Los Alamos

Los Alamos National Laboratory
Los Alamos, New Mexico 87545

DATE: November 2, 1984
IN REPLY REFER TO: Q-6-937 (R649)
MAIL STOP: K557
TELEPHONE: (505) 667-7020
FTS 843-7020

Safety Assessment

Mr. C. O. Thomas, Chief
Standardization & Special
Projects Branch
Division of Licensing
Mail Stop 340-Phil
US Nuclear Regulatory Commission
Washington, DC 20555

Dear Sir:

Enclosed is a revised draft of the sections of the University of Missouri-Rolla Reactor (UMRR) Technical Evaluation Report (TER) for which Los Alamos is responsible.

This version incorporates the additional details supplied by the licensee in their responses to the formal questions. Sections 4, 7, 11, and 14 have been extensively modified. Only minor changes have been required in the balance of the document. The best copies of all referred figures also are included in this revision.

These are rough working papers and are meant solely for your use in the further consideration of this proposed license renewal. If you have any questions about this document, please call me at the above number or, in my absence, call J. E. Hyder on FTS 843-7020.

Sincerely,

C. A. Linder

CAL/jl

Cy: H. N. Berkow, NRC/NRR
M. G. Stevenson/W. L. Kirk, Q-DO, w/o enc., MS E561
L. H. Sullivan/J. R. Ireland, Q-DO/RS, MS K552
R. A. Haarman/W. S. Gregory, Q-6, MS K557
J. E. Hyder, Q-6, MS K557
CRM-4 (2), MS A150
Q-6 File

8411060435 841102
PDR ADOCK 05000123
P PDR

A020
1/1

4. REACTOR

The University of Missouri-Rolla Reactor (UMRR) is an open-pool type reactor using up to 2.7 kg of ^{235}U fuel enriched to ~90%. It is a light-water-moderated, waterand/or graphite-reflected reactor that is authorized to operate at steady-state power levels up to and including 200 kWth. The fuel, core configuration, control rods, and control instrumentation are similar to those of 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 (bridge) that supports the reactor core, control rod systems, reactor instrumentation, and some experimental facilities. A partial reactor core configuration is shown in Fig. 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.

The UMRR generates no electricity and is used primarily 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 (Eppelsheimer, 1958 and University of Missouri-Rolla, 1979 and 1984) and from discussions with licensee personnel. The design and performance characteristics of the UMRR are summarized in Table 4.1.

4.1. Reactor Building

The walls of the building housing the UMRR are insulated steel curtain walls. Weather-stripping of doors and windows and caulking of potential air leakage points minimizes the out-leakage of air typical in this type of construction.

"REVISED"

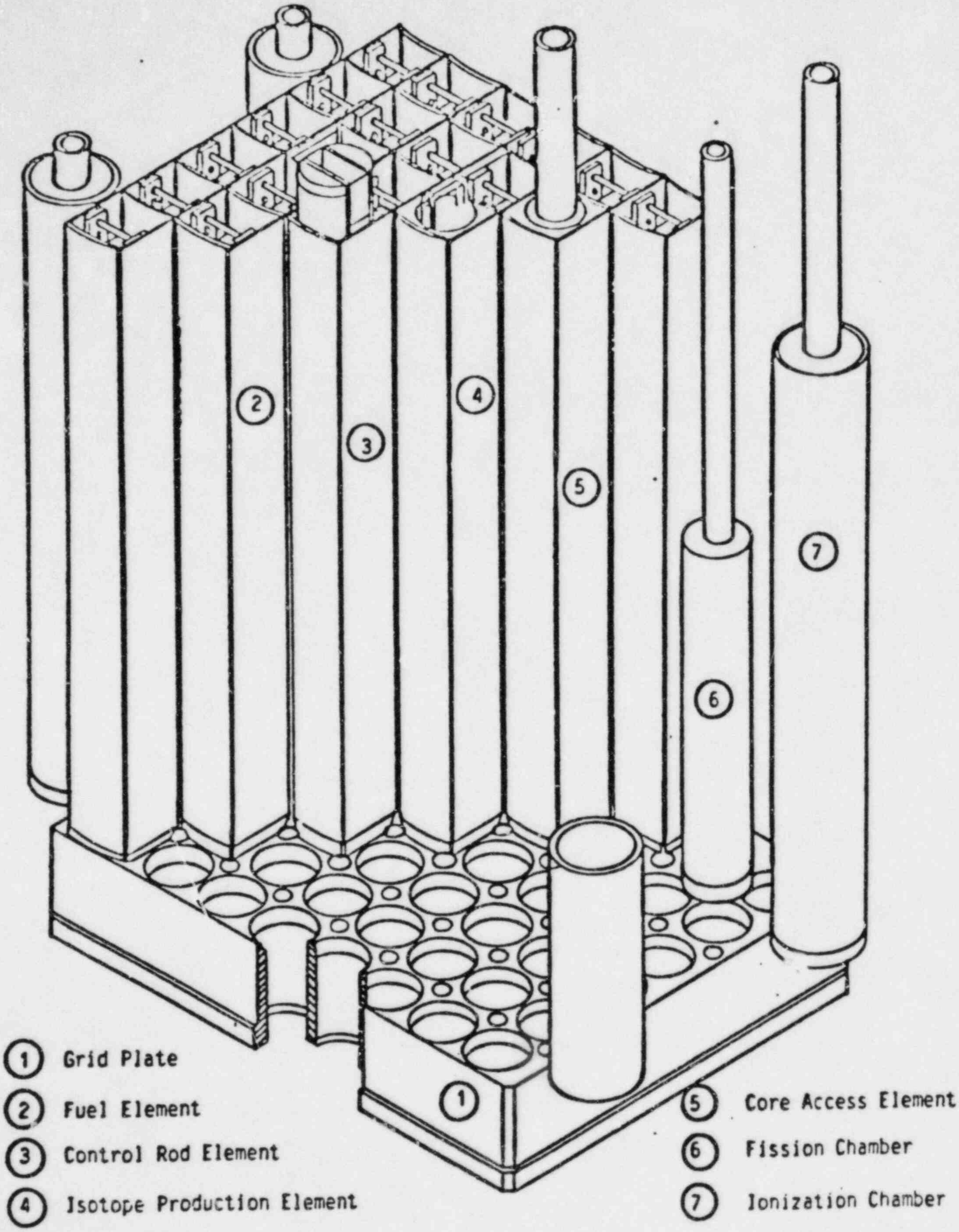


Fig. 4.1. Typical reactor core.

TABLE 4.1

CURRENT UMRR DESIGN AND PERFORMANCE CHARACTERISTICS

General Feature	
Reactor type	Heterogeneous pool
Licensed rated power level	200 kWth
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 (βeff)	0.0075
Temperature coefficient	-1.0×10^{-4}% $\Delta k/k$ per degree Celsius
Void coefficient	-7×10^{-7}% $\Delta k/k$ per cubic centimeter
Average thermal flux at 200 kW, water reflected	1.6×10^{12} n/cm^2 s
Moderator/coolant	H_2O
Reflector	H_2O and graphite
Fuel and Control Elements	
Number of fuel plates	
Fuel elements	10
Control elements	6
Enrichment	~90% ^{235}U
Maximum ^{235}U per plate	17 g
Plate Dimensions	
Thickness	0.06 in. (0.15 cm)
Width	
Fuel core	2.5 in. (6.35 cm)
Plate	3.0 in. (7.62 cm)
Length	24 in. (61 cm)
Cladding thickness (nominal)	0.02 in. (0.05 cm)
Fuel plate temperature (maximum)	315°F (157°C)

"REVISED"

TABLE 4.1 (CONT)

Control Rods and Reactivity Effects	
Material	Boron-stainless steel
Safety	3
Regulating	1 (stainless steel)
Travel	24 in. (61 cm)
Withdrawal speed (maximum)	
Safety	6 in./min (15.2 cm/min)
Regulating	24 in./min (61 cm/min)
Rod worth (current core)	
Safety (single)	2.6% - 3.4% $\Delta k/k$
Safety (ganged)	8.7% $\Delta k/k$
Regulating	0.7% $\Delta k/k$
Maximum allowed rod drop time	600 ms
Coolant	
Type	Light water
Flow	Natural convection
Inlet core temperature (nominal)	68°F (20°C)
Inlet core temperature (maximum)	135°F (57°C)
Outlet core temperature (nominal)	89.6°F (32°C)
Conductivity	<2 μmhos/cm

In addition, all vents in the ventilation system automatically close in the event of a ventilation system shutdown, providing confinement of the building air during abnormal situations.

The building is essentially a rectangular solid 33.5 ft by 39.5 ft by 32.8 ft high (10.2 m by 15 m by 10 m high). An office/reception/entrance area ~260 ft^2 (~24 m^2) was added to the building in 1979/1980.

The main floor contains the reactor room, the control room, counting rooms, and the new office space. The floor at one end of the reactor bay is dropped to provide experimenters with access to the beam tube and the thermal column.

"REVISED"

A normally locked service door, which is sealed by a neoprene gasket, is provided on the opposite end of the reactor bay. The building layout is shown in Figs. 4.2 and 4.3.

4.2. Reactor Core

The core consists of MTR-type fuel elements, four control rod fuel elements, and four control rods. Several different fuel loadings are possible with this reactor, and a 5-in. (12.5-cm)-thick aluminum reactor grid plate containing a 6 by 9 array of holes for positioning the fuel and control elements and experimental apparatus is provided.

4.2.1. Fuel Elements

The fuel and control rod fuel elements are assemblies of fuel-bearing plates. Each plate is a sandwich of aluminum cladding over a uranium-aluminum alloy "meat." The meat is approximately 0.02 in. (0.05 cm) thick, 2.5 in. (6.35 cm) wide, and contains about 17 g ^{235}U. The cladding is 0.02 in. (0.05 cm) thick. The overall dimensions of a fuel plate are approximately 3 in. (7.6 cm) wide, 24 in. (61 cm) long, and 0.06 in. (0.15 cm) thick.

The standard fuel element consists of 10 fuel plates fastened to aluminum side plates so that the finished element has an almost square 3-in. by 3-in. (7.6-cm by 7.6-cm) cross section (Fig. 4.4). A male guide-piece is attached to the bottom end of the fuel element. The guide piece has a circular cross-section and mates with the tapered positioning holes in the grid plate. A handle is attached to the top end of the fuel elements and provides a means for inserting and removing the fuel element. The overall length of a fuel element is about 36 in. (91 cm). Two half-elements (five fueled and five dummy plates) are also available.

There are four control rod fuel elements that are identical to the standard elements with the exception that the center four fueled plates have been removed and replaced with 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 inadvertant lifting of the fuel element.

"REVISED"

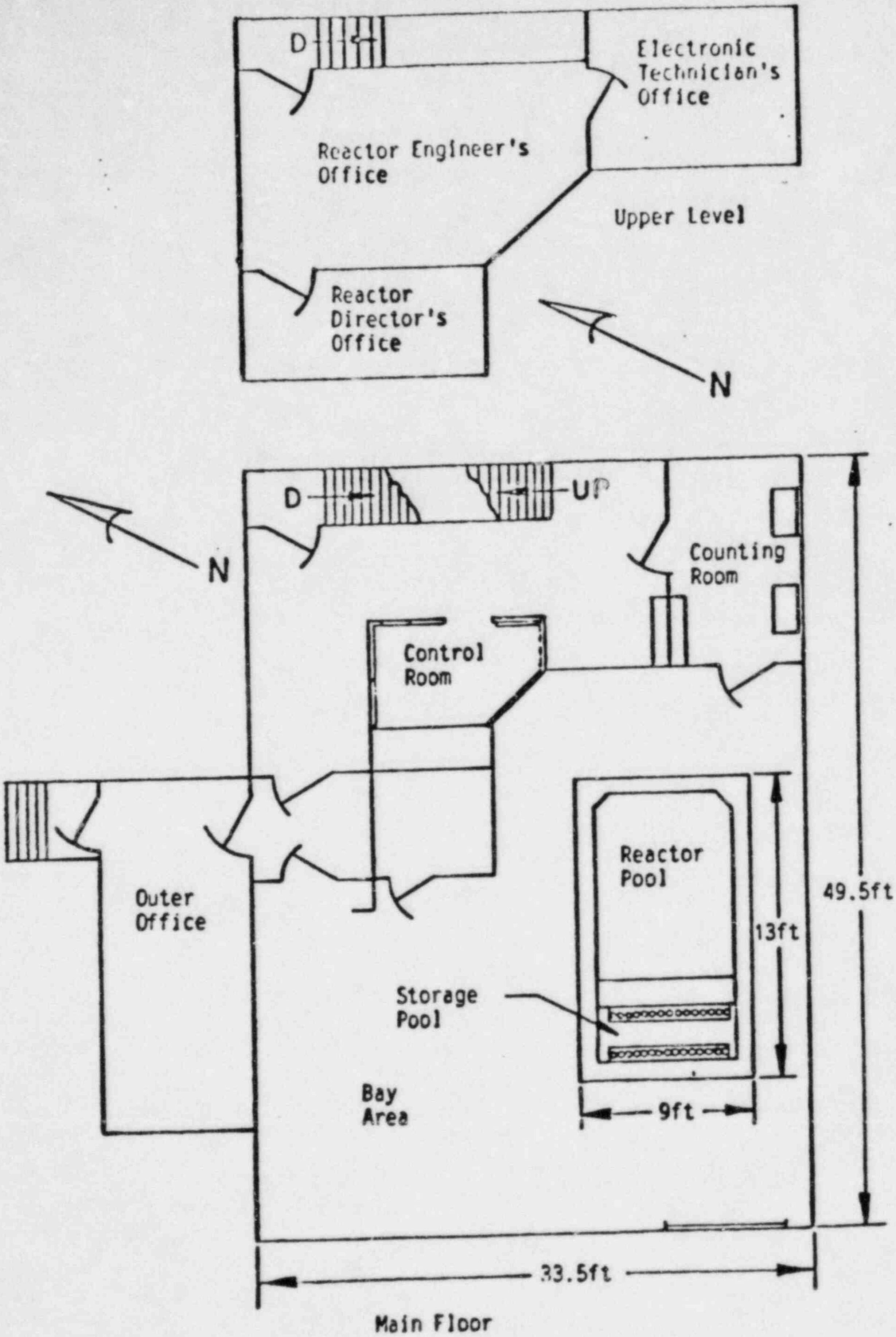


Fig. 4.2.
UMRR facility layout - Main floor and upper level.

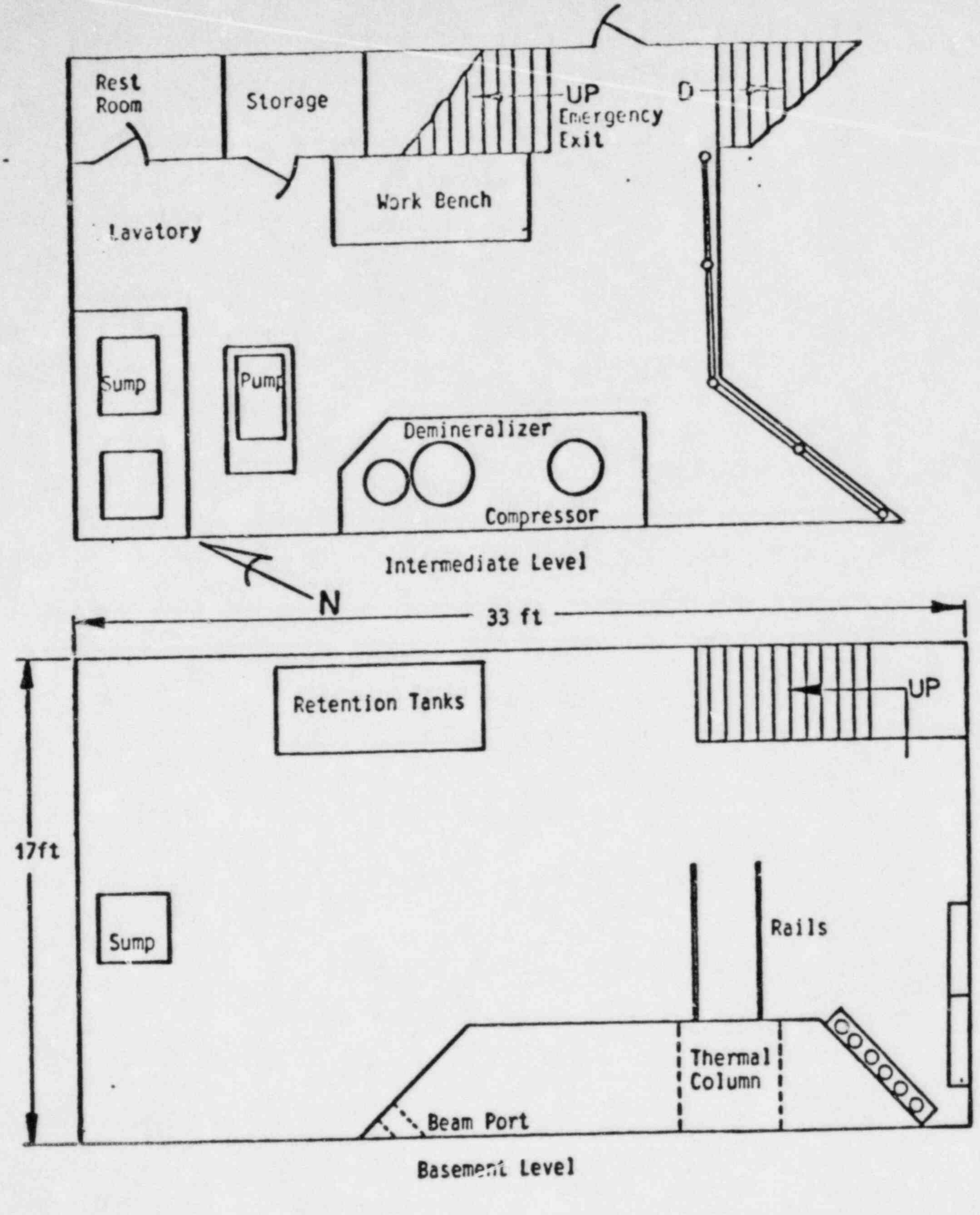


Fig. 4.3.
UMRR facility layout. Intermediate and basement levels.

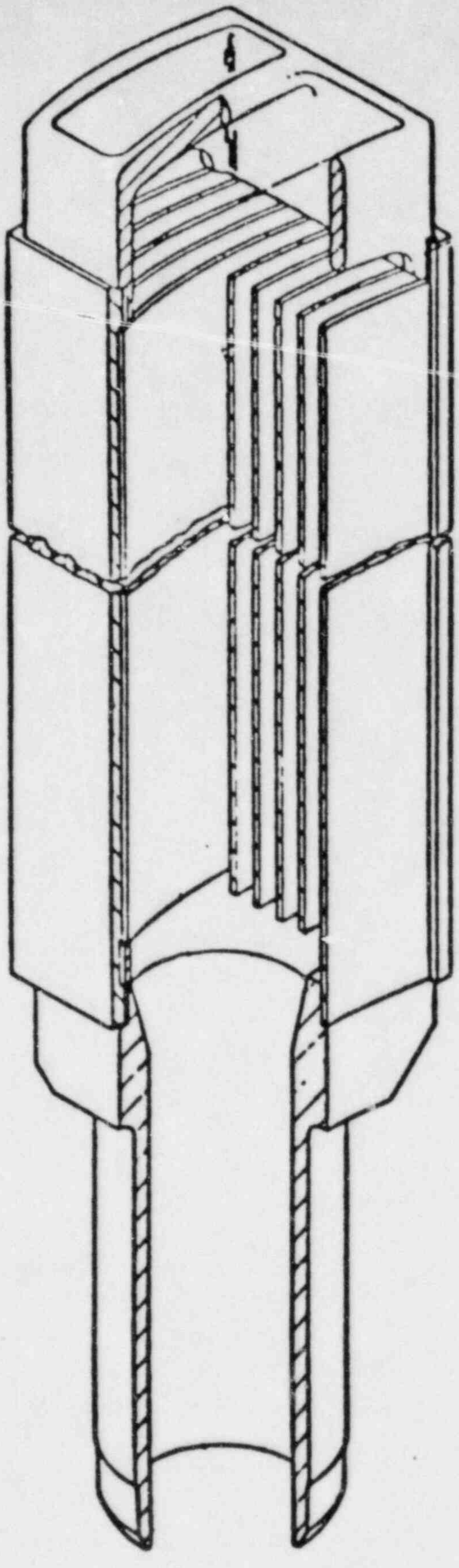


Fig. 4.4.
MTR Fuel Element.

4.2.2. Control Rods

The reactivity and power level in the UMRR are controlled by three safety rods and one regulating rod. All four rods fit into a central gap provided in special control rod fuel elements (discussed in Sec. 4.2.1).

The safety rods, which are used for coarse control, are made of boron stainless steel. The absorbing section is about 0.87 in. (2.22 cm) thick, 2.25 in. (5.72 cm) wide, and 24 in. (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 ganged worth of the three safety rods is about 8.7% $\Delta k/k$. Each safety rod is moved in and out of the core by an individual electro-mechanical system. The drive mechanisms, which are actuated from the control console, are located on the reactor bridge. The rod, which contains the absorber section, is suspended from the drive mechanism by an electromagnet. During normal operation, the safety rods are driven either in or out at a rate of 6 in./min (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 0.07-in. (0.17-cm)-thick stainless-steel tube with a 2.25-in. by 0.8-in. (5.72-cm by 2.03-cm) cross section and an effective poison length of about 24 in. (61 cm). The rod is open at the top and bottom to allow free circulation of water through it to eliminate the possibility of trapping air in the rod with a resultant variable void condition.

The regulating rod has a reactivity worth of about 0.7% $\Delta k/k$, which varies with core loading. The regulating rod is permanently fixed to its drive mechanism and travels in either direction at a speed of 24 in./min (61 cm/min). The regulating rod can be operated manually or automatically for servo-control of the reactor power level.

"REVISED"

4.3. Reactor Pool

The reactor pool is about 19 ft (5.79 m) long, 9 ft (2.74 m) wide, and 27 ft (8.23 m) deep and holds about 34 100 gal (129 000 L) of water. The pool walls are of ordinary reinforced concrete. The internal walls and floor of the pool have several coats of protective vinyl paint to minimize 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; a fuel storage space is on the opposite end. The fuel storage space is formed by a reinforced concrete bulkhead extending 16 ft (4.88 m) above and 3.5 ft (1.07 m) below the pool floor and located 2 ft (0.61 m) from the main pool wall. The fuel storage space is designed so that there will be at least 16 ft (4.88 m) of water above stored fuel elements at all times and adequate shielding for personnel working in a drained pool. There is no drain system built into this storage pool.

4.4. Reactor Support Structure

The reactor grid plate is supported by an aluminum tower assembly hung from a bridge that spans the pool (Fig. 4.5 and 4.6). The bridge is about 11 ft (3.35 m) long and 4.5 ft (1.37 m) wi nd is wheel-mounted on tracks located on the top of the pool walls and para to the long axis of the reactor pool. The bridge can be moved along its rails for a distance of approximately 6 ft (1.83 m) from its normal operating position. In the normal operating position, the tower assembly is adjacent to 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 fixed; the bottom of the core is about 3.3 ft (1 m) above the pool floor. With this core elevation, the top of the active fuel region is about 20.7 ft (6.3 m) below the surface of the water when the pool is full.

4.5. Reactor Instrumentation

The reactor instrumentation is similar to that found at research reactor installations at other laboratories. The initial control console and associated

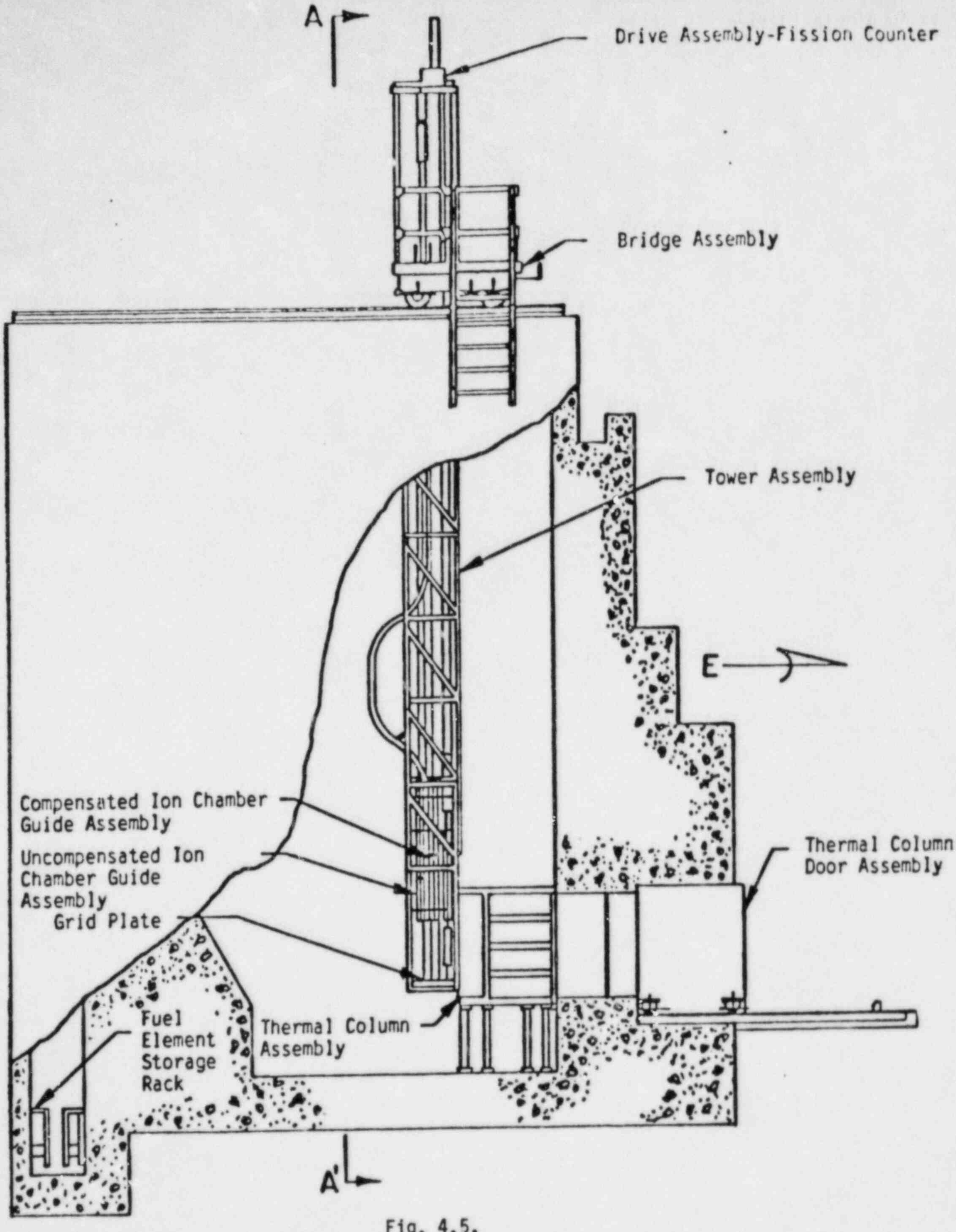
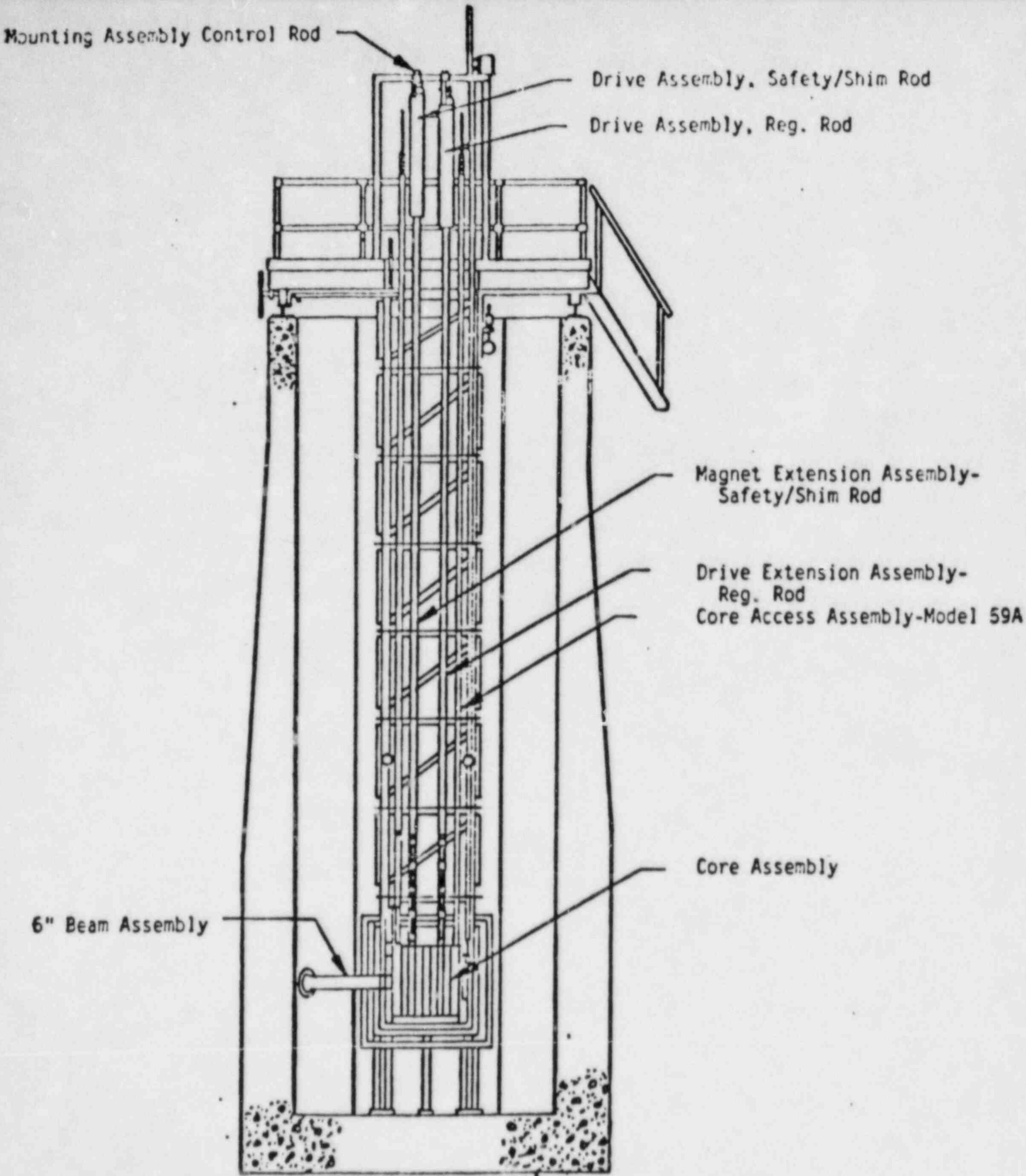


Fig. 4.5.
UMRR Cross section.



SECTION A-A'

Fig. 4.6.
UMRR Section.

instruments were typical of those of several research reactors built by the same vendor. During the past few years, instruments have been improved or replaced to provide state-of-the-art equipment.

The nuclear instrumentation provides the operator with necessary information for proper manipulation of the controls. The following instrument channels are provided to monitor reactor parameters and are discussed in more detail in Sec. 7.

(1) Count-rate or start-up channel (fission chamber)
(2) Linear power and automatic control channel
(3) Log power and period channel
(4) Two safety channels
(5) Core inlet temperature

4.6. Biological Shield

The reactor core is shielded in the lateral direction by pool water, by the concrete walls of the pool, and by earth shielding on three sides. Vertical shielding is provided by about 21 ft (6.4 m) of water above the core and 1 m of water between the core and the pool floor. The pool walls are 1.5-ft (0.46-m)-thick reinforced concrete except at the beam tube and thermal column end (Fig. 4.3), where the thickness is 6.5 ft (1.98 m). The increase in wall thickness extends above the floor of the main operating level with the thickness decreasing in steps (Fig. 4.5). Earth shielding augments the water and concrete shielding on the other three sides of the pool.

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

4.7. Dynamic Design Evaluation

The reactor is provided with redundant rapid-response controls and nuclear instrumentation (Sec. 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.

"REVISED"

The licensee, the reactor vendor, and Los Alamos 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 Sec. 14.2.

4.7.1. Shutdown Margin

The proposed Technical Specifications prescribe a minimum reactivity shutdown margin of 1.0% $\Delta k/k$ in a cold, xenon-free core with the highest worth control (safety) rod fully withdrawn and the highest worth unsecured (movable) experiment (see Sec. 4.7.3) 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% $\Delta k/k$. The maximum worth of a movable experiment is limited by the Technical Specifications to 0.4% $\Delta k/k$. Therefore, as long as the total excess reactivity loaded into the core, including that resulting from experiments in addition to the maximum worth movable experiment and all other experiments, is no more than 4.3% $\Delta k/k$ (8.7 - 3.4 - 1.0), the shutdown margin certainly can be achieved. 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 conditions of operation even if both (1) the most reactive safety rod were to fail to insert and (2) the maximum worth movable experiment were totally displaced simultaneously.

4.7.2. Excess Reactivity

Maximum excess reactivity in the UMRR core for normal operation is limited to 1.5% $\Delta k/k$ by the Technical Specifications. This amount provides for the effect at 200 kW of the negative power detect of reactivity, the negative reactivity effect of xenon at equilibrium at 200 kW, and about 1.0% $\Delta k/k$ additional for experiments, uranium burnup, and operational flexibility. Although the fundamental criterion is maintaining 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.

"REVISED"

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 area. The worth of a fuel element in such a position is less than 1.5% $\Delta k/k$. [The analysis of step reactivity insertions (Sec. 14.2.1) indicates that a step insertion of 1.5% $\Delta k/k$ will not result in fuel or core damage.] Thus, addition of fuel elements 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.

4.7.3. Experiments

The licensee's Technical Specifications provide limitations on the reactivity worths of secured and movable experiments and on reactivity insertion rates for experiments with moving parts. Los Alamos has analyzed these limitations on the basis of information provided by the licensee in the preliminary hazards summary report (Eppelsheimer, 1958), the revised SAR (University of Missouri, 1979 and 1984), and the proposed Technical Specifications.

The proposed Technical Specifications limit a single secured experiment to 0.7% $\Delta k/k$. This worth is less than β_{eff} (that is, less than 1.0\$) 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 about 2—3 s, which would initiate a period scram (set point <5 s).

"REVISED"

The proposed 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 a 0.4% $\Delta k/k$ per experiment. This is well below the 1.5% $\Delta k/k$ step reactivity insertion that the licensee and the reactor vendor (Eppelsheimer, 1958) have 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. Los Alamos has reviewed the licensee's proposed Technical Specification limitations on experiments with moving parts, finds them to be more conservative than the limitations on movable experiments, and therefore concludes that the safety analysis of movable experiments is applicable to experiments with moving parts.

We have reviewed the proposed limitations on the worth of movable and secured experiments and concludes that they are conservative and provide reasonable assurance that failure of single experiments resulting in positive reactivity insertions would not result in damage to the fuel or reactor components. However, the simultaneous removal of four movable experiments, each with worths at or close to the Technical Specifications limit of 0.4 % $\Delta k/k$, has the potential for a step reactivity insertion in excess of the 1.2 % $\Delta k/k$ that has demonstrated not to result in damage to the UMRR MTR-type fuel. However, Los Alamos considers the probability of such a four-fold coincidence to be negligibly small.

4.7.4. Assessment

On the basis of the information presented above, Los Alamos 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 of 0.05% $\Delta k/k$ per second unless the total worth of the experiments is <0.05% $\Delta k/k$, and (4) operation in compliance with the Technical Specifications minimum shutdown margin requirements 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, we believe that the 1.0 % $\Delta k/k$ shutdown margin with the most reactive rod fully withdrawn and the maximum worth movable rod in its most reactive position is sufficient to

"REVISED"

ensure that the reactor can be shut down adequately under all likely conditions. Further, Los Alamos 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 and also specify the level of licensed operators required for direct supervision of unlicensed personnel (students and operator trainees) as a function of total excess reactivity.

4.8. Functional Design of Reactivity Control Systems

4.8.1. Control Element Drives

The control rods are driven by electro-mechanical linear actuators. An actuator is essentially a ball-bearing-type screw driven through a gear reduction unit by a low-inertia reversible servo-motor. The drives for the safety rods are coupled to the control element by means of electromagnets. The regulating rod control element is attached permanently to the drive mechanisms. 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 mechanism. The three safety rods can be operated individually or as a gang. If electrical power is removed from the electromagnets, the safety rods fall into the core by force of gravity.

All control rods have control-console-mounted electronic position indicators that are accurate to ±0.05 in. (1.27 mm). The safety rods have control-console-mounted "insert limit," "shim range," and "withdraw limit" annunciator lights, and an annunciator that lights 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.

4.8.2. Scram-logic Circuitry

The UMRR is equipped with a scram-logic safety system that receives signals from core instrumentation (neutron flux density detectors) and other reactor

"REVISED"

parameters to initiate a scram by removing power from the safety rod magnets and/or the safety amplifer.

The reactor conditions that can initiate these scrams are

(1) high reactor power,
(2) short 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 Sec. 7.

4.8.3. Assessment

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

In addition to the electro-mechanical safety controls for both normal and abnormal operation, the negative bulk temperature coefficient of the moderator provides an inherent backup safety feature.

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

4.9. 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 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 malfunctions of experiments that might lead to the release of contained radioactivity.

4.10. Conclusions

Los Alamos 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 safety-related systems.

Our review of the reactor facility has included studying its specific design, installation, and operational limitations as identified in the original and proposed Technical Specifications revisions and other pertinent documents associated with the reactor. The design features are similar to those of the Bulk Shielding Reactor at Oak Ridge as well as to other pool-type research reactors operating in the US and many other countries of the world. The fuel, which is aluminum-clad high-enriched uranium-aluminum alloy, is used in over 30 NRC licensed research and test reactors in the United States and is very similar to the fuel used in the BORAX and SPERT tests. On the basis of our review of the UMRR and our experience with similar facilities, we conclude that there is reasonable assurance that this reactor is capable of safe operation, as limited by its proposed revised Technical Specifications, for the proposed duration of the license renewal.

"REVISED"

5. REACTOR COOLING

5.1. Reactor Core Cooling

The UMRR core is submerged in a pool containing approximately 3.2×10^4 gal (4.1×10^5 L) of demineralized water and is cooled by natural convection. Currently, the UMRR is operated well below the licensed limits, but even if the reactor were to be operated at 200 kW it would be 24 h before the pool water reached the maximum allowable temperature limit in the Technical Specifications [135°F (57°C)]. The pool water heat is released to the reactor room 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 Fig. 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 make-up water to the pool, raw water is introduced into an open raw water supply tank. Any overflow from this supply tank is released to the campus sewage system. The primary coolant pump takes water 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 inherent possibilites of accidental contamination of the reactor coolant or back flow to the supply system.

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

5.3. Conclusion

Los Alamos concludes that the reactor 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 prevent both corrosion and radioactivity problems associated with coolant contamination.

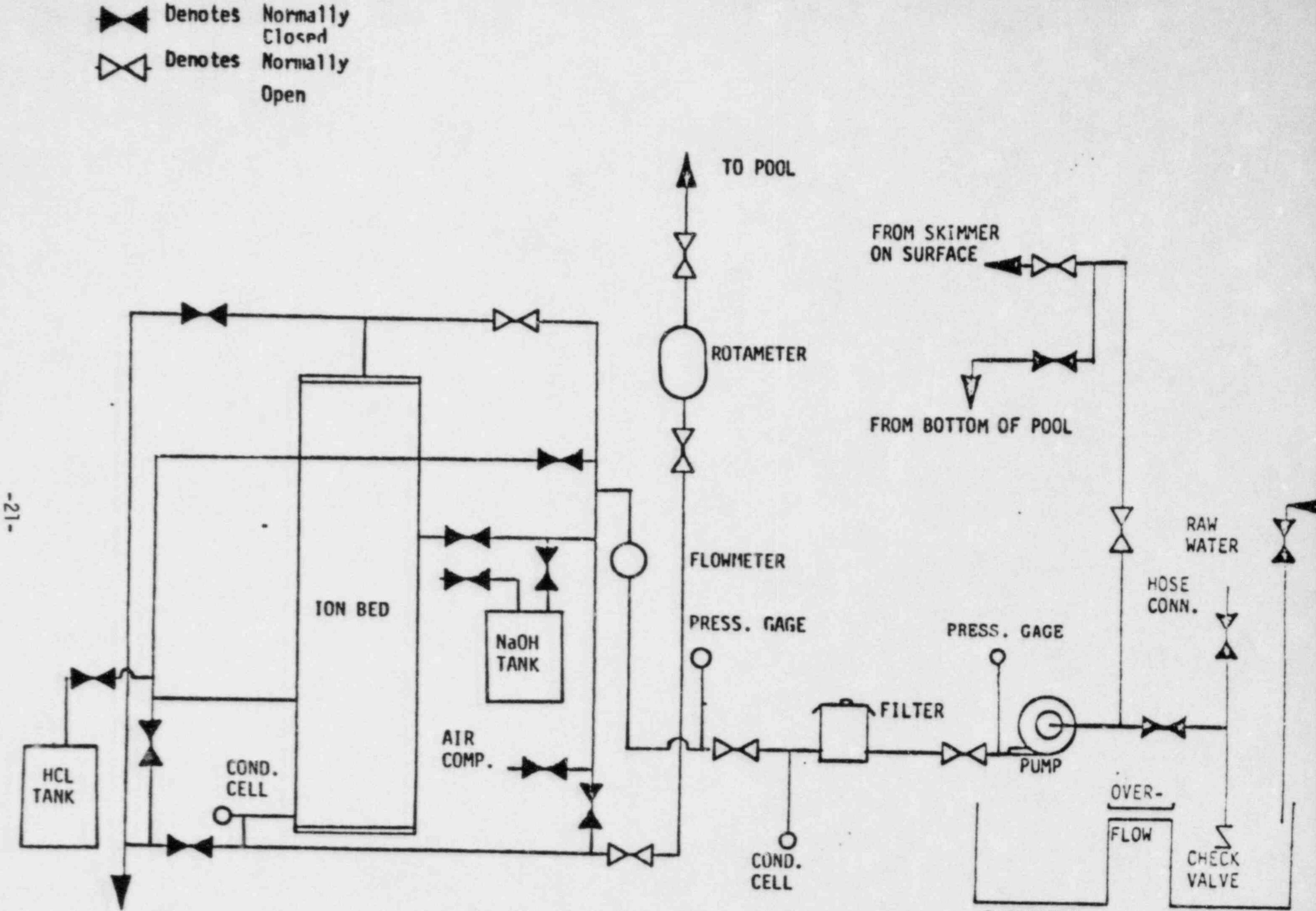


Fig. 5.1. UMRR pool water system.

6. ENGINEERED SAFETY FEATURES

Engineered safety features are systems provided to mitigate the radiological consequences of accidents. The only engineered safety feature provided at the UMRR facility is the ventilation system in that the discharge rate can be controlled and/or shut down.

6.1. Ventilation System

Reactor facility ventilation is accomplished by a system of three exhaust fans that are mounted on the reactor building roof. Air enters the facility through two intakes equipped with fiberglass filters located on the lower level. The discharge rates of the exhaust fans are 5000 ft^3/min (142 m^3/min), 15 000 ft^3/min (425 m^3/min), and 15 000 ft^3/min (425 m^3/min). The controls for the fans are located near the entrance to the control room, and any combination of fans may be used.

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 in the event of an unspecified release within the reactor building. (See Sec. 14 for additional details.)

6.2. Conclusion

The UMRR is a 200-KW pool-type reactor that operates about 30 full-power h/yr. Therefore, the fission product inventory is not large and the inventory at risk in an individual fuel element or in an irradiated sample is even less.

Los Alamos has determined that the operation of the UMRR 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. The ability to control potential release rates only adds to an already safe situation.

7. CONTROL AND INSTRUMENTATION

The control and instrumentation systems at the UMRR 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 both nuclear and process instrumentation and generally is characterized by modern components. The UMRR has a program in operation to replace older instruments with state-of-the-art systems that provide the same functions more reliably.

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 in case of single failure or malfunction of components essential to the safe operation or shutdown of the reactor. (See Tables 7.1 and 7.2.)

7.1.1. Nuclear Control Systems

The reactor is controlled by inserting and withdrawing neutron-absorbing control rods using drive units mounted on the bridge structure over the pool. The regulating rod has a solid coupling and cannot be scrammed. The other three control elements 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 element drives are controlled from the control room by the reactor operator. The control rod systems are discussed in more detail in Sec. 4.2.2.

7.1.2. Supplementary Control Systems

These control systems, also designated as process control systems, are designed to control the various processes involved in reactor operation but do not directly relate to safety. Included in this category are circuits and devices that monitor coolant parameters, such as temperature and conductivity. These control systems assure proper operation of the nonsafety related

"REVISED"

"REVISED"

TABLE 7.1

SAFETY SYSTEM CHANNELS

Function/Situation	Detector	Unit Initiating Action	Resulting Action	Annunciation	Set Point[1]
Manual Scram	Operator	Scram Button	Scram	Yes	Operator
Period	Compensated Ion Chamber	Log-N and Period Amplifier	Scram	Yes	<5 s
Power	Uncompensated	Safety Amplifier	Scram	Yes	>300 kW
Bridge Motion	Motion Switch	Motion Switch	Scram	Yes	1.3 cm horizontal travel
Log-N and Period Amplifier	Log-N Period Amplifier	Relay	Scram	Yes	Not in OPERATE position

TABLE 7.2

CONTROL CHANNELS

Function/Situation	Detector	Unit Initiating Action	Resulting Action	Annunciation	Set Points
Power Demand	Compensated Ion Chamber	Linear Recorder	Rundown	Yes	~120% of Selected Scale
Period	Compensated Ion Chamber	Period Recorder	Rundown	Yes	15 s
Regulating Rod Insert Limit on Automatic	Microswitch	Microswitch	Rundown	Yes	0.0
CIC Voltage	DC Relay	DC Relay	Rundown	Yes	400 V
Power	Compensated Ion Chamber	Log-N Recorder	Rundown	Yes	240 kW
High Radiation[2,3,4] GM Tubes at Ram	GM Tubes	Remote Area Monitoring (RAM) System	Rundown	Yes	20 mR/h (0.2 mSv/h) low 30 mR/h (0.3 mSV/h) high
Evacuation Alarm	GM Tubes	RAM	Initiate evacuation sequence; both automatic and manual actuation	Yes	30 mR/h (0.30 Sv/h)
Period 30 s	Compensated Ion Chamber	Period Recorder	Rod Prohibit	Yes	30 s
Any Recorder Off[5]	Relay	Relay	Rod Prohibit	Yes	
Log Count Rate 2 counts/s or Less[3]	Fission Chamber	Log Count Rate System	Rod Prohibit	Yes	2 cps
Safety Rods Below Shim Range	Microswitch	Relay	Regulating Safety Rod Prohibit	No	
Regulating Rod Above Insert Limit[2]	Microswitch	Relay	Prohibit	No	
Reactor Power Deviation	Compensated Ion Chamber	Linear Channel	Servo-prohibit	Yes	~5% of Selected Power Level
Core Inlet Water Temperature	Thermocouple	Relay	Rod Prohibit	Yes	135°F (57°C)

[1]Limiting values; operational set points may be more limiting.
[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.

"REVISED"

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 function is provided through the safety amplifier discussed below.

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 0.06 W to 300 kW and provides the signal for automatic servo-control 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 300 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 derived from a thermal calibration.

A drop in the high voltage to the CICs will result in a reactor power rundown (Table 7.2). Also, if the log-N and period channel amplifier is not in the operating position, a relay in the scram system will prevent closure 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 of assistance in the operation of the facility.

A core inlet water temperature $\geq$135°F (57°C) initiates a rod withdrawal prohibit and an alarm. The coolant core inlet prohibit and alarm are activated by a thermocouple in the pool water below the core grid plate. The conductivity of the pool water flowing to the demineralizer is monitored by a conductivity bridge. Conductivities $\geq$ 2 μmhos activate a reactor console alarm.

Loss of ac power to the console will scram the reactor automatically. The reactor console key in the off position is essentially identical to the loss of ac power to the console and causes a reactor scram if turned off when the reactor is running.

7.2.3. Inhibits and Annunciation

Inhibit signals that will prevent control blade removal (reactor startup) are provided by 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 $\leq$30 s, if the core inlet temperature is $\geq$135°F (57°C) and the safety rods are below the shim range, or if the regulating rod is above the insert limit.

"REVISED"

A control-console-mounted annunciator panel of lights provides the operator with information on the condition 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 acknowledge and reset to restore the annunciator to normal operating condition.

7.2.4. Reactor Safety System

The control and instrumentation systems are interconnected through a safety amplifier. This unit provides current for the electromagnets that support the control-safety blades, as well as high-voltage power for the safety channel ion chambers. Each ion chamber is provided with an independent amplifier circuit that will cause a scram upon receipt of an appropriate trip signal or upon failure. The safety circuit provides for 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 system (CAM). The alarm set points are listed in Tables 7.2 and 7.3.

Single RAMs are located at the reactor bridge, at the demineralizer, and in the equipment room (basement). The monitors alarm both locally and in the control room. All of the RAM alarms 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. The monitor measures the neutron flux level in the beam room and alarms both locally and in the control room.

The CAM detects airborne particulate material and provides an alarm. A continuous sample is drawn from the reactor room through a filter. The air sample

"REVISED"

"REVISED"

TABLE 7.3

INFORMATION CHANNELS

Function/Situation	Detector	Unit Initiating Action	Annunciation	Set Points[1]
Interlock Bypassed	Key Switch	Key Switch	Yes	
Effluent Pool Demineralizer Conductivity 2 μmhos/cm or more	Conductivity Bridge	Relay	Yes	2 μmhos/cm
High Neutron Flux in Beam Room[2]	BF_3 Neutron Detector	Relay	Yes	30 mR/h (0.3 m Sv/h)
Airborne Particulate Material	GM Tube	Building CAM	Yes	Footnote 3

[1] Limiting values, operational set points may be more limiting.
[2] These will be set by measurement during initial increase in power level. The set points will be less than 30 mrem/h.
[3] 50% of limits in 10 CFR 20, Appendix B, Table 1, Col. 1.

stream passes through the particulate detector system and is released to the reactor room. The particulate filter is replaced periodically (generally monthly) and is assayed for gross beta-gamma activity.

7.4. Conclusions

The control and instrumentation systems at the UMRR are well designed and maintained. Redundancy in the important ranges of power measurements is ensured by over-lapping 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 shut down automatically and safely if electrical power is lost.

On the basis of its review of the control and instrumentation systems, Los Alamos has concluded that these systems are adequate to ensure safe operation of the reactor within the limits of the proposed Technical Specifications.

"REVISED"

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 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 Sec. 14.5), no emergency power is supplied except battery-operated emergency lighting for personnel movement during a power outage.

8.3. Conclusions

The above factors leads Los Alamos to conclude that the electrical power system is acceptable for continued operation of the UMRR.

"REVISED"

9. AUXILIARY SYSTEMS

9.1. Fuel Handling and Storage

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

Two storage racks are available in the fuel storage pit that are capable of holding up to 30 fuel elements in two separate 1 x 15 arrays. The fuel elements are oriented in the storage racks in the same manner as in the core (standing). The 3- by 3-in. (7.6- by 7.6-cm) fuel elements are positioned on about 5-in. centers, and the two arrays are separated by 20--24 in. (50.1--70.0 cm), thus ensuring a $K_{effective}$ of <0.9

9.2. Fire Protection System

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

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

In the event that 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

"REVISED"

fire extinguishers located throughout the reactor building at strategically important locations.

The City of Rolla Fire Department, which responds to all campus alarms, is located less than 1 mile from the reactor facility.

9.3. Air Conditioning

A recirculating air conditioner located in the reactor building regulates air for human comfort in the experimental area, equipment room, and reactor room.

9.4. Conclusion

Los Alamos concludes that the fuel handling facilities are appropriate for the reactor size and use. We further conclude that the fire protection equipment and organization are acceptable. Finally, Los Alamos concludes that the ventilation system is adequate to disburse safely the small amount of radioactive gas produced in the UMRR facility. (See Sec. 11.)

10. EXPERIMENTAL PROGRAM

In addition to being an integral part of the nuclear engineering undergraduate and graduate educational programs, the UMRR supports the various experimental programs of the staff and students. 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. Experimental Facilities

The experimental facilities available in the UMRR are shown in Figs. 4.1, 4.5, 4.6, and 10.1 and are described below.

10.1.1. Beam Hole

There is a 6-in. (15.2-cm.)-diam. (nominal) beam hole as shown in Fig. 4.6. The beam hole is lined with stainless steel and has a removable aluminum beam tube and separately removable beam tube extension. Irradiations can be performed within the beam tube or in the external 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 door.

10.1.2. Thermal Column

The thermal column assembly is a 3.6-ft by 3.6-ft by 5-ft (1.1-m by 1.1-m by 1.5-m) cube of graphite extending from the reactor core into the concrete biological shield (Fig. 4.5). The reactor face of the thermal column assembly is covered with a 4-in. (10.2-cm) lead shield. A 4-ft by 4-ft by 5-ft (1.2-m by 1.2-m by 1.5-m) movable concrete thermal column door provides access

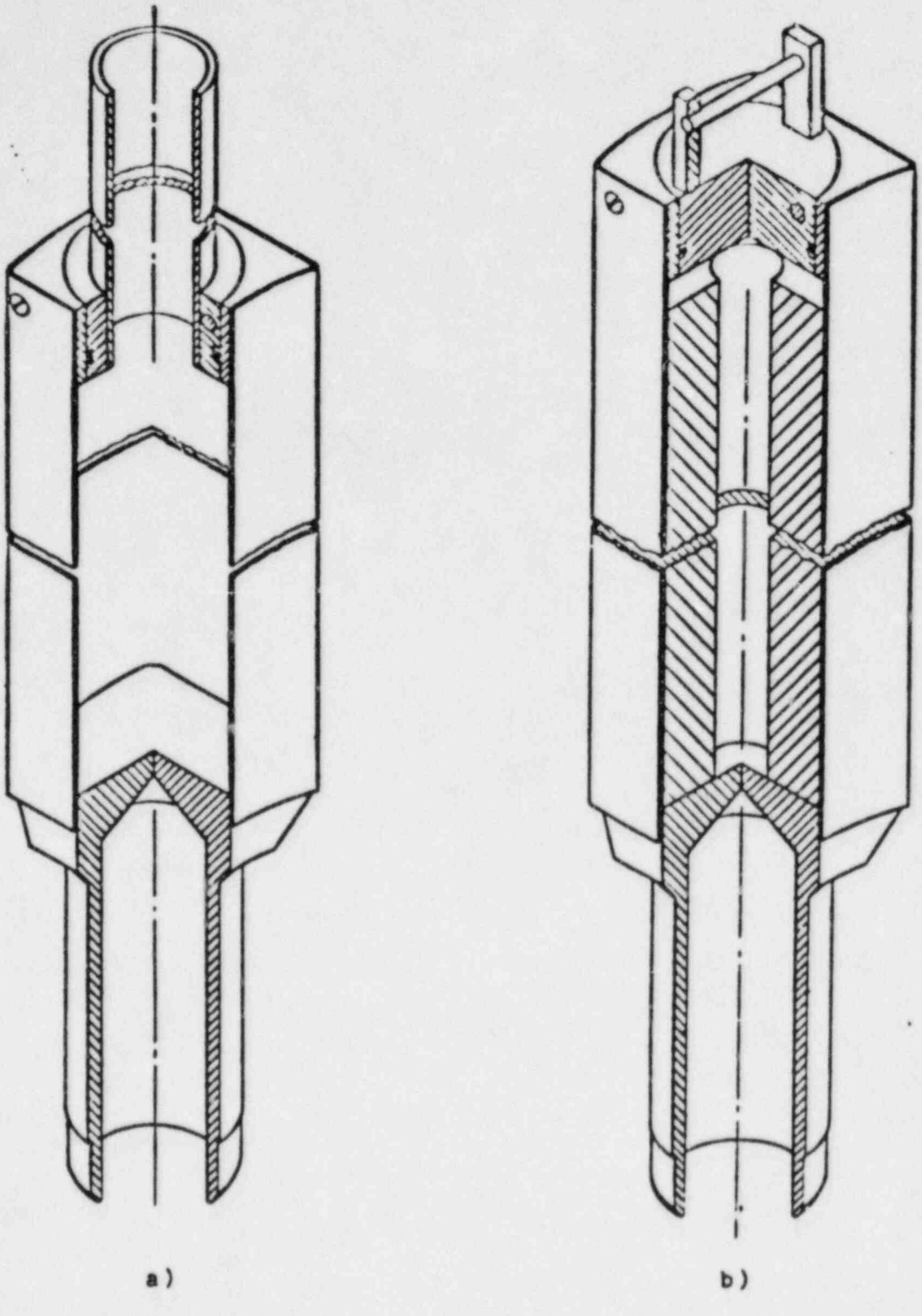


Fig. 10.1. (a) Core access element.
(b) Isotope production element.

to the thermal column as well as additional shielding when in the closed position. The inner face of the door is lined with boral.

There are five irradiation facilities in the thermal column assembly. These are one 8-in. (20.3-cm)-square and four 4-in. (10.2-cm)-square horizontal access ports that are filled with graphite plugs when they are not in use.

10.1.3. Irradiation Elements

The two types of irradiation elements (Fig. 10.1) that can be used for incore irradiations are discussed below. The irradiation elements are designed to fit into the grid plate holes (Fig. 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 an irradiation sample. The top sealing plug is held in place by an aluminum pin that is inserted in a horizontal through-hole. An O-ring seal also permits the use of the isotope production element as a dry irradiation facility.

10.1.3.2. Core Access Facility

The core access element provides unreflected access to the active 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 above the pool water level and is curved to prevent neutron and gamma streaming. Samples are lowered into the core access elements with a leader.

10.1.4. Pneumatic Transfer Facility

The pneumatic (nitrogen gas) 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 to core configurations in which at least one of the sides of the rabbit-tube faces a moderating medium. The tube can be lined with cadmium for experiments not requiring slow neutrons.

10.2. Experimental Review

All proposed new experiments, procedures, and facility changes 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 committee may be from the organization responsible for reactor operations. 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 can be approved by the Reactor Manager. Such changes are subject to RSC approval.

10.3. Conclusions

Los Alamos 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 proposed Technical Specifications ensure acceptable and safe experimental programs. Therefore, we believe 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 reactor operation 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 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 microcuries of radionuclides per year.

The solid waste is collected in specially marked containers. It then 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.

"REVISED"

11.2.2. Liquid Waste

Normal reactor operations produce no radioactive liquid waste. However, many of the research activites conducted within the reactor complex are capable of generating such waste. Liquid waste drains in the reactor room and equipment areas drain 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 purpose of the retention tanks is to allow for additional radioactive decay of these regeneration liquids. The tank contents eventually are released to the lower level sump, pumped to the middle level sump, and released into the sanitary sewer system.

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 guidelines, and the solutions have been retained for several weeks following the regeneration before discharge.

11.2.3. Airborne Waste

The 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 is 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 half-life) ^{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 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.

The UMRR has calculated the release of airborne radioactivity (mostly ^{41}Ar) at less than 25 mCi/yr (0.925 GBq). Both the applicant's and staff's evaluation show that this amount of release would lead to exposures in the unrestricted areas that are well within the limits specified in 10 CFR 20.

11.3. Conclusion

Los Alamos concludes that the waste management activities at 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 Los Alamos review has followed the methods of ANSI/ANS 15.11, 1977, "Radiological Control at Research Reactor Facilities."

Because ^{41}Ar is the only significant radionuclide released by the reactor to the environment during normal operations, we have reviewed the history, current practices, and future expectations of operations with regard to this radionuclide. We conclude that the 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, our calculations of the dose beyond the limits of the reactor facilities 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.

12.1. ALARA Commitment

The Office of Radiation Safety has implemented the policy that operations are to be conducted in a manner to keep all radiation exposures as low as reasonably achievable (ALARA). A training film has been developed that explains the hazards of radiation and discusses techniques of minimizing exposures in detail. 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. The routine health physics-type activities at the reactor can be 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.

Los Alamos believes that the radiation safety support is adequate for the research efforts within this reactor facility.

"REVISED"

12.2.2. Procedures

Detailed written procedures have recently been revised to address the Radiation Safety support that is expected to be provided to the routine operations of the University's research reactor facility. These procedures identify the interactions between the operational and experimental personnel. They also specify numerous administrative limits and action points as well as appropriate responses and corrective actions if these limits or action points are reached or exceeded. Copies of these procedures are readily available to the operational and research staffs 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 yr. The level of retraining given is determined by the examination results.

"REVISED"

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 clean-up systems, and radioactive gases (primarily ^{41}Ar).

The fission products 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 area and its discharge to the atmosphere, where it diffuses further before reaching unrestricted 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

"REVISED"

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 is also a constant particulate air monitor in the reactor room 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 to minimize personnel exposures and limit 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, TLDs and self-reading 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 personnel annual exposure history for the last 5 yr is given in Table 12.1. These data indicate that both the management of reactor operations and the radiation protection program are effective in limiting personnel exposures at the UMRR.

"REVISED"

TABLE 12.1

NUMBER OF INDIVIDUALS IN EXPOSURE INTERVAL

Whole-body exposure range (rem or 10^{-2} Sv)	Number of individuals in each range 1979	1980	1981	1982	1983
No measurable exposure	44	44	38	47	31
Measurable exposure less than 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
Number of individuals monitored	44	60	41	48	39

12.6. Effluent Monitoring

12.6.1. Airborne Effluents

As discussed in Sec. 11, airborne effluents from the reactor facility consist principally of low concentrations of ^{41}Ar. The small amount of ^{41}Ar released into the reactor room is diluted by the almost 60 000-ft^3 (1700-m^3) volume of air. A measured concentration in the reactor room after 3.5 h of full-power operation with only the small exhaust fan in operation (1/7 of the normal exhaust rate) (see Sec. 6.1) was 5 x 10^{-8} μCi/mL (1.85 x 10^{-6} kBq/mL). In actual operation, this concentration is seldom achieved. Reactor room air is normally discharged at a rate of about 35 000 ft^3/min (1000 m^3/min) near the top of the reactor building, resulting in additional dilution before reaching unrestricted areas.

12.6.2. Liquid Effluent

The reactor generates very limited radioactive liquid waste during routine operations. However, leaks in the primary coolant system do have the potential for releases, 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 lead to the lower level sump. The periodically generated waste liquid produced by the regeneration of the demineralizer is collected in two 300-gal (1136-L) waste storage tanks.

12.7. Potential Dose Assessments

Natural background radiation levels in the Rolla, Missouri, area result in an exposure of about 105 mrems/yr (1.05 mSv/yr) to each individual residing there. At least an additional 8% [approximately 8 mrems/yr (0.08 mSv/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 exposure of those individuals.

Conservative calculations by the staff based on the amount of ^{41}Ar released from the reactor facility stack during normal operations predict a maximum annual exposure of less than 1 mrem (0.01 mSv/yr) in the unrestricted areas.

12.8. Conclusion

Los Alamos considers that radiation protection currently receives appropriate support from the University administration. We conclude that (1) the program is staffed and equipped properly, (2) the reactor health physics staff has adequate authority and lines of communication, (3) the procedures are integrated correctly into the research plans, and (4) surveys verify that operations and procedures achieve ALARA principles.

Additionally, Los Alamos concludes that the University of Missouri-Rolla radiation protection program is acceptable because we have found no instances of reactor-related exposures of personnel above applicable regulations.

Furthermore, we consider 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.

14. ACCIDENT ANALYSIS

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

Among the accidents postulated, the one with the greatest potential effect on the environment 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 clad failure and would not result in a release of radioactivity.

The failure of a fueled experiment is designated 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. Thus, we assumed that the accident occurs but did not try to describe or evaluate the mechanical details of the accident or the probability of its occurrence. Only the consequences are evaluated.

The potential accidents or effects that have been evaluated:

(1) failure of a fueled experiment,

(2) rapid insertion of reactivity,

(3) loss of coolant, and

(4) fuel handling.

They are discussed in the following sections.

14.1. Failure of a Fueled Experiment

As mentioned above, the failure of a fueled experiments is defined as the MHA for this reactor. Los Alamos evaluated the failure of a fueled experiment based on the semi-infinite cloud model outlined in NRC Regulatory Guides 1.25 and 1.109. It was assumed conservatively that 100% of the noble gases and 50%

of the halogens would be released from the experiment upon total failure (AEC report TID 14844). An irradiation time of 8 h was assumed along with an experiment fission power of 100 W.

Additionally, it is assumed that the fission products are released into the reactor building instantaneously and dispersed uniformly within the air. It is assumed that a person within the reactor building would be exposed to the radioactivity for 5 min before being alerted and evacuated from the reactor building. The free air volume of the facility is ~6 x 10^4 ft^3 (1700 m^3). For evaluating inhalation volumes, a breathing rate of 3.47 x 10^{-4} m^3/s is assumed. The computed doses in the reactor building are given in Table 14.1.

For a person just outside the building, the doses were computed assuming (1) that all the redionuclides released to the building in accident were released over the same time period that the individual(s) at risk were being exposed, (2) the dispersion factor (χ/Q) was 0.01 s/m^3, and (3) that there was no radioactive decay during the release. The computed doses for outside the reactor building are given in Table 14.2. Potential exposure to individuals in the unrestricted area would be less than those forming the bases of 10 CFR 20. The above analysis is conservative for a number of reason.

1. No credit was taken for dissolution, chemical combination, washout or plateout of radionuclides in the pool or reactor building.

2. No decrease in source strength resulting from radioactive decay was assumed.

3. Unfavorable atmospheric dispersion conditions were assumed (minimum dispersion factor).

4. It was assumed in the case of onsite exposure that 5 min would be required to exit the building and that for offsite exposure the individual at risk would be exposed to the entire discharge plume.

5. The semi-infinite cloud model results in doses that may be high by an order of magnitude.

TABLE 14.1

RADIATION DOSES WITHIN UMRR REACTOR BUILDING[a]

Element	Beta dose[b] (mrem or 10^{-2}mSv)	Gamma dose[b] (mrem or 10^{-2} mSv)	Gamma dose[c] (mrem or 10^{-2} mSv)	Thyroid dose commitment[c] (rem or 10^{-2} Sv)
I	154	682	29	14.8
Kr	374	600	30	d
Xe	568	347	18	d

[a]Experiment Fission Power = 100 W, irradiation time = 8 h, and evacuation time = 300 s.
[b]Semi-infinite cloud model
[c]Finite-cloud model
[d]Thyroid doses from krypton and xenon are negligible.

TABLE 14.2

RADIATION DOSES FOR ENVIRONMENT OUTSIDE UMRR REACTOR BUILDING[a]

Element	Beta dose[b] (mrem or 10^{-2} mSv)	Gamma dose[b] (mrem or 10^{-2} mSv)	Thyroid dose commitment[b] (rem) or 10^{-2} Sv)
I	9	39	0.8
Kr	21	34	c
Xe	32	20	c

[a]Assumes exposure time = release time = 2h, $\chi/Q = 10^{-2}$ s/m^3, and no decay once in environment.
[b]Semi-infinite cloud model
[c]Thyroid doses from krypton and xenon negligible.

Based on the above analysis, Los Alamos concludes that fueled experiments can be used at the UMRR facility in accordance with the limitations stated in the Technical Specifications without undue risk to public health and safety.

14.2. Rapid Insertion of Reactivity (Nuclear Excursion)

The licensee has analyzed potential transients that might result from a rapid insertion of reactivity. Los Alamos also evaluated potential transients resulting from a 1.5% $\Delta k/k$ ramp insertion of reactivity during startup conditions.

14.2.1. Step Insertion of Reactivity

At the UMRR, the 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 in the central position of the core will cause a reactivity change of about 0.7% $\Delta k/k$. A fuel handling accident will not result in a reactivity insertion greater than 1.5% $\Delta k/k$. From the above, the analysis will assume a step insertion of reactivity of 1.5% $\Delta k/k$.

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[1-4] demonstrated that no mechanical damage or high fuel temperatures

TABLE 14.3

UMRR VS SPERT-I FUEL DATA

Geometry	UMRR Plate	SPERT-1 Plate
Length	24.0 in. (61 cm)	24.0 in. (61 cm)
Width	3.0 in. (7.6 cm)	3.0 in. (7.6 cm)
Thickness	0.06 in. (0.15 cm)	0.06 in. (0.15 cm)
Water gap	0.25 in. (0.63 cm)	0.25 in. (0.45 cm)

"REVISED"

TABLE 14.3 (CONT)

Fuel	UMRR Plate	SPERT-1 Plate
Material	U_3O_8-Al	U-Al
Enrichment (%)	93	100
Weight fraction of U	0.36	0.24
Thickness	0.2 in. (0.51 mm)	0.2 in. (0.51 mm)
Cladding		
Material	Al	Al
Thickness	0.2 in. (0.51 mm)	0.2 in. (0.51 mm)

occurred for a step insertion of 1.5% Δk/k. Based on 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% Δk/k.[1] Thus, we conclude that a step insertion of reactivity of 1.5% Δk/k will not result in fuel or core damage.

14.2.2. Ramp Insertion of Reactivity

During startup it is possible for all three safety/shim rods to be ganged, which would provide a maximum ramp insertion rate of 0.064% Δk/k s. If the interlock failed on the regulating rod, all four rods could be withdrawn simultaneously, providing a maximum ramp insertion rate <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.[3] Assuming an insertion of 1.5% Δk/k at a rate of 0.08% Δk/k/s from critical at 5 W and using the results of SPERT I Tests No. 2733 and No. 2727, a period of about 0.08 s, a peak power of about 6.0 MW, and a maximum fuel temperature of about 248°F (120°C) would occur initially.[3] The reactor power then would oscillate about 0.5 MW, which the licensee's heat transfer analysis has shown would not result in damage to the fuel. These results do not depend on the safety system working initially. Thus, the staff concludes that no fuel damage will result as a result of a maximum ramp insertion of reactivity of 1.5% $\Delta k/k$ at a rate of 0.08% $\Delta k/k/s$.

14.3. Loss of Coolant

A loss of coolant is considered extremely unlikely because of the design and construction of the reactor pool. If the pool does drain, the loss of water (moderator) would shut down the reactor and the removal of decay heat would occur by natural convection of ambient air. The initial decay heat at shutdown from full power is 14 kWth. The decay power will decrease rapidly; a fuel temperature of 797°F (425°C) is the maximum expected if there is no decrease in heat source strength during this transient. Thus, Los Alamos concludes that no fuel damage will result from a loss of coolant.

14.4. Fuel Handling

The staff has analyzed 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 inside, each fuel element is moved individually using a manual handling tool and put into the storage space within the reactor pool. If a fuel element was dropped inadvertently during transfer, sufficient mechanical distortion of the end fittings as to prohibit continued use as a fuel element possibly could occur; however, sufficient damage to strip cladding from one or more fuel plates with subsequent release of fission products is not credible. Experiments at the Curtiss-Wright Research Reactor[5] have shown that the worth of an outside fuel element is less than 1.5% $\Delta k/k$. Therefore, if a fuel element was dropped next to a barely subcritical core, the resulting reactivity insertion would not be greater than 1.5% $\Delta k/k$ with consequences less than those analyzed in Sec. 14.2.1.

The reactor room is not provided with an overhead hoist and, because of its construction (steel curtain walls), cannot be equipped with one that could handle an irradiated fuel shipping cask. The truck access door is not large enough to accomodate a large-capacity crane. Because of these limiting physical aspects of the facility, there are no in-pool operations involving fuel casks, and the potential for dropping a cask on the core does not exist. As noted in Section 9, irradiated fuel is stored in one end of the reactor pool. If it becomes necessary to ship irradiated fuel, procedures would have to be developed for transfer of fuel to a fuel shipping cask located outside the building and would incorporate measures designed to eliminate the dropping of a small fuel transfer cask on the stored irradiated fuel elements.

Los Alamos concludes, on the basis of the above considerations, that fuel handling accidents will not lead to release of fission products to the reactor building or the environment because of fuel cladding failures.

14.5. Conclusion

Los Alamos has reviewed the potential transients for the UMRR. Based on the review, the most significant event that is postulated to result in a 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 the limits of 10 CFR 20. Therefore, we conclude that the design of the facility together with 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.

References

1. D. S. Eppelsheimer, "Preliminary Hazards Evaluation, The Curators of the University of Missouri, School of Mines and Metallurgy 10 kW Training Reactor, Rolla, Missouri" (December 26, 1958).

2. University of Missouri, "Hazards Analyses University of Missouri-Rolla Reactor" (December 14, 1979).

3. University of Missouri, "Final Safety Analysis Report" (1984).

4. J. R. Dietrich, "Experimental Investigation of the Self-Limitation of Power During Transients in a Sub-cooled Water-Moderated Reactor, BORAX-1 Experiments," Argonne National Laboratory report AECD-3668 (1954).

5. W. E. Nyer, S. G. Forbes, F. L. Bentzen, G. O. Bright, F. Schroeder, and T. R. Wilson, "Experimental Investigations of Reactor Transients," Idaho Operations Office report IDO 16285 (April 20, 1956).

6. R. W. Miller et al., "Report on the SPERT I Destructive Test Program on an Aluminum, Plate-Type, Water-moderated Reactor," IDO-16883 (June 1964).

7. M. R. Zeisaler, "Non-Destructive and Destructive Transient Tests of the SPERT I-D, Fully Enriched, Aluminum-Plate-Type Core: Data Summary Report," IDO-16886 (November 27, 1963).

8. S. G. Forbes et al., "Instability in the SPERT I Reactor Preliminary Report," IDO-16309 (October 10, 1956).

9. M. C. Edlund and L. C. Noderer,"Analysis of Borax Experiments," Oak Ridge National Laboratory report CF-57-7-92 (July 19, 1957).

10. "Hazards Evaluation Report, Curtiss-Wright Research Reactor," Curtiss—Wright Corporation, Nuclear Power Department, Research Reactor and Physics Division report CWR-400-2 (undated).

"REVISED"