NUREG-1086

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

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ABSTRACT

This Safety Evaluation Report for the application filed by the University of Missouri-Rolla for a renewal of Operating License R-79 to continue to operate a research reactor has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is owned by the University of Missouri and is located on the campus in Rolla, Missouri. On the basis of its technical review, the staff concludes that the reactor facility can continue to be operated by the university without endangering the health and safety of the public or the environment.

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1 INTRODUCTION

The University of Missouri at Rolla (UMO/licensee) submitted a timely application to the U.S. Nuclear Regulatory Commission (NRC/staff) for renewal of the Class 104 Operating License (R-79), for its open-pool-type research and training reactor. The application, with supporting documentation, was transmitted by letter dated October 15, 1979, as supplemented, requesting renewal of the license for a period of 10 years. The licensee is permitted to operate the reactor within the conditions authorized in the existing license, as amended, in accordance with Title 10 of the Code of Federal Regulations, Paragraph 2.109 (10 CFR 2.109), until NRC action on the renewal request is completed.

The renewal application references information regarding the original design of the reactor facility and contains information about modifications to the facility made since initial licensing.

The application also includes a revised Final Safety Analysis Report (FSAR), information for an environmental impact assessment, financial information, an Operator Requalification Program, revised Technical Specifications, and a revised Physical Security Plan which is protected from public disclosure under 10 CFR 2.790(d)(1) and 10 CFR 9.5(a)(4).

The staff's technical review with respect to issuing a renewal operating license to the UMO has been based on visits to the facility and on the information contained in the renewal application and supporting documents, plus responses to requests for additional information. This material is available for review at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555. This Safety Evaluation Report (SER) was prepared by R. E. Carter, Project Manager, Division of Licensing, Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission. Major contributors to the technical review include the project manager and J. E. Hyder, C. C. Thomas, and C. Linder of the Los Alamos National Laboratory under contract to NRC.

The purpose of this SER 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 SER will serve as the basis for renewal of the license for operation of the UMRR facility at power levels up to and including 200 kW thermal. 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). Because there are no specific accident-related regulations for research reactors, the staff has at times compared calculated hypothetical radiation dose values with related standards in 10 CFR 20, "Standards for Protection Against Radiation," for employees and the public.

The UMRR was initially licensed for operation at 10 kWt in December 1961 as an open-pool-type reactor, with fuel of the Materials Testing Reactor (MTR) type.

In 1966 the licensee initiated a program of upgrading the reactor facility, and the license was amended to permit operation at power levels up to and including 200 kWt. The reactor has been operated at those power levels since 1967.

1.1 Summary and Conclusions of Principal Safety Considerations

The staff's evaluation considered the information submitted by the licensee, past operating history recorded in annual reports submitted to the Commission by the licensee, reports by the Commission's Office of Inspection and Enforcement, and onsite observations. In addition, as part of its licensing review, the staff obtained laboratory studies and analyses of credible accidents postulated for plate-type reactors. The principal safety matters reviewed for the UMRR and the conclusions reached follow:

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

(2) 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. The staff performed conservative analyses of the most serious credible accidents and determined that the calculated potential radiation doses outside of the reactor room would not exceed 10 CFR 20 guidelines for persons in unrestricted areas.

(3) The licensee's 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.

(4) 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).

(5) The licensee's 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.

(6) The financial data provided by the licensee are such that the staff has determined that the licensee has sufficient revenues to cover operating costs and eventually to decommission the reactor facility.

(7) The licensee's program for providing for the physical protection of the facility and its special nuclear material complies with the requirements of 10 CFR 73.

(8) The licensee's procedures for training reactor operators and the plan for operator requalification are acceptable. These procedures give reasonable assurance that the reactor facility will be operated with competence.

(9) The licensee's Emergency Plan provides reasonable assurance that the licensee is prepared to assess and respond to emergency events.

1.2 Reactor Description

The UMRR is a heterogeneous, swimming-pool-type nonpower 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 and to allow insertion of graphite reflector elements to displace reflector water. Each fuel element consists of several thin metal plates assembled into a unit about 7.6 cm by 7.6 cm with an active fuel length of ~0.61 m. Fuel elements of this general configuration were first designed for and used in the Materials Testing Reactor (MTR) and thus are referred to as MTR-type fuel elements. Four of the fuel elements were fabricated with the four 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 neutron and gamma-ray fluxes are suspended near the core. The control console, from which the operator can observe the reactor room and the top structures of the reactor through a large window, is located in a small room adjacent to the reactor room. The control console consists of typical read-out and control instrumentation.

1.3 Reactor Location

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 and 290 km, respectively.

1.4 Shared Facilities and Equipment and Special Location Features

The reactor building is separate from other buildings on the 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 approximately 50 other research reactors operating in the United States and at least 25 reactors operating in foreign countries. The control and instrumentation systems, while different in detail, are based on the same operating principles used for these 75 other research or test reactors.

1.6 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 Department of Energy (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 universities and other government agencies operating nonpower 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.

2 SITE CHARACTERISTICS

2.1 Geography

The site for the UMRR is near the eastern edge of the campus of the University of Missouri at Rolla, Phelps County, Missouri. There are few large centers of population in central Missouri, and the nearest large city is St. Louis, 161 km to the northeast.

The general terrain of and around the campus is characterized as largely hilly and rolling, with the reactor site itself being in a relatively flat area.

The location of the UMRR within the campus is at the center of the concentric circles shown on the map of Rolla in Figure 2.1. The nearest off-campus residential area is approximately 100 m from the reactor building. Figure 2.2 shows the location of Rolla with respect to other major cities of the state.

2.2 Demography

The daytime population within ~0.2 km of the reactor site is normally 8,000 people, including students and university staff. Because the campus is within the Rolla city limits, the population within 0.4 km during working hours is normally ~9,000 people. Most of the population of Rolla, ~14,000, reside within 2 km of the reactor site.

2.3 Nearby Industrial, Transportation, and Military Facilities

There is no large industry, heavily traveled transportation route, or military installation in or near Rolla. There is a large military base, Ft. Leonard Wood, some 40 km southwest of Rolla. In addition to the university, Rolla is the headquarters for the Missouri Geological Survey and also home of a United States Bureau of Mines research division and important divisions of the United States Geological Survey concerned with topographic mapping and water resources.

Because there are no industrial, military, or major transportation facilities in the near vicinity of the reactor site that could directly or indirectly cause accidental damage to the reactor, the staff concludes that such accidents need not be hypothesized and evaluated.

2.4 Meteorology

The general climate in the Rolla area of Missouri is of a continental midwestern type, not influenced by any local mountains or large bodies of water. Temperatures have a continental range with hot summers to generally mild winters ranging over 38°C (100°F) to -20°C (-4°F). Detailed surface wind and precipitation data have been obtained from a Civil Aeronautics Administration (CAA) station at Vichy, some 21 km north of Rolla and at the same elevation. There seems to be no reason not to rely on these data for Rolla, which indicate relatively constant southwesterly prevailing winds with various wind conditions.

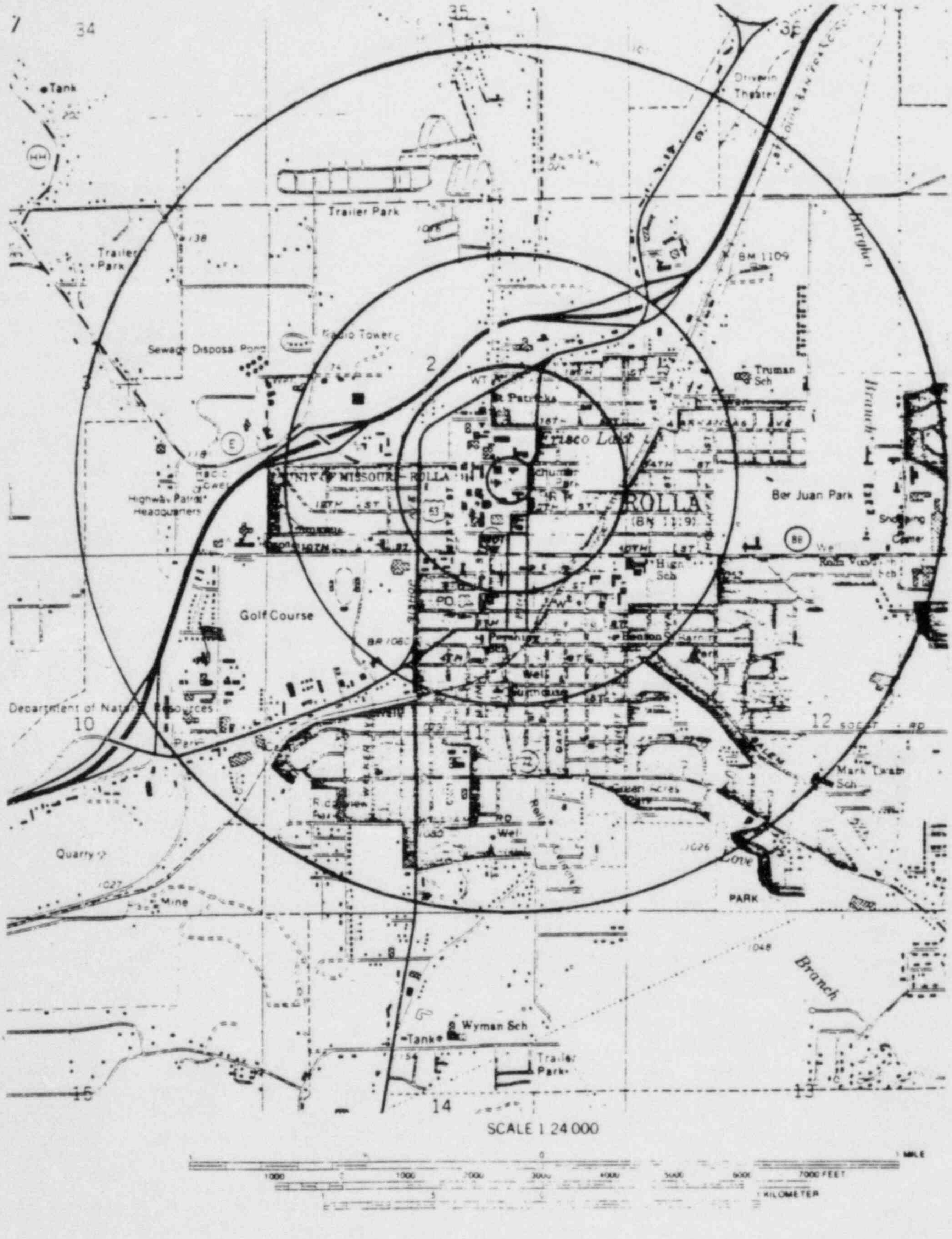

Figure 2.1 Map of Rolla area

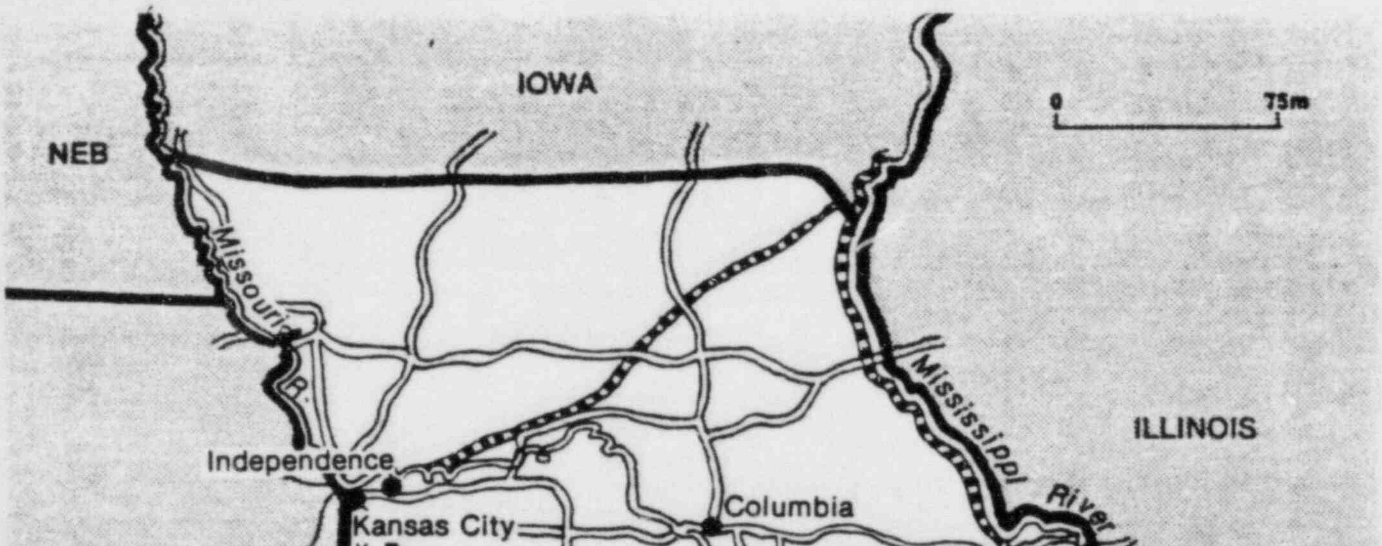

Figure 2.2 Map of the State of Missouri

There is little variation of the most frequent winds from day to night or during periods of precipitation or low visibility. These general conclusions apply for all seasons. The data on winds during precipitation are relevant in considering washout of airborne particulates, and the data on winds during low visibility are relevant in periods of atmospheric stability.

Wind data at higher elevations have been obtained at the airport in St. Louis. Considering the general topography of the area, it seems reasonable to assume these data also apply at Rolla. The general wind flow is from the west-northwest quadrant, with velocities increasing steadily as the elevation increases above the land surface.

Tropical disturbances generally do not influence the weather in Rolla, and even though tornados occur frequently in some of the midwestern states, their frequency and intensity in the Rolla area are not high.

On the basis of the meteorological data presented in the licensee's SAR, the staff concludes that the meteorological conditions at the site do not pose a significant risk of damage to the reactor nor render the site unacceptable for the facility.

2.5 Geology

The UMRR site is located within the Central Stable Region which is characterized by gentle arches, domes and sedimentary basins, resulting from tectonic episodes in the Paleozoic Era approximately 225 million years before present (mybp). The site is on the northwest flank of the Ozark uplift approximately 75 km west of the St. Francois Mountains, which are comprised of Precambrian igneous and metamorphic rocks. Numerous geologic structures, such as folds and faults, have been documented in the site region. Some of these have mapped lengths of up to 60 km and strike northwest-southeast with the overall structural fabric of the region. The closest observed fault is ~20 km south of the site, but the main concentration of longer faults terminates ~35 km west of the site.

Recent geological and geophysical studies have revealed a buried late Precambrian rift beneath the Upper Mississippian embayment area. The rift has influenced the tectonics and geologic history of the area since Precambrian time (570 mybp) and is presently associated with the contemporary earthquake activity of the New Madrid seismic zone. The rift formed as a result of continental breakup and has been reactivated by compressional stresses related to plate tectonic interactions. The closest approach of the New Madrid seismic zone to the UMRR site is ~200 km distant.

The seismic implications of the numerous northwest trending faults, the St. Francois Mountains, and the New Madrid seismic zone to the UMRR site are discussed in Section 2.7 of this report.

The sedimentary rock section in the Rolla area averages about 510 m (1,700 ft) in total thickness. This section consists largely of Paleozoic dolomites and magnesium limestones, but with some sandstone and shale members (see Figure 2.3). The Cambrian Lamotte Formation, a basal sandstone, usually is encountered in deep wells. The Lamotte unconformably overlies Precambrian metamorphic and igneous rocks.

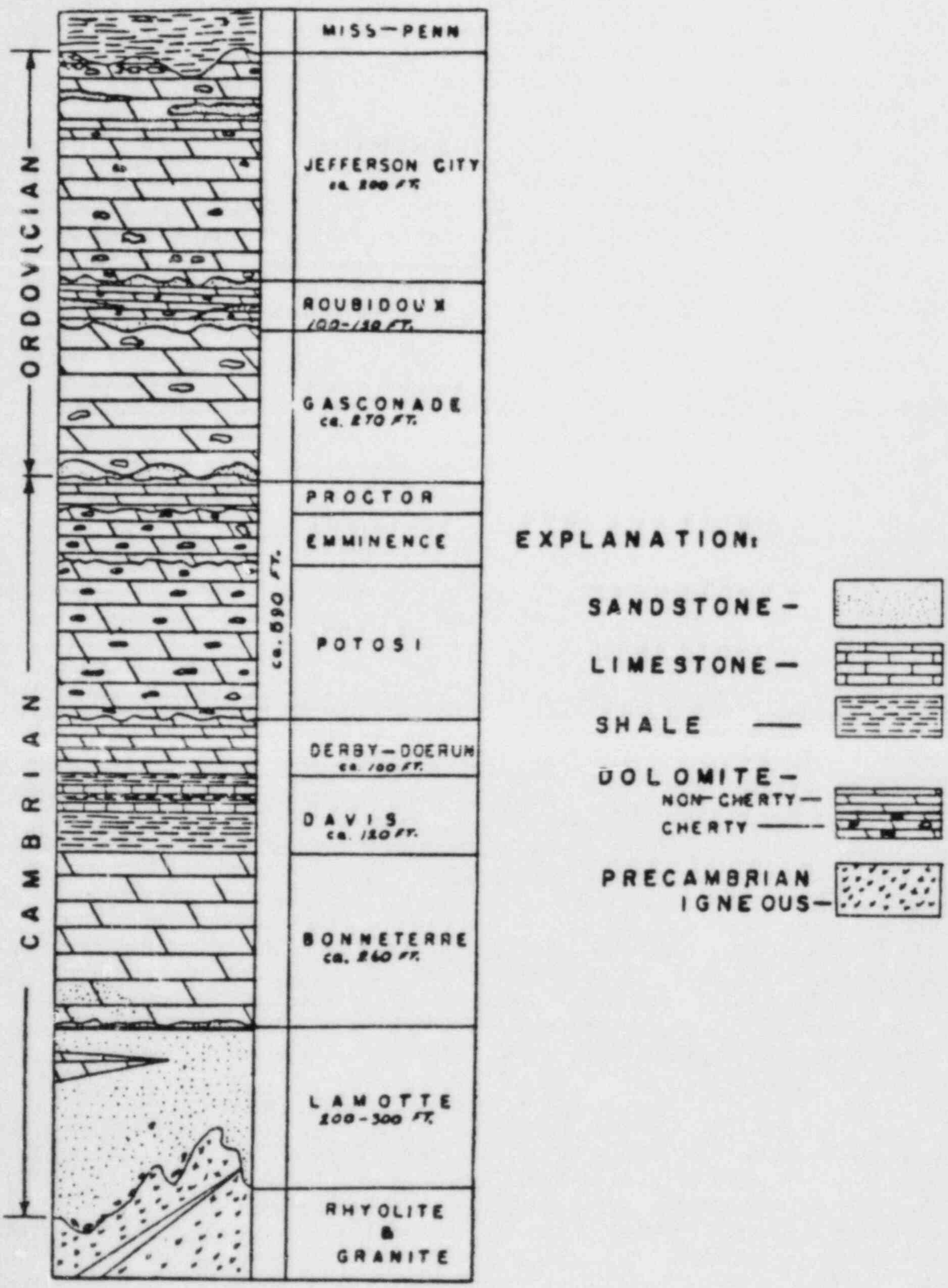

Figure 2.3 Geologic column, Rolla area, Missouri

The geographical center of the Ozark uplift lies to the southeast of Rolla. Consequently, the regional dip in the Rolla area is toward the northwest, with a very gentle gradient of less than 1°. In places, however, sink holes developed in the Gasconade, Roubiodoux, and Jefferson City formations (see Figure 2.4) cause high local dips and even faulting.

The sink holes were caused by collapse of old solution channels in the carbonate rocks. Surface exposures of sink holes at Rolla ordinarily show solidly compacted fillings of clay shale and sandstone of Pennsylvanian age. In view of the fact that sink holes have been observed and mapped at the site, there is a possibility that future collapse of a solution cavity in the plant area could affect plant operations. However, as discussed in Section 14, loss of coolant or fuel damage does not lead to unacceptable radiation levels in unrestricted areas.

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

2.6 Hydrology

The drinking water for the city of Rolla comes mainly from wells that are cased for varying depths from the surface. These wells, in turn, are fed by aquifers that generally run within submerged geologic formations, with occasional outcropping in local streams. Surface waters at the reactor site are drained into streams that flow toward the east, eventually emptying into the Meramec River. There are no known uses of this river's water for drinking until it feeds underground deep-driven wells in the suburbs of St. Louis some 150 km from Rolla. The Meramec River finally joins the Mississippi River about 19 km south of St. Louis.

On the basis of the above information, the staff concludes that the hydrological conditions at the reactor facility do not render the site unacceptable for the research and training reactor location.

2.7 Seismology

The UMRR site is located within the Central Stable Region. There is no known relationship between historic seismicity and the structures in this region. The maximum historic earthquakes in the western section of the region were Modified Mercalli (MM) intensity VII. The area near Rolla has low seismicity; within 50 km of Rolla there have been no historic earthquakes (USGS Open-File Report 84-225).

Nuttli and Brill (NUREG/CR-1577) have delineated seismic source zones for the Central United States based on the distribution of historic earthquakes and knowledge of the tectonic features. The St. Francois uplift is the earthquake source zone nearest the UMRR site (about 75 km east of the site). The largest historic earthquake in the St. Francois zone was a magnitude (m_b) 5.0, maximum MM intensity VI, in 1977.

The other area of significance is the New Madrid seismic zone (see Section 2.5) whose closest approach is about 200 km southeast of the site. Major earthquakes of maximum MM intensity XI-XII occurred in the seismic zone near the community

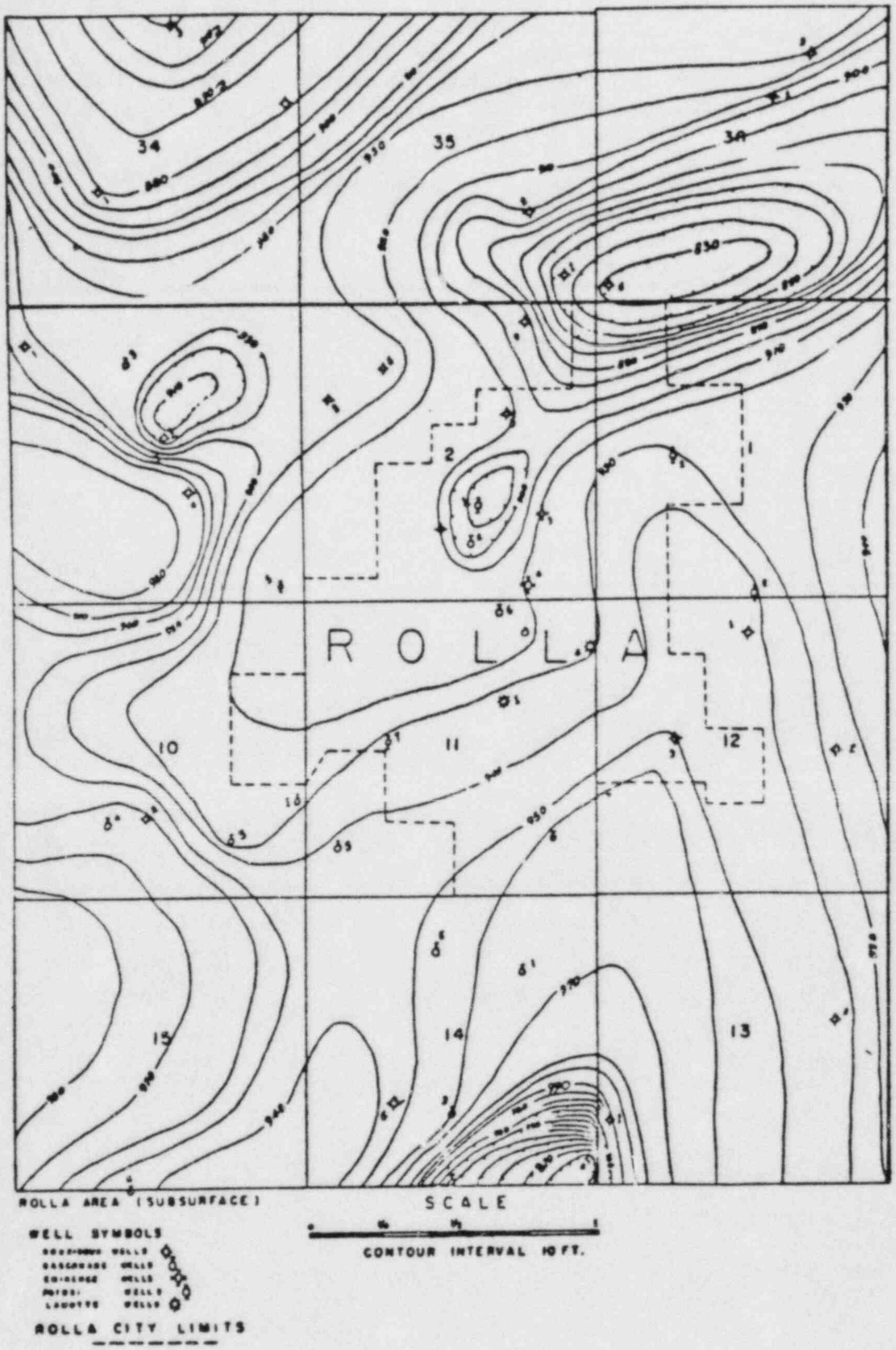

Figure 2.4 Rolla area subsurface

of New Madrid in 1811 and 1812. The epicentral region of these events continues to be the center of frequent earthquakes, as reported by St. Louis University in the Central Mississippi Valley Earthquake Bulletins (NUREG/CR-3768). More recent locations of microearthquakes show alignments along zones that are within a buried northeast trending rift zone.

The staff concludes that the history of earthquake activity, with no damaging historic earthquakes near Rolla, supports the conclusion that the risk of seismic-induced hazards to the UMRR is not significant.

2.8 Conclusion

The staff has reviewed and evaluated the UMRR site for both natural and manmade hazards and concludes that there are no significant risks associated with the site that make it unacceptable for the continued operation of the reactor.

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Reactor Building

The building housing the UMRR is constructed of insulated steel curtain walls. 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, all 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 of approximately 0.5 mm of water is maintained within the reactor building to control outleakage of air.

The building is essentially a rectangular structure 15 m by 10 m by 10 m high. An office/reception/entrance area ($\sim$24 m^2) was added to the building in 1979/1980. The main floor contains a reactor room, control room, counting rooms, and the new office space. The floor at one end of the reactor bay is lower to provide experimentalists with access to the beam tube and the thermal column. There is a normally locked service door, which is sealed by a neoprene gasket, on the opposite end of the reactor room. The building layout is shown in Figures 3.1 and 3.2.

3.2 Wind Damage

The Rolla, Missouri, area experiences few extreme wind conditions such as tornados or inland hurricanes. Further, the reactor building is constructed of a steel frame and poured concrete floor, and the reactor pool is formed by a poured concrete biological shield that is reinforced. On this basis, the staff concludes that wind or storm damage to the UMRR facility is very unlikely.

3.3 Water Damage

The reactor site is situated on gently sloping terrain, but well above any flood plain. Therefore, the staff concludes that there is reasonable assurance that significant damage to the reactor because of flooding is not likely and that the site is suitable from this perspective.

3.4 Seismic-Induced Reactor 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. In the event of an earthquake causing catastrophic damage to the reactor building and/or the reactor pool, water might be released. However, Section 14 of this SER 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. On the basis of these considerations, the staff concludes that the risk of radiological hazard resulting from seismic damage to the reactor facility is not significant.

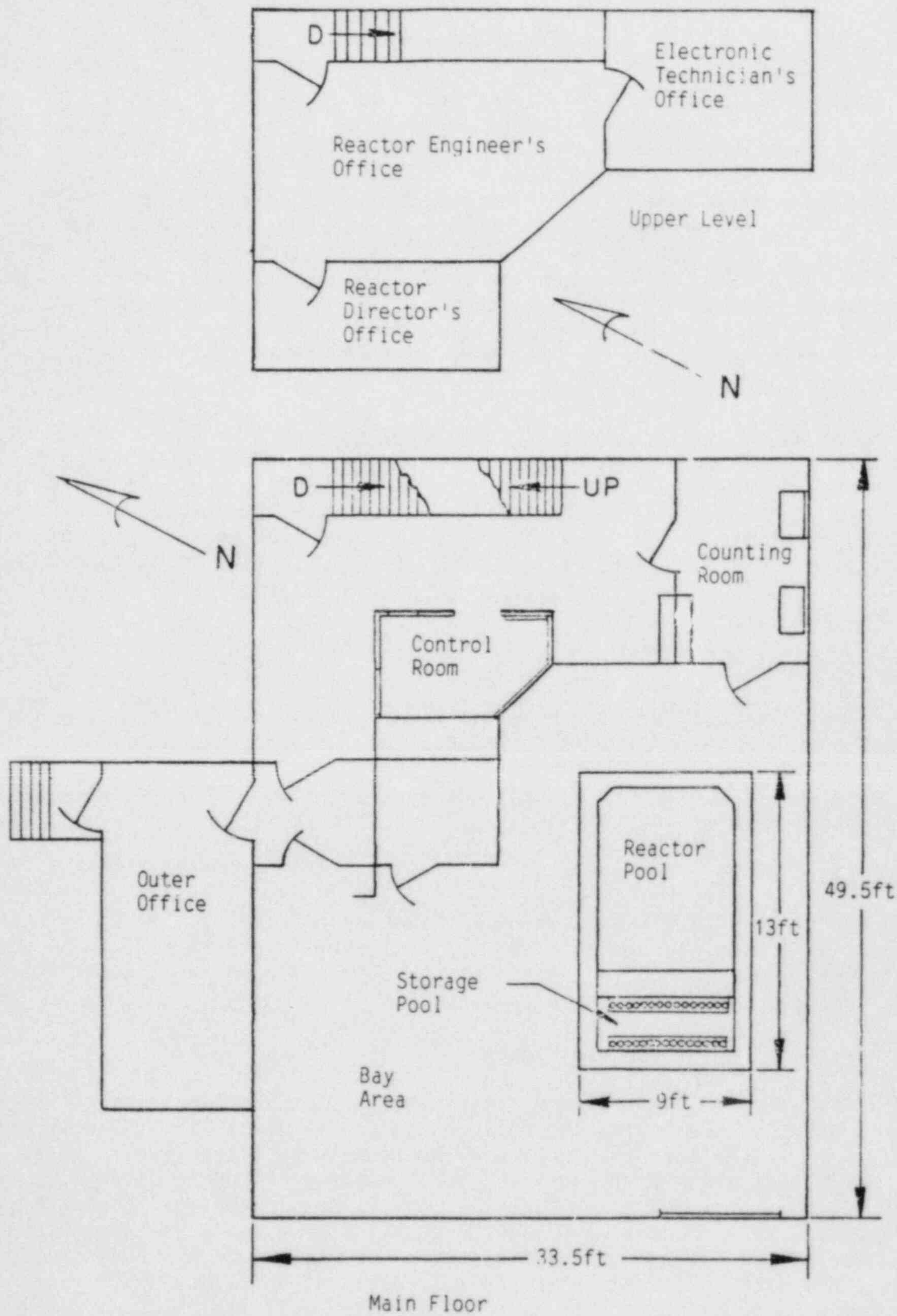

Figure 3.1 UMRR upper level and main floor layouts

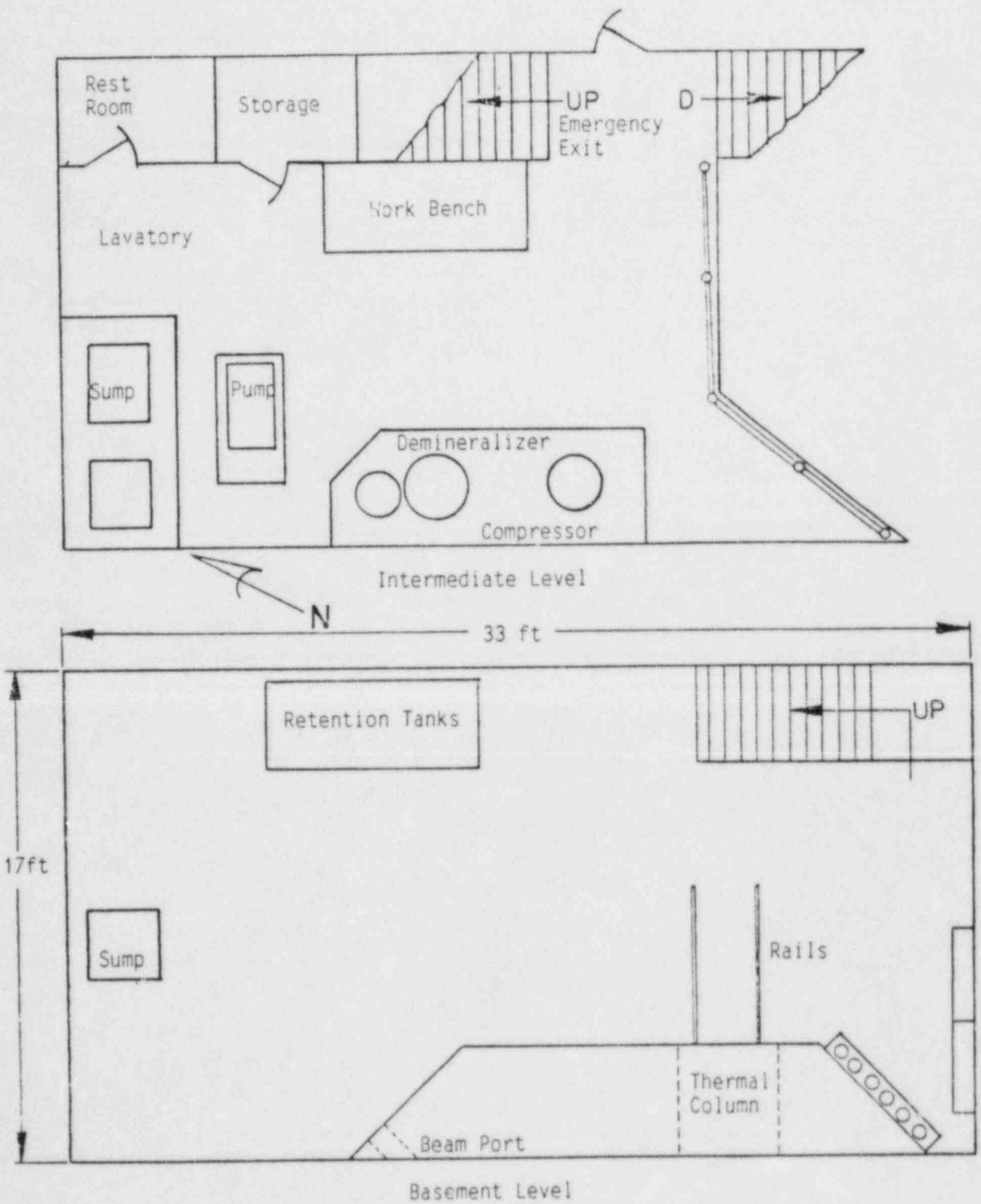

Figure 3.2 UMRR intermediate level and basement level layouts

3.5 Mechanical Systems and Components

The mechanical systems important to safety are the neutron-absorbing control rods suspended from the reactor superstructure. The motors, gear boxes, electromagnets, switches, and wiring are all 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.

The recent history of operation of the UMRR indicates few malfunctions of electromechanical systems, no persistent malfunction of any one component, and most malfunctions were one of a kind (i.e., few repeats). (See Inspection Reports and licensee reports to the Commission.) On the basis of the above information, the staff concludes that there has not been significant deterioration of this equipment with time or with operation and there is reasonable assurance that continued operation of the UMRR facility will not increase the risk to the public.

3.6 Conclusion

The UMRR was designed and built to withstand all credible and probable wind and water events associated with the site. A large seismic event has a small likelihood of occurring and the consequences of such an event would not pose a significant radiological hazard to the public (see Section 14). There is no evidence of significant deterioration of systems or components. Therefore, the staff concludes that the construction of the facility is acceptable and that continued operation, as proposed, will not cause significant radiological risk to the public.

4 REACTOR

The University of Missouri-Rolla reactor (UMRR) is an open-pool type with a core containing 2.7 kg of ^{235}U fuel enriched to ~90%. It is a light-water moderated, water- and graphite-reflected reactor that is licensed to operate at power levels up to and including 200 kWt. The fuel, core configuration, control rods, and control instrumentation are similar to some 75 research reactors operating throughout the world. At least 30 MTR-type reactors have been evaluated and licensed by the AEC/NRC.

The reactor core is immersed in a vinyl-painted, reinforced-concrete, water-filled, open-topped pool. The pool is spanned by a movable structure that supports the reactor core, control rod systems, reactor instrumentation, and some experimental facilities. A partial reactor core configuration is shown in Figure 4.1.

Reactor control is achieved by inserting or withdrawing neutron-absorbing control elements suspended from the drive mechanisms. Heat generated by fission is transferred from the fuel to the pool water. Cooling is provided by natural convection of the water within the pool.

The UMRR is used for class instruction, student experiments, reactor operator training, research, and radioisotope production. The discussion in the following sections is based on information obtained from licensee reports and during discussions with licensee personnel. The design and performance characteristics of the UMRR are summarized in Table 4.1

4.1 Reactor Core

The core consists of MTR-type fuel elements, four control rod elements, and four control rods. Several different core configurations are possible with this reactor, and there is a 12.7-cm-thick aluminum reactor grid plate containing a 6 by 9 array of holes for positioning the fuel and control rod elements and experimental apparatus. Graphite elements also can be loaded around the core to act as a neutron reflector.

It is not necessary to plug positions in the grid plate not containing fuel, control rod, reflector or experimental elements during operation because convective-flow cooling water tends to pass up through the fueled elements rather than through the empty holes. The grid plate also contains a series of small holes interspaced between the main positioning holes to provide coolant flow between the elements.

4.1.1 Fuel Elements

The fuel and control rod elements are assemblies of fuel-bearing plates. Each plate is a sandwich of aluminum cladding over a uranium oxide-aluminum "meat." The meat is approximately 0.05 cm thick, 6.35 cm wide, and contains about 17 g ^{235}U. The cladding is 0.05 cm thick. The overall dimensions of a fuel plate are approximately 7.62 cm wide, 61 cm long, and 0.15 cm thick.

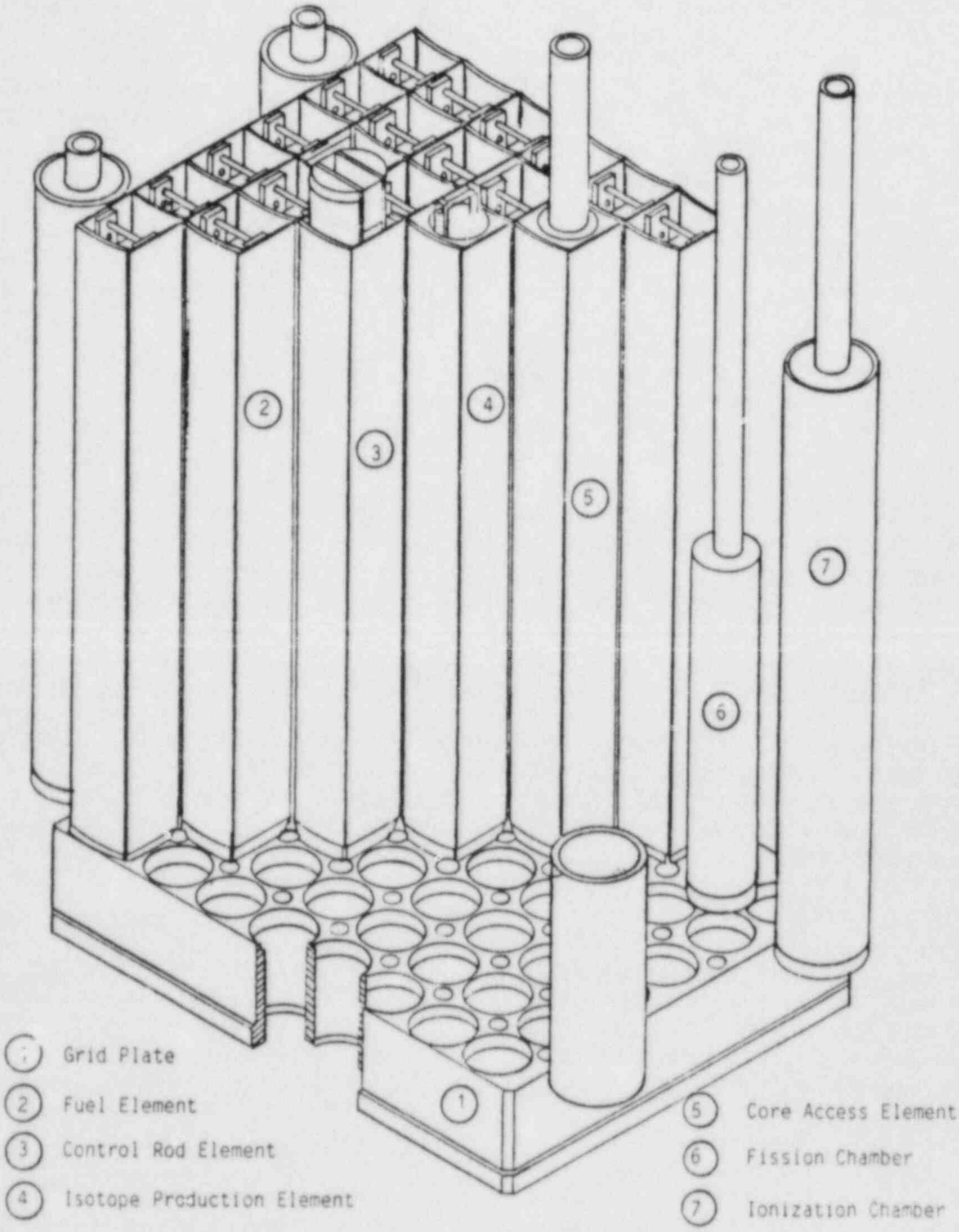

Figure 4.1 Typical reactor core

Table 4.1 Current University of Missouri-Rolla reactor design and performance characteristics

Item	Characteristic
General Feature	
Reactor type	Heterogeneous-pool
Licensed rated power level	200 kW thermal
Excess reactivity	1.5% $\Delta k/k$
Clean-cold critical mass	2.7 kg ^{235}U, water-reflected
Effective prompt neutron lifetime	4.5×10^{-5} s
Effective delayed neutron fraction (β_{eff})	0.0075
Temperature coefficient (measured)	-1.8×10^{-4} $\Delta k/k$ per °C
Void coefficient (measured)	-7×10^{-7} $\Delta k/k$ per cm^3
Average thermal flux at 200 kW	1.6×10^{12} $n/cm^2 \cdot s$
Moderator/coolant	H_2O
Reflector	H_2O and graphite
Fuel and Control Elements	
Number of fuel-bearing plates	
fuel elements	10;5
control elements	6
Enrichment	~90% ^{235}U
Maximum ^{235}U per fuel-bearing plate	17 g
Plate Dimensions	
Thickness	0.15 cm
Width	
fuel core	6.35 cm
plate	7.62 cm
Length	61 cm
Cladding thickness	0.05 cm
Control Rods and Reactivity Effects	
Material	Boron stainless steel
Number	
Safety	3
Regulating	1
Travel	61 cm
Withdrawal speed (maximum)	
Safety	15.2 cm/min
Regulating	61 cm/min
Rod worth	
Safety (single)	2.6-3.4% $\Delta k/k$
Safety (combined)	8.7% $\Delta k/k$
Regulating	0.7% $\Delta k/k$

Table 4.1 Continued

Item	Characteristic
<u>Coolant</u>	
Type	Light water
Flow	Natural convection
Inlet core temperature (nominal)	20°C
Outlet core temperature (nominal)	32°C
Conductivity	$\leq$2 micromhos/cm

The standard fuel element consists of 10 fuel plates fastened with aluminum side plates so that the finished element has an almost square 7.62 cm (3 in.) by 7.62 cm cross section (Figure 4.2). A male guide piece is attached to the bottom end of the fuel element. The guide piece has a circular cross section that mates with the tapered positioning hole in the grid plate. A handle is attached to the top end of the fuel element, providing a means for insertion and removal of the fuel element. The overall length of a fuel element is about 0.91 m. Two half elements (five fueled plates and five dummy plates) also are available for use in adjusting reactivity.

There are four fuel elements for the control rods that are identical to the standard elements, with the exception that the center four fueled plates have been removed and replaced with two guide plates. The guide plates prevent the control rod from coming in contact with a fuel plate. In addition, the fuel plate spacing is somewhat closer than a standard element, and the control rod guide structure prevents lifting of the fuel elements inadvertently.

The licensee also has obtained several standard TRIGA-MTR conversion fuel elements in accordance with an intent to convert to low-enrichment (<20% ^{235}U) fuel. These elements consist of an aluminum guide piece, four stainless-steel clad TRIGA fuel rods, and an aluminum handle. The fuel rods contain a zirconium hydride moderator homogeneously mixed with uranium. Before loading the reactor with TRIGA fuel or mixed MTR-TRIGA fuel, the licensee must prepare an appropriate safety analysis and obtain authorization from the NRC.

4.1.2 Control Rods

The reactivity and power level in the UMRR are controlled by three safety rods and one regulating rod. All four control rods fit into central gaps provided in special fuel elements, as discussed in Section 4.1.1. The rods and their elements could be located in any core position, within the Technical Specifications limits on reactivity.

The three safety rods, which are used for coarse control, are made of boron stainless-steel clad with aluminum. The absorbing section is about 2.22 cm thick, 5.72 cm wide, and 61 cm long. The boron content is about 1.5 to 1.7% natural boron. The reactivity worth of each safety rod varies with the core loading and configuration and is typically about 3% $\Delta k/k$ with a maximum worth of about 3.4% $\Delta k/k$. For a normal core loading, the combined worth of the three safety rods is about 8.7% $\Delta k/k$. Each safety rod is moved in and out of the

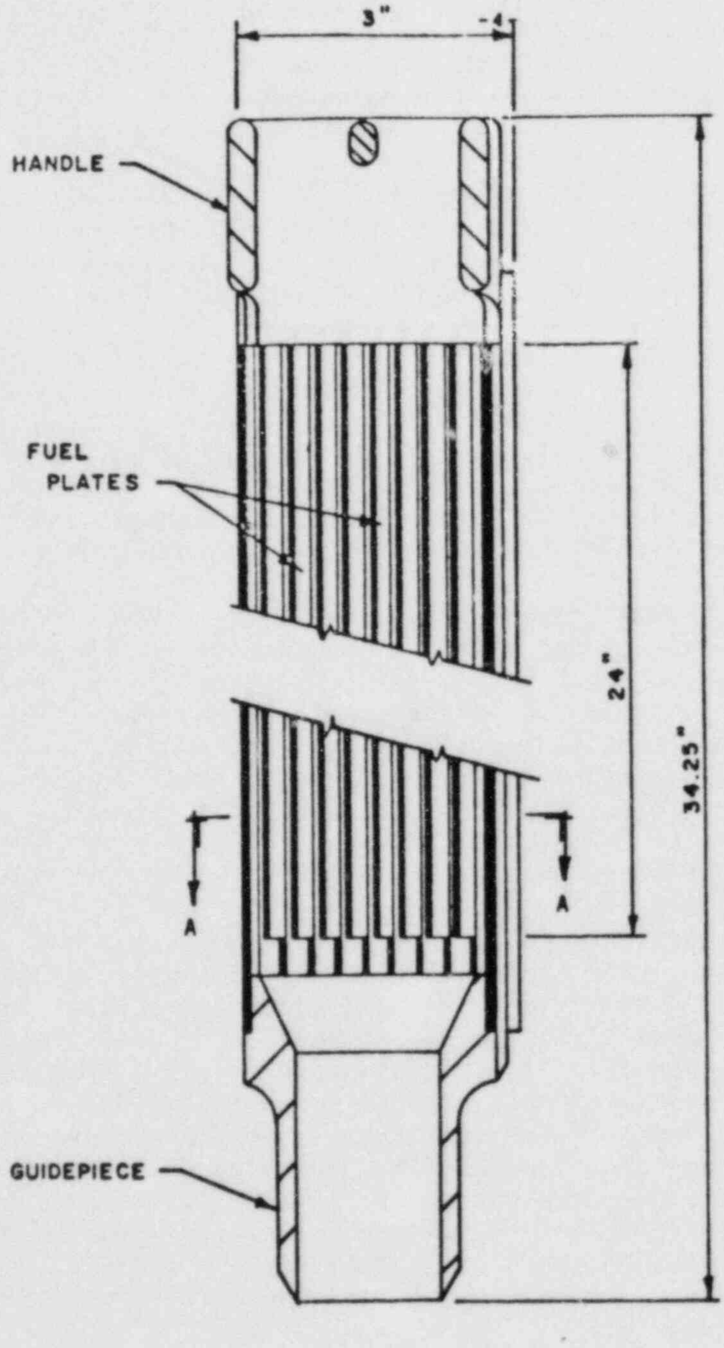

Figure 4.2 UMRR fuel element

core by an individual electromechanical system. The drive mechanisms are located on the reactor bridge and are actuated from the control console. The rods containing the absorber section are suspended from the drive mechanisms by an electromagnet. During normal operation, the safety rods are driven in or out at a rate of 15.2 cm/min. When a scram signal is received, the magnets are deenergized and the safety rods drop freely into the core. Means are provided for automatic or manual scrams, blade reversal, and blade inhibits to maintain the reactor in a safe operating range and for safe shutdown.

The regulating rod, which is used for fine control, is a flattened, ~0.17-cm-thick stainless-steel tube with a 5.72-cm by 2.03-cm cross section and an effective poison length of ~61 cm. The tube is open at the top and bottom allowing free circulation of water through it to eliminate the possibility of trapping air with a resultant variable void condition.

The regulating rod has a reactivity worth of about 0.7% $\Delta k/k$, which varies somewhat with core loading. The regulating rod is permanently fixed to its drive mechanism and travels in either direction at a speed of 61 cm/min. The regulating rod can be operated manually or automatically for servocontrol of the reactor power level.

4.2 Reactor Pool

The reactor pool is ~5.79 m long, 2.74 m wide, and 8.23 m deep and holds 130 m^3 of water. The pool walls are of ordinary reinforced concrete. The internal sides and floor of the pool have several coats of protective vinyl paint to prevent leaching of minerals from the concrete into the water.

A beam tube and a thermal column are located at one end of the reactor pool, and a fuel storage space is located at the opposite end. The fuel storage space is formed by a reinforced concrete bulkhead extending 4.88 m above and 1.07 m below the pool floor and located 0.61 m from the main pool wall. The fuel storage space is designed so that there will be at least 4.88 m of water above stored fuel elements at all times, providing adequate shielding for personnel working in a drained pool. There is no drain system built into this storage pool.

4.3 Reactor Support Structure

The reactor grid plate is supported by an aluminum tower assembly hung from a bridge that spans the pool (refer to Figures 4.3 and 4.4). The bridge is ~3.35 m long and 1.37 m wide and is wheel-mounted on tracks located on the top of the pool walls parallel to the long axis of the reactor pool. The bridge can be moved along its rails for a distance of ~1.83 m from its normal operating position. In the normal operating position, the tower assembly is in contact with the thermal column and the beam tube. Stops are provided on the bridge rails to limit bridge travel within the pool area. The reactor's vertical position is not variable; the bottom of the core is ~1 m above the pool floor. With this core elevation, the top of the active fuel region is >6 m below the surface of the water when the pool is full.

4.4 Reactor Instrumentation

The reactor instrumentation is discussed in Section 7 of this report.

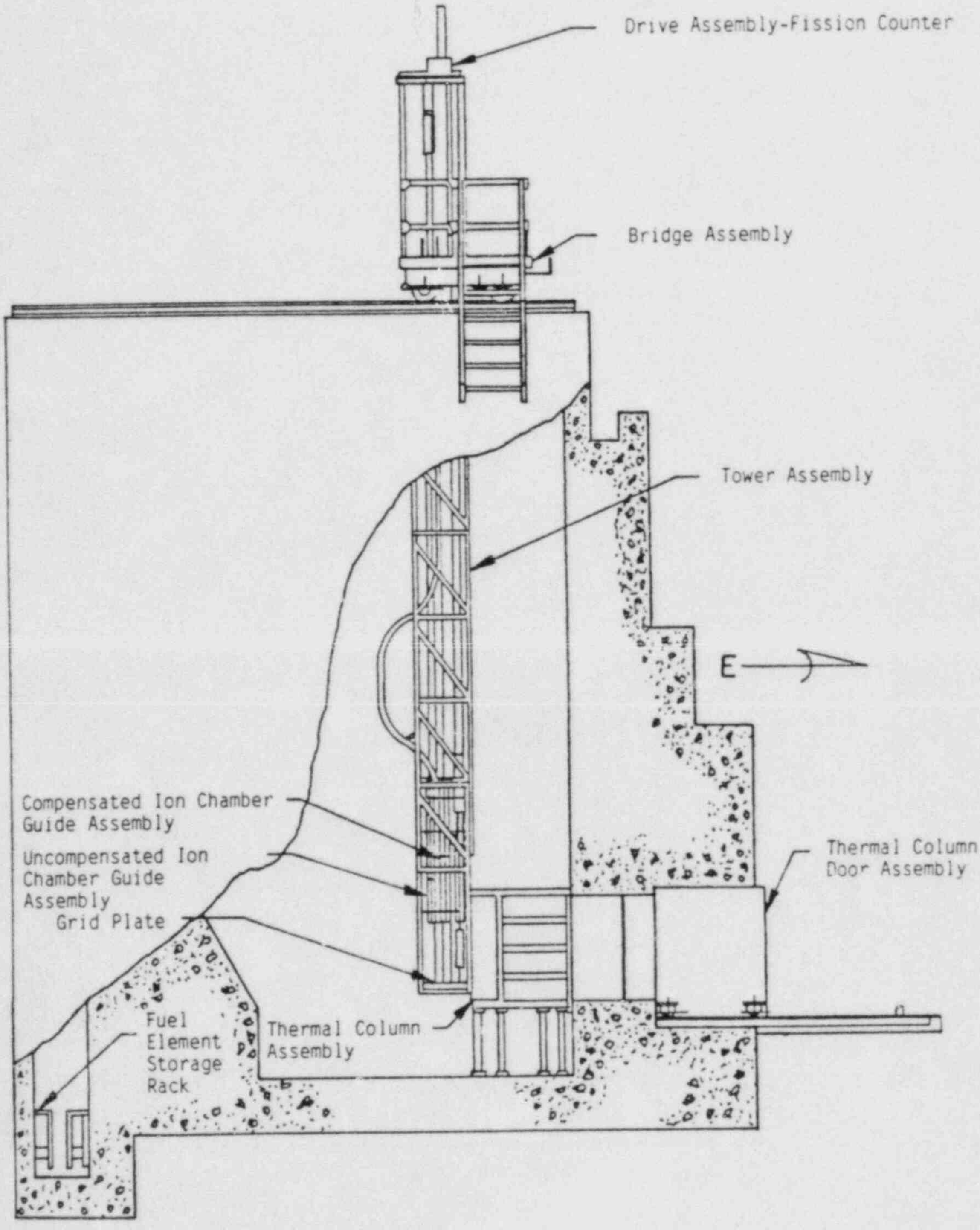

Figure 4.3 Cutaway view of UMRR pool

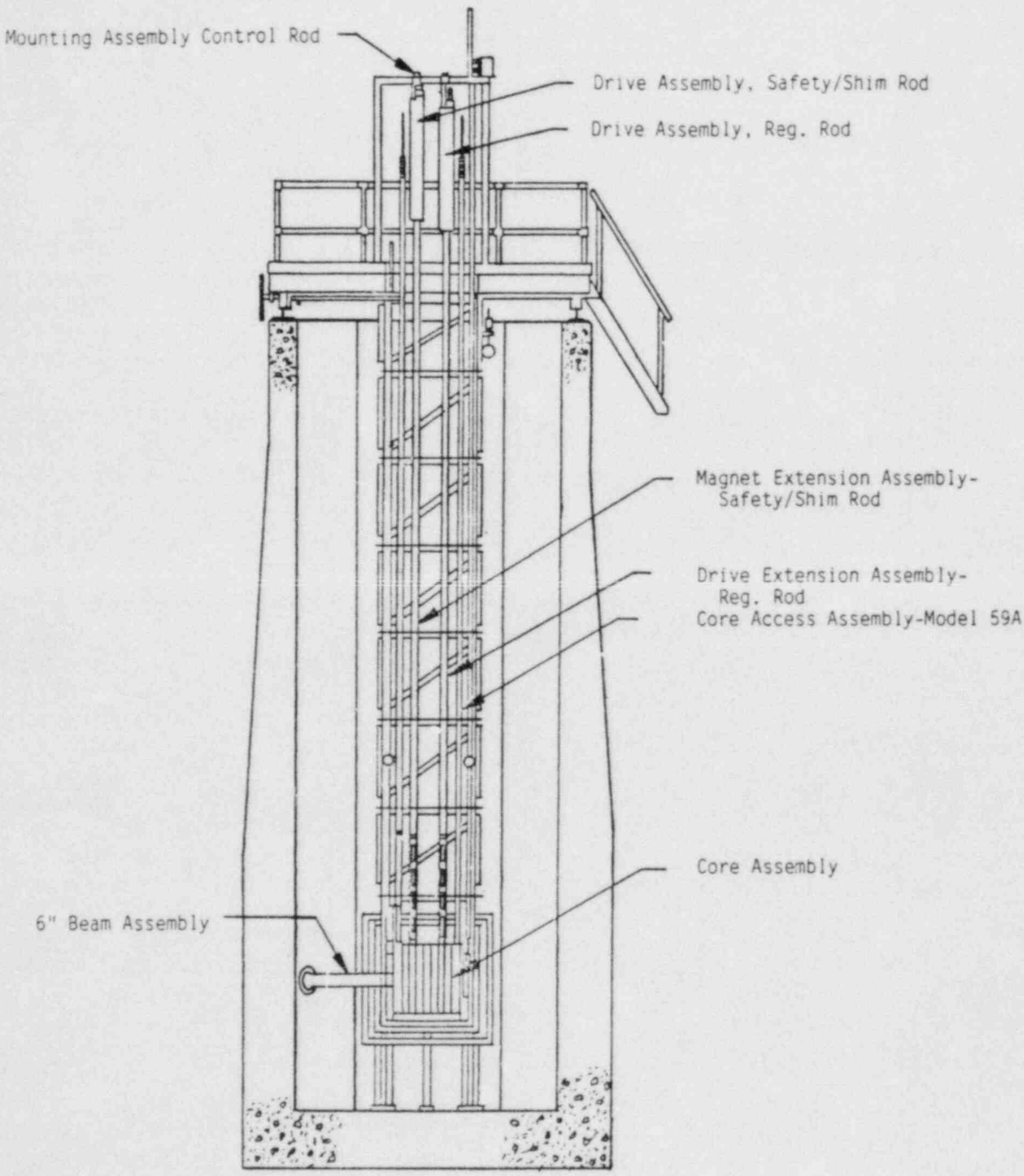

Figure 4.4 Vertical cross-sectional view of UMRR

4.5 Biological Shield

The reactor core is shielded in the lateral direction by pool water, by the concrete walls of the pool, and on three sides by earth shielding. Vertical shielding is provided by about 6 m of water above the core and 1 m between the core and the pool floor. The pool walls are 0.46-m-thick reinforced concrete except at the beam tube and thermal column end (Figures 4.3 and 4.4), where the thickness is 1.98 m (6.5 ft). The increase in wall thickness extends above the floor of the main operating level with the thickness decreasing in steps as shown in Figure 4.3. The other three sides of the pool are below ground level and backed by earth shielding.

The staff concludes that the shielding was designed adequately to reduce external radiation exposure rates to acceptable levels.

4.6 Dynamic Design Evaluation

The reactor is provided with redundant and diverse rapid-response controls and nuclear instrumentation (Section 7) 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 licensee, reactor vendor, and the staff have performed analyses of reactor dynamic behavior initiated by various changes in reactivity. A detailed evaluation of reactivity insertions by means of the control rods is discussed in Section 14.4.

4.6.1 Shutdown Margin

The Technical Specifications prescribe a minimum reactivity shutdown margin of 1.0% $\Delta k/k$ in a cold, xenon-free core with the highest worth safety control rod fully withdrawn and the highest worth unsecured (movable) experiment (see Section 4.6.3) in the core in its most reactive state. Depending on the core loading, the reactivity worth of this maximum safety rod ranges from about 3% to 3.4% $\Delta k/k$, and the total worth of all safety rods is about 8.7% $\Delta k/k$. Therefore, the shutdown margin limitation would allow up to 4.3% $\Delta k/k$ ((8.7 - 3.4) - 1.0) excess reactivity, including the maximum worth movable experiment in addition to any nonmovable experiments, to be loaded into the reactor. 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 (1) the most reactive safety rod were to fail to insert and (2) the maximum worth movable experiment were totally displaced simultaneously.

4.6.2 Excess Reactivity

Maximum excess reactivity in the UMRR core is normally limited to 1.5% $\Delta k/k$ by the Technical Specifications. This amount provides for the negative power defect (of reactivity) at 200 kW, the negative reactivity effect of xenon at equilibrium at 200 kW, and $\sim$1.0% $\Delta k/k$ for experiments, uranium burnup, and operational flexibility. Although the fundamental criterion is to maintain ensured capability to shut the reactor down, hence the minimum shutdown margin, imposing a limit on the total excess reactivity as well helps ensure that the SAR analyses are applicable to the operational core.

To provide sufficient excess reactivity for accurate control rod total and differential worth measurements, the licensee's Technical Specifications permit an excess reactivity of 3.5% $\Delta k/k$ 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 to the periphery of the core. The worth of one fuel element in such a position is less than 1.5% $\Delta k/k$. The analyses, in Section 14.2.2, of rapid reactivity insertions indicate that a step insertion of 1.5% $\Delta k/k$ will not result in fuel or core damage. Thus, addition of fuel elements, one at a time, to the periphery of the core to increase the core excess reactivity will not have consequences more severe than those analyzed in the step reactivity insertion accident.

4.6.3 Experiments

The licensee's Technical Specifications provide limitations on the reactivity worth of secured and movable experiments and on reactivity insertion rates for experiments with moving parts. The staff has analyzed these limitations on the basis of information provided by the licensee in the preliminary hazards summary report, the revised SAR, and the revised Technical Specifications.

The Technical Specifications limit a single secured experiment to 0.7% $\Delta k/k$. This is less than β_{eff} for the UMRR, and thus failure of the features designed to meet the criteria for a secured experiment and its subsequent movement would not result in prompt criticality. Furthermore, a step increase in reactivity of 0.7% $\Delta k/k$ would result in a stable reactor period of <3 s, which would initiate a period scram (set point <5 s).

The Technical Specifications (1) define a movable experiment as one that can be inserted, removed, or manipulated while the reactor is critical and (2) limit the reactivity of such experiments to 0.4% $\Delta k/k$ per experiment. This is well below the 1.5% $\Delta k/k$ step reactivity insertion that the licensee has determined, on the basis of the BORAX and SPERT experiments (Dietrich, 1954; Nyer, 1956), would not result in damage to the UMRR MTR-type fuel elements. The staff has reviewed the limitations on experiments with moving parts, provided in the Technical Specifications, and finds them to be more conservative than the limitations on movable experiments. Therefore, the staff concludes that the safety analysis of movable experiments is applicable to experiments with moving parts.

The staff has reviewed the limitations on the worth of movable and secured experiments, provided in the Technical Specifications, and concludes that they are conservative and provide reasonable assurance that reactivity insertions caused by failure of single experiments would not result in damage to the fuel. Furthermore, the simultaneous removal of four movable experiments with a total worth at, or close to, the limit of 1.2% $\Delta k/k$ has the potential for a step reactivity insertion less than the 1.5% $\Delta k/k$ that the licensee has demonstrated would not result in damage to the UMRR MTR-type fuel. However, the staff considers that the probability of such a fourfold coincidence is negligibly small.

4.6.4 Conclusions

On the basis of the considerations presented above, the staff concludes that (1) a limitation on reactivity worth of each secured experiment of 0.7% $\Delta k/k$, (2) a movable experiment limitation of 0.4% $\Delta k/k$ per experiment with a total reactivity worth limitation of 1.2% $\Delta k/k$ for all experiments, (3) a limitation on reactivity insertion rates of experiments with moving parts, and (4) operation in compliance with minimum shutdown margin requirements of the Technical Specifications provides assurance that these experiments will not lead to a reactivity insertion that will cause fuel damage that would pose a threat to the health and safety of the public. In addition, the staff concludes that the 1.0% $\Delta k/k$ shutdown margin limitation is sufficient to ensure that the reactor can be adequately shut down under all likely conditions. Further, the staff notes that the licensee's operating procedures limit the total excess reactivity levels that can be in the core during operations by students, operator trainees, licensed reactor operators, and senior operators. The procedures also specify acceptable qualifications of licensed operators required for direct supervision of unlicensed personnel (students and operator trainees).

4.7 Functional Design of Reactivity Control Systems

4.7.1 Control Rod Drives

The control rods are driven by electromechanical linear actuators. An actuator is essentially a ball-bearing-type screw driven through a gear reduction unit by a low-inertia reversible servomotor. The drives for the control rods are coupled to the control element by means of electromagnets. The regulating rod control element is attached rigidly to its drive mechanism. The drive mechanisms are actuated by switches from the control console. The limits of stroke of the control elements are set by adjustable cam-operated microswitches mounted on the rod drive mechanisms. The three control rods can be operated individually or together. If electrical power is removed from the electromagnets, the control rods fall into the core by force of gravity.

All control rods have control-console mounted electronic position indicators that are accurate to ± 1.27 mm. The three control rods have control-console mounted "insert limit," "shim range," and "withdraw limit" annunciator lights and an annunciator light that indicates when the rod is in contact with its magnet. The regulating rod has "insert limit" and "withdraw limit" annunciator lights as well as a pair of lights that indicate the direction of the rod movement.

4.7.2 Scram-Logic Circuit

The UMRR is equipped with a scram-logic safety system that receives signals from core instrumentation (neutron flux detectors) and other reactor parameters to initiate a scram by removing power from the safety rod magnets and/or the safety amplifier. The reactor parameters that can initiate these scrams are (1) high reactor power, (2) short reactor period, (3) bridge movement, (4) log N and period amplifier inoperative, and (5) operator manual scram. The safety system is discussed in more detail in Section 7.

4.7.3 Conclusion

The UMRR is equipped with a safety and control system typical of nonpower reactors, incorporating multiple control rods and multiple and redundant sensors that can initiate scrams. There is sufficient redundancy of control rods that the reactor can be shut down safely even if the most reactive control rod fails to insert on receiving a scram signal.

In addition to the electromechanical safety controls for both normal and abnormal operation, the negative bulk temperature coefficient of the moderator provides an inherent backup feature that would decrease reactivity during temperature increases.

In accordance with the above and the details presented in Section 7, the staff concludes that the reactivity control systems of the UMRR are designed and function adequately to ensure safe operation and safe shutdown of the reactor under all operating conditions.

4.8 Operational Practices

The University of Missouri-Rolla has implemented a preventive maintenance program that is supplemented by a detailed preoperational checklist to ensure that the reactor is not operated at power unless the appropriate safety-related components are operable. The reactor is operated by NRC-licensed personnel in accordance with explicit operating procedures, which include specified responses to any reactor control signal. All proposed new experiments involving the use of the UMRR are reviewed by the Nuclear Safeguards Committee for potential effects on the reactivity of the core or damage to any component of the reactor, as well as for possible malfunction of the experiment that might lead to the release of contained radioactivity.

4.9 Conclusions

The staff review of the reactor facility has included studying its specific design, installation, and operational limitations as identified in the Technical Specifications and other pertinent documents associated with the reactor. The design features are similar to those of the Bulk Shielding Reactor at Oak Ridge National Laboratory as well as to other pool-type research reactors operating in the United States and many countries of the world. The fuel, which is aluminum-clad, highly enriched uranium oxide-aluminum, is similar to that used in more than 30 NRC-licensed research and test reactors and to fuel used in some of the BORAX and SPERT tests (Dietrich, 1954; Nyer, 1956). The staff concludes that the UMRR is designed and built according to good industrial practices. It consists of standardized components representing many reactor-years of operation and includes redundant and diverse safety-related systems. On the basis of its review of the UMRR and its experience with similar facilities, the staff concludes that there is reasonable assurance that this reactor is capable of continued safe operation, as limited by its Technical Specifications.

5 REACTOR COOLING

5.1 Reactor Core Cooling

The UMRR core is submerged in a pool containing about 130 m^3 of demineralized water and is cooled by natural convection. Currently the UMRR is usually operated well below the licensed limits, but even if the reactor were to be operated continuously at 200 kW for 24 hours, the pool water would not reach the maximum allowable temperature set forth in the Technical Specifications (57°C). Heat from the pool water is dissipated into the reactor room primarily by evaporation and discharged to the environment by the ventilation system.

5.2 Coolant Purification System

The reactor coolant purification system is shown schematically in Figure 5.1. About 30 gal/min of pool water are pumped through a filter, a mixed resin bed demineralizer, and back into the pool. When it is necessary to add makeup water to the pool, raw water is introduced into a raw water supply tank. The primary coolant pump takes suction from this tank and pumps it through the filter and demineralizer and into the pool. This system for adding makeup water avoids having a raw water supply pipe attached to the system with the possibilities of accidental contamination of the reactor coolant or of backup of pool water into the campus water supply system.

HCl and NaOH and a dedicated air compressor are available for ion bed regeneration.

5.3 Conclusion

The staff concludes that the core cooling system is adequate to prevent fuel element overheating under all normal and likely off-normal operating conditions and that the coolant purification system can mitigate both corrosion and radioactivity problems associated with coolant contamination.

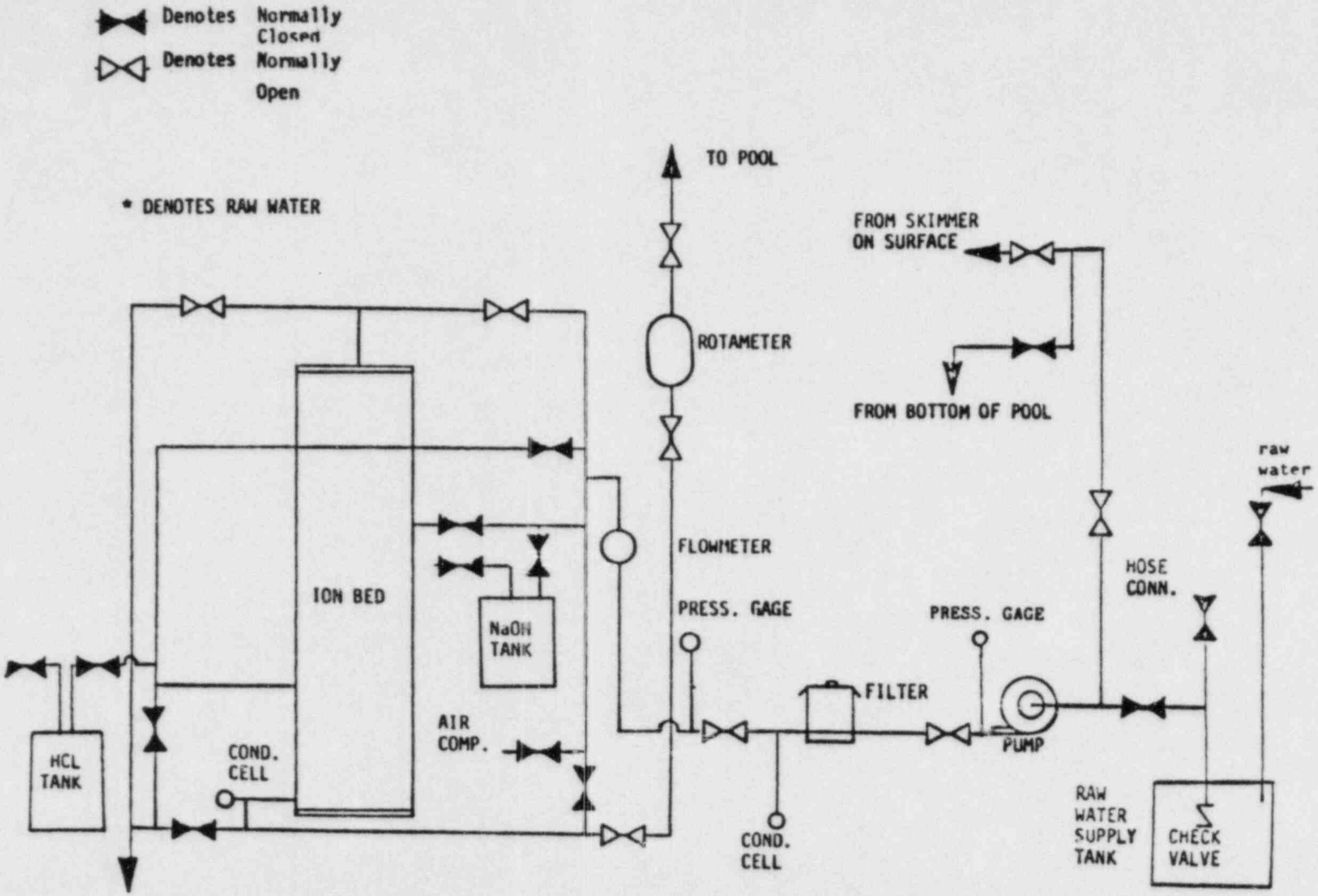

Figure 5.1 UMRR coolant system

6 ENGINEERED SAFETY FEATURES

Engineered safety features are systems provided to mitigate the radiological consequences of accidents. The single engineered safety feature at the UMRR facility is the ventilation system that can control the discharge rate of potentially contaminated reactor room air to the environment.

6.1 Ventilation System

A system of three exhaust fans that are mounted on the reactor building roof provides ventilation for the building. Air enters the building through two intakes equipped with fiberglass filters located on the lower level. The discharge rates of the exhaust fans are 142 m^3/min (5,000 ft^3/min), 425 m^3/min (15,000 ft^3/min), and 425 m^3/min (15,000 ft^3/min), respectively. The controls for the fans are located near the entrance to the control room, and any combination of fans may be selected.

The exhaust ducts and intakes are equipped with louvers that close automatically when the fans are turned off. Other building openings are not sealed; thus, some air movement caused by atmospheric pressure changes and temperature differentials would continue. However, there would be no sudden or large discharge of radioactive material to the environment in the event of a release within the reactor building (see Section 14 for additional discussion).

6.2 Conclusion

The UMRR is a 200-kW pool-type reactor whose operation has averaged the equivalent of $\sim$30 full-power hours per year. Therefore, the fission product inventory is low and the radioactivity inventory at risk in an individual fuel element or in an irradiated sample is even less.

The staff has determined that the operation of the UMRR even without any engineered safety features would not pose a significant radiological hazard to the public or to the environment in the event of an accident (see Section 14). Therefore, the staff concludes that the ability to control potential releases of airborne radioactivity adds conservatism to an already safe situation.

7 CONTROL AND INSTRUMENTATION

The control and instrumentation systems at the UMRR facility are similar to those in wide use for research reactors in the United States. Control of the nuclear fission process is achieved by using three control-safety (scrammable) rods and one regulating rod. The instrumentation system, which is interlocked with the control system, is composed of nuclear and process instrumentation and generally is characterized by state-of-the-art components. The licensee has a program in operation to replace older instruments with state-of-the-art systems that provide the same functions more reliably. The control and instrumentation systems are summarized in Table 7.1.

7.1 Control System

The control system is composed of both nuclear and process control equipment in which safety-related components are designed for redundant operation so that single failure or malfunction of components will not prevent the safe operation or shutdown of the reactor.

7.1.1 Nuclear Control Systems

Control of the reactor is achieved in the standard way by inserting and withdrawing nuetron-absorbing control rods by the use of control drive units mounted on the bridge structure over the pool. The regulating rod has a solid coupling and cannot be scrammed. The other three safety controls are supported by electromagnets so that any electrical power interruption will result in the elements falling by gravity into the core, causing a reactor scram. The control rod drives are controlled from the control room by the reactor operator. The control rod systems are discussed in more detail in Section 4.1.2.

7.1.2 Supplementary Control Systems

These control systems, also called process control systems, are designed to control the various processes involved in reactor operation, but do not relate directly to safety. Included in this category are circuits and devices that monitor coolant parameters such as temperature and conductivity. These control systems ensure proper operation of the non-safety-related systems and provide the operator with information on the status of these systems and related reactor parameters.

7.2 Instrumentation System

The instrumentation system is composed of both nuclear control and process instrumentation circuits. The electronics system contains both solid-state and tube-type components and provides annunciation and/or indication in the control room. The automatic scram functions through the safety amplifier discussed below.

Table 7.1 Safety and control instrumentation

Situation	Detector	Unit Initiating Action	Resulting Action	Annun-ciation	Limiting Values(1)
Manual scram	Operator	Scram button	Scram	Yes	Operator
Period ≤5 s	Compensated ion chamber	Log-N and period amplifier	Scram	Yes	5 s
High reactor power	Uncompensa-ted ion chamber	Safety amplifier	Scram	Yes	300 kW
Bridge motion	Motion switch	Motion switch	Scram	Yes	1.3 cm horizontal travel
Log-N and period amplifier NOT operative	Log-N period amplifier	Relay	Scram	Yes	
Power demand	Compensated ion chamber	Linear recorder	Rundown	Yes	≤120% of selected scale
Period ≤15 s	Compensated ion chamber	Period recorder	Rundown	Yes	15 s
Regulating rod insert limit on automatic	Microswitch	Microswitch	Rundown	Yes	0.0
Low CIC voltage	DC relay	DC relay	Rundown	Yes	400 V
≥120% full power	Compensated ion chamber	Log-N recorder	Rundown	Yes	240 kW
High radiation (2,3,4) at RAM Points	GM Tubes	Remote area monitoring system	Rundown	Yes	20 mR/h low 30 m R/h high
Period ≤30 s(3)	Compensated ion chamber	Period recorder	Rod prohibit	Yes	30 s
Any recorder off (5)	Relay	Relay	Rod prohibit	Yes	
Log count rate ≤2 CPS(3)	Fission chamber	Log count rate system	Rod prohibit	Yes	2 cps

See notes at end of table

Table 7.1 Continued

Situation	Detector	Unit Initiating Action	Resulting Action	Annunciation	Limiting Values(1)
Safety rods below range or regulating rod above insert limit(3)	Microswitch	Relay	Rod prohibit	No	
Reactor power deviation ±5% of selected power level	Compensated ion chamber	Linear channel	Servoprohibit	Yes	Power level
Core inlet water temperature ≥57°C	Thermocouple	Relay	Rod prohibit	Yes	57°C
Interlock bypassed	Key switch	Key switch		Yes	
Effluent pool demineralizer conductivity ≥2 micromhos per cm	Conductivity bridge	Relay		Yes	2 micromhos/cm
High neutron flux in beam room(4)	BF_3 neutron detector	Relay		Yes	30 mR/h
Evacuation alarm	GM tubes	Remote area monitoring (RAM) system	Initiate evacuation sequence; both automatic and manual actuation	Yes	30 mR/h
Airborne particulate radioactivity(6)	GM	Building continuous air monitor		Yes	

(1)Limiting values, operational set points may be more conservative.
(2)Radiation detector on the reactor bridge causes building alarm.
(3)Indicates that the situation may be key bypassed around safety circuitry.
(4)These will be set by measurement during initial increase in power level. The set points will be less than 30 mrem/h.
(5)The drive motor on startup channel recorder may be off.
(6)50% of limits in 10 CFR 20, Appendix B, Table 1, column 1.

7.2.1 Nuclear Instrumentation

This instrumentation provides the operator with the necessary information for proper manipulation of the nuclear controls.

(1) Log count rate or startup channel. This channel receives data from a movable fission chamber. Its primary purpose is to monitor the reactor power during startup.

(2) Linear-N power or linear power channel. This channel receives data from an electrically compensated ion chamber (CIC). This channel monitors the reactor power level in the range of <0.1 W to greater than 300 kW and provides the signal for automatic servocontrol of reactor power.

(3) Log-N power channel. This channel also receives data from a CIC and monitors the reactor power level from a few watts to greater than 200 kW. This channel also provides the signal to the period amplifier for indication of the reactor period and for period scram.

(4) Safety channels. Two uncompensated ion chambers provide signals for two independent channels, which give the redundancy to scram the reactor in response to reactor power above the set point.

All neutron-sensing chambers are located in the pool outside of the core and are independently adjustable over a limited distance to allow calibration of their respective channels to the reactor thermal power.

A drop in the high voltage to the CICs will result in a reactor power rundown (Table 7.1). Also, if the log-N and period channel amplifier mode switch is not in the operating position, a relay in the scram system will prevent reset of the scram circuit. Movement of the mode switch from the operating position when the reactor is operating will result in a scram.

7.2.2 Process Instrumentation

The process instrumentation monitors nonnuclear parameters and provides, as appropriate, rod withdrawal prohibits and/or alarm signals as well as information to assist in the operation of the facility.

Core inlet water temperature $\geq$57°C initiates a rod withdrawal prohibit and an alarm. The coolant core inlet prohibit and alarm is activated by a thermocouple below the core grid plate. The conductivity of the pool water flowing to the demineralizer is monitored by a conductivity bridge. Conductivity $\geq$2 micromhos activates a reactor console alarm.

Loss of ac power to the console will scram the reactor automatically by removing power from the rod-holding magnets. The reactor console key in the off position causes an essentially identical loss of ac power to the console and causes a reactor scram if turned off when the reactor is operating.

7.2.3 Inhibits and Annunciation

Inhibit signals that will prevent control rod removal (reactor startup) are provided if there is a low neutron count-rate in the startup channel; if the chart recorders are inoperable on the log count rate, linear-N, or log-N instruments; if the period is <30 s; if the core inlet temperature is >57°C; if the safety rods are below the shim range; or if the regulating rod is above the insert limit.

A control console-mounted annunciator panel of lights provides the operator with information on conditions of important variables related to reactor operation. The annunciator is energized continuously through the main power disconnect switch. Following annunciation of an event, the condition must be corrected, and the operator must reset to restore the annunciator to normal operating condition. Table 7.1 summarizes the various inhibit and annunciation functions.

7.2.4 Reactor Safety System

The control and instrumentation systems are interconnected through a safety amplifier. This unit supplies current for the electromagnets that support the control-safety rods, as well as high voltage for the ion chambers. Each ion chamber is provided with an independent amplifier circuit that will cause a fast scram on receipt of an appropriate trip signal or on failure. The safety circuit provides a scram by interrupting the dc current in the holding magnets or by turning off the ac power supply for the magnets.

7.3 Radiation Monitoring Instruments

The radiation monitoring system consists of fixed-position remote area monitors (RAMs), a neutron flux monitor, and a continuous air monitoring (CAM) system. The alarm set points are listed in Table 7.1.

Single RAMs are located at the reactor bridge, the demineralizer, and in the basement equipment room. The monitors alarm both locally and in the control room. All of the RAM alarms also initiate a reactor power rundown. The reactor bridge RAM has dual alarms with the lower set point initiating the rundown and the higher set point activating the building evacuation sequence.

A BF_3 neutron monitor is located in the beam room. This instrument monitors the neutron flux level in the beam room and alarms both locally and in the control room.

The CAM system detects radioactive airborne particulate material. A continuous sample is drawn from the reactor room air. The air sample stream passes through the particulate detector and is released back to the room. The particulate filter is replaced periodically and is assayed for gross beta-gamma radioactivity.

7.4 Conclusions

The control and instrumentation systems at the UMRR facility are well designed and maintained in an acceptable manner. Redundancy in the important ranges of reactor power measurements is ensured by overlapping ranges of the log-N and linear power channels.

The licensee's performance specifications for the individual components used throughout the system exceed the minimum acceptable. This helps to ensure system reliability and decreases the chances of simultaneous multicomponent failures.

The control system is designed so that the reactor is automatically and safely shut down if electrical power is lost (also see Section 8).

On the basis of its review of the control and instrumentation systems, the staff concludes that these systems are adequate to ensure safe operation of the reactor within the limits of the Technical Specifications and the other licensee conditions.

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 Main Power

A 110/220 V distribution panel in the reactor building is fed from a campus substation.

8.2 Emergency Power

No emergency electrical power is provided 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 Section 14.2.3), no emergency power is supplied except battery-operated emergency lighting for personnel movement during a power outage.

8.3 Conclusion

On the basis of the considerations above, the staff concludes that the electrical power system is acceptable for safe operation of the UMRR.

9 AUXILIARY SYSTEMS

9.1 Fuel Handling and Storage

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

Two storage racks are available in the fuel storage pit and each is capable of holding up to 30 fuel elements in two 1 by 15 arrays. The fuel elements stand in the storage racks in the same manner as in the core. Recent tests by the licensee have demonstrated that a factor of k_{eff} <0.6 exists for each side of the storage rack when fully loaded and that there is no measurable neutronic coupling between the two units. Therefore, there is reasonable assurance that an inadvertent criticality in the storage pit cannot occur.

9.2 Fire Protection System

The fire protection system gives warning in the event of a fire or smoke development within the reactor building. If smoke or fire is detected, audible and visual alarms are actuated inside and outside the reactor building and a remote alarm is transmitted to the campus police station.

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

In the event that the electrical power is lost to the reactor building, there is a backup battery system that will give an audible fire or smoke alarm to personnel in the reactor building and at the campus police station. There are also eight hand-held fire extinguishers located throughout the reactor building at strategically important locations.

The Rolla Fire Department, located <1.6 km from the reactor facility, responds to all calls.

9.3 Air Conditioning

A recirculating air conditioner located in the reactor building regulates building air for human comfort and assists in dissipating reactor pool heat.

9.4 Conclusions

The staff concludes that the fuel-handling facilities are appropriate for the reactor size and use and that the fire protection equipment and protective organization are acceptable. The staff also concludes that the function of the air-handling system is acceptable to dissipate the small amount of heat energy transferred to the reactor pool water during operation.

10 EXPERIMENTAL PROGRAM

In addition to being an inte part of the nuclear engineering undergraduate and graduate educational pro , the UMRR supports the various experimental programs of the staff and studen.. Most of the experimental work uses the neutrons available from the reactor to induce radioactivity in various materials. These irradiated materials may be foils or small samples to evaluate reactor parameters or material composition (neutron activation analysis), or they may be used as tracers in various studies.

10.1 <u>Experimental Facilities</u>

10.1.1 Beam Hole

There is a 6-in.-diameter (nominal) beam hole (shown in Figure 4.4). The beam hole is lined with stainless steel and has a removable aluminum beam tube and a separately removable beam tube extension. Irradiations can be performed within the tube or in the radiation beam emerging from the beam tube extension. The beam hole can be sealed with a blind flange whenever the beam tube and beam tube extension are removed. Biological shielding is provided by neutron and gamma absorbing liners and plugs. In addition, an outer shielding door is provided to cover the opening of the beam hole. A sealing gasket allows the outer shielding door to be used as a watertight cover.

10.1.2 Thermal Column

The thermal column assembly is a 1.1- by 1.1- by 1.5-m cube of graphite blocks extending from the reactor core into a cavity in the concrete biological shield (see Figures 4.3 and 4.4). The reactor face of the thermal column assembly is covered with a 10-cm lead shield. A 1.2- by 1.2- by 1.5-m movable concrete thermal column door provides additional shielding, when in the closed position, as well as access to the thermal column. The inner face of the door is lined with Boral to reduce thermal neutron activation in the concrete.

There are five irradiation facilities in the thermal column assembly. These are one 20.4-cm^2 and four 10.2-cm^2 horizontal access ports that are normally filled with graphite plugs.

10.1.3 Irradiation Elements

The two types of irradiation elements (Figure 10.1) that can be used for in-core irradiations are discussed below. The irradiation elements are designed to fit into the grid plate holes (see Figure 4.1).

10.1.3.1 Isotope Production Element

The isotope production element is essentially a graphite reflector element with an aluminum-lined central access hole that can accommodate a neutron source or a sample to be irradiated. The top sealing plug is held in place by an aluminum pin that is inserted in a horizontal through-hole. An O-ring seal permits use of the isotope production element as a dry irradiation facility.

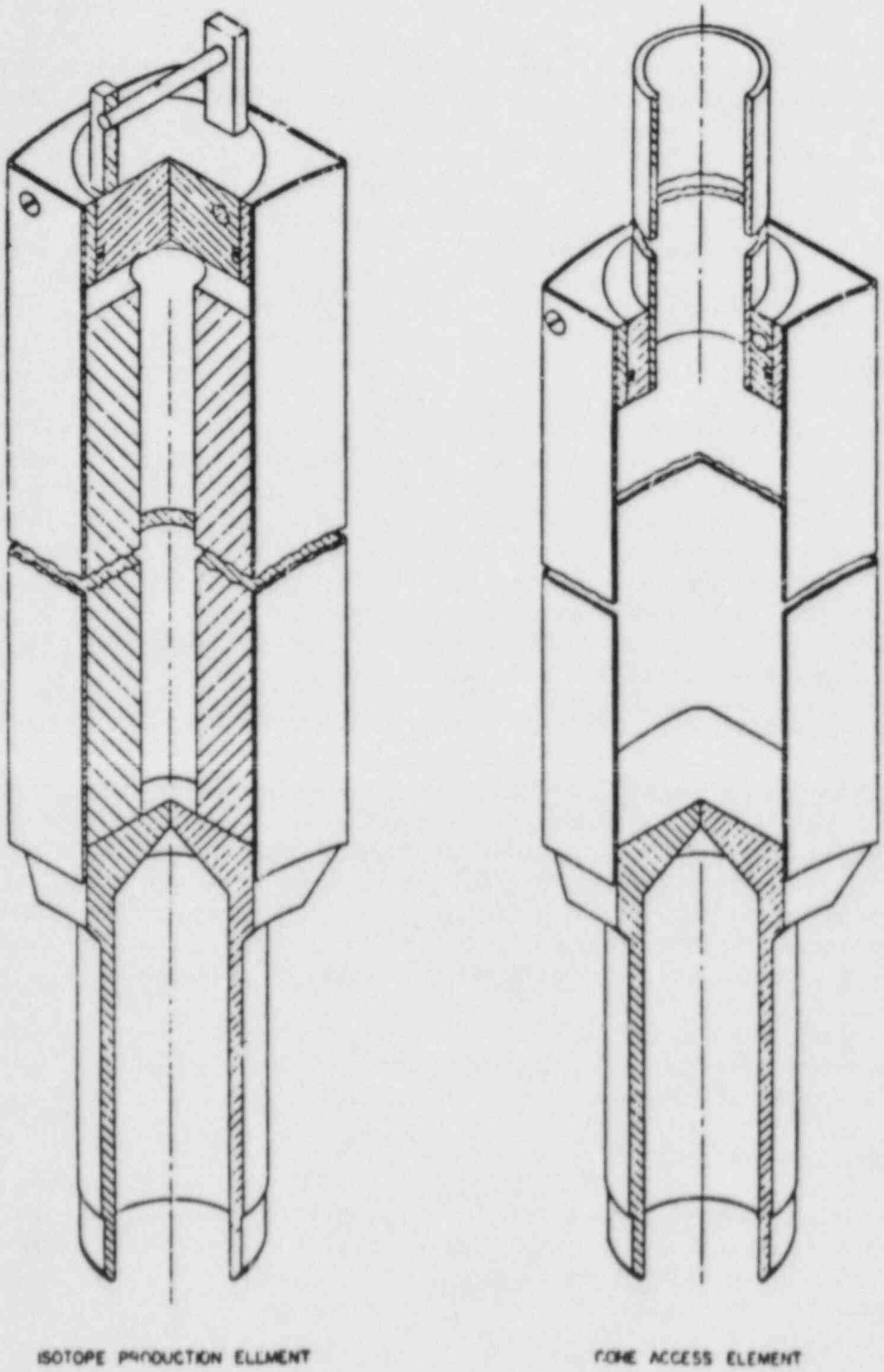

Figure 10.1 Two types of irradiation elements

10.1.3.2 Core Access Facility

The core assess element provides the capability to irradiate samples in the core lattice and, like the isotope production element, has an O-ring seal that permits its use as a dry irradiation facility. The core access element is basically an unfueled fuel element. The top sealing plug is provided with an aluminum tube that extends from core level to above the pool water level and is curved to prevent neutron and gamma ray streaming. Samples are lowered into the core access elements with a leader.

10.1.4 Pneumatic Transfer Facility

The pneumatic transfer facility (rabbit-tube), which can be used to transfer samples in and out of the core rapidly, fits into the grid plate in a manner similar to the irradiation elements. The rabbit-tube position is limited by Technical Specifications to core configurations in which at least one of the sides of the rabbit-tube is not adjacent to fuel. The tube can be cadmium lined for experiments in which it is desired to eliminate slow neutrons.

10.2 Experimental Review

All proposed new experiments, as well as procedures and facility changes for such experiments, must be approved by the Radiation Safety Committee (RSC). The RSC is composed of the Radiation Safety Officer and at least four other members having expertise in reactor operation, reactor safety or research use of radioisotopes. No more than two members of the RSC may be from the organization responsible for reactor operations. The review by the RSC is carried out to

(1) provide assurance that accidents causing changes in composition and geometry of the experiments would not cause unsafe changes in reactivity

(2) provide assurance of mechanical integrity, chemical compatibility, and adequate protection of the reactor against any other potential hazards

(3) provide assurance that materials to be irradiated will be enclosed in a manner appropriate for their physical state so that no uncontrolled releases of radioactive material will occur

(4) provide assurance that experiments involving potentially explosive materials are designed to prevent damage to the reactor

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.

10.3 Conclusion

The staff concludes that the design of the UMRR experimental facilities combined with the detailed review, the administrative procedures, and the limitations for experiments delineated in the Technical Specifications ensure acceptably safe experimental programs. Therefore, the staff considers that reasonable provisions have been made so that experimental programs and facilities do not pose a significant risk of radiation exposure to the staff, students, or the public.

11 RADIOACTIVE WASTE MANAGEMENT

The major radioactive waste generated by the reactor operations is activated gases, principally ^{41}Ar. 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.

11.1 ALARA Commitment

The UMRR is operated with the philosophy of minimizing the release of radioactive materials to the environment (as low as reasonably achievable). The university administration, through the Radiation Safety Officer, instructs all operating and research personnel to develop procedures to limit the generation and subsequent release of radioactive materials.

11.2 Waste Generation and Handling Procedures

11.2.1 Solid Waste

The disposal of high-level radioactive waste in the form of spent fuel is not anticipated during the term of this license renewal. Therefore, 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.

11.2.2 Liquid Waste

Normal reactor operations produce no radioactive liquid waste. However, some of the research activities conducted within the reactor complex are capable of generating such waste. Liquid waste drains from the reactor room and equipment areas into the lower level (basement) sump.

The largest volume of potentially contaminated water is produced by the regeneration of the demineralizer. This periodically generated effluent is first discharged to two 300-gal retention tanks. The retention tanks allow for adequate radioactive decay of these regeneration liquids. The contents of the tanks 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 10 CFR 20, Appendix B, Column 2, will not be exceeded.

Grab samples are collected at the time of the regeneration during both the acid and caustic washes. All samples have been below 10 CFR 20 guideline values, and the solutions may be retained for several weeks following the regeneration before discharge.

11.2.3 Airborne Waste

The principal potential airborne waste is composed of ^{41}Ar and neutron-activated dust particulates. These are produced by the irradiation of air in the pool water and air and airborne particulates in the thermal column and other experimental facilities. The air is swept constantly from the experimental area and from above the reactor pool and discharged into the environment through a stack. Another activation product that can be airborne is ^{16}N, produced within the coolant passing through the core of the reactor. To decrease the ^{16}N gas that becomes airborne, a jet of water is forced over the surface of the core. This increases the transport time of the short-lived (7.1 s) ^{16}N 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 ^{16}N is well below the limits prescribed by 10 CFR 20. No fission products escape from the fuel cladding during normal operations, but a small quantity of gaseous fission products from uranium contamination in the pool is detectable in the reactor room air.

The UMRR staff has estimated the release of airborne radioactivity (mostly ^{41}Ar) at <25-mCi/yr. The licensee's and the staff's evaluations show that this amount of release would lead to radiation exposures in the unrestricted areas that are well within the limits specified in 10 CFR 20 (see Sections 12.6, 12.7).

11.3 Conclusion

The staff concludes that the waste management activities at the UMRR facility have been conducted and are expected to continue to be conducted in a manner consistent with 10 CFR 20 and with the ALARA principle. Among other guidance, the staff review has followed the methods of ANSI/ANS 15.11, 1977, "Radiological Control at Research Reactor Facilities."

Because ^{41}Ar is the only significant 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 this radionuclide. The staff concludes that the potential doses in unrestricted areas, as a result of actual releases of ^{41}Ar, have never exceeded or even approached the limits specified in 10 CFR 20, when averaged over a year. Furthermore, the staff's 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 ^{41}Ar release would not be significant even if there were major changes in the operating schedule of the UMRR.

12 RADIATION PROTECTION PROGRAM

The University of Missouri-Rolla has a structured radiation safety program with a health physics staff equipped with radiation detection instrumentation to determine, control, and document occupational radiation exposures at its reactor facility. The licensee has made significant changes and improvements in its radiation protection program and procedures during 1983 and 1984.

12.1 ALARA Commitment

The Radiation Safety Committee has been instructed by the Office of the Vice Chancellor of UMO (N. K. Smith letter of July 2, 1984) to implement the policy that operations are to be conducted in a manner to keep all radiation exposures ALARA. All proposed experiments and procedures at the reactor are reviewed for ways to minimize the potential exposures of personnel. All unanticipated or unusual reactor-related exposures will be investigated by both the health physics and the operations staffs to develop methods to prevent recurrences.

12.2 Health Physics Program

12.2.1 Health Physics Staffing

The normal radiation safety staff at the University of Missouri-Rolla consists of two professional health physicists and a part-time technician. This staff provides radiation safety support to the entire university complex, including many radioisotope laboratories. Some routine health-physics-type activities at the reactor are performed by the operations staff. The formal health physics staff is available for consultation and assistance when needed. Monthly surveys (audits) are conducted in the reactor areas by the health physics staff.

The staff concludes that the current radiation safety program is adequate and acceptable for the activities conducted at this reactor facility.

12.2.2 Procedures

Detailed written procedures that address the radiation safety support that is expected to be provided to the routine operations of the UMRR facility have recently been revised. These procedures identify the interactions between the operational and experimental personnel. They also specify numerous administrative limits and action points, as well as the appropriate responses and corrective action if these limits or action points are reached or exceeded. Copies of these procedures are readily available to the operational and research staff's and to the administrative and radiation safety personnel.

12.2.3 Instrumentation

The University of Missouri-Rolla has acquired a variety of detecting and measuring instruments for monitoring potentially hazardous ionizing radiation. The instrument calibration procedures and techniques ensure that any credible type of radiation and any significant intensities will be detected promptly and measured correctly.

12.2.4 Training

All reactor-related personnel are given an indoctrination in radiation safety before they assume their work responsibilities. Additional radiation safety instructions are provided to those who will be working directly with radiation or radioactive materials. A training film explaining the ALARA concept, general government rules and regulations, and basic university-wide radiation safety procedures forms the basis for more detailed job-specific instructions. The training program is designed to identify the particular hazards of each specific type of work to be undertaken and methods to mitigate their consequences. Retraining in radiation safety is provided as well. As an example, all reactor operators are given an examination on health physics practices and procedures at least every 2 years. The level of retraining given is determined by examination results.

12.3 Radiation Sources

12.3.1 Reactor

Sources of radiation directly related to reactor operations include radiation from the reactor core, ion exchange columns, filters in the water cleanup systems, and radioactive gases (primarily ^{41}Ar).

The fission products from the fuel are contained within the fuel's aluminum cladding. Radiation exposures from the reactor core are reduced to acceptable levels by water and concrete shielding. The ion exchange resins and filters are changed routinely before high levels of radioactive materials have accumulated, thereby limiting personnel exposure.

Personnel exposure to the radiation from chemically inert ^{41}Ar is limited by dilution and prompt removal of this gas from the reactor building and its discharge to the atmosphere, where it diffuses and is diluted further before reaching potentially occupied areas.

12.3.2 Extraneous Sources

Sources of radiation that may be considered as incidental to the normal reactor operation, but associated with reactor use, include radioactive isotopes produced for research, activated components of experiments, and activated samples or specimens.

Personnel exposure to radiation from intentionally produced radioactive material as well as from the required manipulation of activated experimental components is controlled by rigidly developed and reviewed operating procedures that use the normal protective measures of time, distance, and shielding.

12.4 Routine Monitoring

12.4.1 Fixed-Position Monitors

The UMRR facility has several fixed-position radiation monitors: one on the bridge above the reactor, another near the water purification system, and the third near the thermal column on the lower level. All monitors have adjustable alarm set points and read out in the control room as well as locally. There also is a continuous air monitor to detect radioactive particulates in the reactor room air that reads out locally and is recorded on the control room auxiliary panel.

12.4.2 Experimental Support

The health physics staff participates in experiment planning by reviewing all proposed procedures for methods of minimizing personnel exposures and limiting the generation of radioactive waste. Approved procedures specify the type and degree of radiation safety support required by each activity.

12.5 Occupational Radiation Exposures

12.5.1 Personnel Monitoring Program

The University of Missouri-Rolla personnel monitoring program is described in its Radiation Safety Procedures. 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, thermoluminescent dosimeters (TLDs) and self-reading pocket ion chambers are available. Instrument dose rate and time measurements are used to administratively keep occupational exposures below the applicable limits in 10 CFR 20.

Visitors are provided with self-reading ion chambers for monitoring purposes.

12.5.2 Personnel Exposures

The UMRR facility personnel annual exposure history for the last 5 years is given in Table 12.1. These data indicate that both the management of reactor operations and the personnel protection program are effective in limiting radiation exposures at the UMRR facility.

Table 12.1 Number of individuals in exposure interval

	Number of individuals in each range				
Whole-body exposure range (rem)	1979	1980	1981	1982	1983
No measurable exposure	44	44	38	47	31
Measurable exposure < 0.1	0	13	1	1	8
0.1 to 0.25	0	3	2	0	0
Over 0.25	0	0	0	0	0
Total number of individuals monitored	44	60	41	48	39

12.6 Effluent Monitoring

12.6.1 Airborne Effluents

As discussed in Section 11, airborne radioactive effluents from the reactor facility consist principally of low concentrations of ^{41}Ar. The small amount of ^{41}Ar released into the reactor room at low concentration is diluted further by the 1,700-m^3 volume of room air, which at full power is continuously discharged at about 1,000 m^3/min near the top of the reactor building, resulting in additional dilution before reaching potentially occupied, unrestricted areas. Recent measurements of the concentration of ^{41}Ar in the reactor room air were made after 3.5 hours of operation at 200 kW with reduced exhaust air flow of 140 m^3/min. The measured result was 5×10^{-8} μCi/ml. This measured value is a long-run equilibrium concentration determined by the interplay of the constant production rate and the constant removal rate from the room by the exhaust fan. The higher exhaust rate mentioned above (10^3 m^3/min) would, therefore, lead to a full-power equilibrium concentration of $<10^{-8}$ μCi/ml, which is well below the guideline values of 10 CFR 20, Appendix B, for unrestricted areas.

12.6.2 Liquid Effluents

The reactor generates very limited radioactive liquid waste during routine operations. However, potential leaks in the primary coolant system could lead to uncontrolled small releases to the environment, and experimental activities associated with reactor usage also may generate radioactive liquids. The major source (volume) of liquid waste is from regeneration of the demineralizer system.

All drains in the reactor bay area lead to the lower level sump. The periodically generated waste liquid produced by the regeneration of the demineralizer is collected in two 300-gal (1.13 m^3) waste storage tanks.

If the concentrations of radioactive materials in the waste are less than the guideline values of 10 CFR 20.303, the liquids are discharged directly to the sanitary sewer.

12.7 Potential Dose Assessments

Natural background radiation levels in the Rolla, Missouri, area result in a dose of about 105 mrems/yr to each individual residing there. At least an additional 8% (~8 mrems/yr) will be received by those living in a brick or masonry structure. Any medical diagnosis X-ray examination will add to the natural background radiations, increasing the total accumulative annual dose of those individuals.

Calculations by the staff, based on a conservatively assumed 100 mCi/yr of ^{41}Ar released during normal operations from the reactor facility stack, predict a maximum annual whole-body dose of less than 1 mrem in the nearest unrestricted areas resulting from this source. The staff has relied on the methods of NUREG-0851 for these estimates.

12.8 Conclusions

The staff considers that radiation protection currently receives appropriate support from the university administration. The staff concludes that (1) the program is properly staffed and equipped, (2) the reactor health physics staff has adequate authority and lines of communication, (3) the procedures are integrated adequately into the research plans, and (4) surveys verify that operations and procedures are consistent with ALARA principles.

The staff concludes that the effluent monitoring programs currently conducted by university personnel provide reasonable assurance that significant releases of radioactivity will be promptly identified to predict maximum potential exposures to individuals in the unrestricted area. These predicted maximum levels during normal operations are well within applicable regulations and guidelines of 10 CFR 20.

Additionally, the staff concludes that the University of Missouri-Rolla radiation protection program is acceptable because the staff has found no instances of reactor-related exposures of personnel above applicable regulations and no unidentified significant releases of radioactivity to the environment. Furthermore, the staff considers that there is reasonable assurance that the personnel and procedures will continue to protect the health and safety of the public during routine reactor operations.

13 CONDUCT OF OPERATIONS

13.1 Overall Organization

Responsibility for the safe operation of the reactor facility is vested within the chain of command shown in Figure 13.1. The Reactor Director is delegated responsibility for overall facility operation.

13.2 Training

Most of the training of reactor operators is done by in-house personnel. The licensee's Operator Requalification Program has been reviewed, and the staff concludes that it meets the applicable regulations (10 CFR 50.54(i-1) and Appendix A of 10 CFR 55) and is consistent with the guidance of ANS 15.4.

13.3 Emergency Planning

10 CFR 50.54(q) and (r) require that a licensee authorized to possess and/or operate a research reactor shall follow and maintain in effect an emergency plan that meets the requirements of Appendix E of 10 CFR 50. As part of the application for license renewal, the licensee submitted an Emergency Plan following the then existing guidance. By letter dated October 27, 1982, the licensee transmitted a revised Emergency Plan in fulfillment of the requirements of revised applicable regulations (RG 2.6, Rev. 1, March 1982; ANSI/ANS 15.16, 1981 Draft). By letter dated August 27, 1984, the NRC transmitted its approval of the Emergency Plan to the licensee.

13.4 Operational Review and Audits

The Radiation Safety Committee (RSC) provides independent review and audit of facility activities. The Technical Specifications outline the qualifications and provide that alternate members may be appointed by the Chairman. The RSC must review and approve plans for modifications to the reactor, new experiments, and proposed changes to the license or to procedures. The RSC also is responsible for conducting audits of reactor facility operations and management and for reporting the results thereof to the Chancellor of the University of Missouri-Rolla.

13.5 Physical Security Plan

The UMRR facility has established and maintains a program to protect the reactor and its fuel and to ensure its security. The NRC staff has reviewed the Physical Security Plan and concludes that the plan, as amended, meets the requirements of 10 CFR 73.67 for special nuclear material of moderate strategic significance. The UMRR facility's inventory of special nuclear material for reactor operation falls within that category.

Both the Physical Security Plan and the staff's evaluation are withheld from public disclosure under 10 CFR 2.790(d)(1). Amendment No. 6 to the facility Operating License No. R-79, dated August 5, 1981, incorporated the Physical Security Plan as a condition of the license.

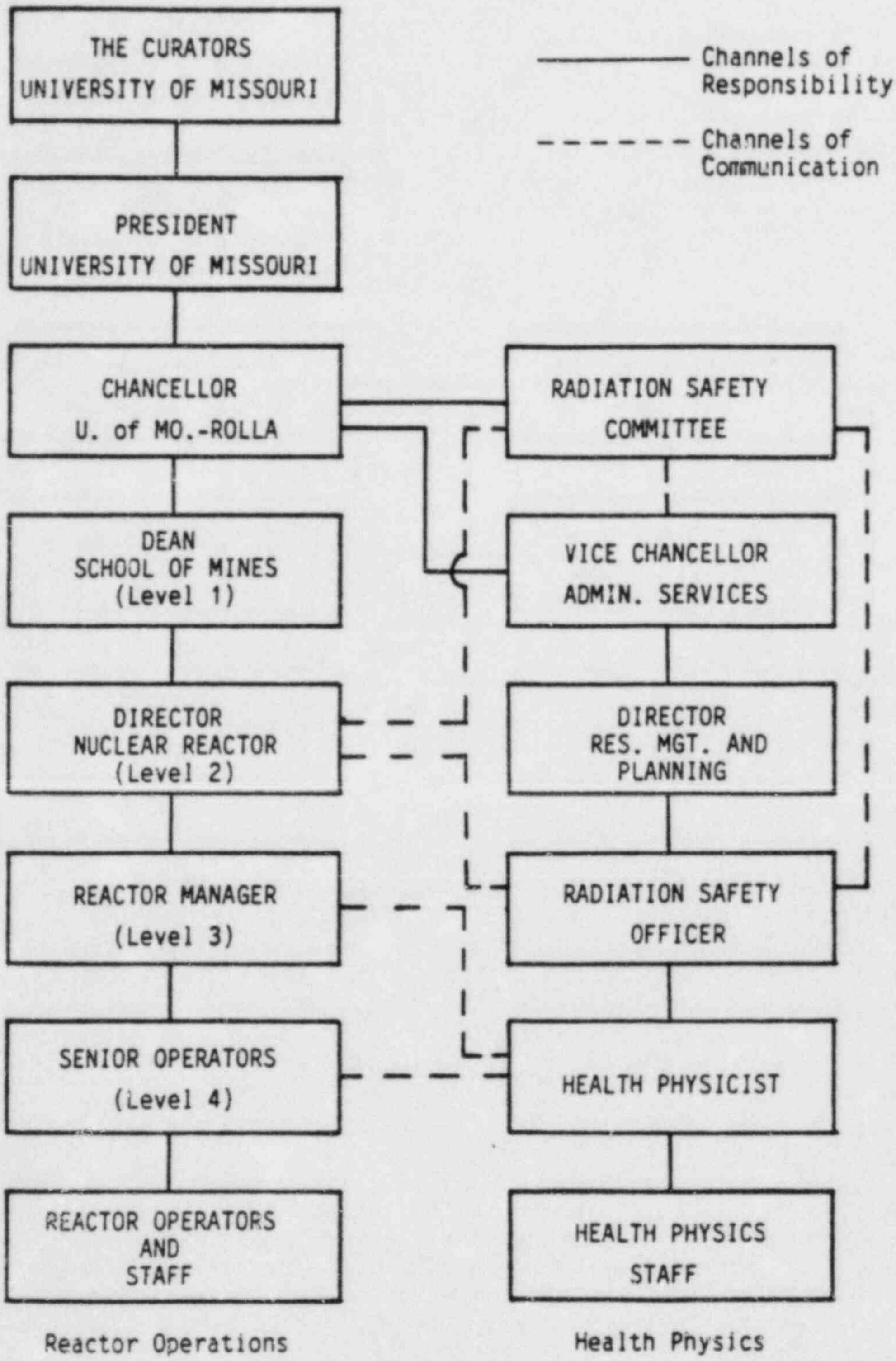

Figure 13.1 Organizational structure of the University of Missouri related to the UMRR facility

13.6 Conclusion

On the basis of the above, the staff concludes that the licensee has sufficient experience, management structure, and procedures to provide reasonable assurance that the reactor will be managed in a way that will cause no significant risk to the health and safety of the public.

14 ACCIDENT ANALYSIS

14.1 General Summary

In establishing the limiting safety system settings and the limiting conditions for operation for the UMRR, the licensee and the staff analyzed potential transients to ensure that these events would not result in the safety limits being exceeded. Other hypothetical accidents and their potential effects on the reactor core and the health and safety of the public also were analyzed.

Among the accidents postulated, the one with the greatest potential radiological effects in the unrestricted area is the failure of a fueled experiment and the subsequent release of its fission product inventory. None of the reactor transients or other accidents analyzed posed a significant risk of fuel cladding failure and, therefore, would not result in a release of radioactivity.

The failure of a fueled experiment is designated by the staff as the maximum hypothetical accident (MHA) for the UMRR. An MHA is defined as an accident for which the risk to public health and safety is greater than from any other credible event. The staff assumes that the accident occurs but does not try to describe or evaluate the mechanical details of the accident or the probability of its occurrence. Only the consequences of the MHA are evaluated.

14.2 Accidents Analyzed

The following potential accidents have been evaluated: failure of a fueled experiment, rapid insertion of reactivity, loss of coolant, and fuel handling. They are discussed in the following sections.

14.2.1 Failure of a Fueled Experiment

Because of its potential for release of fission products to the unrestricted environment, the failure of a fueled experiment is defined as the MHA for this reactor. Both the staff and the licensee evaluated the failure of a fueled experiment being irradiated in a sealed capsule close to the reactor core beneath more than 6 m of water. The staff's analysis was based on methods outlined in NRC Regulatory Guides 1.25 and 1.109, NUREG-0851, and AEC Report TID 14844 (March 23, 1962). It was assumed conservatively that 100% of the noble gases and 50% of the halogens would be released from the experiment on total failure (see NUREG-0772). An irradiation time of 8 hours was assumed along with an experiment fission power of 100 W. This is approximately the power that could be generated in a gram of ^{235}U near core center at full power.

Additionally, it was assumed that the halogen and noble gas fission products are released instantaneously into the reactor room, with no plateout or entrapment in pool water, and dispersed uniformly within the air. It is assumed that a person within the reactor room would be exposed to the radioactivity for 5 minutes before being alerted and evacuated. The free air volume of the reactor room is 1,700 m^3, and it was assumed that the exhaust fans were not in operation. For evaluating inhalation volumes, an average breathing rate of

3.47×10^{-4} m^3/s was assumed. The computed doses in the reactor room are given in Table 14.1. The whole-body dose computations are based on assuming that the room full of airborne radioactive gases and particulates can be adequately simulated by a hemisphere of the same volume with the recipient of the dose located at the center of the sphere. Doses to the body surfaces resulting from beta rays would be of the same order of magnitude as the whole-body gamma doses.

Table 14.1 Radiation doses within UMRR building*

Element	Whole-body gamma dose (mrem)	Committed thyroid dose (rem)
I	29	15
Kr	30	**
Xe	18	**

*Experiment fission power = 100 W; irradiation time = 8 hours and exposure time = 300 s.
**Thyroid dose from krypton and xenon inhalation is negligible

Doses also were computed for a point just outside the reactor building, assuming the same fuel failure scenario. However, in this case it was assumed that the exhaust fans are operating and that the most exposed individual remains in place throughout the time required to remove essentially all of the contaminated air from the reactor room. It also was assumed that there is additional dispersion (χ/Q) of the exhausted air before the dose recipient is immersed in it, with an equivalent χ/Q of 10^{-2} s/m^3. The potential doses computed for this hypothetical accident are given in Table 14.2, assuming a semi-infinite cloud of gases. The whole-body doses estimated for a realistic finite cloud would be at least a factor of ten smaller (NUREG-0851).

Table 14.2 Radiation dose for environment outside UMRR building*

Element	Whole-body gamma dose (mrem)	Committed thyroid dose (rem)
I	39	0.1
Kr	34	**
Xe	20	**

*Assumes exposure time = release time; $\chi/Q = 10^{-2}$ s/m^3, no decrease in source strength by radioactive decay once in the environment.
**Thyroid dose from krypton and xenon inhalation is negligible.

The licensee also designated the failure of a fueled experiment as the MHA and provided an independent analysis. The basic scenario was similar to the staff's, but somewhat different conservative assumptions were used. The maximum potential doses calculated by the licensee agreed acceptably with those calculated by the staff.

On the basis of the above conservative analyses, the staff concludes that fueled experiments can be performed at the UMRR facility in accordance with the limitations stated in the Technical Specifications without undue risk to public health and safety. Even though a conservatively high fission product release was assumed, the computed maximum accidental radiation doses to an individual in the restricted and unrestricted areas would be below the guideline values in 10 CFR 20, Appendix B, which form the basis for routine operation.

14.2.2 Rapid Insertion of Reactivity (Nuclear Excursion)

The licensee has analyzed potential transients that might result from a rapid insertion of reactivity. The staff reviewed these analyses and also evaluated potential transients resulting from a 1.5% $\Delta k/k$ step insertion of reactivity and from a ramp insertion of reactivity during reactor startup.

14.2.2.1 Step Insertion of Reactivity

The UMRR Technical Specifications limit the maximum reactivity worth of a movable experiment to 0.4% $\Delta k/k$. The flooding of the isotope production element or core access element positioned in the central position of the core might cause a reactivity change of about 0.7% $\Delta k/k$. Inadvertently inserting a fuel element into a vacancy at the periphery of the core will result in a reactivity insertion no greater than 1.5% $\Delta k/k$. Although it is not considered credible to have a step insertion of reactivity equal to the limit on excess reactivity of 1.5% $\Delta k/k$ provided in the Technical Specifications, for purposes of analysis the licensee and the staff assumed that a step insertion of reactivity of 1.5% $\Delta k/k$ occurs.

The UMRR fuel geometry and composition are very similar to the SPERT-I D-12/25 core (Table 14.3). Excursion experiments at the Borax and SPERT facilities (Miller et al., 1964; Edlund and Noderer, 1957) demonstrated that no mechanical damage or damaging fuel temperatures occurred for a step insertion of 1.5% $\Delta k/k$. On the basis of these experiments and the similarity to the SPERT-I D-12/25 core, a period of about 0.007 s, a peak power of about 630 MW, an energy release of about 16 MW-s, and a maximum fuel temperature of 490°C would occur for a step insertion of reactivity of 1.5% $\Delta k/k$. Because the SPERT reactor survived these conditions with no observed fuel damage, the staff concludes that a step insertion of reactivity of 1.5% $\Delta k/k$ will not result in fuel or core damage at the UMRR.

14.2.2.2 Ramp Insertion of Reactivity

During startup it is possible for all three safety-control rods to be withdrawn simultaneously, which could cause a maximum ramp insertion rate of 0.06% $\Delta k/k/s$. If the interlock failed on the regulating rod, all four control rods could be withdrawn simultaneously, providing a maximum ramp insertion rate of

Table 14.3 UMRR vs. SPERT-I fuel data

Item	UMRR Plate	SPERT-I Plate
Geometry		
Length (cm)	61	61
Width (cm)	7.6	7.6
Thickness (cm)	0.15	0.15
Water gap (cm)	0.63	0.45
Fuel		
Material	U_3O_8-Al	U-Al
Enrichment (%)	93	fully enriched
Weight fraction of U	0.03	0.24
Thickness (mm)	0.51	0.51
Cladding		
Material	Al	Al
Thickness (mm)	0.51	0.51

0.08% Δk/k/s. The boiling ramp tests at the SPERT facility for the SPERT I core demonstrated that ramp insertions of reactivity up to 2.5% Δk/k at rates up to 0.35% Δk/k/s resulted in no damage to the fuel (Forbes et al., 1956). Assuming that a total insertion of 1.5% Δk/k at a rate of 0.08% Δk/k/s from just critical at ~5 W and that protective circuits do not function and relying on the results of SPERT I Tests, Nos. 2733 and 2727, a period of about 0.08 s, a peak power of about 6.0 MW, and a maximum fuel temperature of about 120°C would occur initially (Forbes et al., 1956). The reactor power then would oscillate around 0.5 MW, unless automatic or manual protective action occurred. Because no fuel damage occurred at SPERT under the more severe reactivity conditions, the staff concludes that no fuel damage would result at UMRR from a hypothetical maximum ramp insertion of reactivity of 1.5% Δk/k at a rate of 0.08% Δk/k/s.

14.2.3 Loss of Coolant

A loss-of-coolant accident (LOCA) is considered unlikely because of the design and construction of the reactor pool. If the pool were to drain, the loss of water (moderator) would shut down the reactor and the removal of radioactive decay heat would occur by natural convection of ambient air. The licensee has analyzed this event with the following results. The initial decay power at shutdown, following a long operating time at full power, is 14 kW. If this decay power remained constant, the fuel temperature would rise to a maximum of 425°C during this transient. This temperature is safely below the melting point (660°C) of aluminum. Furthermore, the decay power, actually decreases relatively rapidly after the fission process stops, so the probable maximum temperature from a LOCA would be even lower than 425°C. The staff concurs with the licensee's analysis and concludes that no fuel damage would result from a LOCA at the UMRR.

14.2.4 Fuel Handling

The staff has considered an accident in which a fuel element is dropped during fuel manipulation so that it occupies a position on the periphery of the core. During core unloading, which always proceeds from the outside to the center, each fuel element is moved individually, using a hand-operated tool, into the storage space within the reactor pool. If a fuel element were inadvertently dropped during transfer, sufficient mechanical distortion of the end fittings to prohibit continued use as a fuel element could possibly occur. However, sufficient damage to strip cladding from one or more fuel plates with subsequent release of a large fraction of the fission products is not credible. Experiments at the UMRR facility have shown that the worth of a fuel element in a vacancy at the side of the core is less than 1.5% $\Delta k/k$. Therefore, if a fuel element were dropped next to a just-critical core, the resulting reactivity insertion would be no greater than 1.5% $\Delta k/k$ with consequences no more serious than those analyzed in Section 14.2.2.

Because there are no in-pool operations involving a fuel transfer cask, the potential for dropping a cask on the core does not exist. However, if in the future there were operations involving a fuel transfer cask and it were dropped, the staff concludes that the core supporting structure that suspends the core assembly from the bridge and the control rod drives would serve as a protective barrier between the falling cask and the fuel elements. Although considerable damage to the reactor structures, control rods, and fuel elements could occur, failure of fuel cladding and subsequent release of significant quantities of fission products is not considered likely.

The staff concludes, on the basis of the above considerations, the credible fuel damaging accidents will not lead to release of hazardous quantities of fission products to the reactor building or the environment because of fuel cladding failures.

14.3 Conclusion

The staff has reviewed the potential credible accidents for the UMRR. On the basis of the review, the only event that is postulated to result in a significant release of fission products to the environment is the total failure of a fueled experiment. The analysis has demonstrated that even if this unlikely event should occur, the resultant doses would be below 10 CFR 20 guideline values. Therefore, the staff concludes that the design of the facility and the controls of the Technical Specifications provide reasonable assurance that the UMRR can continue to be operated without significant risk to the health and safety of the public.

15 TECHNICAL SPECIFICATIONS

The licensee's Technical Specifications, evaluated in this licensing action, define certain features, characteristics, and conditions governing the continued operation of the UMRR facility. These Technical Specifications will be explicitly included in the renewal license as Appendix A. Formats and contents acceptable to the NRC have been used in the development of these Technical Specifications, and the staff has reviewed them using the ANSI/ANS 15.1-1982 standard as a guide.

On the basis of its review, the staff concludes that normal reactor operation within the limits of the Technical Specifications will not result in offsite radiation exposures in excess of 10 CFR 20 limits. Furthermore, the limiting conditions for operation, surveillance requirements, and engineered safety features will limit the likelihood of malfunctions and mitigate the consequences to the public of off-normal or accident events.

16 FINANCIAL QUALIFICATIONS

The UMRR is owned and operated by a state educational institution in support of its role in education and research. On the basis of financial information supplied by the licensee in its submittal of October 15, 1979, as supplemented, the staff concludes that funds will be made available, as necessary, to support continued operations and eventually to shut down the facility and maintain it in a condition that would constitute no risk to the public. The licensee's financial status was reviewed and found to be acceptable in accordance with the requirements of 10 CFR 50.33(f).

17 OTHER LICENSE CONSIDERATIONS

17.1 Prior Reactor Utilization

Previous sections of this SER concluded that normal operation of the reactor causes insignificant risk of radiation exposure to the public, and that only an off-normal or accident event could cause some measurable exposure. Even a maximum hypothetical accident would not lead to a dose to the most exposed individual greater than applicable guideline values of 10 CFR 20.

The staff concluded in its SER for the original Operating License that the reactor was initially designed and constructed to operate safely. During its current review, the staff considered whether prior operation would cause significant degradation in the capability of components and systems to continue to perform their safety function. Because fuel cladding is the component most responsible for preventing release of fission products to the environment, possible mechanisms that could lead to detrimental changes in integrity were considered. Prominent among the considerations were the following: (1) radiation degradation of cladding integrity, (2) high fuel temperature or temperature cycling leading to changes in the mechanical properties of the cladding, (3) corrosion or erosion of the cladding leading to thinning or other weakening, (4) mechanical damage resulting from handling or experimental use, and (5) degradation of safety components or systems.

The staff's conclusions regarding these parameters, in the order in which they were identified above, follow.

(1) Nearly identical fuel has been laboratory tested elsewhere, and has been exposed in similar irradiation conditions to much higher total radiation doses in operating reactors, such as at the Oak Ridge Research Reactor and the Omega West Reactor (Los Alamos National Laboratory). No significant degradation of cladding has resulted in any of these reactors.

(2) The power density, coolant flow rates, and maximum temperatures reached in the UMRR fuel are far below similar parameters in some other nonpower reactors using similar fuel. No damage has occurred during normal operations in any of these reactors.

(3) The coolant flow rate at the UMRR is much lower than used at several higher powered research reactors using MTR-type fuel. No erosion problems have been observed. At the UMRR facility, corrosion is kept to a reasonable minimum by careful control of the conductivity of the primary coolant water.

(4) The fuel is handled as infrequently as possible, consistent with required surveillance. Any indications of possible damage or degradation are investigated immediately, and damaged fuel would be removed from service in accordance with Technical Specifications. All experiments placed near the core are isolated from the fuel cladding by a water gap and at least one barrier or encapsulation.

(5) The UMRR personnel perform regular preventive and corrective maintenance and replace components, as necessary. Nevertheless, there have been some malfunctions of equipment. However, the staff review indicates that most of these malfunctions have been random one-of-a-kind incidents, typical of even good quality electromechanical instrumentation. There is no indication of significant degradation of the instrumentation, and the staff further has determined that the preventive maintenance program would lead to adequate identification and replacement before significant degradation occurred. Therefore, the staff concludes that there has been no apparent significant degradation of safety equipment and, because there is strong evidence that any future degradation will lead to prompt remedial action by the UMO, there is reasonable assurance that there will be no significant increase in the likelihood of occurrence of a reactor accident as a result of component malfunction.

18 CONCLUSIONS

Based on its evaluation of the application as set forth in the previous sections, the staff has determined that

(1) The application for renewal of Operating License R-79 for its research reactor filed by the University of Missouri, dated October 15, 1979, as supplemented, complies with the requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I.

(2) The facility will operate in conformity with the application as supplemented, the provisions of the Act, and the rules and regulations of the Commission.

(3) There is reasonable assurance (a) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public; and (b) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter I.

(4) The licensee is technically and financially qualified to engage in the activities authorized by the license in accordance with the regulations of the Commission set forth in 10 CFR Chapter I.

(5) The renewal of this license will not be inimical to the common defense and security or to the health and safety of the public.

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13. ABSTRACT (200 words or less)

This Safety Evaluation Report for the application filed by the University of Missouri-Rolla for a renewal of Operating License R-79 to continue to operate a research reactor has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is owned by the University of Missouri and is located on the campus in Rolla, Missouri. On the basis of its technical review, the staff concludes that the reactor facility can continue to be operated by the university without endangering the health and safety of the public or the environment.

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