scrams, power would drop sharply (prompt drop) to less than 200 kW and would then continue to decay. Therefore, as a result of the decreasing power, the cladding wall temperature in the hot channel during the power excursion would remain significantly below the saturation temperature. Consequently, the safety margin available between the wall temperature and the melting temperature of cladding is even larger than the steady-state heat transfer calculation used in this analysis indicates. It is concluded that the failure of a moveable experiment could not endanger the integrity of the reactor fuel.

13.1.7 Loss of Normal Electrical Power

Loss of electric power to the facility creates no hazard. The nuclear instrumentation initiates a reactor SCRAM upon the loss of building power. Electric power is not required for the continued safe shutdown of the reactor. Personnel safety lighting is installed at various locations throughout the facility in order to provide a minimal amount of lighting to allow personnel to safely move about the building.

13.1.8 External Events

The reactor pool is mostly underground and is constructed of reinforced concrete. A direct hit by a tornado on the facility would not be expected to result in core damage as the core is located near the bottom of the pool and is afforded protection from the reactor bridge and superstructure. A tornado would not be expected to damage the pool integrity.

An earthquake could, at worst, destroy the pool integrity causing a massive loss of pool water. Section 13.1.3 shows that an instantaneous loss of pool water would not result in core damage.

13.1.9 Mishandling or Malfunction of Equipment - Reactor Startup Accident

The reactor startup accident scenario assumes control rods are continuously withdrawn while the reactor is subcritical or critical and at zero power. According to [13-6], both cases are quite similar. They exhibit similar power traces, the governing parameter being the reactivity insertion rate. The startup accident postulated here assumes an uncontrollable simultaneous withdrawal of all three shim/safety rods and the regulating rod. Typical maximum reactivity insertion rates calculated at the central portion of each rod are shown below:

Regulating rod	[redacted]
Shim/Safety rod No. 1	
Shim/Safety rod No. 2	
Shim/Safety rod No. 3	
Total	

It should be noted that the above maximum insertion rates are only available along a short portion of the total distance each control rod can travel.

A startup accident has been evaluated assuming that no protective actions occur during the ramp reactivity insertion. A computer analysis has been carried out for this scenario using the computer program PARET [13-17]. PARET has been designed for use in predicting the course and consequences of reactivity accidents in small reactors. It is based on coupled thermal-hydraulics and point kinetics equations solved in channel-type geometry. Two channels, the hot and average reactor channel, have been evaluated in this analysis. Results of the analysis show [13-18] that the startup accident is self-limiting due to a strong negative reactivity feedback caused mainly by increase in moderator temperature and voiding. The maximum clad temperature, reached briefly after the onset of boiling, during this power transient is 147° C (296.6°F).

This accident was also analyzed for the HEU core in the Preliminary Hazards Evaluation [13-1]. In that analysis data obtained from the BORAX and SPERT experiments [13-8, 13-9, 13-10], in which self-shutdown behavior was investigated, were used. It was shown that no damage to the fuel would occur up to about 2.5% $\Delta k/k$ of the total reactivity insertion and that there would be ample time for the reactor operator to take a corrective action before this point has been reached.

There are a number of protective actions which would be activated to terminate a reactor startup accident including the reactor period and high power trips. As the amount of inserted reactivity is continuously increased during the withdrawal of control rods, the reactor period would become shorter activating the "Period <5 sec" scram. Ultimately, if the reactor power exceeded 150% of full power, the "Full Power >150 %" scram would be activated. Hence equipment diversity provides multiple safety barriers which would prevent a startup failure from progressing.

The analysis of a startup accident has shown that a startup failure can not possibly develop into a serious accident. No adverse consequences are to be expected to the health and safety of the public or to the reactor staff from this type of accident.

13.2 Summary and Conclusions

Several postulated accidents have been analyzed, all of which show that the health and safety of the public and the reactor staff are sufficiently protected. The maximum hypothetical accident associated with the facility was evaluated using very conservative assumptions and was shown to result in doses that were below annual limits set in 10CFR20 for normal operations. It should be noted that the Atomic Safety and Licensing Appeal Board (ASLAB) issued a ruling that 10CFR20 standards were "unduly restrictive" for evaluation of accident scenarios. The fact that the consequences of the maximum hypothetical accident at the UMR Reactor Facility are even below 10CFR20 standards demonstrates the low risk potential associated with the facility.

13.3 References

[13-1] Preliminary Hazards Evaluation, The Curators of the University of Missouri, School of Mines and Metallurgy, 10 kW Training Reactor, Rolla, Missouri, December 26, 1958.

[13-2] Hazards Summary Report for the University of Missouri at Rolla Nuclear Reactor, November 1, 1965.

[13-3] Safety Analysis Report for the University of Missouri-Rolla Reactor, Rolla, Missouri, September 27, 1984.

[13-4] Lamarsh, J.R., Introduction to Nuclear Engineering, Addison-Wesley Pub., 1977.

[13-5] Thompson, T.J., "Accidents and Destructive Tests", The Technology of Nuclear Reactor Safety, MIT Press, 1964.

[13-6] Lewis, E.E., Nuclear Power Reactor Safety, J. Wiley, 1977.

[13-7] PARET computer code. Argonne National Laboratory. Argonne, Illinois.

[13-8] Nyer, W.E. et al, "Transient Experiments with the SPERT-1 Reactor," Nucleonics, June 1956.

[13-9] Silver, E.G., "SPERT Program Status Report," Nuclear Safety, Vol. 5, No. 2, 1963.

[13-10] Whitener, H.L. et al, "Report on SPERT-1 Destructive Test Results," ANS Trans., Vol. 6, No. 1, June 1963.

[13-11] Gambill, W.R., "Collapse of Parallel-Plate Fuel Assemblies, Nuclear Safety, Vol. 1, No. 1.

[13-12] Straka, M., "Reactivity Worth of the Flooded Isotope Production Element in a Core of the UMR Reactor", UMRR/88-1, Nuclear Reactor Facility, University of Missouri-Rolla, 1988.

[13-13] Little, W.W. Jr. and Harde, R.W., "2DB User's Manual – Rev. 1", BNWL-831, Rev. 1, 1969.

[13-14] Straka, M. and Covington, L., "Study of Neutron Physics: Conversion of the University of Missouri-Rolla Reactor to Low Enriched Fuel," Trans. Am. Nucl. Soc., Vol. 55, November 1987.

[13-15] Covington, L., "A Neutronic and Thermal-hydraulic Study of the Conversion of the University of Missouri-Rolla Reactor to Low Enriched Uranium Fuel," MS Thesis, University of Missouri-Rolla, December 1988.

[13-16] Glasstone and Sesonske, Nuclear Reactor Engineering, 3rd ed. Van Nostrand Reinhold. New York 1981.

[13-17] Woodruff, W.L., "A Kinetics and Thermal-hydraulics Capability for the Analysis of Research Reactors," J. of Nuclear Technology, Vol. 64, February 1984.

[13-18] Straka, M., "Reactivity Accident Analysis of the UMRR-LEU Core", UMRR/88-2, Nuclear Reactor Facility, University of Missouri-Rolla, 1988.

14. TECHNICAL SPECIFICATIONS

14.1 Introduction

14.1.1 Purpose and Scope

This document constitutes the Technical Specifications for the University of Missouri-Rolla Reactor Facility (License No. R-79). This document was written using the guidance provided in ANSI/ANS-15.1-1990 [14-1].

The Technical Specifications represent the agreement between the licensee and the U.S. Nuclear Regulatory Commission on administrative controls, equipment availability, and operational parameters.

Specific limitations and equipment requirements for safe reactor operation and for dealing with abnormal situations, typically derived from the Safety Analysis Report (SAR), are called specifications. These specifications represent a comprehensive envelope for safe operation. Only those operational parameters and equipment requirements directly related to preserving that safe envelope are listed.

Included are the "Specifications" and the "Bases" for the Technical Specifications. These bases provide the technical support for the individual technical specifications and are included for information purposes only. They are not part of the specifications and do not constitute limitations or requirements to which the licensee must adhere.

14.1.2 Definitions

channel - the combination of sensor, line, amplifier, and output devices which are connected for the purpose of measuring the value of a parameter.

channel calibration - an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include a channel test.

channel check - a qualitative verification of acceptable performance by observation of channel behavior, or by comparison of the channel with other independent channels or systems measuring the same variable.

channel test - the introduction of a signal into the channel for verification that it is operable.

confinement - a closure on the overall facility which controls the movement of air into it and out through a controlled path.

control rod - a device fabricated from neutron absorbing material which is used to establish neutron flux changes.

direct supervision - in visual and audible contact.

excess reactivity - that amount of reactivity that would exist if all control rods were fully withdrawn from the core.

experiment - any apparatus, device, or material installed in or near the core or which could conceivably have a reactivity effect on the core and which itself is not a core component or experimental facility.

experimental facility - any structure or device associated with the reactor that is intended to guide, orient, position, manipulate, or otherwise facilitate a multiplicity of experiments of similar character.

explosive material - any solid or liquid that is categorized as a Severe, Dangerous, or Very Dangerous Explosion Hazard in Sax's Dangerous Properties of Industrial Materials [14-2], or is given an Identification of Reactivity (Stability) index 2, 3, or 4 by the National Fire Protection Association in its publication 704-M, Identification System for Fire Hazards of Materials [14-3] or enumerated in the CRC Handbook for Laboratory Safety [14-4] published by the Chemical Rubber Co.

fueled experiment - any experiment that contains U-235, U-233 or Pu-239 in greater than trace quantities, not including the normal reactor fuel elements.

measured value - the value of a parameter as it appears on the output of a channel.

mode - when the reactor is positioned as close as possible to the thermal column it is in the T mode and when it is moved away from the thermal column and reflected by water it is in the W mode.

movable experiment - an experiment which is intended to be moved in or near the core or into and out of the reactor while the reactor is operating.

operable - a component or system which is capable of performing its intended function.

operating - a component or system which is performing its intended function.

protective action - the initiation of a signal or the operation of equipment within the reactor safety system in response to a variable or condition of the reactor facility having reached a specified limit.

reactivity worth of an experiment - the maximum absolute value of the reactivity change that results from the experiment being inserted into or removed from its intended position.

reactor facility - that portion of the Reactor Building that constitutes the confinement but which does not include the front office area.

reactor operating - whenever the reactor is not secured or shutdown.

reactor operator - an individual who is licensed to manipulate the controls of the reactor

reactor secured - whenever (1) all shim/safety rods are fully inserted, (2) the console key is in the OFF position and is removed from the lock, (3) no experiments worth more than 0.4% $\Delta k/k$ are near the core, and (4) no in-core work is in progress involving fuel, and maintenance of the core structure, control rods, or control rod drive mechanisms.

reactor shutdown - when the reactor is subcritical by at least one dollar in the reference core condition with the reactivity worth of all installed experiments included.

reference core condition - reactivity condition of the core when it is at 20°C and the reactivity worth of xenon is negligible (<0.30 dollars).

regulating rod - a low reactivity-worth control rod used primarily for fine control to maintain an intended power level. Its position may be varied either by manual control or by the automatic servo-controller.

secured experiment - any experiment, experimental facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment may normally be subjected.

senior reactor operator - an individual who is licensed to direct the activities of licensed reactor operators. Such an individual is also a reactor operator. A senior reactor operator is also referred to as a senior operator.

shall, should and may - the word "shall" is used to denote a requirement; the word "should" to denote a recommendation; and the word "may" to denote permission, which is neither a requirement nor a recommendation.

shim/safety rods - high reactivity-worth boron containing control rods used primarily to provide coarse reactor control. They are connected electromagnetically to their drive mechanisms and have scram capabilities.

shutdown margin - the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition with the maximum worth scrammable rod and any non-scrammable control rod in their fully withdrawn positions and that the reactor will remain subcritical without further operator action.

surveillance time intervals:

two-year (interval not to exceed 30 months).
annually (interval not to exceed 15 months).
semiannually (interval not to exceed 7 ½ months).
quarterly (interval not to exceed 4 months).
monthly (interval not to exceed 6 weeks).
weekly (interval not to exceed 10 days).
daily (must be done during the working day).

unscheduled shutdown - any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions which could adversely affect safe operation, not including shutdowns which occur during testing or check-out operations.

14.2 Safety Limits and Limiting Safety System Settings

14.2.1 Safety Limits

Applicability: This specification applies to the temperature of the fuel cladding.

Objective: To ensure that the integrity of the fuel cladding is maintained in order to guard against an uncontrolled release of fission products.

Specification: The safety limit shall be on the temperature of fuel element cladding, which shall be less than 580°C (1076°F).

Bases: The melting temperature of the aluminum alloy used for cladding in the fuel element fabrication is 588°C (1076°F). In order to maintain the fuel element integrity the cladding temperature must not exceed 580°C (1076°F). The maximum cladding temperature associated with full power (200 kW_t) operations is only about 90°C. Furthermore, calculations show that cladding temperatures associated with a reactor power of 4.5 MW would only be about 140°C, still well under the safety limit.

14.2.2 Limiting Safety System Settings

Applicability: This specification applies to the set points for the safety channels monitoring reactor thermal power, P.

Objective: To ensure that automatic protective action is initiated to prevent the maximum fuel cladding temperature from exceeding the safety limit.

Specifications: The limiting safety system setting shall be on reactor thermal power, P, which shall be no greater than 300 kW_t, or 150% of full power.

Bases: Reactor cooling is provided by natural convection in the reactor pool. Therefore, the only parameter which can be used to limit the fuel cladding temperature is the reactor power. The Safety Analysis Report (Section 4.6) shows that at a reactor power of 300 kW_t, the maximum cladding temperature is well below 105°C (221°F). This temperature is much lower than the temperature at which fuel element damage could occur. Therefore, an extremely large safety margin exists between the limiting safety system set point and the safety limit.

14.3 Limiting Conditions for Operations

14.3.1 Reactor Core Parameters

Applicability: These specifications apply to the reactivity condition of the reactor and the reactivity worths of control rods and experiments.

Objectives: To ensure that the reactor can be operated safely and to ensure that it can be shut down at all times.

Specifications: The reactor shall not be operated unless the following conditions exist:

(1) The maximum excess reactivity for reference core conditions with secured experiments and experimental facilities in place shall 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 purposes of control rod calibration 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% $\Delta k/k$.

(2) The minimum shutdown margin under reference core condition with secured experiments and experimental facilities in place and with the highest worth control rod and the regulating rod fully withdrawn shall be no less than 1.0% $\Delta k/k$.

(3) The excess reactivity limit (Section 14.3.1(1)) and shutdown margin limit (Section 14.3.2(2)) may be temporarily exceeded following a core configuration change under the following conditions:
  1. reactor power is limited to 2 kW_t,
  2. reactor operations are limited to the measurement of excess reactivity, control rod worths and shutdown margin, and
  3. the reactor is immediately shutdown upon discovery of excess reactivity or shutdown margin being in violation of the limits specified in Section 14.3.1(1) or Section 14.3.1(2). In such an instance, a core configuration change shall be implemented with the intent of meeting the limits specified in Section 14.3.3(1) and Section 14.3.1(2).

(4) The reactor shall be operated only when all lattice positions internal to the active fuel boundary are occupied by either a fuel element, control rod fuel element, or by an experimental facility.

<u>Bases:</u>

(1) A sufficient excess reactivity is needed to provide for temperature effect override, xenon override, and operational and experimental flexibility. The limit of 1.5% $\Delta k/k$ on excess reactivity is to assure that the operational characteristics of a reactor core are such as analyzed in the Safety Analysis Report. It has been shown in Chapter 13 of the SAR that a stepwise reactivity insertion of 1.5% $\Delta k/k$ does not adversely affect the health and safety of the public and the reactor staff. The limit of 3.5% $\Delta k/k$ allows for the complete, direct calibration of the highest worth shim/safety rod. In the accident analysis performed in Chapter 13 of the SAR it was concluded that no credible physical mechanism exists which could possibly lead to a sudden release of this amount of reactivity. Past experience has shown that it takes about five working days to perform control rod calibrations.

(2) The minimum shutdown margin provides assurance that the reactor can be shut down from any operating condition and remain shut down after cool down and xenon decay, even if one control rod should become stuck in the fully withdrawn position.

(3) This specification provides for operational flexibility during measurements of excess reactivity and shutdown margin.

(4) This specification precludes the possibility of having an internal vacancy into which a fuel element could be inadvertently inserted.

14.3.2 Reactor Control and Safety Systems

14.3.2.1 Reactor Control Systems

Applicability: This specification applies to the instrumentation that must be operable for safe operation of the reactor.

Objective: To require that sufficient control information and automatic protective signals are available to the operator to ensure safe operation of the reactor.

Specification: The reactor shall not be operated unless the channels described in Table 14.1 are operable. Values listed are the limiting set points. For operational convenience the actual set points may be on more restrictive values.

Table 14.1-Control Channels.

Channel	Set Point	Function
Reactor Power	120%	Rundown
Reactor Period	15 s	Rundown
Radiation Area Monitors[1]	20 mrem/hr	Rundown[1]
Core Inlet Pool Water Temperature	135°F	Rod Withdrawal Prohibit
Startup Count Rate[1]	2 cps	Rod Withdrawal Prohibit[1]
Reactor Period [1]	30 s	Rod Withdrawal Prohibit[1]
Recorder Off	Not applicable	Rod Withdrawal Prohibit

Bases: The 30 second reactor period rod withdrawal prohibit serves to establish a reasonable and conservative limit for normal operations. The 15 second reactor period rundown provides an additional layer of period protection and prevents the reactor from reaching the scram setpoint of 5 seconds described in Table 14.2. The 120% reactor power rundown provides an

[1] These functions may be key bypassed at the reactor console by the Senior Operator on Duty as provided for in the Standard Operating Procedures

additional layer of protection designed to prevent the LSSS (150% power) from being reached. The radiation area monitors 20 mrem/hr rundown provides for reactor shutdown in the unusual event that radiation levels at any of three gamma RAM locations reaches the setpoint. The startup interlock, which requires a neutron count rate of at least 2 counts per second (cps) before the reactor is operated, ensures that sufficient neutrons are available for proper operation of the startup channel, and for a controlled approach to criticality. The core inlet pool water temperature rod withdrawal prohibit provides protection to keep the demineralizer resins below their suggested temperature limit of 140°F (60°C). The recorder off rod withdrawal prohibit assures that the strip chart recorders are on during reactor operations.

14.3.2.2 Reactor Safety Systems

Applicability: This specification applies to the reactor safety system channels.

Objective: To stipulate the reactor safety system channels that must be operable to ensure that the limiting safety system settings are not exceeded during normal operation.

Specification: The reactor shall not be operated unless the safety system channels presented in the Table 14.2 are operable. Values listed in the table are the limiting set points. For operational convenience the actual set points may be on more restrictive values.

Table 14.2-Safety System Channels.

Channel	Set Point	Function
Manual Scram Button	Not applicable	Scram
Reactor Power	300 kW_t	Scram
Reactor Period	5 s	Scram
Bridge Motion	Not applicable	Scram
Log N & Period Not Operative	Not applicable	Scram

Bases: Power channels are provided to ensure that the power level is limited to protect against abnormally high fuel temperatures. The manual scram allows the operator to shut down the reactor if an unsafe or abnormal condition arises. The period scram is provided to ensure that the power level does not increase on a period less than 5 seconds. The bridge motion scram shuts the reactor down in the event that the bridge is moved. The Log N and Period not operative scram shuts the reactor down if the Log N and Period Channel are in a not operative condition.

14.3.2.3 Shim/Safety Rod Drop Times

Applicability: This specification applies to the time from the receipt of a safety signal to the time it takes for a shim/safety rod to drop from the fully withdrawn to the fully inserted position (free-drop time).

Objective: To ensure that the reactor can be shut down within a specified period of time.

Specification: The reactor shall not be operated unless the free-drop time for each of the three shim/safety rods is less than 1 second.

Bases: Shim/safety rod drop times as specified will ensure that the safety limit will not be exceeded in a worst-case delayed critical transient which has been analyzed in Chapter 13, Section 13.1.9.

14.3.3 Coolant System

Applicability: This specification applies to the water in the reactor pool which serves as the reactor coolant.

Objective: To ensure that adequate cooling is provided for the reactor core at all times and to ensure that there is sufficient biological shielding available. The objective of the water quality requirement is to minimize corrosion of the fuel element cladding and to minimize neutron activation of dissolved materials.

Specification:

(1) The reactor shall not be operated unless there is at least 16 feet (4.88 m) of water above the core.

(2) The resistivity of the pool water shall be greater than 0.2 megaohm-cm as long as there are fuel elements in the pool. This requirement may be waived for a period of up to three weeks once every three years.

(3) The minimum temperature of the reactor pool should be no less than 15.5°C (60°F) when the reactor is operated.

Bases:

(1) This provision is primarily to assure sufficient depth of water for shielding but also provides assurance that a natural convection flow path will be available.

(2) Experience with water quality control at this and many other reactor facilities have shown that maintenance within the specified limit provides acceptable control of the corrosion rate. (See Chapter 5, Section 5.2 for further information.) The provision that allows this requirement to be temporarily waived is to provide for operational flexibility in the unlikely event that the demineralizer becomes inoperable. The three week period should be sufficient to make repairs.

(3) The reactor core has a negative moderator reactivity effect that provides an increase in excess reactivity when the reactor pool is at lower temperatures and lower reactivity at higher pool temperatures. Maintaining a minimum reactor pool temperature of 15.5°C (60°F) or greater will assure the excess reactivity will not significantly increase and shutdown margin decrease.

14.3.4 Confinement

Applicability: This specification applies to the capability of isolating the reactor facility from the unrestricted environment when necessary.

Objective: To minimize exposure to the public resulting from airborne activity potentially released into the reactor facility and to be consistent with the ALARA principle.

Specification: The reactor shall not be operated unless the reactor truck door is closed and the ventilation intake and exhaust duct louvers are operable or secured in a closed position. At least one of the three facility vent fans must be operable.

Basis: This specification ensures that the reactor facility can be quickly isolated in the case of an unexpected release of airborne radioactivity from the reactor or associated experimental facilities.

14.3.5 Ventilation System

Applicability: This specification applies to the ventilation fans and the associated intakes and exhausts.

Objective: To provide for normal building ventilation and the reduction of airborne radioactivity within the reactor bay during reactor operation.

Specification: A ventilation fan with a rated capacity of at least 4,500 cubic feet per minute (cfm) (127.4 m^3/min) shall be turned on within ten minutes after the reactor reaches full power.

Bases: Experience has shown that during normal operation this specification is sufficient to maintain radioactive gaseous effluents below 10 CFR 20 (Appendix B) limits. Chapter 11, Section 11.1.1 shows that releasing the air does not unduly expose the public. The ten minute provision provides for operational flexibility.

14.3.6 Radiation Monitoring Systems and Radioactive Effluents

14.3.6.1 Radiation Monitoring Systems

Applicability: This specification applies to the gamma radiation area monitoring instrumentation.

Objective: To provide protection against excessive radiation levels for personnel in the reactor building.

Specifications: The reactor shall not be operated unless the Radiation Area Monitors (RAMs) located at the reactor bridge, at the demineralizer, and in the basement experimental area are operable. Table 14.3 specifies the approximate locations, set points and functions. Values listed are the limiting set points. For operational convenience the actual set points may be on more restrictive values.

The reactor may be operated with one or more of the RAM channels being inoperable under the following conditions:

1. The period of operations with the RAM Channel(s) being inoperable does not exceed one week.
2. A portable gamma radiation instrument is placed in the same vicinity as the inoperable RAM detector(s) with a local audible alarm set point of 20 mrem/hr or less.
3. If the inoperable channel is the Bridge RAM, the control room operator must be able to visually monitor the radiation level of the portable unit.

Table 14.3-Radiation Area Monitors.		
Location	**Set Point**	**Function** [2]
Reactor Bridge	20 mrem/hr 50 mrem/hr	Rundown Building Evacuation
Demineralizer	20 mrem/hr	Rundown
Basement Experimental Area	20 mrem/hr	Rundown

Bases: The RAMs provide information to operating personnel about the radiation level above the reactor pool, at the demineralizer, and in the basement experimental area. It ensures that in the case of a failure of an experiment or a significant drop in the pool water level the appropriate action can be automatically initiated.

A detailed discussion of the rationale for the RAM detector locations, setpoints, and functions is presented in Chapter 7, Section 7.4.

14.3.6.2 Radioactive Effluents

Applicability: This specification applies to radioactive effluents released from the reactor facility

14.3.6.2(1) Airborne Effluents

Objective: To ensure that exposure to the public resulting from the routine release of radioactive airborne effluents will not endanger the health and safety of the public.

Specification: The activity of Ar-41 released from the facility shall not exceed the limits of 10 CFR 20, Appendix B, Table 2.

Bases: The bases for this specification are given in Chapter 11 of the SAR.

[2] These functions may be key bypassed at the reactor console by the Senior Operator on Duty as provided for in the Standard Operating Procedures.

14.3.6.2(2) Liquid Effluents

Objective: To ensure that exposure to the public resulting from the release of radioactive liquid effluents will not endanger the health and safety of the public.

Specification: The activity of liquids released from the facility shall not exceed 10 CFR 20 limits.

Bases: The bases for this specification are given in Chapter 11 of the SAR.

14.3.7 Experiments

Applicability: These specifications apply to experiments run in conjunction with the reactor.

Objectives: To ensure the reactor can be shut down at all times, that the reactor fuel will not be damaged, that the limiting conditions for operation will not be exceeded, and that a malfunction of an experiment will not result in undue radioactivity release to the environment.

14.3.7.1 Reactivity Limits

Specifications: The reactor shall not be operated unless the following conditions exist:

(1) Experiments worth more than 0.4% Δk/k shall be:
  a) a secured experiment,
  b) inserted and removed with the reactor shut down, and
  c) inserted and removed from the reactor with a procedure approved by the Radiation Safety Committee.

(2) The sum of the absolute values of all experiments shall be no greater than 1.2% Δk/k.

(3) Experiments having moving parts shall not have a continuous insertion rate greater that +0.05% Δk/k per second. This requirement does not apply to the experiment's insertion to or removal from the core.

Bases:

(1a) This limit is provided in order to prevent a moveable experiment from inserting a large reactivity insertion into the operating reactor. An analysis of this reactivity limit is given in Chapter 13, Section 13.1.6.2.

(1b) In order to not accidentally insert too much reactivity when the reactor is operating, such experiments need to be positioned or removed only when the reactor is shut down.

(1c) Thorough Radiation Safety Committee review of such a procedure provides assurance that such experiments will take reactor and personnel safety, and the environment into proper account.

(2) The total reactivity of 1.2% Δk/k places an acceptable upper limit on the worth of all experiments. This limit is lower than the reactivity for which an accident analysis was performed in Chapter 13, Section 13.1.2. It was shown in this analysis that the maximum fuel cladding temperature would not exceed the safety limit should an accident occur.

(3) This specification allows for certain reactor kinetics experiments to be performed, while maintaining constraint upon the rate of change of reactivity insertions. It is well within the envelope of the reactivity insertion rate which was analyzed in Chapter 13, Section 13.1.9. Results have shown that the health and safety of the public and the reactor staff would not be endangered in such an accident.

14.3.7.2 Materials

Specifications:

(1) All materials to be irradiated in the reactor shall be either corrosion resistant in reactor pool water or encapsulated within corrosion resistant containers.

(2) Explosive material shall not be allowed in or near the reactor unless specifically approved by the Radiation Safety Committee. Experiments reviewed by the Radiation Safety Committee in which the material is potentially explosive, either while contained or if it leaked from the container, shall be designed to prevent damage to the reactor core or to the control rods or instrumentation, and to prevent any changes in reactivity. Known explosives in the amount greater than 25 milligrams shall not be irradiated in or near the reactor core. In addition the pressure shall be calculated or experimentally determined such that it will not cause the sample container to fail.

(3) Fueled experiments shall not be allowed in or near the reactor unless specifically approved by the Radiation Safety Committee. Fueled experiments in the amount which would generate a power greater than 100 W shall not be irradiated at the UMRR facility. Fueled experiments which generate more than 1 W power shall be irradiated in the reactor pool at least 4.88 m (16 ft) deep under the pool water surface. Fueled experiments which generate less than 1 W power may be irradiated anywhere in the facility.

(4) Cooling shall be provided to prevent the surface temperature of an experiment being irradiated from exceeding the boiling point of the reactor pool water.

Bases:

(1) The requirement concerning either corrosion resistant materials or corrosion resistant

encapsulation provides assurance that irradiation samples will not contaminate the pool water.

(2) Special case-by-case precautions must be taken before irradiation of explosive materials will be allowed. The quantities are restricted to very small masses. Most likely such irradiations would be done at the far end of the beam tube or of the thermal column. In which case, the potential for core damage or reactivity changes would be very small.

(3) Special case-by-case precautions must be taken before irradiation of fueled experiments. The Radiation Safety Committee must determine whether there are any unreviewed safety questions. Section 13.1.1 of the Safety Analysis Report addresses the impact of the failure of a fueled experiment.

(4) Samples or containers irradiated in the pool are in contact with a large heat sink. However, in order to assure that departure from nucleate boiling does not occur, adequate heat removal must be provided.

14.3.7.3 Failure and Malfunction

Specifications:

Experiments shall be designed such that they will not contribute to the failure of other experiments, core components, or cause other perturbations that may interfere with the safe operation of the reactor. Experiments shall be designed such that no credible reactor transient could cause the experiment to fail in such a way as to contribute to a reactor accident.

Bases:

Experiments which could adversely affect proper operation of the control rods must be avoided. Control over the reactor core must be maintained should an experiment fail.

14.4 Surveillance Requirements

Allowable surveillance time intervals shall not exceed the times shown in the definition Section 14.1.2. The maximum intervals on surveillance frequencies indicated are to provide operational flexibility and are not to be used to reduce frequency. The established frequencies are to be maintained over the long term.

Surveillance requirements (except those specifically required for safety when the reactor is shutdown) may be deferred during reactor shutdown; however, they must be completed prior to reactor startup unless reactor operation is required for the performance of the surveillance. Such surveillance should be performed as soon as practical after reactor startup.

14.4.1 Reactor Core Parameters

14.4.1.1 Excess Reactivity, Rod Worth, and Shutdown Margin Measurements

Applicability: This specification applies to the reactor core.

Objective: To assure that the requirements of specification 14.3.1 are not violated.

Specifications: Following a change in core configuration the following steps shall be performed:

1. A licensed operator shall visually confirm that all internal grid plate positions are occupied prior to taking the reactor critical.

2. The excess reactivity of the core shall be measured. If the excess reactivity is found to be outside of the limits specified in Section 14.3.1, the reactor shall be shutdown and the core configuration changed with the intent of complying with the limits specified in Section 14.3.1. If the excess reactivity is found to be acceptable, then:
   i) the control rod worths shall be measured, and
   ii) the shutdown margin shall be determined.

Bases:

1) Visual inspection of the reactor core is the most reliable way to assure that all internal positions are occupied and that no space exists for rapid insertion of a fuel element (see Chapter 13, Section 13.1.2).

2) An experimental determination of the excess reactivity and shutdown margin is necessary in order to preclude operating the reactor without adequate shutdown capability.

14.4.2 Reactor Control and Safety Systems

14.4.2.1 Shim/Safety Rods

Applicability: This specification applies to the surveillance requirements for the shim/safety rods.

Objectives: To ensure that the control rods are capable of performing their function and to establish that no significant physical degradation in the rods has occurred.

Specifications:

1) Shim/safety rod drop times shall be measured
   a) semiannually
   b) for a particular control rod whenever the magnet assembly is disassembled, reassembled or if the control assembly is moved to a new grid position.

2) The shim/safety rods shall be visually inspected annually for pitting and cracking and whenever rod drop times exceed the limiting conditions for operation (Section 14.3.2.3 of these specifications).

Bases:

1) Rod drop time measurements are required to assure the reactor can be quickly shutdown.

2) The visual inspection of the shim/safety rods and measurement of their drop times are made to determine whether they are capable of performing properly and to detect any gradual degradation in rod performance.

14.4.2.2 Safety Channels

Applicability: This specification applies to the surveillance requirements for the reactor safety system channels listed in Section 14.3.2.2 for the reactor.

Objective: To ensure that the reactor safety system channels are operable.

Specifications:

1) A channel test of each of the reactor safety system channels shall be performed before each day's operation or before each operation expected to extend more than one day, except for the bridge motion monitor which shall be done weekly.

2) A channel calibration of the reactor power range safety channel and period channel shall be performed annually.
3) The thermal power shall be experimentally verified annually.

Bases:

1) The daily channel tests will ensure that the safety channels are operable.

2) The annual calibration will correct for any long-term drift of the channels.

3) The annual verification of thermal power will correct for drift and ensure operation within the requirements of the license.

14.4.2.3 Maintenance

Applicability: This specification applies to the surveillance requirements following maintenance of control or safety systems.

Objective: To ensure that a system is operable before being used after maintenance has been performed.

Specification: Following maintenance or modification of a control or safety system or component, it shall be verified that the system is operable either before it is returned to service or during its initial operation.

Bases: The intent of the specification is to ensure that work on the system or component has been properly performed and that the system or component has been properly reinstalled or reconnected. Correct operation of some systems, such as power range monitors, cannot be verified unless the reactor is operating. Operation of these systems will be verified during their initial operation following maintenance or modification.

14.4.3 Coolant System

Applicability: This specification applies to the surveillance of coolant water quality.

Objective: To ensure that water quality does not deteriorate over extended periods of time even if the reactor is not operated.

Specification:

1) The resistivity of the coolant water shall be measured at least once every two weeks when the reactor is operated.

2) If the rector is not operated, conductivity shall be measured monthly.

Bases: Section 14.3.3 of these specifications establishes water quality requirements. This section ensures that the water quality is not permitted to deteriorate over extended periods of time even if the reactor does not operate. The demineralizer resins should be regenerated in order to improve the water quality. If that is not sufficient, then the resins should be replaced.

14.4.4 Confinement

Applicability: This specification applies to the surveillance requirements for confinement of the reactor bay.

Objective: To ensure that the closure equipment to the reactor bay is operable.

Specifications: A test shall be performed quarterly to assure that the following equipment is operable or can remain permanently closed: bay door, ventilation inlet and exhaust duct louvers, and the personnel security door.

Bases: Quarterly surveillance of this equipment will verify that the confinement of the reactor bay can be maintained, if confinement is needed.

14.4.5 Ventilation Systems

Applicability: This specification applies to the ventilation fans and associated closure devices.

Objective: The objective is to ensure that the ventilation fans and closure devices perform their function satisfactorily.

Specification: Ventilation fans and intake/exhaust louvers shall be visually checked quarterly for proper operation.

Bases: Quarterly surveillance is to ensure proper exchange of air through the reactor facility to reduce the buildup of radioactive gases within the reactor bay.

14.4.6 Radiation Area Monitors and Radioactive Effluents

14.4.6.1 Radiation Area Monitors

Applicability: This specification applies to the gamma Radiation Area Monitors required by Section 14.3.6.1 of these specifications.

Objectives: To ensure that the radiation area monitors are operating properly.

Specifications:

1) A channel check shall be performed on each gamma RAM Channel daily before reactor startup.

2) Calibration of the radiation area monitors shall be performed annually.

Bases: Adequate radiation control requires operable monitors, and experience has shown that an annual calibration of the monitoring systems is adequate to ensure their proper functioning within the specified limits.

14.4.6.2 Radioactive Effluents

14.4.6.2(1) Airborne Effluents

Applicability: This specification applies to the surveillance of the air in the reactor building while the reactor is operated.

Objective: To verify the method used to calculate the airborne effluents.

Specifications: An experimental verification of calculated release values shall be performed every five years and when a change in licensed power occurs.

Bases: This is to ensure that the airborne radioactive effluents will be properly accounted. The basis for this specification is given in Chapter 7, Section 7.4.

14.4.6.2(2) Liquid Effluents

Applicability: This specification applies to the surveillance of liquid radioactive effluents.

Specifications: Before any release of potentially radioactive liquid effluent, samples shall be drawn and analyzed.

Bases: This is to ensure that radioactive liquid effluents will be properly analyzed before being released to the unrestricted environment. The basis for this specification is given in Chapter 11, Section 11.2.3.3.

14.5 Design Features

Only those design features of the facility describing materials of construction and geometric arrangements , which if altered or modified would significantly affect safety and which are not included in sections 14.2, 14.3 or 14.4 of the Technical Specifications, are included in this section.

The Safety Analysis Report contains the details necessary for establishing criteria for the following Technical Specifications.

14.5.1 Site and Facility Description

14.5.1.1 Location

The Nuclear Reactor Building is located on the east side of the University of Missouri-Rolla campus in Rolla, Missouri, near 14th Street and Pine Street.

14.5.1.2 Description

The reactor is housed in a steel-framed, double-walled building designed to restrict leakage. Air and other gases may be exhausted through vents in the reactor bay ceiling 9.1 m (30 ft) above grade. The Reactor Building free volume is approximately 1700 cubic meters.

14.5.2 Reactor Coolant System

The reactor is cooled by natural convection of light water. The core is submerged in the reactor pool assuring a pathway for natural convection flow. The pool also serves as a heat sink, neutron moderator and reflectors, and radiation shield.

14.5.3 Reactor Core and Fuel

14.5.3.1 Core Configurations

Various core configurations that meet the requirements of Section 14.3.1 may be used to accommodate experiments.

14.5.3.2 Fuel Elements

1) Plate fuel elements of the MTR type are used. The overall dimensions of each element are approximately 7.6 × 7.6 × 91.4 centimeters (3 × 3 × 36 in). The active length of fuel is approximately 24 inches and the fuel is clad in aluminum alloy. The fuel elements have 18 fuel plates joined to two side plates. The whole assembly is joined at the bottom to a cylindrical nose piece that fits into the core grid plate.

   The fuel meat is U_3Si_2 dispersed in an aluminum matrix and is enriched to approximately 20% U-235.

2) Control rod fuel elements are similar to the elements described in (1) with the exception that the center eight plates have been removed and have been replaced with guide plates such that the control rod cannot come in contact with fuel plates.

3) Half fueled elements have nine LEU fueled plates (either the front ones or the rear ones as appropriately marked) and nine dummy (or unfueled) plates.

4) An irradiation fuel element has six fuel plate positions left unoccupied (plate positions 11 through 16), plates 10 and 17 are unfueled and all the others (1 through 9 and 18) are fueled.

14.5.3.3 Control Rods

1) Poison sections of the three shim/safety rods are stainless steel and initially contained approximately 1.5% natural boron. The rods dimensions of 5.7 × 2.2 cm (2¼ × 7/8 in) and are approximately 83.8 cm (33 in) long.

2) The poison section of the regulating rod is a stainless steel oval-shaped tube, 25 inches long, having a wall thickness of 0.065 inches, and is mechanically coupled to the rod drive.

14.5.3.4 Control Rod Drive Mechanisms

1) The shim/safety rod drives have a maximum vertical travel of 24 inches and a withdrawal rate of approximately 6-inches per minute. The shim/safety rods are magnetically coupled to the drive mechanisms and drop into the core, by gravity, upon a scram signal.

2) The regulating rod drive has a maximum vertical travel of 24 inches and a withdrawal rate of approximately 24 inches per minute. The regulating rod is mechanically coupled to its rod drive and does not respond to a scram signal.

3) Lights are provided on the operator's console to indicate upper limit, lower limit, and shim range for each shim/safety rod.

14.5.4 Fissionable Material Storage

The fuel storage pit, which is located below the floor of the reactor pool and at the end opposite from the core, is capable of storing the complete low-enriched uranium fuel inventory. The neutron multiplication factor of the fully loaded storage pit shall not exceed 0.9 under any conditions.

14.6 Administrative Controls

14.6.1 Organization

14.6.1.1 Structure

The Nuclear Reactor Facility is a part of the School of Materials, Energy & Earth Resources of the University of Missouri-Rolla. The organizational structure is shown in Figure 14.1.

14.6.1.2. Responsibility

The Dean of the School of Materials, Energy & Earth Resources is the individual responsible for the reactor facility's licenses (Level 1).

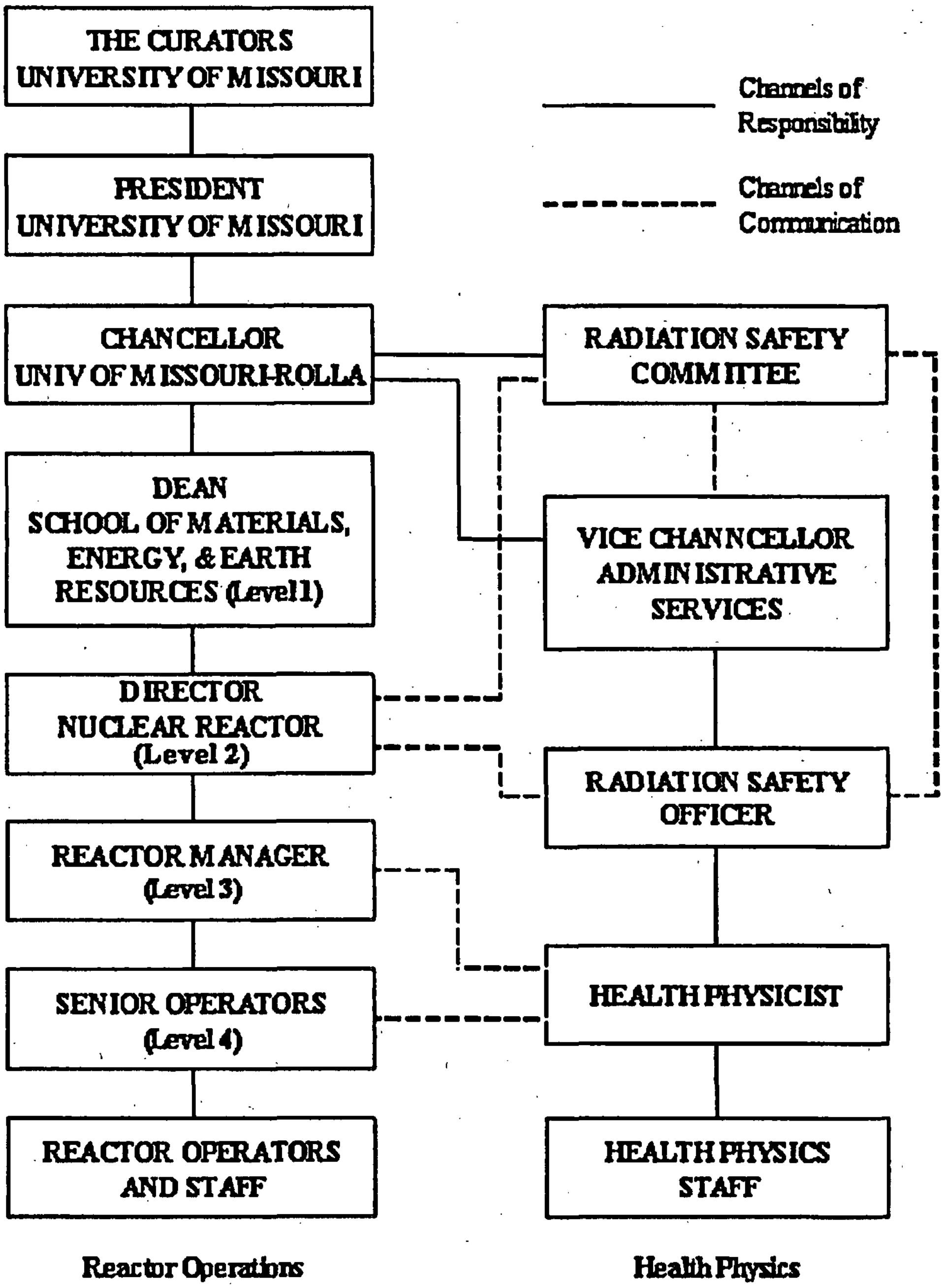
The Director of the Nuclear Reactor Facility is the contact person for the NRC and has overall responsibility for management of the facility (Level 2). The Director shall have a minimum of six years of nuclear experience. The Director shall have a Bachelor's (or higher) degree in engineering or science. Equivalent education or experience may be substituted for a degree. The degree may fulfill four years of the six years of nuclear experience required. As soon as reasonably possible after being assigned to the position, the Director shall obtain and maintain a NRC Senior Operators License.

The Reactor Manager (Level 3) shall be responsible for the day-to-day operation and for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license and the provisions of the Radiation Safety Committee. During periods when the Reactor Manager is absent, his responsibilities may be delegated to a Senior Operator (Level 4).

A Health Physicist who is organizationally independent of the Reactor Facility operations group, as shown in Figure 14.1, shall be responsible for radiological safety at the facility. The Health Physicist may also be the Radiation Safety Officer.

14.6.1.3 Staffing

1) When the reactor is operating the following staffing conditions shall be met:
    a) At least two persons, one of whom is a licensed Senior Reactor Operator, shall be present in the Reactor Building.
    b) A licensed Reactor Operator or Senior Reactor Operator shall be present in the control room.
2) All rearrangements of the core, fuel movement, and associated Health Physics monitoring, shall be supervised by a licensed Senior Operator.



igure 14.1-

Organizational Structure of the UM System Related to the UMRR Facility.

14.6.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall meet or exceed the requirements of American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4 (1978), Sections 4-6. [14-5]

14.6.2 Review and Audit

There shall be a committee that reviews and audits reactor operations to ensure that the facility is operated in a manner consistent with public safety and within the terms of the facility license. The Committee shall be referred to as the Radiation Safety Committee and shall report to the Chancellor of the campus.

14.6.2.1 Composition and Qualifications

The Committee shall be composed of at least three members, one of whom shall be the Radiation Safety Officer of the campus. No more than two members will be from the organization responsible for reactor operations. At least three members of the committee shall collectively represent a broad spectrum of expertise in areas relating to reactor safety and research use of radioisotopes. Qualified approved alternates may serve in the absence of regular members.

14.6.2.2 Charter and Rules

1) A quorum of the Committee shall consist of at least one half of the voting members where the operating staff does not constitute a majority.

2) The Committee shall meet at least once each calendar year. Minutes of all meetings shall be disseminated to Committee members and to other responsible personnel as designated by the Committee Chairman.

3) The Committee shall have a written statement, or charter, defining such matters as the authority of the Committee, the subjects within its purview, and other such administrative provisions as are required for effective functioning of the Committee.

14.6.2.3 Review Function

As a minimum, the Radiation Safety Committee shall:

1) Review in accordance with 10CFR50.59 untried experiments and tests that are significantly different from those previously used or tested in the reactor, as determined by the Facility Director.

2) Review in accordance with 10CFR50.59 changes to the reactor core, reactor systems or design features that may affect the safety of the reactor.

3) Review all proposed amendments to the facility license and Technical Specifications.

4) Review reportable occurrences and the actions taken to identify and correct the cause of the occurrences.

5) Review significant operating abnormalities or deviations from normal performance of facility equipment that affect reactor safety.

This same Committee may have other responsibilities, for example oversight of the campus byproduct material license. The Committee may assign sub-committees to act on its behalf provided that said sub-committees report in writing all actions they take.

14.6.2.4 Audit Function

The Radiation Safety Committee will arrange for a knowledgeable and impartial individual (or individuals) to review reactor operations and audit the operational records for compliance with reactor procedures, Technical Specifications, and license provisions. An impartial individual is one who is not directly affected by the findings or recommendations of the audit and has no reason to be biased concerning the review. These audits shall be performed annually.

14.6.3 Radiation Protection Program

The Campus Health Physicist shall be responsible for implementing the radiation protection program at the reactor facility.

14.6.4 Operating Procedures

The reactor staff shall prepare and utilize written procedures for at least the items listed below. These procedures shall be adequate to ensure the safe operation of the reactor, but should not preclude the use of independent judgment and action should the situation require such.

1) Startup, operation, and shutdown of the reactor.

2) Installation or removal of fuel elements, control rods, experiments, and experimental facilities.

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

4) Emergency conditions involving potential or actual release of radioactivity, including provisions for evacuation, re-entry, recovery, and medical support.

5) Preventive and corrective maintenance operations that could have an effect on reactor safety.

6) Periodic surveillance (including testing and calibration) of reactor instrumentation and safety systems.

7) Radiation control procedures which shall be maintained and made available to all operations personnel.

8) Implementation of emergency and physical security plans.

Substantive changes to approved procedures having safety significance shall be made only with the approval of the Radiation Safety Committee. Changes that do not change the original intent of the procedures may be made with the approval of the Facility Director.

14.6.5 Experiments Review and Approval

The reactor staff shall perform a thorough review of all proposed experiments in order to assure that they meet the requirements of Sections 14.3.7 of these specifications.

Following the reactor staff review and approval, any proposed untried experiments will be forwarded to the Radiation Safety Committee for its review.

14.6.6 Required Actions

14.6.6.1 Action to be taken in the Case of Safety Limit Violation

1) The reactor shall be shut down, and reactor operations shall not be resumed until authorized by the NRC.

2) The safety limit violation shall be promptly reported to the Director of the Reactor Facility.

3) The safety limit violation shall be reported to the NRC.

4) A safety limit violation report shall be prepared. The report shall describe the following:

    a) Applicable circumstances leading to the violation including, when known, the cause and contributing factors.

b) Effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public.

c) Corrective action to be taken to prevent recurrence.

5) The report shall be reviewed by the Radiation Safety Committee and any follow-up report shall be submitted to the NRC when authorization is sought to resume operation of the reactor.

14.6.6.2 Actions to be Taken in Response to Certain Occurrences

The following actions shall be taken if an event of the type identified in Section 14.6.7.2(1)b or 14.6.7.2(1)c occurs:

1) Reactor conditions shall be returned to normal or the reactor shall be shut down. If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed unless authorized by Level 2 or designated alternates.

2) The occurrence shall be reported to the Director and to the NRC (see 14.6.7.2).

3) The occurrence shall be reviewed by the Radiation Safety Committee at their next scheduled meeting.

14.6.7 Reports

14.6.7.1 Operating Reports

An annual progress report will be made by May 30 of each year to the NRC Document Control Desk with a copy to the Regional Administrator, which provides the following information:

1) A narrative summary of reactor operating experience including the energy produced by the reactor or the hours the reactor was critical, or both

2) The unscheduled shutdowns including, where applicable, corrective action taken to preclude recurrence.

3) Tabulation of major preventive and corrective maintenance operations having safety significance.

4) A summary of changes to the facility or procedures, which affect reactor safety, and performance of tests or experiments carried out under the conditions of Section 50.59 of 10 CFR 50.(6)

5) A summary of the nature and amount of radioactive effluents released or discharged to environs beyond the site boundary. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or

diffusion is less than 25 percent of the concentration allowed, a statement to this effect is sufficient.

6) A summary of exposures received by facility personnel and visitors where such exposures are greater than 25 percent of that allowed.

14.6.7.2 Special Reports

1) There shall be a report not later than the following working day by telephone to the NRC Project Manager and the regional NRC office, to be followed by a written report that describes the circumstances of the event within 14 days of any of the following:

a) Violation of safety limits (see 14.6.6.1)
b) Release of radioactivity from the site above allowed limits (see 14.6.6.2)
c) Any of the following: (see 14.6.6.2)

i) Operation with actual safety-system settings for required systems less conservative than the limiting safety-system settings specified in the technical specifications.

ii) Operation in violation of limiting conditions for operation established in the technical specifications unless prompt remedial action is taken.

iii) A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns.

NOTE: Where components or systems are provided in addition to those required by the technical specifications, the failure of the extra components or systems is not considered reportable provided that the minimum number of components or systems, specified or required, perform their intended reactor safety function.

iv) An unanticipated or uncontrolled change in reactivity greater than one dollar. Reactor trips resulting from a known cause are excluded

v) Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or containment boundary (excluding minor leaks) where applicable, which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both

vi) An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

2) A written report shall be submitted within 30 days to the NRC Document Control Desk, with a copy to the Regional Administrator of the following:
   a) Significant changes in the transient or accident analyses as described in the SAR.

   b) Permanent changes in facility organization involving Level 1 or 2 personnel.

14.6.8 Records

Records may be in the form of logs, data sheets, or other suitable forms. The required information may be contained in single or multiple records, or a combination thereof.

14.6.8.1 Records to be retained for a Period of at Least Five Years

1) Normal reactor facility operation (but not including supporting documents such as checklists , log sheets, etc., which shall be maintained for a period of at least one year)
2) Principal maintenance operations
3) Reportable occurrences
4) Surveillance activities required by the Technical Specifications.
5) Reactor facility radiation and contamination surveys where required by applicable regulations
6) Experiments performed with the reactor
7) Fuel inventories, receipts, and shipments
8) Approved changes in operating procedures
9) Records of meeting minutes and audit reports of the Radiation Safety Committee

14.6.8.2 Records to be retained for at Least One Requalification Cycle

Regarding retraining and requalification of licensed operations personnel, the records of the most recent complete requalification cycle shall be maintained.

14.6.8.3 Records to be retained for the Life of the Facility

1) gaseous and liquid radioactive effluents released to the environment,
2) radiation exposures for all personnel monitored,
3) Updated, corrected, and as-built drawings of the facility.

14.7 REFERENCES

[14-1] "American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," American Nuclear Society, LaGrange Park, Illinois (1990).

[14-2] Sax's Dangerous Properties of Industrial Materials, Richard J. Lewis, Sr. Van Nostrand-Reinhold Co. New York, NY (1996).

[14-3] Identification System for Fire Hazards of Materials, Publication 704, National Fire Protection Association, Batterymarch Park, Quincy, MA (1980).

[14-4] CRC Handbook for Laboratory Safety, Norman V. Steere, Chemical Rubber Company, Cleveland, OH. (1971).

[14-5] "Selection and Training of Personnel for Research Reactors," ANSI/ANS-15.4-1977, American Nuclear Society, LaGrange Park, IL (1978).

15. FINANCIAL QUALIFICATIONS

15.1 Operation of the Facility

The UMR Reactor Facility is part of the University of Missouri system and is a state supported institution. Funding is provided by the University of Missouri-Rolla. Annual operating costs provided by the University for the five year period between 1999 and 2003 are typical and are presented in Table 15.1.

Table 15.1-Operating Costs for the UMR Reactor.

Description	1998-99	1999-00	2000-01	2001-02	2002-03
Salary & Wages	$122,878	$135,876	$107,551	$135,511	$135,511
Employee Benefits	$27,505	$30,942	$22,693	$26,303	$29,585
E&E (Expenses & Equip.)	$18,127	$17,814	$18,089	$17,214	$16,414
Total Amount of Operational Expenses	$168,510	$184,632	$148,333	$179,028	$181,510

Information provided in Table 15.1 shows that the majority of operating expenses are for staff salaries. The reactor is relatively inexpensive to operate. Expenses and equipment (E&E) funding is less than $20,000 per year. Funding the level shown in Table 15.1 is adequate for the safe operation of the facility. Not reflected in table 15.1 are additional monies obtained from external sources acquired by submitting proposals to various organization's (e.g. government agencies, industry, etc). External funding provides resources to expand activities and to upgrade facilities.

Table 15.2 provides the projected annual operating cost for the years 2004-2008. Funds to cover the basic operating cost will be provided by the University.

Table 15.2-Projected 5 Year Annual Costs for the UMR Reactor.					
Description	**2003-04**	**2004-05**	**2005-06**	**2006-07**	**2007-08**
Salary & Wages	**$136,905**	**$142,381**	**$148,076**	**$154,000**	**$160,159**
Employee Benefits	**$26,012**	**$27,052**	**$28,135**	**$29,260**	**$30,430**
E&E (Expenses & Equip.)	**$16,500**	**$16,830**	**$17,167**	**$17,510**	**$17,860**
Total Amount of Operational Expenses	**$179,417**	**$186,264**	**$193,378**	**$200,769**	**$208,450**

15.2 Decommissioning of the Facility

The University has committed to provide financial resources as needed to decommission the facility at the end of the facility's operating life. A statement of intent [15-1] as required by 10CFR50.75 from the Vice President of Administrative Affairs has been provided to the NRC that assures funds for decommissioning. The cost of decommissioning was estimated at $850,000 in 1990 [15-1] and has been adjusted using guidance in NUREG-1307 [15-2] to be about $1,925,000 in the Year 2005.

15.3 References

[15-1] McGill, James T., "Statement of Intent as Required by 10CFR50.75 Assuring That Funds for Decommissioning will be Obtained When Necessary" Letter to the Nuclear Regulatory Commission from James T. McGill. Vice President For Administrative Affairs, University of Missouri, License R-79, Docket 50-123. July 9, 1990.

[15-2] NUREG-1307, "Report on Waste Burial Charges-Escalation of Decommissioning Waste Disposal Costs at Low-Level Waste Burial Facilities" US Nuclear Regulatory Commission, Rev. 10, 2002.