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

LICENSE RENEWAL APPLICATION
SAFETY ANALYSIS REPORT
VOLUME 1
CHAPTERS 1 - 9

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University of Missouri, Columbia

Missouri University Research Reactor (MURR)

Safety Analysis Report

MU Project # 000763

August 18, 2006

For the:

Curators of the University of Missouri

Preface

This Safety Analysis Report of the Missouri University Research Reactor (MURR) describes the facility in detail and includes modifications performed to the facility as of August 18, 2006. This document is being submitted to the U.S. Nuclear Regulatory Commission (NRC) in support of an application request for a twenty-year renewal of the Class 104c Amended Facility License No. R-103 (NRC Docket No. 50-186).

The format and content of this document follows the guidance of NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors." Efforts were directed on ensuring that each chapter contained the technical content and necessary information requested by NUREG-1537, including the updating of regional demographic, hydrological, seismological, and meteorological data, and performing accident analyses using newer, updated computer codes and methodologies.

The preparation of this document represents the collective cooperation, support and efforts of many individuals, including members of the facility staff and external organizations. The following is a list of personnel that were directly involved in the preparation of this report. Although some individuals were asked to contribute more time and resources than others, each played a key role, and this document could not have been completed without them.

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(Detailed contents are given at the beginning of each chapter.)

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1.1 Glasstone, S. and Sesonske, A., "Nuclear Reactor Engineering," New York: Van Nostrand Reinhold Company, 1967.

1.0 THE FACILITY

1.1 Introduction

The submittal of this Safety Analysis Report (SAR) by The Curators of the University of Missouri supports an application to the U.S. Nuclear Regulatory Commission (NRC) for authorization to continue operation of the Missouri University Research Reactor (MURR). The MURR is a pressurized, reflected, open pool-type, light water moderated and cooled, heterogenous system designed for operation at a maximum steady-state power level of 10 Megawatts thermal [MW]. The application is for a twenty-year renewal of the Class 104c Amended Facility License No. R-103 (NRC Docket No. 50-186).

The MURR is owned by The Curators of the University of Missouri and is operated by a uniquely qualified staff who promote basic and applied research in neutron-related science and engineering, and provides an educational opportunity for students in these fields. MURR also provides radiation and isotope production services for public and private consumers.

This document only addresses the safety issues associated with the operation of the MURR reflecting the as-built condition of the reactor facility, and includes the experience observed in the operation and performance of the reactor systems. Accident scenarios are analyzed in Chapter 13, Accident Analyses.

1.1.1 Location of the Facility

The MURR Facility is situated on a 7.5-acre (3.0-hectare) lot in the central portion of the University Research Park, an 84-acre (34.0-hectare) tract of land approximately one mile (1.6 km) southwest of the MU main campus. The campus is situated in the southern portion of Columbia; a city with a current population of approximately 91,885 people (based on 2000 U.S. Census Bureau statistics and adjusted by an annual population growth rate of 1.4%). Columbia is the county seat and largest city in Boone County, Missouri.

The University Research Park consists of low occupancy research buildings. Personnel in facilities located within 1,500 feet (457.2 m) can be rapidly evacuated, if required. Therefore, the University Research Park is adequately suited for the location of the MURR.

1.1.2 Purpose of the Facility

The MURR is a multi-disciplinary research and education facility providing a broad range of analytical, radiographic, and irradiation services to the research community and the commercial sector. Scientific programs include research in archaeometry, epidemiology, health physics, human and animal nutrition, nuclear medicine, radiation effects, radioisotope studies, radiotherapy, and nuclear engineering; and research techniques including neutron activation analysis, neutron and gamma-ray scattering, and neutron interferometry. MURR staff generate and nurture extensive collaborations with outside researchers. Research groups are made up of both MURR staff and researchers from the University of Missouri System's four campuses (primarily from the Columbia campus). The breadth and quality of the research programs and the available facilities and equipment are comparable to those found in the U.S. National Laboratories.

The MURR has six types of experimental facilities designed to provide these services: the Center Test Hole (Flux Trap); the Pneumatic Tube System; the Graphite Reflector Region; the Bulk Pool Area; the (six) Beamports; and the Thermal Column. The first four types provide areas for the placement of sample holders or carriers in different regions of the reactor core assembly for the purpose of material irradiation. Some of the material irradiation services include transmutation doping of silicon, isotope production for the development of radiopharmaceuticals and other life-science research, and neutron activation analysis. The six beamports channel neutron radiation from the reactor core to experimental equipment which is used primarily to determine the structure of solids and liquids through neutron scattering. The graphite thermal column is designed for the purpose of performing neutron radiographs and large sample irradiations.

The MURR also participates in a U.S. Department of Energy (DOE) program to provide the availability of university reactor facilities to non-reactor-owning colleges and universities. The MURR also provides support to institutions with reactors, but which operate at power levels too low to adequately perform required experiments. Reactor sharing projects include work in fields such as anthropology, archaeology, animal science, analytical epidemiology-nutrition, crystallography, geology, materials science, physics, nuclear analysis development, and biochemistry.

The MURR represents a major research and education resource for the University, the State of Missouri, and the nation. It is a facility which enhances the international reputation of the University, and provides the catalyst for MU to be the leader in the education of future generations of nuclear scientists and engineers.

The MURR experimental facilities and program are described in greater detail in Chapter 10, Experimental Facilities and Utilization.

1.2 General Description of the Facility

1.2.1 Introduction

The MURR is a heterogenous, pressurized, reflected, open pool-type, which is light water moderated and cooled. It accommodates an experimental position (flux trap) through the center of the core which is external to the reactor pressure vessel. A flux trap-type reactor is characterized by a thin fuel region adjacent to a good moderator which thermalizes the neutrons and causes the thermal flux to peak in a region (center test hole) accessible for experiments. It also provides relatively high beam tube currents resulting from the high power density. The reactor is designed to operate at a maximum thermal power of 10 MW with forced cooling, or up to 50 kW in the natural-convection mode.

The reactor is located eccentrically within a cylindrically-shaped, aluminum-lined pool approximately 10 feet (3.0 m) in diameter and 30 feet (9.1 m) deep. The reactor fuel is covered by approximately 23½ feet (7.2 m) of shielding water during reactor operation. The pool liner is surrounded by and anchored to a reinforced concrete edifice (biological shield). The biological shield is a massive bulk shield structure varying in thickness from 3 feet (0.9 m) to 7½ feet (2.3 m), with the smaller dimension at the top. The reactor pool allows direct visual and mechanical access to the reactor core and its assembly, thereby facilitating experimental use, inspection, maintenance, and fuel handling.

The reactor pool, control room, and a limited number of offices are enclosed in a multistory, reinforced concrete containment building with inside dimensions of approximately 67 feet (20.4 m) by 62 feet (18.9 m) by 64½ feet (19.7 m) high, and with a space extending to the north at below grade level that is 15 feet (4.6 m) high by 37 feet (11.3 m) deep by 40 feet (12.2 m) wide. Completely encircling the reactor containment building and extending one story above grade level is a structure containing laboratories and supporting facilities. Surrounding the reactor containment building on three sides below grade level is an excavated area which contains process equipment, demineralizer columns for the primary and pool coolant systems, and radioactive liquid waste retention and disposal facilities.

The facility ventilation supply and exhaust systems maintain the reactor containment building at a slightly negative pressure with respect to the surrounding laboratory building to prevent the spread of radioactive contamination. These systems also maintain concentrations of radioactive gases in the laboratory and containment buildings at levels which are below the Title 10, Chapter I of the Code of Federal Regulations, Part 20 (10 CFR 20) limits for restricted areas, and they also ensure that minimum concentrations of radioactive gases are released to the unrestricted environment by providing maximum dilution of potentially contaminated air.

Normal electrical power and domestic cold water are supplied to the MURR from the campus' electrical power and water supply systems. The design of the reactor does not require electrical power or domestic cold water to safely shut down or to maintain an acceptable shutdown condition. The MURR contains the compressed air, demineralized water, and sewer utilities required for operation. In addition, the reactor facility has communication systems, a closed circuit television system, a fire protection system, radiation monitoring systems, and a radioactive liquid waste retention and disposal system.

The reactor containment and laboratory buildings are shown in Figures 1.1 through 1.9. Figures 1.1 through 1.5 show floor plans of all five levels of the reactor facility.

1.2.2 Reactor

The reactor core consists of eight identical fuel assemblies, each occupying a 45° segment of a cylindrical annulus. Each fuel assembly comprises 24 circumferential plates containing uranium enriched to approximately 93% in the isotope ^{235}U as the fuel material. The active fuel length is

[redacted]

The reflector region consists of two concentric right circular annuluses surrounding the reactor pressure vessel. The inner reflector annulus is a 2.71-inch (6.9-cm) thick solid ring of beryllium metal. The outer reflector annulus consists of vertical elements of graphite canned in aluminum, having a total thickness of 8.89 inches (22.6 cm). The graphite elements are designed to accept the source ends of the six beamports.

The reactivity of the reactor is controlled by five control rods in the form of curved blades that travel vertically in the space between the reactor pressure vessel and the beryllium reflector. The blades are positioned by drive mechanisms mounted on the upper bridge over the reactor pool. Four of the control blades serve as shim rods (blades), each occupying 72° of the arc around the pressure vessel. The fifth control blade functions as a regulating rod (blade), occupying approximately 18° of the arc.

The reactor utilizes three coolant systems: primary, pool, and secondary. The reactor core is cooled by the circulation of pressurized primary coolant through two heat exchangers located in a mechanical equipment room (Room 114) external to the reactor containment building. Cooling of the beryllium and graphite reflectors, the control blades, and the center test hole (flux trap) is accomplished by drawing pool coolant down through these regions to a heat exchanger also located in Room 114. The heat from the primary and pool coolant systems is then transferred from the heat exchangers to the secondary coolant system. This heat is then dissipated to the atmosphere through a cooling tower.

There are two demineralizer loops associated with the reactor: one serving the primary coolant system and one serving the pool coolant system. The demineralizer loops serve to reduce the inventory of radioactive nuclides present in the primary and pool coolant, and also to maintain a primary grade level of water quality which limits corrosion to essential components. The reactor also has an emergency decay heat removal system which enhances decay heat removal from the core following an emergency shutdown accompanied by a loss of primary coolant flow.

The reactor control room is located on the third level of the reactor containment building at the elevation of the reactor pool surface and the operating (upper) bridge. The reactor control room contains the process and reactor instrumentation needed to monitor and control all aspects of reactor operation from a single location. Two major instrument cabinets provide these functions: the reactor control console and the instrument panel. The reactor control console provides the necessary instrumentation and controls required for control rod movements, including both manual and automatic control options. All process equipment is operated from the instrument panel.

Selected operating information is displayed or recorded on the reactor control console and the instrument panel. Recorded operating information provides historical operating data which can

[redacted]

FIGURE 1.1
BELOW GRADE LEVEL PLAN (BASEMENT)

[redacted]

FIGURE 1.2
GRADE LEVEL PLAN

[redacted]

FIGURE 1.3
THIRD LEVEL PLAN

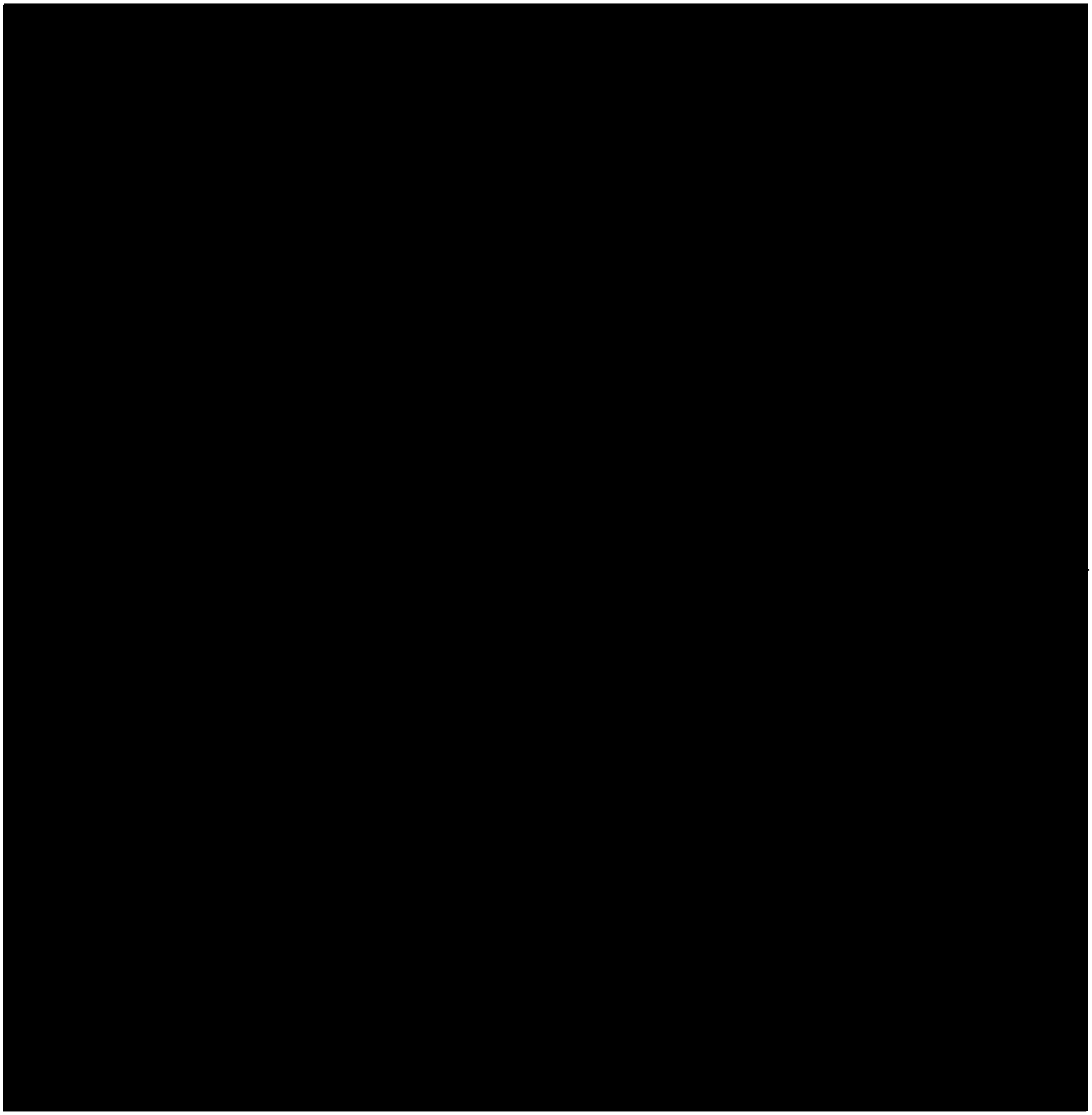


FIGURE 1.4
FOURTH LEVEL PLAN

[redacted]

FIGURE 1.5
FIFTH LEVEL PLAN

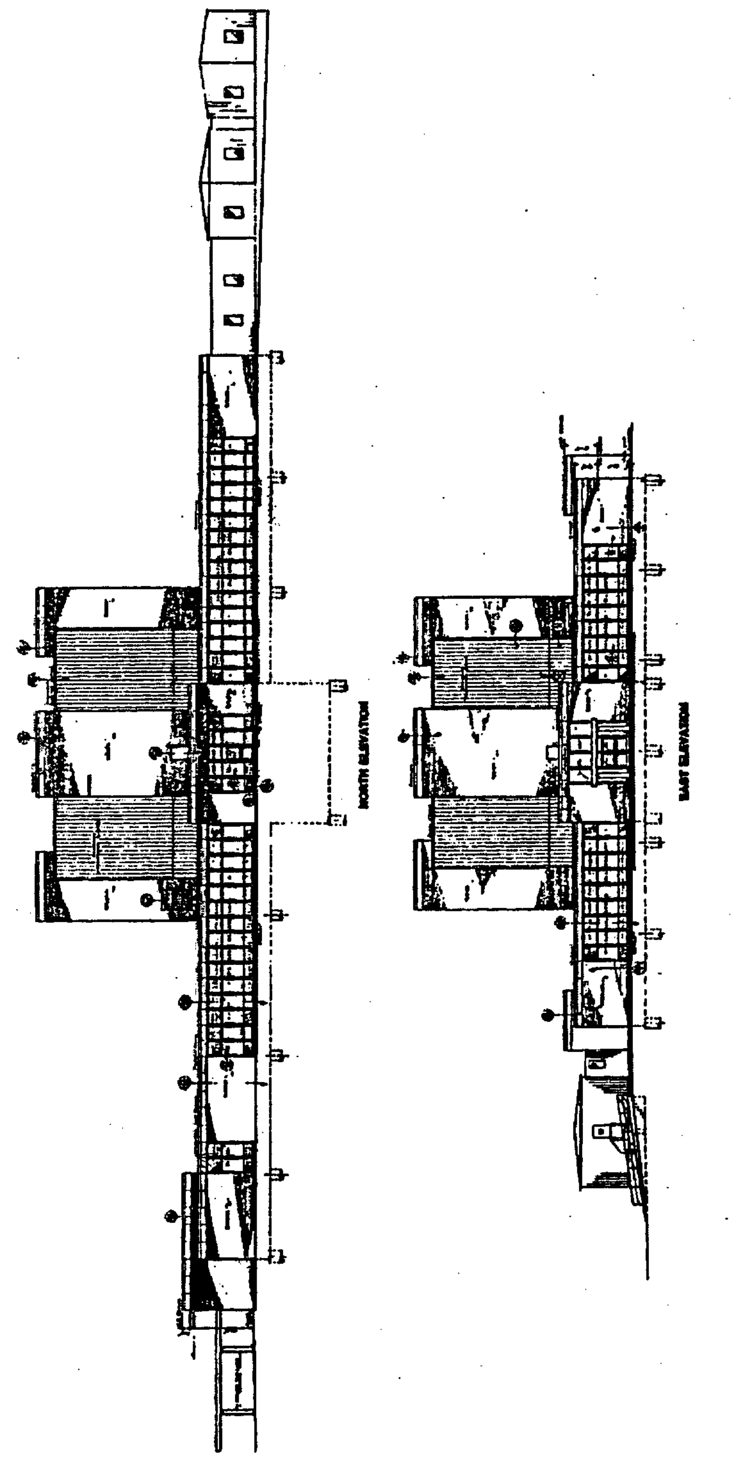
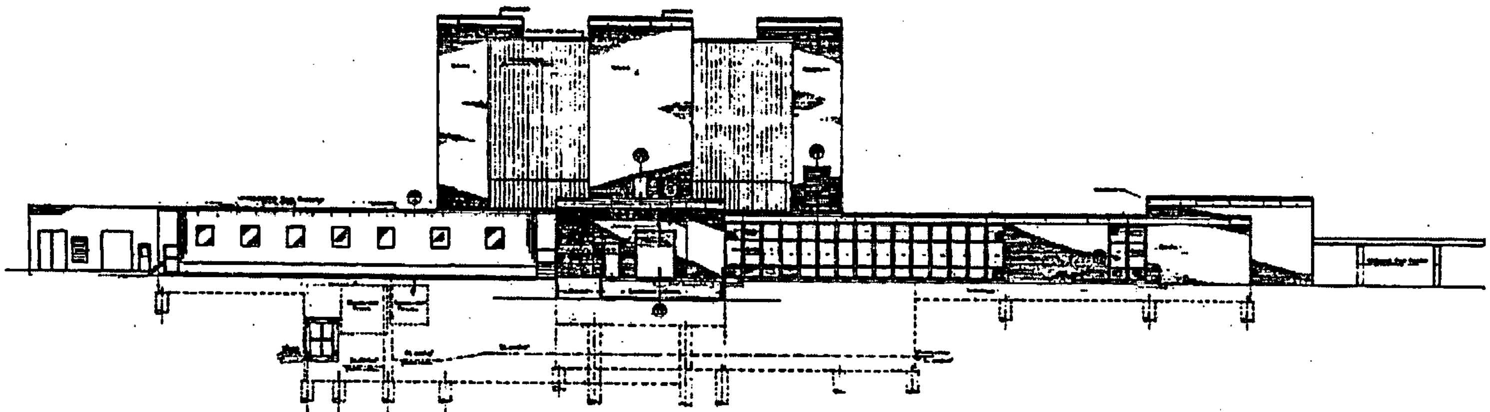
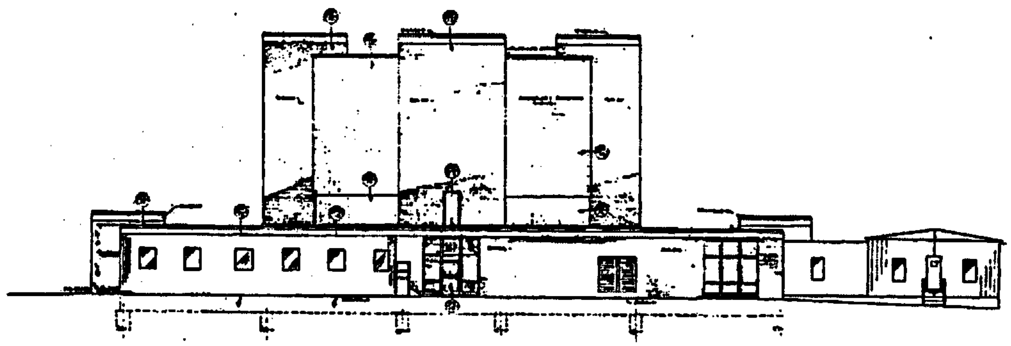


FIGURE 1.6
NORTH AND EAST ELEVATIONS



SOUTH ELEVATION



WEST ELEVATION

FIGURE 1.7
SOUTH AND WEST ELEVATIONS

[redacted]

FIGURE 1.8
BUILDING SECTIONS A-A AND B-B

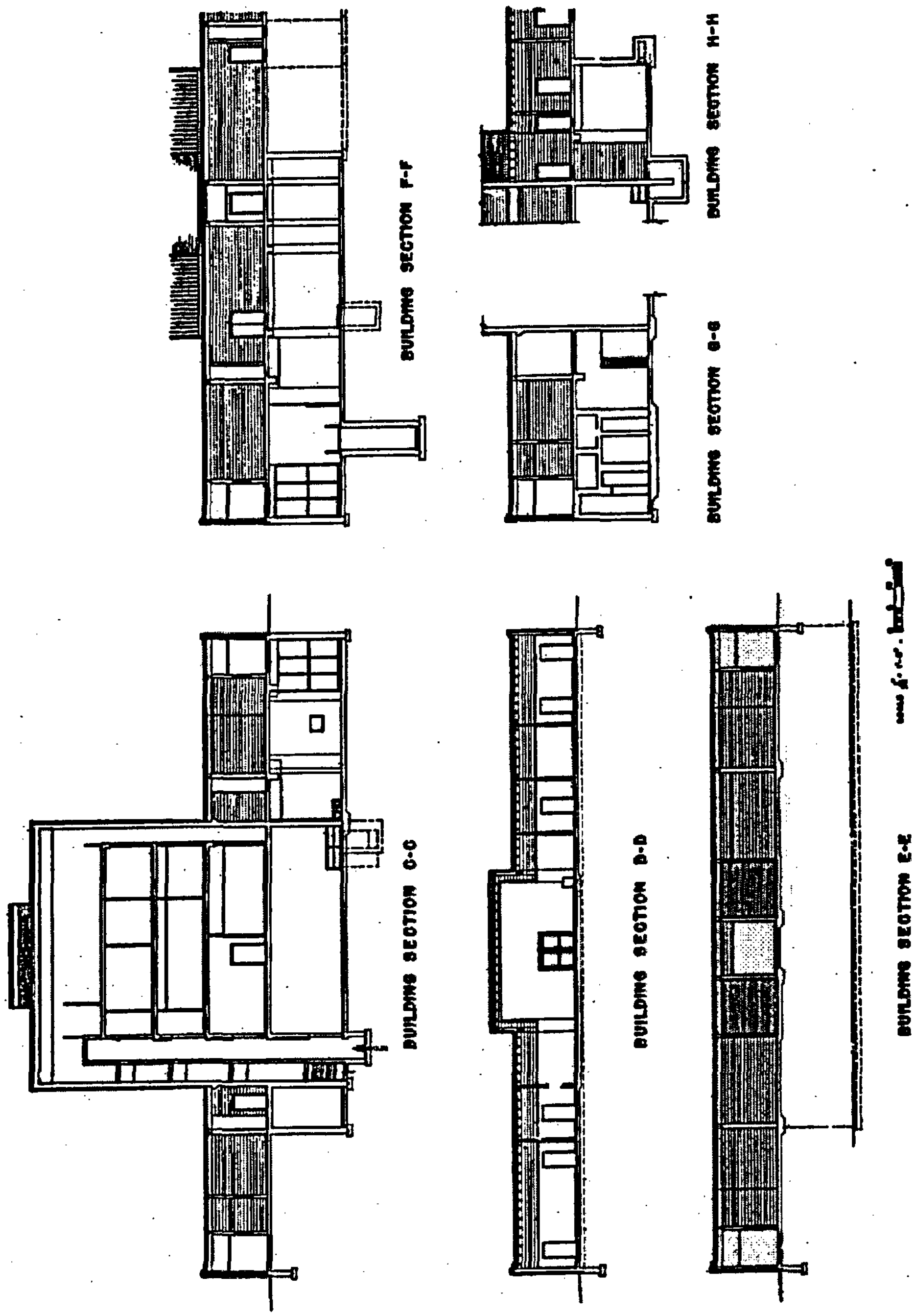


FIGURE 1.9
BUILDING SECTIONS C-C, D-D, E-E, F-F, G-G, AND H-H

be referenced at a later time. Some of this operating information includes reactor power level, and primary and pool coolant system temperatures and flows. Table 1-1 is a summary of the principal design parameters for the MURR.

TABLE 1-1
SUMMARY OF REACTOR PARAMETERS

Power	
Initial Thermal Power	5 MW
Maximum Thermal Power	10 MW
Average Power Density at 5 MW	151 kW/liter
Average Power Density at 10 MW	303 kW/liter
Average Thermal Neutron Flux (n/cm^2 - sec)	
In Flux Trap at 5 MW	3.1×10^{14}
In Flux Trap at 10 MW	6.2×10^{14}
In Reactor Core at 5 MW	2.3×10^{13}
In Reactor Core at 10 MW	4.6×10^{13}
Reactor Core	
Geometry	Annular
Innermost Fuel Plate Center Radius	2.795 inches (7.099 cm)
Outermost Fuel Plate Center Radius	5.785 inches (14.694 cm)
[redacted]	
Cladding of Fuel Assemblies	Aluminum
Number of Fuel Assemblies	8
Fuel Plates Per Fuel Assembly	24
Cooling Gap Thickness	0.080 inches (2.032 mm)
Fuel Plate Thickness	0.050 inches (1.270 mm)
Fuel Material Thickness	0.020 inches (0.508 mm)
Cladding Thickness	0.015 inches (0.381 mm)

TABLE 1-1
SUMMARY OF REACTOR PARAMETERS (cont)

Primary Coolant	
Total Flow Rate at 5 MW	1,875 gpm (7,098 lpm)
Total Flow Rate at 10 MW	3,750 gpm (14,195 lpm)
Coolant Pressure at 5 MW	75 psia (0.52 MPa)
Coolant Pressure at 10 MW	85 psia (0.59 MPa)
Maximum Design Pressure	125 psig (0.86 MPa)
Reactor Core Inlet Temperature	120 °F (48.9 °C)
Reactor Core Outlet Temperature	137 °F (58.3 °C)
Demineralizer Flow Rate	50 gpm (189 lpm)
Pool Coolant	
Total Flow Rate at 5 MW	600 gpm (2,271 lpm)
Total Flow Rate at 10 MW	1,100 gpm (4,164 lpm)
Mixed (Bulk) Pool Temperature	100 °F (37.8 °C)
Pool Inlet Temperature	99 °F (37.2 °C)
Pool Outlet Temperature	105 °F (40.6 °C)
Demineralizer Flow Rate	50 gpm (189 lpm)
Assorted Characteristics	
Measured Reactivity Requirements at 5 MW	0.0555 Δk/k
Measured Rod Worth (four shim blades)	0.1655 Δk
Reactor Core Lifetime (UAL_x)	1,200 MWD
^{235}U Consumed	1.51 Kg
Experimental Facilities	No.
Center Test Hole (Flux Trap)	1
6.0-inch I.D. Beamport	3
4.0-inch I.D. Beamport	3
Pneumatic Tube System (Transfer Terminals)	Up to 4
Graphite Thermal Column	1
Graphite Reflector Irradiation Positions	15 - Typical
Bulk Pool Irradiation Positions	3 - Typical

TABLE 1-1
SUMMARY OF REACTOR PARAMETERS (cont)

Reflector	
Inner Reflector Material	Beryllium
Inner Reflector Thickness	2.71 inches (6.88 cm)
Inner Reflector Height	37.0 inches (93.98 cm)
Outer Reflector Material	Graphite
Outer Reflector Thickness	8.89 inches (22.58 cm)
Outer Reflector Height	36.0 inches (91.44 cm)
Coolant	Pool Water
Control Blades	
Location	Outside Vessel
Type	Curved Plate
Material	Boral
Cladding	Aluminum
Blade Thickness	0.250 inches (6.35 mm)
Shim	4
Regulating	1
Coolant	Pool Water
Flux Trap	
Inner Diameter	3.3125 inches (8.41 cm)
Coolant	Pool Water
Water Island	
Thickness	5 inches (12.70 cm)
Coolant	Pool Water

1.2.3 Reactor Containment Building

The reactor and biological shield are housed in a five-level poured concrete building which extends one level below grade. This structure also contains the reactor control room, office spaces, a 15-ton overhead rectilinear crane, and a 2-ton capacity passenger elevator which services four levels. Access to the reactor containment building is available through a personnel airlock at grade level or a heavy equipment entry door at the below-grade level.

The reactor containment building is designed to withstand a peak internal pressure of 2.0 psig (13.8 kPa above atmosphere) which is in excess of the pressure predicted for any potential accident. Utility lines are brought in through a water-filled trap designed to maintain a positive seal against a 2.0-psig (13.8-kPa above atmosphere) internal pressure differential and a 1.7-psig (11.7-kPa above atmosphere) external pressure differential. If either pressure differential is momentarily exceeded, water level within the trap will elevate and create a path to equalize pressure. The water trap will re-seal the containment building once the pressure transient has terminated.

In order to minimize potential leakage points from the containment building, penetrations through the concrete structure are limited. In addition to the pedestrian and the heavy equipment entries previously mentioned, the penetrations include supply and exhaust air ducts, and two steel penetration plates which allow electrical lines and the pneumatic tube system to enter and exit the containment building. The pedestrian and the heavy equipment entries, and the supply and exhaust ducts, include sealable closures which ensure that sufficient integrity can be maintained to prevent the accidental release of radioactivity to the environment.

Although space is provided within the reactor containment building for both neutron beam and in-core irradiation experiments, the additional laboratories and offices required to support the experiment program are located in the laboratory building.

1.2.4 Laboratory Building

The laboratory building is a one-level 33,184-ft^2 (3,083-m^2) rectangular-shaped building of poured concrete and block and brick construction which completely surrounds, and is an integral part of, the reactor containment building. The laboratory building, as shown in Figure 1.2, contains electronic and machinery shops, offices, conference rooms, laboratories, and auxiliary electrical and mechanical equipment which support the operation of the reactor and the surrounding facility.

Surrounding the reactor containment building on three sides below grade level is an excavated area with approximately 9,000 ft^2 (836 m^2) of floor space. This area contains a 15-ton capacity hydraulic freight elevator and a hot cell which is used to open and seal material irradiation canisters. This level also includes shielded spaces which contain reactor process equipment, demineralizer columns for the primary and pool coolant systems, and the radioactive liquid waste retention and disposal system.

Five temporary office buildings (TOBs) and an industrial building [6,257 ft^2 (581 m^2)] provide additional office and laboratory space for the facility. Four of the TOBs and the industrial building are attached to the west side of the laboratory building by connecting corridors and the other TOB is attached to the south side.

1.3 Comparison With Similar Facilities

The uranium-aluminide dispersion UAl_x fuel system currently used at the MURR was developed at the Idaho National Engineering Laboratory (INEL) in southwest Idaho. The UAl_x dispersion fuel system was used at the Materials Test Reactor (1952-1970) and the Engineering Test Reactor (1957-1982) prior to its initial use at the MURR in August 1971 and continues to be used in the Advanced Test Reactor (1967-present), a high flux, high power test reactor located at the Test Reactor Area (TRA). The excellent performance of UAl_x dispersion fuels has been demonstrated for over the past thirty-five years at the MURR and the Advanced Test Reactor where over 3,950 fuel elements made of this fuel material have been used.

1.4 Summary and Conclusions on Principal Safety Considerations

1.4.1 Introduction

The analyses presented in this SAR demonstrate that the MURR can be operated safely, and that it will not constitute an undue hazard or risk to the health and safety of the reactor facility staff or the general public. With nearly forty years of established operating experience, the MURR has earned an enviable record of safe and reliable operation. This section summarizes the design features, inherent or passive safety features, and the engineered safety systems which promote safe operation and shutdown of the reactor and which prevent or mitigate the consequences of certain identifiable accidents.

In order to confirm the inherent stability and safety of the MURR, this document will show that, based on past operating experience, the reactor systems are of proven design, and that reactor safeguards were observed and implemented in the design, construction, and operation of the reactor. Reactor safeguards are steps which are taken to ensure, as far as reasonably possible, that the reactor can be operated safely and without risk to the health and safety of the staff or the general public (Ref. 1.1). The four principal elements of reactor safeguards include:

(a) Site Selection:
Selection of a site such that, in normal operation, the radioactivity of effluents released to the environment will not result in the contamination of air or water to levels exceeding the limits of 10 CFR 20, Appendix B, Table 2;

(b) Engineering Design:
Detailed engineering design of all parts of the reactor, and of the overall system, to minimize the possibility of accidents either as a result of human or mechanical failure, and

to deal with the consequences should an accident occur during either normal or abnormal operation; the reactor design will normally include inherent safety features as well as mechanical safeguards;

(c) Confinement or Containment:
Confining or containing radioactivity to prevent the uncontrolled release to surrounding areas if an accident involving the release of fission products should occur; and

(d) Operating Procedures:
Written operating procedures providing methods and detailed guidance in the operation and utilization of the reactor and associated systems to ensure safety and performance within the limits of the Technical Specifications; an Emergency Plan containing a detailed description of the elements of advanced planning to contend with emergency situations connected with the operation of the reactor facility, with this Plan focusing primarily on situations that may cause or may threaten to cause radiological hazards affecting the health and safety of reactor facility staff or the general public.

Chapter 2, Site Characteristics, contains a detailed discussion on the features and the characteristics of the site selected for the MURR. The operating procedures, both normal and emergency, are discussed in Chapter 12, Conduct of Operations. The principal safety features of the reactor and the reactor containment building and its support structure (laboratory building) are summarized below.

1.4.2 Reactor

The MURR is a heterogenous, pressurized, reflected, open pool-type reactor, which is light water moderated and cooled. As such, it has many of the inherent safety features characteristic of a heterogenous, water-moderated, small compact core reactor: a negative void coefficient and a negative temperature coefficient of reactivity.

One type of fuel assembly is designed for use in the reactor core. This design has proven to be reliable with more than 35 years of continuous operation at the MURR, and at other test and research reactors, with no significant failures. The average power density is 303 kW/liter and the average heat flux is 1.72×10^5 BTU/ft^2-h. The maximum fuel surface temperature calculated at the hot spot is sufficiently low so that nucleate boiling is not expected under normal conditions. These power density and heat flux conditions are very mild for this type of fuel, therefore, the likelihood of any significant thermal and/or hydraulic problems occurring are very small.

The drop times of each of the four shim blades are measured at intervals as specified in the Technical Specifications. Insertion of the shim blades to the 20% withdrawn position in less than 0.7 seconds ensures that the reactor will be promptly shut down when required. In the nearly 40 years of performing this surveillance, to date, the shim blades have never failed to meet this specification.

The control rod drives are electrically-driven ball screw mechanisms of the same type used on a number of research reactors designed by General Electric.

The reactor is located near the botto[redacted] of an aluminum-lined pool which is surrounded by and anchored to a reinforced concrete edifice (biological shield). The purpose of the biological shield is to provide radiation shielding to keep exposures to personnel working in and around the reactor facility as low as is reasonably achievable (ALARA). In addition, the immense size of the biological shield also provides excellent protection against natural phenomena that could result in damage to the reactor core assembly.

The beryllium reflector effectively decouples the reactor core from experiment variations in the beamports, bulk pool and graphite reflector irradiation positions, the pneumatic tube system, and the thermal column. The only experimental facility not decoupled from the reactor in this manner is the center test hole (flux trap). Also, these experimental facilities are similar to many other systems that have been used throughout the United States. They have well-established operating experience with no new significant design alterations required.

The abnormal conditions or postulated accident events or categories analyzed in this SAR include the following:

(1) Maximum Hypothetical Accident (MHA);
(2) Insertion of Excess Reactivity;
(3) Loss of Primary Coolant;
(4) Loss of Primary Coolant Flow;
(5) Mishandling or Malfunction of Fuel;
(6) Experiment Malfunction;
(7) Loss of Electrical Power;
(8) External Events; and
(9) Mishandling or Malfunction of Equipment.

The MHA is an accident condition which postulates the melting of four fuel plates in the reactor core and the subsequent release of fission products to the primary coolant system. This accident is selected to postulate conditions which lead to consequences worse than those resulting from any other credible accident. The circumstances that lead to the MHA are immaterial to the analysis. The MHA for the MURR is consistent with the assumed MHA at other similar research reactor facilities.

In all of the other proposed accident events or categories listed above, fuel and cladding temperatures remain at levels below those required to produce cladding failure, and thus, no release of fission products would occur. Chapter 13, Accident Analyses contains a detailed discussion of each accident scenario listed above.

The anti-siphon system, an engineered safety feature, functions as a backup system to the various safety instrumentation and equipment (e.g., pressure sensors, pump and valve interlocks, etc.). The anti-siphon system ensures that the reactor core does not become uncovered during a loss of primary coolant accident. Redundancy is incorporated into the system to ensure no single component or circuit failure will render any portion of the anti-siphon system inoperative.

The MURR has an emergency decay heat removal system capable of removing decay heat from the reactor core following an emergency shutdown accompanied by a primary loop isolation, or, in the event of a loss of primary coolant flow.

The effects of argon-41, the principal radionuclide in the gaseous effluents (approximately 99% of the total) released through the facility exhaust stack during normal operation of the reactor, have been evaluated. Using the most probable wind direction, the annual exposure from this isotope was determined at two different distances: 150 meters north to the emergency planning zone (EPZ) boundary, and at the nearest residence in relation to the facility (760 meters north). The maximum average annual dose at these two locations was calculated at 0.7 mrem/y and 4.2 mrem/y, respectively. The difference in relative plume height at these sites is what leads to this difference in dose rates. The release to the atmosphere results in a maximum downwind concentration below the limits of 10 CFR 20.

1.4.3 Reactor Containment and Laboratory Buildings

The reactor is housed in a building specifically designed for operation of the reactor. The reactor containment building is a five-level poured concrete structure with 12-inch thick reinforced exterior walls. Centered on the outside of each wall are two pilaster support columns closed at the back to form a tower. The pilaster support columns, comprising the side walls of the towers, are built into the main building structure to achieve the requisite structural strength to withstand an internal pressure of 2.0 psig (13.8 kPa above atmosphere). This design value is in excess of the pressure predicted for any potential accident. In addition, the massive concrete walls also provide protection to the reactor from natural phenomena. The containment system, an engineered safety feature, is designed to completely isolate the reactor containment building to prevent or mitigate an uncontrolled release of radioactive materials to the environment during an accident. Redundancy is incorporated into the system to ensure that no single component or circuit failure will render any portion of the containment system inoperative. The design maximum leakage rate of 10% of the contained volume [over a 24-hour period at 2.0 psig (13.8 kPa above atmosphere)] of the containment building, as verified by periodic surveillance testing, ensures that this objective is met. Thus, the reactor containment building is an adequately suited structure for housing the reactor.

The facility ventilation supply and exhaust systems maintain the reactor containment building at a slightly negative pressure with respect to the surrounding laboratory building to prevent the spread of radioactive contamination. These systems also maintain concentrations of radioactive gases in the laboratory and containment buildings at levels which are below the 10 CFR 20 limits for restricted areas. Air from the laboratory building is directed through stainless steel filter housings, each containing a bank of pre-filters and a bank of high efficiency particulate air (HEPA) filters, prior to being discharged to the atmosphere through the ventilation exhaust stack. Air from the mechanical equipment room (Room 114) passes through a pre-filter, a HEPA filter, and an activated charcoal filter.

To ensure that all personnel working in the reactor facility are cognizant of the operating status of the reactor, "REACTOR ON" lights are installed throughout the reactor containment building and at the entrance to the personnel airlock.

All spent fuel shipping cask lifting equipment, including the 15-ton capacity overhead rectilinear crane, is rigorously maintained, including preventive maintenance and magnetic particle testing (MT) as appropriate. A dye penetrant inspection is also performed on the shipping cask. Therefore, the dropping of a spent fuel shipping cask during fuel transfers is considered a highly unlikely event.

Area radiation monitoring equipment has been installed at strategic locations in the reactor facility to monitor radiation levels. Each radiation meter is equipped with an adjustable setpoint trip that initiates an audible and visual alarm upon detection of a high radiation level.

1.5 Summary of Operations

Since September 1, 1977, the MURR has had an operating schedule of 24 hours a day, 6½ days per week (averaging approximately 150 hours per week). The weekly half-day shutdown is required for refueling the reactor, changing samples in the flux trap, and performing any corrective or preventative maintenance. This operating schedule equates to operating at full power (10 MW) for 90% of all available hours in the year. The MURR plans to continue operation on this schedule.

1.6 Compliance With the Nuclear Waste Policy Act of 1982

The Nuclear Waste Policy Act of 1982, Section 302(b)(1)(B) states that the NRC may require, as a precondition to renewing an operating license for a research reactor under Section 104 of the Atomic Energy Act of 1954, as amended (the Act), that the applicant shall have entered into an agreement with the Department of Energy (DOE) for the disposal of high-level radioactive wastes and spent nuclear fuel. DOE provides fuel assistance for the MURR, by purchasing the fuel from the fuel fabricator, with MURR paying a portion of the cost to the DOE. The DOE has informed the NRC that universities and other government agencies operating non-power reactors have entered into contracts with the DOE which state that the DOE retains title to the fuel and is obligated to take the spent fuel for storage or reprocessing. The Curators of the University of Missouri have entered into such a contract with the DOE; hence the applicable requirements of the Nuclear Waste Policy Act of 1982 are satisfied.

1.7 Facility Modifications and History

1.7.1 Introduction

The MURR first achieved criticality on October 13, 1966. The reactor was originally designed for 10-MW operation, but was initially licensed to operate at only 5 MW until reactor utilization and operating experience were sufficient to justify full power operation. Prior to operation at 5 MW, an extensive low power testing and calibration program was performed to verify that the critical physics and plant parameters were in compliance with the facility operating license. In 1974, additional cooling equipment was added and the process instrumentation and safety systems were modified as required to facilitate operation of the reactor at the full design power of 10 MW.

Since achieving initial criticality, the reactor has operated safely for over 272,000 hours in support of the primary mission of providing the maximum flux, or current, of neutrons to the maximum number of users without endangering the health and safety of facility workers and the general public.

The following is a chronology of the significant events and their corresponding dates in the history of the MURR.

- February 3, 1959 — University of Missouri President Elmer Ellis announces the University will construct a research reactor.
- November 21, 1961 — Construction Permit No. CPRR-68 is issued to The Curators of The University of Missouri by the U.S. Atomic Energy Commission (AEC) authorizing the University to construct a 10-MW research reactor.
- July 16, 1963 — Initial on-site construction begins.
- October 11, 1966 — Facility License No. R-103 is issued to The Curators of The University of Missouri by the AEC authorizing the University to operate the 10-MW research reactor at steady-state power levels up to a maximum of 5 MW.
- October 13, 1966 — The reactor establishes initial criticality with a 5.2-kg uranium aluminum alloy fuel core.
- June 30, 1967 — The reactor operates at 5 MW for the first time.

- July 14, 1969 — The reactor starts a 5-MW, 100-hour-per-week operating schedule.

- August 13, 1971 — Use of the 6.2-kg uranium aluminide dispersion UAl_x fuel core begins.

- July 9, 1974 — The AEC issues Amendment No. 2 to Facility License No. R-103 authorizing the University to operate the reactor at steady-state power levels up to a maximum of 10 MW.

- July 18, 1974 — The reactor is upgraded to 10 MW.

- September 1, 1977 — The reactor starts a 10-MW, 150-hour-per-week operating schedule.

1.7.2 Original Design and Construction

The original design, construction, construction supervision, and testing of the MURR Facility required major contributions from six contractors: two for design and construction, and four for construction and testing.

The reactor preliminary design study, final design, and design specifications were provided by the Internuclear Company of St. Louis, Missouri. The preliminary design study was the basis for the AEC Construction Permit (CPRR-68) which granted permission to The Curators of The University of Missouri to construct the research reactor facility at the location specified by the original license application. Subsequently, Internuclear completed all plans and specifications for the reactor core configuration, fuel assemblies, reactor pressure vessel, control elements, in-pool equipment and piping, cooling equipment, water treatment equipment, instrumentation and controls, experimental facilities, and the shield design. During reactor final design and construction, Internuclear reviewed all reactor shop drawings to ensure compliance with the design specifications. Internuclear also conducted "in-plant" or site inspections of installed equipment.

The design of the laboratory and reactor containment buildings was provided by C.L.T. Gabler and Associates of Detroit, Michigan. This architectural firm furnished the plans and specifications for the entire facility, including all structural, electrical, and heating and cooling. The reactor biological shield structural design was also furnished by this contractor.

Final engineering design, fabrication and the installation of the reactor system were accomplished by the General Electric Company, Atomic Power Equipment Department, San Jose, California. The B.D. Simon Construction Company of Columbia, Missouri was a subcontractor to General Electric to assist in the installation of the reactor system.

The general building construction and installation of equipment were also provided by B.D. Simon Construction Company. The mechanical work was provided by the Natkin Company of Kansas City, Missouri. The electrical work was provided by C.J. Hervey Company of St. Louis, Missouri. Both companies were subcontractors to the B.D. Simon Company.

1.7.3 License Amendments

Amendments to the facility license must conform to the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the regulations of the NRC as stated in 10 CFR. Activities authorized by the issuance of any amendment are conducted in compliance with the regulations of the NRC and with reasonable assurance that the health and safety of facility personnel and the general public will not be endangered.

Prior to the issuance of an amendment, a Safety Evaluation (SE) is conducted by the Office of Nuclear Reactor Regulation to verify (1) that the amendment does not involve a significant increase in the probability or consequences of accidents previously evaluated, nor does it create the possibility of a new or different kind of accident from any accident previously evaluated, does not involve a significant reduction in a margin of safety, and does not involve a significant hazards consideration; (2) that there is reasonable assurance that the health and safety of facility personnel or the general public will not be endangered by the proposed activities; and (3) that such activities will be conducted in compliance with the NRC regulations, and the issuance of the amendment is not inimical to the common defense and security or the health and safety of facility personnel or the general public.

The following thirty-three amendments have been issued by the NRC to the Amended Facility License No. R-103 for the MURR Facility.

Amendment 1 - Date of Issuance: April 23, 1968
Amendment 1 authorized the MURR to increase the amount of contained ^{235}U that it can receive, possess, and use from 15 to 45 kilograms. This amendment ensured that a sufficient fuel reserve was maintained in storage for continuity of operation of the reactor at 5 MW.

Amendment 2 - Date of Issuance: July 9, 1974
Amendment 2 authorized the MURR to:

1. Operate at a steady-state power level up to 10 MW;
2. Receive, possess and use a 100-curie source of antimony-beryllium; and
3. Incorporate into revised Technical Specifications as Change No. 10.

This amendment authorized operation of the reactor at 10 MW, as it was originally designed and constructed, after additional heat exchanger equipment was installed.

Amendment 3 - Date of Issuance: February 14, 1975
Amendment 3 authorized the MURR to incorporate into revised Technical Specifications as Change No. 11 which permitted the MURR to operate with increased fuel burnup. The

amendment also increased the allowable surveillance interval for uranium aluminide dispersion UAl_x fuel elements.

Amendment 4 - Date of Issuance: September 7, 1976

Amendment 4 authorized the MURR to:

1. Possess up to 20 grams of ^{239}Pu as neutron filters used in conjunction with the operation of neutron spectrometers;
2. Permit the use of up to 60 kilograms of source material under Facility Operating License R-103 rather than under the existing Source Material License No. SUD-287; and
3. Revise certain portions of the Technical Specifications to correct editorial errors and incorporate changes in the facility management structure.

Amendment 5 - Date of Issuance: May 18, 1977

Amendment 5 authorized the MURR to increase the possession limit of ^{235}U from 45 to 49 kilograms for an interim period from the effective date until January 1, 1978. This amendment allowed the transfer of new, unirradiated fuel elements from the U.S. Nuclear Corporation to the MURR prior to the shutdown of the Oak Ridge, Tennessee facility.

Amendment 6 - Date of Issuance: July 8, 1977

Amendment 6 authorized the MURR to increase the possession limit of unirradiated ^{235}U from 19 to 20.5 kilograms. This amendment accounted for the storage of three half-loaded alloy fuel elements which were used to provide a conservative method for initial criticality approach when new core configurations were established.

Amendment 7 - Date of Issuance: July 29, 1977

Amendment 7 authorized the MURR to change the fuel inspection intervals specified in the Technical Specifications. This amendment reduced the number of fuel inspections required to be performed to coincide with the increased operating schedule.

Amendment 8 - Date of Issuance: February 24, 1978

Amendment 8 authorized the MURR to change the limitations on fueled experiments specified in the Technical Specifications. This amendment allowed expansion of the experimental program by revising the restrictions placed on fueled experiments.

Amendment 9 - Date of Issuance: March 23, 1978

Amendment 9 authorized the MURR to incorporate control blade surveillance requirements into the Technical Specifications. This amendment formalized a facility practice of inspecting a control blade (shim blade) prior to use.

Amendment 10 - Date of Issuance: July 13, 1978

Amendment 10 authorized the MURR to perform the reactor containment building leakage rate test at 1.0 psig (6.9 kPa above atmosphere) using the makeup flow technique. This amendment lowered the maximum required test pressure from 2.0 psig (13.8 kPa above atmosphere) to

1.0 psig (6.9 kPa above atmosphere) to avoid unnecessarily stressing the containment structure or evacuating the seal trench which would invalidate the test results.

Amendment 11 - Date of Issuance: January 26, 1979

Amendment 11 authorized the MURR to revise its Security Plan.

Amendment 12 - Date of Issuance: July 5, 1979

Amendment 12 authorized the MURR to change the limitations on gaseous radioactivity release specified in the Technical Specifications. This amendment increased the Technical Specification limits for gaseous radioactivity release to the environment to support the increased operating schedule from between 90 and 100 hours per week to approximately 150 hours per week.

Amendment 13 - Date of Issuance: March 5, 1981

Amendment 13 authorized the MURR to change the facility organizational structure specified in the Technical Specifications to reflect the current administrative organization of the facility.

Amendment 14 - Date of Issuance: April 14, 1981

Amendment 14 authorized the MURR to change the definition of "Reactor Secured" specified in the Technical Specifications. This amendment allowed the dummy load test connectors to be installed or removed from the electrical control circuitry of the control rod drive mechanisms without being considered "work" on the control rods. This allowed greater flexibility in the scheduling and performing of maintenance, with a consequent increase in the efficiency in both manpower and reactor utilization.

Amendment 15 - Date of Issuance: August 12, 1983

Amendment 15 authorized the MURR to revise its security plan.

Amendment 16 - Date of Issuance: October 20, 1988

Amendment 16 authorized the MURR to:

1. Increase the amount of depleted uranium that may be received, possessed, and used from 20 to 50 kg; the depleted uranium is used in conjunction with filtered beam research for neutron capture therapy;
2. Receive, possess, and use up to 40 grams of plutonium in the form of a rod which is sealed in a stainless steel can; the sealed rod is used for studies with neutron diffraction of the crystal structure and magnetic phases of plutonium in the temperature range of 20 to 650 K;
3. Remove the requirement that the pool coolant circulation pumps operate in parallel; testing has shown that either pump will provide the minimum Technical Specifications for pool flow and differential pressure across the reactor reflector region;
4. Change the facility organizational structure specified in the Technical Specifications to reflect the current administrative organization of the facility; and
5. Correct a typographical error and replace references to the U.S. Atomic Energy Commission (AEC) with NRC in the Technical Specifications.

Amendment 17 - Date of Issuance: February 16, 1989
Amendment 17 authorized the MURR to correct an error which occurred in the requested change to the Technical Specifications that was issued as Amendment No. 16 on October 20, 1988.

Amendment 18 - Date of Issuance: May 8, 1989
Amendment 18 authorized the MURR to shift administrative authority for the MURR from the MU system to the MU at Columbia. This amendment allowed the MURR to be administered by the campus upon which it is located since decisions concerning personnel and budgets were made locally; this allowed the university to develop a close relationship between the campus and the MURR research programs.

Amendment 19 - Date of Issuance: June 6, 1990
Amendment 19 authorized the MURR to temporarily increase the possession limit for ^{235}U from 45 to 60 kilograms for an interim period from the effective date until May 31, 1991. The two spent fuel shipping casks with the capability to transport MURR spent fuel were removed from service for various reasons. This amendment allowed the MURR to retain spent fuel on site until a suitable shipping cask was found, thus maintaining uninterrupted reactor operation.

Amendment 20 - Date of Issuance: August 1, 1990
Amendment 20 authorized the MURR to use extended life aluminide fuel (ELAF) in the reactor core. This amendment allowed the MURR to use a new fuel element design which will significantly reduce the fuel cycle cost and reduce the amount of ^{235}U needed per Megawatt Day (MWD) of energy produced.

Amendment 21 - Date of Issuance: May 8, 1991
Amendment 21 authorized the MURR to temporarily increase the possession limit for ^{235}U from 60 to 75 kilograms for an interim period from the effective date until December 31, 1992. The 60-kilogram limit was a temporary limit authorized by Amendment No. 19. The two spent fuel shipping casks with the capability to transport MURR spent fuel were removed from service for various reasons. The task to find and become an authorized user of a new shipping cask was not completed by the May 31, 1991 expiration date. This amendment allowed the MURR to retain spent fuel on site until a suitable shipping cask was found, thus maintaining uninterrupted reactor operation.

Amendment 22 - Date of Issuance: August 17, 1992
Amendment 22 rescinded the temporary possession limit increases in special nuclear material granted by Amendment Nos. 19 and 21 and clarified the special nuclear material and byproduct possession clauses of the Amended Facility License. A spent fuel shipping cask was acquired by DOE and made available to the MURR. This amendment re-established the possession limit for ^{235}U at 45 kilograms.

Amendment 23 - Date of Issuance: May 17, 1993
Amendment 23 authorized the MURR to add to the Technical Specifications the requirement to have written procedures for the shipping of byproduct material produced under the Amended Facility License.

Amendment 24 - Date of Issuance: July 21, 1993
Amendment 24 authorized the MURR to temporarily increase the possession limit for ^{235}U from 45 to 60 kilograms for an interim period from the effective date until May 31, 1994. This amendment was required because the Department of Energy Savannah River Plant (SRP) was not able to accept spent fuel from the MURR on a schedule that would allow the MURR to continue operation without exceeding the current possession limit of 45 kilograms.

Amendment 25 - Date of Issuance: March 2, 1994
Amendment 25 authorized the MURR to change the due date of the facility annual report to the NRC. The revision of 10 CFR 20, "Standards for Protection Against Radiation," which became effective January 1, 1994, stated dose and effluent release limits in annual terms. This amendment allowed the annual report period to coincide with the regulations.

Amendment 26 - Date of Issuance: August 5, 1994
Amendment 26 authorized the MURR to temporarily increase the possession limit for ^{235}U from 45 to 60 kilograms for an interim period from the effective date until May 31, 1995. This amendment was required because the Department of Energy Savannah River Plant (SRP) was not able to accept spent fuel from the MURR on a schedule that would allow the MURR to continue operation without exceeding the current possession limit of 45 kilograms.

Amendment 27 - Date of Issuance: November 4, 1994
Amendment 27 authorized the MURR to install a single pool coolant system heat exchanger in place of the two existing heat exchangers. This amendment was required because the two existing heat exchangers had reached the end of their design operational lifetime. The two existing heat exchangers were replaced by a single heat exchanger of plate-type design.

Amendment 28 - Date of Issuance: March 15, 1995
Amendment 28 authorized the MURR to:

1. Increase the permanent ^{235}U possession limit from 45 to 60 kg; and
2. Extend the temporary increase in the ^{235}U possession limit to May 31, 1997, at an increased possession limit of 75 kg; this temporary authorization was due to expire on May 31, 1995.

This amendment was required because the Department of Energy Savannah River Plant (SRP) could not receive spent fuel from the MURR on a continuous basis to maintain the fuel inventory at the MURR below 45 kilograms.

Amendment 29 - Date of Issuance: March 6, 1997
Amendment 29 authorized the MURR to change the aspects of the administrative structure having oversight authority for the MURR and corrected an inconsistency in the Technical Specifications concerning meeting requirements of the Reactor Advisory Committee (RAC).

Amendment 30 - Date of Issuance: February 26, 1998
Amendment 30 authorized the MURR to clarify the Technical Specifications requirements for procedures for the preparation for shipment of byproduct material produced under the Amended Facility License.

Amendment 31 - Date of Issuance: September 20, 1999
Amendment 31 authorized the MURR to conduct movable and unsecured experiments in the center test hole of the reactor flux trap.

Amendment 32 - Date of Issuance: October 19, 2001
Amendment 32 changed the reactor operating license expiration date from November 21, 2001, to October 11, 2006. This amendment recaptures the construction time between the issuance date of Construction Permit No. CPRR-68 and the issuance of the operating license.

Amendment 33 - Date of Issuance: January 29, 2004
Amendment 33 revised the management organization, as per the Technical Specifications, for the MURR.

1.7.4 Reactor Systems Modifications

The following are the significant modifications made to reactor license-related systems since the construction of the reactor facility in 1966. Modifications to the reactor systems are performed, when necessary, to ensure continuity of operation of the reactor, maintain compliance with applicable regulations, and/or further enhance the capabilities of the experimental facilities. Safety Evaluations (SEs) are performed on all system modifications, providing the bases for the determination as to whether the modification does or does not involve a question as defined in Title 10, Chapter I of the Code of Federal Regulations, Part 50.59 (10 CFR 50.59).

- August 1967 - Installation of the Reactor Containment Building Ventilation Backup Doors

  Installation of the backup doors, in addition to the two electric-motor-driven doors located in the reactor containment building ventilation supply and exhaust plenums, incorporated further redundancy into the reactor containment system, thus ensuring compliance with the Criterion 54 of 10 CFR 50, Appendix A and the Institute of Electrical and Electronic Engineers (IEEE-279), Single Failure Criterion.

- January 1974 - Installation of Reactor Containment Building Exhaust Valve 16B

  Installation of Valve 16B in series with the existing reactor containment building exhaust Valve 16A, incorporated further redundancy into the reactor containment system, thus ensuring compliance with Criterion 54 of 10 CFR 50 Appendix A and IEEE-279, Single Failure Criterion.

- July 1974 - Installation of Additional Heat Exchanger Capability to Facilitate 10-MW Operation

  The following components were added to the primary, pool, and secondary coolant systems to facilitate the upgrade in power to 10 MW: primary coolant heat exchanger HX503B, pool coolant heat exchanger HX521B, primary coolant circulation pump P501B, pool coolant circulation pump P508B, and secondary coolant circulation pump SP-3. Piping, valves, and instrumentation were installed, or modified as necessary to support the addition of the heat exchangers and the circulation pumps.

- July 1974 - Modification of the Anti-Siphon System

  The anti-siphon system was modified such that any gaseous fission products that might result from the meltdown of four fuel plates (MHA) will be retained in a closed volume, thus reducing the exposure levels inside the reactor containment building.

- July 1974 - Installation of In-pool Heat Exchanger Isolation Valve 546B

  Installation of isolation valve 546B in parallel with the existing in-pool heat exchanger isolation valve 546A incorporated redundancy into the reactor decay heat removal system, thus ensuring compliance with IEEE-279, Single Failure Criterion.

- April 1990 - Installation of a 275-kW Emergency Power Diesel Generator

  A 344-kVA (275-kW at 0.8 Power Factor) diesel generator was installed to replace the existing emergency power generator (277/480-volt, 45-kW at 0.8 PF, driven by a Ford 292 industrial engine) due to a lack of reliability and a shortfall in the capacity of the emergency electrical power system to meet future demands. This modification allowed the October 18, 1990 Facility Ventilation Exhaust System upgrade to be accomplished.

- April 1990 - Installation of a 15-kVA Uninterruptible Power Supply

  A 15-kVA Uninterruptible Power Supply was installed to provide precise, regulated and transient-free 120-VAC electrical power to the reactor instrumentation and control system to reduce spurious (false) reactor scrams caused by voltage fluctuations or momentary interruptions in electrical power.

- October 1990 - Facility Ventilation Exhaust System Upgrade

  The laboratory building exhaust fans (EF-13 and EF-14) were replaced by two 36,500 SCFM fans, each driven by a 2-speed, 100-HP motor. The existing fans (16,500 SCFM each), due to an increase in the number of operable laboratory fume hoods, could no longer provide the required air flow (125 Linear Ft/Min) across each laboratory fume hood necessary to maintain the operational standards for a safe working environment.

- March 1995 - Replacement of the Two Pool Coolant System Heat Exchangers with a Single Heat Exchanger

  The two existing pool coolant system tube-and-shell-type heat exchangers (HX521A and HX521B) were replaced by a single heat exchanger (HX521) of plate-type design. This replacement was necessitated by the degradation of the heat transfer capability of the heat exchangers since they were approaching the end of their design operational lifetime.

- April 2000 - Installation of an Additional Stack Monitor to the Off-Gas Radiation Monitoring System

  In order to provide operational flexibility and increased reliability, an additional stack monitor was installed in parallel to the existing MURR stack monitor. The new stack monitor is functionally equivalent to the current monitor. The system is designed to measure the airborne concentrations of radioactive particulate, iodine, and noble gases exiting the facility through the ventilation exhaust stack while also providing audible and visual alarm functions on high activity or abnormal flow through the radiation detection equipment.

- June 2002 - Replacement and Upgrade of Substation "B" 1,000-kVA Transformer

  The facility's primary electrical 1,000-kVA transformer was replaced and upgraded to a 2,000-kVA transformer. This replacement was required since the transformer was approaching the end of its design operational lifetime. Additionally, the transformer was relocated outside the facility, thus removing a major heating load from inside the laboratory building.

- November 2004 - Installation of a Fire Detection/Suppression System

  A new fire detection/suppression system was installed in the facility. The new system, as the name implies, provides two functions: (1) a detection system that provides an early warning of an actual or potential fire condition by a combination of heat, smoke, and remote manual devices, and (2) a suppression system that incorporates a normal sprinkler system with a pre-action system used in areas with sensitive electronic equipment, and a deluge, non-freezing system used in the cooling tower.

- <u>February 2005 - Replacement and Upgrade of Substation "A" 500-kVA Transformer</u>

  The cooling tower's primary electrical 500-kVA transformer was replaced and upgraded to a 1,500-kVA transformer. This replacement was required since the transformer was approaching the end of its design operational lifetime. Additionally, the transformer was relocated outside the building, thus removing a major heating load from inside the cooling tower.

CHAPTER 2

SITE CHARACTERISTICS

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2.0 SITE CHARACTERISTICS

This chapter discusses and describes the geographical, geological, seismological, hydrological, and meteorological characteristics of the reactor facility site and vicinity in conjunction with present and projected population distributions, industrial facilities and land use, and site activities and controls.

The conclusion reached in this chapter and throughout the SAR is that the site is well suited for the location of the facility when considering the relatively benign operating characteristics of the reactor including the Maximum Hypothetical Accident (MHA).

2.1 Geography and Demography

2.1.1 Site Location and Description

The Missouri University Research Reactor (MURR) is situated on a 7.5-acre lot in the central portion of the University Research Park, an 84-acre tract of land approximately one mile (1.6 km) southwest of the University of Missouri at Columbia's main campus (Ref. 2.1). [redacted] Approximate distances to the University property lines from the reactor facility are 2,400 feet (732 m) to the north, 4,800 feet (1,463 m) to the east, 2,400 feet (732 m) to the south, and 3,600 feet (1,097 m) to the west.

The Columbia campus is located in the southern portion of Columbia, the county seat and largest city in Boone County, Missouri. Boone County is located in the central part of the state as shown in Figure 2.1 and consists of an area of approximately 683 square miles (1,769 square km). The county lies between 38° 40' and 39° 15' North Latitude and between 92° 5' and 92° 35' West Longitude and is approximately 41 miles (66 km) in its greatest north-to-south length and 22 miles (35.4 km) in its greatest east-to-west width. The southwestern border is formed by the Missouri River, and the southeastern border is formed almost entirely by one of its small tributaries (Cedar Creek). These bodies of water follow rather sinuous courses, producing a great deal of irregularity in the county's outline (Ref. 2.2).

In addition to the MURR, there are several low occupancy research buildings situated within the University Research Park. The Science Instrument Shop and the Botany Research Green House are located just to the south of the reactor facility. The Research Park Development Building, Dalton Research Center, the Laboratory Animal Center, the United States Department of Agriculture (USDA) Biological Control of Insects Research Laboratory, and the Psychology Animal Research Laboratory (Marx Building) are located to the north. Personnel in facilities located within 1,500 feet (457 m) can be rapidly evacuated if required. The University Research Park is therefore adequately suited for the location of the MURR.

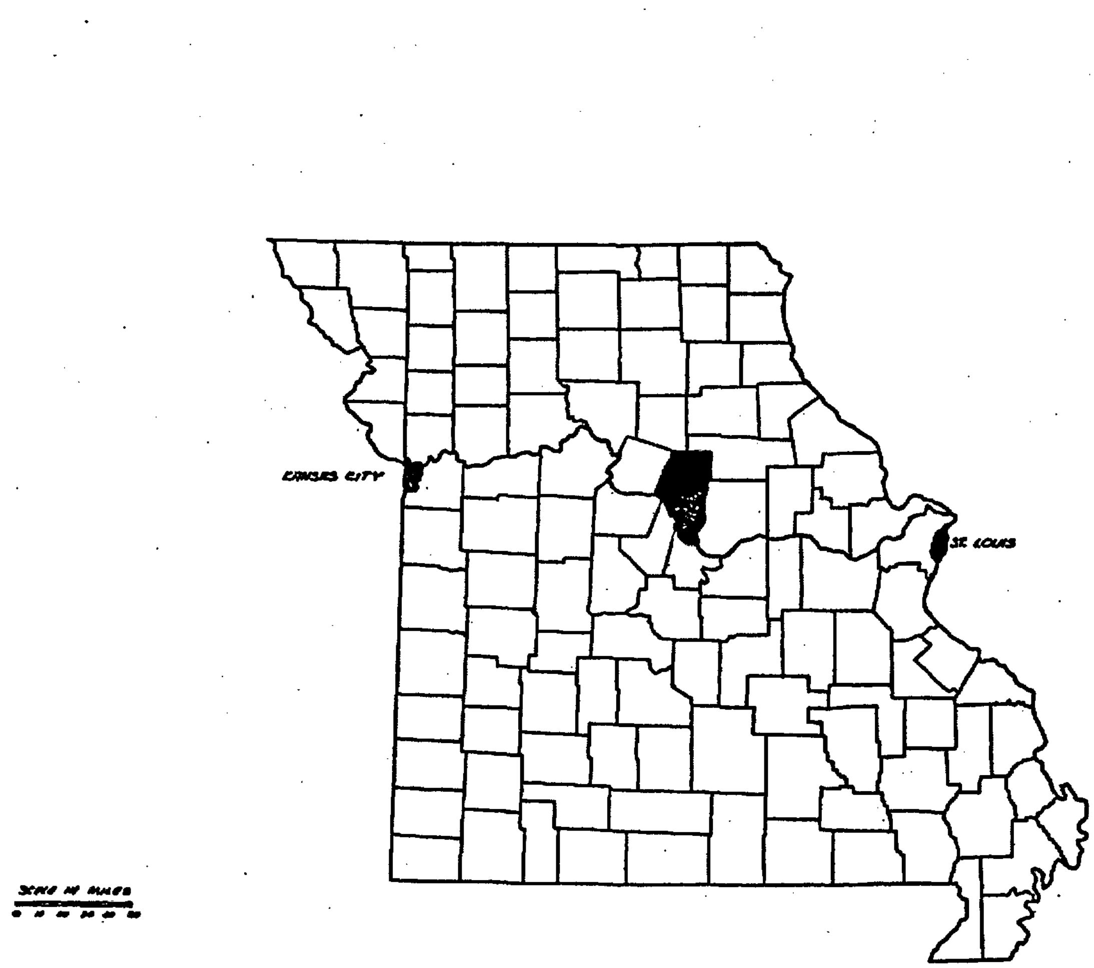


FIGURE 2.1
MAP OF MISSOURI SHOWING BOONE COUNTY

There are two major highways which extend across the county and intersect at Columbia. Interstate Highway 70 (I-70) extends east-to-west across the middle width of the county and provides connections between St. Louis and Kansas City. U.S. Highway 63 extends north and south across the mid-length of the county, connecting with Jefferson City to the south and with U.S. Highway 24 at Moberly to the north. The reactor facility is located approximately 3.25 miles (5.2 km) southwest of this intersection.

The location of the major cities, railroads, and highways in Boone County are shown in Figure 2.2.

2.1.2 Operational Boundaries

There are three areas of concern regarding the normal operation, safety, and emergency actions associated with the reactor facility: the area within the operations boundary; the area within the site boundary; and the Emergency Planning Zone (EPZ).

The operations boundary consists of the outer walls of the Research Reactor Facility (laboratory and reactor containment buildings). The area within this boundary is a "restricted access" area where the Reactor Facility Director has direct authority and control over all activities, normal and emergency. The adjacent reactor cooling tower building is also included within the restricted access area. A tunnel connects the cooling tower building to the laboratory building basement. There are pre-established evacuation routes and procedures known to personnel frequenting this area. The operations boundary is within the site boundary.

The site boundary consists of the following: Stadium Boulevard; Providence Road (Route K);[1] the MU Recreational Trail; and the MKT Nature and Fitness Trail. The area within these boundaries is owned and controlled by the University of Missouri and may be frequented by people unacquainted with the operation of the reactor. The Reactor Facility Director has authority to initiate emergency actions in this area, if required.

In addition, an Emergency Planning Zone (EPZ) has been established for which emergency plans have been developed to ensure that prompt and effective actions can be taken to protect the public in the event of an accident. MURR's EPZ is the area bounded by a 150-meter radius from the reactor facility ventilation exhaust stack and lies completely within the site boundary (Figure 2.3).

[1]Providence Road crosses University of Missouri property separating the University Research Park from another MU-owned track of land lying to the east. The road runs north and south, with the closest point of approach being approximately 1,300 feet (400 meters) east of the reactor facility. MU has the authority to determine all activities, including the exclusion or removal of personnel and property, and to temporarily secure the flow of traffic on this road during an emergency.

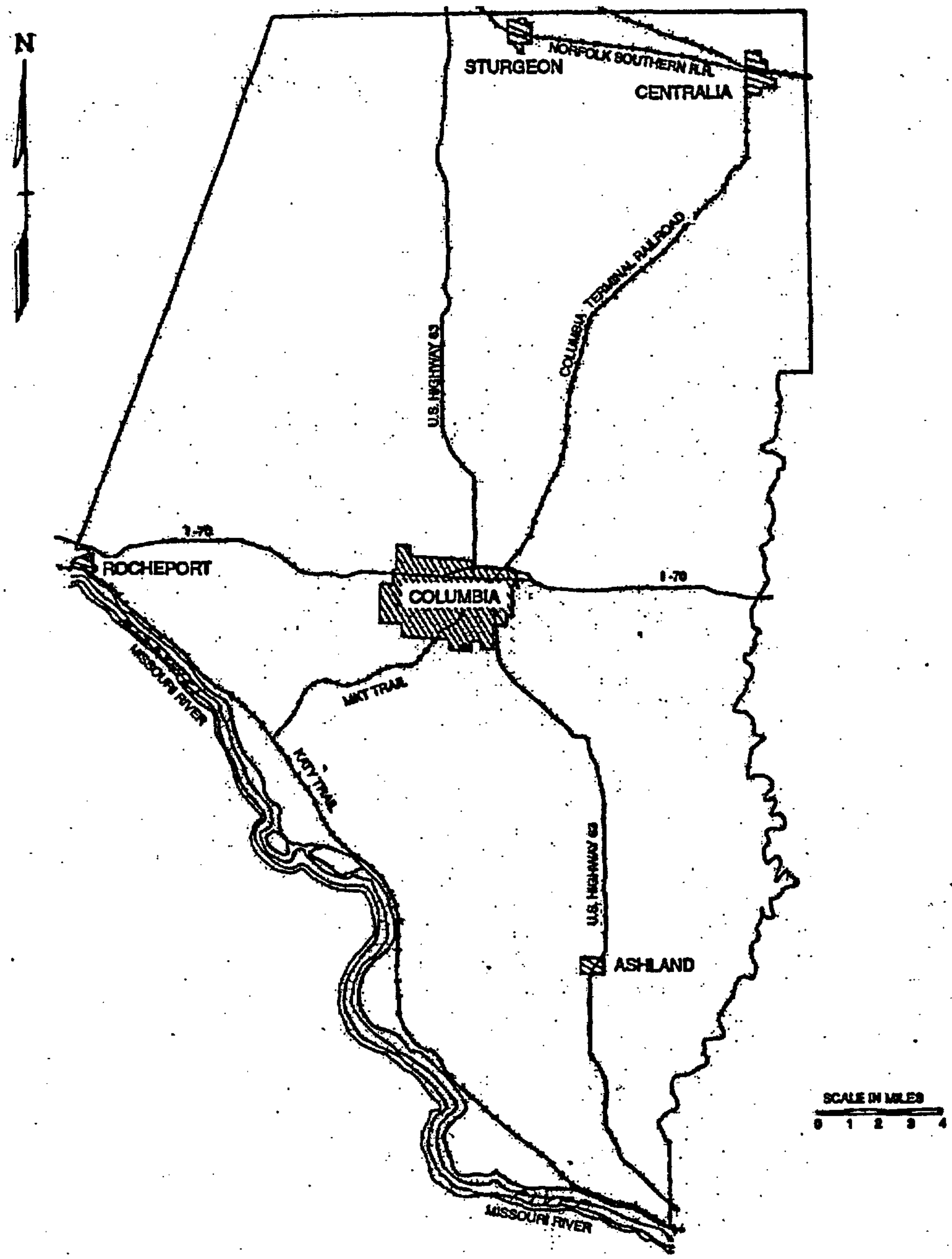


FIGURE 2.2
LOCATION OF MAJOR CITIES, RAILROADS, AND HIGHWAYS
IN BOONE COUNTY, MISSOURI

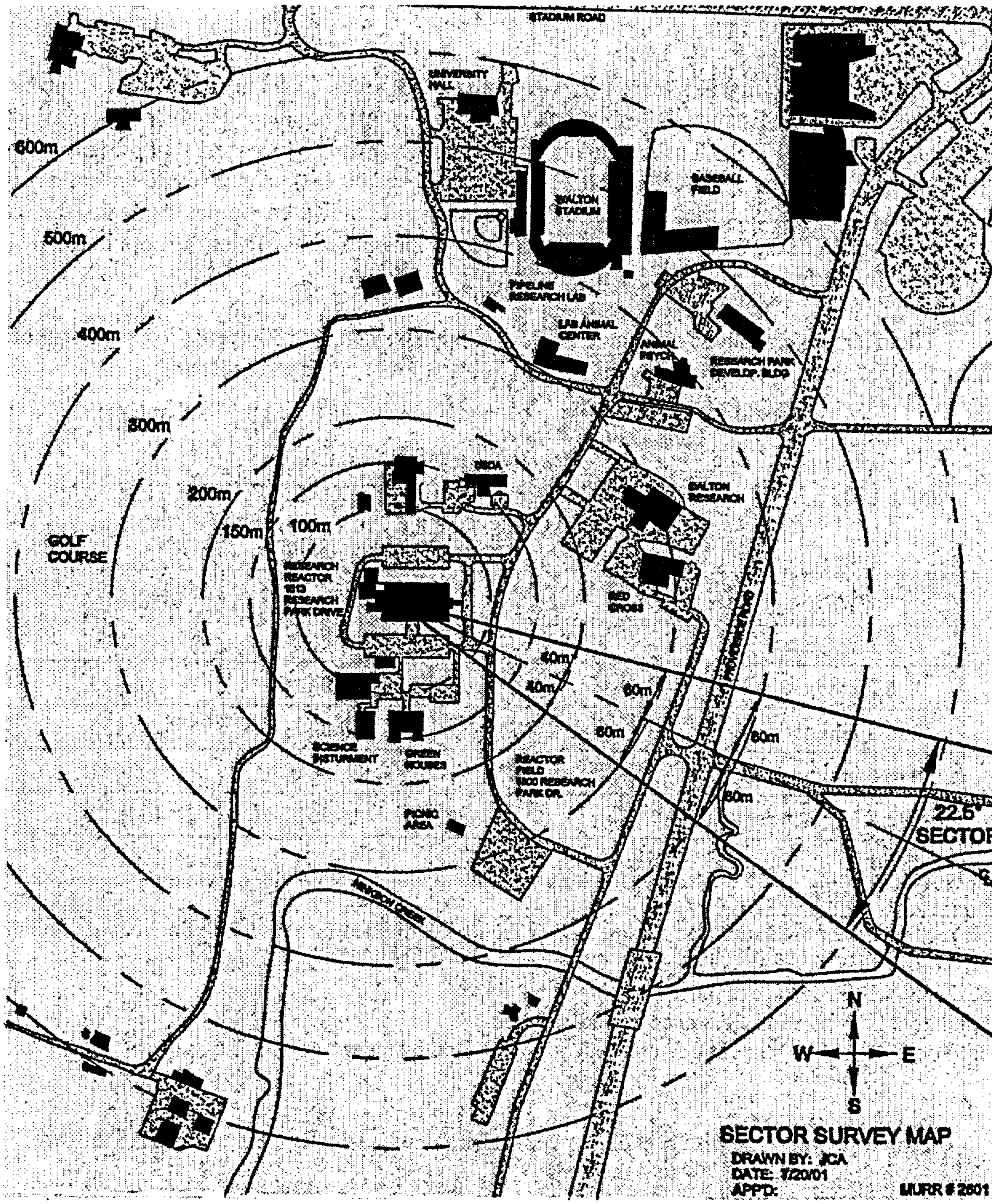


FIGURE 2.3
EMERGENCY PLANNING ZONE (EPZ)

2.1.3 Population Distribution

The reactor facility is situated approximately 1.5 miles (2.4 km) south-by-southwest of downtown Columbia, Missouri. The City of Columbia currently has a population of approximately 91,885 (based on 2000 U.S. Census Bureau statistics and adjusted by an annual population growth of 1.4%), an increase of 57% since 1970. Approximately 100,192 people reside within a 5-mile (8-km) radius of the reactor facility (Table 2-1). This 5-mile radius encompasses nearly the entire city as well as parts of the outlying metropolitan area. Approximately 70% of the population resides within the 2 to 6 kilometer (1.2 to 3.7 mile) range.

The nearest permanent residence is located approximately 2,500 feet (762 m) north of the reactor facility near Stadium Boulevard (State Highway 740) on Brandon Road.

TABLE 2-1
POPULATION DISTRIBUTION WITHIN 8 KILOMETERS OF THE REACTOR FACILITY (2006 Projected)[d]

Distance From Facility[a]	Area (sq. km)	Population[b]	Population[c]
0 to 1 kilometer	3.14	255	255
1 to 2 kilometers	9.42	16,060	16,315
2 to 4 kilometers	37.70	39,820	56,135
4 to 6 kilometers	62.83	29,506	85,641
6 to 8 kilometers	87.96	14,551	100,192
Total	201.06		100,192

[a]This represents the area surrounding the reactor facility as divided into a series of concentric annuluses (rings).
[b]Population within each concentric ring.
[c]Cumulative population to a distance of 8 kilometers.
[d]Based on 2000 U.S. Census Bureau statistics and adjusted by an annual population growth of 1.4%.

Existing land uses within each concentric ring can generally be described as follows:

0 to 1 kilometer: There is very little residential development within 1 kilometer of the reactor facility site. Most of the land is owned and operated by the University of Missouri. Recreational areas include a golf course to the west and a park to the south. Also located within this area are three major University sports venues; Memorial Stadium/Faurot Field (62,000 seats), Mizzou Arena (15,061 seats), and Hearnes Center (13,300 seats);

1 to 2 kilometers: The major residential areas are located to the north, northwest, and south. A shopping center, business district, two hospitals, and a large portion of the University of Missouri at Columbia's main campus are located within this area. With the exception of a small area to the southeast, there is no room for any substantial residential or nonresidential (industrial, commercial, or business) development;

2 to 4 kilometers: The major residential areas are located in the northern half and the southwest. A shopping center, business district, two hospitals, two colleges, three high schools, three middle/junior high schools, and nine elementary schools are located in this area. Recreational areas include two golf courses and eight parks. The downtown area of Columbia, which consists mainly of government offices and retail, commercial, and business uses, is located to the northeast. Development should continue within this area, probably to the south of the reactor facility;

4 to 6 kilometers: Most residential development is within the northern half. Three shopping centers, two hospitals, one middle/junior high school, three elementary schools, and an industrial park are located in this area. Recreational areas include two golf courses and five parks. Substantial amounts of land exist for residential or nonresidential development; and

6 to 8 kilometers: The only substantial residential development is to the northeast. A shopping center, two middle/junior high schools, and four elementary schools are located in this area. Recreational areas include one park. Substantial amounts of land presently exist for residential or nonresidential development.

Population growth within each concentric ring surrounding the reactor facility is difficult to project. The land use pattern of the city is shaped by market forces and available or proposed infrastructure (i.e., roadways, sewers, water, electricity, and fire and police protection). The City of Columbia tends to promote a mix of land uses instead of focusing on the separation of uses (Ref. 2.28). However, based strictly upon the availability of land for development, most of the projected residential growth should occur within the 4 to 6- and 6 to 8-kilometer (2.5 to 3.5- and 3.5 to 5-mile) rings.

Table 2-2 shows the historical and the projected population growth for the City of Columbia. Population growth for the city is projected to be approximately 1.4% annually. This value was provided by the Chamber of Commerce and is based on economic development within the region. Student enrollment is difficult to project but should remain relatively constant.

Due to the seasonal occupation of certain buildings, significant population variations do occur. Approximately 8,300 of the nearly 30,000 college students reside in residence halls or Greek houses. These buildings are located within 4 kilometers (2.5 miles) of the reactor facility and would only be occupied when school is in session. Memorial Stadium, located approximately 1 km (0.6 miles) northeast of the reactor facility, with a seating capacity of 62,000, is occupied five or six afternoons per year during football season for approximately four hours. The Hearnes and Mizzou

TABLE 2-2
HISTORICAL AND PROJECTED POPULATION GROWTH
FOR THE CITY OF COLUMBIA

Population	Year							
	1950	1960	1970	1980	1990	2000	2006	2026
Resident	25,335	36,650	58,557	62,061	69,101	84,531	91,885	121,340
Student[a]	10,909	13,191	24,184	27,228	29,496	30,000	30,000	30,000
Total	36,244	49,841	82,741	89,289	98,597	114,531	121,885	151,340

[a]Student enrollment includes the University of Missouri, Stephens College, and Columbia College (Prior to 1970, Columbia College was identified as Christian College).

Arenas are, at most, occupied several times per week but are rarely occupied at the same time as Memorial Stadium.

2.1.4 Physiography

Boone County lies along the northern Ozark Border of the state. It is superior in agricultural productivity to the Ozark uplands to the south, yet inferior to the plains of northern and western Missouri. Surface configuration is the primary consideration when studying the agricultural pattern. Crop production is confined primarily to valley bottoms and lower slopes. Corn, soybeans, and winter wheat, in general, are the most prevalent crops of the region. Livestock also has significant importance because much of the land is poorly adapted for row cropping (Ref. 2.3).

The county contains a variety of topographic features, ranging from rugged maturely dissected hills to flat flood plains and flat uplands, and can be divided into four physiographic regions: the Centralia uplands, a flat upland plain 850 feet (259 m) above sea level; the Columbia dissected uplands; the Ashland hills; and the Missouri River flood plain, the latter being in the extreme southern tip of the county (Ref. 2.7).

The Centralia uplands are located in the northeastern portion of the county, between the town of Sturgeon and the city of Centralia. This flat upland plain is lightly dissected by the upper portions of the Silver Fork Stream and its tributaries. The area averages from 850 to 900 feet (259 to 274 m) above sea level and extends southward from Centralia to near the towns of Hallsville and Murry. In this same general area, portions of the flat upland plain gradually slope upward to the highest point in the county, some 940 feet (286.5 m) above sea level.

The Columbia dissected uplands are located southwest of the Centralia uplands. This land surface is more completely dissected in a wide zone extending to within a few miles of the Missouri River. Over most of this region the interstream areas stand at about the same elevation so that the skyline is remarkably uniform.

The Ashland hills region is three to five miles (4.8 to 8 km) wide and lies between the Columbia dissected uplands and the flood plain of the Missouri River. It parallels the Missouri River and extends some distance up the tributary valleys. In this rugged area, the local relief ranges up to approximately 250 feet (76 m) and there is very little flat upland or lowland. Several of the slopes are very steep, including numerous vertical bluffs along the Missouri River and some of its major tributaries.

The Missouri River flood plain varies in width, but along much of the southwestern part of the county, it is about 2 miles (3.2 km) wide. The lowest point in the county is on the flood plain at the extreme southern tip, 540 feet (164.6 m) above sea level.

All of Boone County eventually drains into the Missouri River. Perche Creek is the largest stream in the county, entering near the northwestern corner of the county and flowing southward across the western half. On the Missouri River flood plain, it meanders southeastward and parallels the river for about 5 miles (8 km) before flowing into it near a location where the river approaches a steep bluff. The majority of the northern half of Boone County, including the City of Columbia and the reactor facility site, lies within the Perche Creek drainage basin. Principal tributaries of Perche Creek near the MURR include the Hinkson, Grindstone, Flat Branch, and Hominy Creeks.

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.1 Industry

There are a number of small industrial activities in the area, but no major industrial facilities that need be of concern, from a safety standpoint, for the reactor facility. The area's economy is based primarily on government (which includes the University of Missouri), retail, and services with smaller contributions by such things as medical facilities (hospitals and clinics), insurance, agriculture, and manufacturing. Agriculture is important in the upland area where the soils are fertile and the terrain is relatively flat. The limestone deposits in the state form the basis for a number of quarrying operations which produce road aggregate, agricultural limestone, sand, and building stone. Several quarries are located in Boone County. The closest operation is approximately 3 miles (4.8 km) northwest of the reactor facility.

The University of Missouri at Columbia is the area's largest employer, employing approximately 14,500 people. Columbia is also the home of Stephens College (employing approximately 200 people) and Columbia College (employing approximately 220 people).

2.2.2 Transportation

2.2.2.1 Highway Transportation

The City of Columbia is at the cross-roads of two major highways: Interstate Highway 70 (I-70), and U.S. Highway 63. I-70 extends east-to-west across the middle width of Boone County and provides connections between St. Louis and Kansas City. I-70 then continues westward from Kansas City where it intersects Interstate Highway 15 (I-15) in Utah and eastward from St. Louis where it terminates in Baltimore, Maryland. U.S. Highway 63 extends north and south across the mid-length of the county, connecting with Jefferson City to the south and with U.S. Highway 24 at Moberly to the north.

The reactor facility is located on Research Park Drive approximately 3.5 miles (5.6 km) southwest of the I-70/U.S. Highway 63 intersection.

In addition to these two main highways, there are numerous state highways, state routes, and secondary roads making most sections of Boone County easily accessible.

2.2.2.2 Rail Transportation

There is only one railway which provides transportation services for the City of Columbia. The Columbia Terminal (COLT) is a 21.4-mile (34.4-km) long freight-only branch-line from the Norfolk Southern, a railroad which runs east-to-west through northern Boone County. The COLT, owned by the City of Columbia and operated by the Water and Light Department, originates in Centralia and generally runs in a southwesterly direction through the towns of Brown Station and Hallsville prior to terminating near the center of Columbia, approximately 1.7 miles (2.7 km) from the reactor facility.

The COLT provides service for industrial land uses along the Route B corridor in northeast Columbia. Other land uses served include the City of Columbia Municipal Power Plant and a commercial lumber facility to the north of downtown. The vast majority (97%) of the rail traffic is inbound. In fiscal year 2004, 2,150 carloads of freight were transported on this track, but the typical usage is about 1,500 cars per year. The primary freight included coal for the City Power Plant, chemicals, petroleum, steel for several manufacturing facilities, and lumber for several commercial facilities.

The Columbia metropolitan area has no passenger rail service. The nearest passenger train service is operated by AMTRAK in Jefferson City approximately 26 miles (41.8 km) south of the reactor facility.

2.2.2.3 Water Transportation

The Missouri River is one of the largest river systems in the United States and the largest river in Boone County. The river originates in southwestern Montana and generally flows in an easterly and southeasterly direction before entering the Mississippi River in eastern Missouri a length of some 2,700 miles (4,345 km). The river lies approximately 7.3 miles (11.7 km) to the west of the reactor facility and forms the southwestern border of Boone County.

The Missouri River is the only river system in Boone County large enough for commercial navigation; however, there are no ports which directly service the City of Columbia. Therefore, any commercial/barge traffic on the Missouri River is beyond consideration for creating a credible reactor facility accident scenario.

TABLE 2-3
MAJOR LAND TRANSPORTATION ROUTES WITHIN 5 MILES
OF THE REACTOR FACILITY

Name	Location from Reactor Facility		
	Miles	Km	Direction
Interstate Highway 70 (I-70)	2.5	4.0	N
State Highway 763	2.3	3.7	N
U.S. Highway 63	2.4	3.9	E
State Highway 740	0.4	0.6	N
State Highway 163	0.2	0.3	E
Columbia Terminal (COLT) Railroad	1.7	2.7	NE

2.2.2.4 Air Transportation

The Columbia metropolitan area is served by the Columbia Regional Airport, located approximately 10 miles (16 km) southeast of the reactor facility. The airport is situated on approximately 1,314 acres and is owned and operated by the City of Columbia. It is the sole public use airport located in Boone County for which records are kept. The airport has two aircraft runways: a 6,501 by 150 feet (1,982 by 46 m) concrete strip which supports most of the commercial air traffic, and a smaller 4,401 by 75 feet (1,341 by 23 m) cross-wind runway primarily for private aircraft.

The Columbia Regional Airport averages around 37,300 commercial, private, and military flight operations yearly.

2.2.3 Military Facilities

No military bases, missile sites, or military firing ranges are located within 5 miles (8 km) of the reactor facility. Fort Leonard Wood and Whiteman Air Force Base (AFB) are the closest major military facilities. Fort Leonard Wood is located approximately 86 miles (138 km) south of the reactor facility; Whiteman AFB is located approximately 67 miles (108 km) to the west.

Fort Leonard Wood, a subordinate of the U.S. Army Training and Doctrine Command, is situated on 63,000 acres in the Ozarks region of south-central Missouri approximately 130 miles (209 km) west of St. Louis and approximately 80 miles (128.7 km) east of Springfield. The facility is the home of the Army Engineer School and Regiment, and a major initial entry training center. As such, the post trains around 32,000 soldiers annually.

Whiteman AFB is located in the wooded, rolling hills of west-central Missouri approximately 60 miles (96.6 km) southeast of Kansas City and two miles (3.2 km) south of Knob Noster. Whiteman AFB is the home of the 509th Bomb Wing, which operates and maintains a division of the Air Force's B-2 bombers. At present, Whiteman AFB employs approximately 3,800 military and civilian personnel.

2.2.4 Analysis of Potential Accidents at Facilities

There are no nearby industrial, transportation, or military facilities with the potential of causing a credible accident (which could prevent a safe reactor shutdown or result in a release of radioactive material from the reactor facility) that would exceed the general public exposure limits of Title 10, Chapter I of the Code of Federal Regulations, Part 20 (10 CFR 20).

Although shipments of hazardous materials may occur by highway and rail, accidents involving these materials have no impact on the safe operation of the MURR, even if an evacuation is required. The facility may be placed in a safe condition in a timely manner if an evacuation of the facility is required.

The basic design and structure of the facility provides significant protection for the reactor. As described in Chapter 1, The Facility, and Chapter 6, Engineered Safety Features, the reactor is housed in a five-level poured concrete building with 12-inch (0.3-m)-thick reinforced exterior walls. In addition, the reactor core is [redacted] surrounded by a massive reinforced concrete edifice (biological shield) approximately 31 feet (9.4 m) in height and varying in thickness from 3 to 7½ feet (0.9 to 2.3 m). The immense size of the biological shield provides excellent protection against natural phenomena that could result in damage to the reactor core assembly.

2.3 Meteorology

2.3.1 General and Local Climate

Central Missouri's location within the North American continent places it in the Humid Continental - Warm Summer climatic zone. This type of climate has a characteristic long, warm summer with moderate relative humidity. The winters are cool to cold and mark a period of lower precipitation than during the remainder of the year. Because of its geographical location well inland, the region is subject to significant seasonal and daily temperature variations. Air masses moving over the state during the year will include cold continental polar air from Canada, warm and humid maritime tropical air from the Gulf of Mexico and the Caribbean Sea, and dry eastward flowing air masses from the Rocky Mountains located to the west. Prolonged periods of extreme hot or cold temperatures are unusual.

The general geostrophic airflow pattern and the prevailing jet stream track shuttle precipitation-producing mid-latitude cyclones (lows) across the state from west-to-east throughout the year. Consequently, precipitation events in all seasons move through from a westerly direction.

The prevailing surface wind direction in Missouri is from the south. However, most winds with speeds in excess of 15 knots will tend to be from the north or northwest.

Temperatures rarely exceed 100 °F (38 °C) in the summer and rarely fall below 0 °F (-18 °C) in the winter. The mean maximum temperatures in Columbia range from 36.6 °F (2.6 °C) in January to 88.6 °F (31.4 °C) in July. The city's average relative humidity is 70%.

Precipitation in the Columbia area averages approximately 39 inches (99 cm) per year. The city has measurable amounts of precipitation about 100 days per year, with thunderstorms and associated lightning occurring on about half of those days. Hail may be expected as a product of these storms on two to three days per year. Spring, summer, and early fall precipitation occurs in the form of rain showers and thunderstorms. In central Missouri, severe thunderstorms typically occur during the period from mid to late spring through early summer. Wind speeds of up to 60 miles (97 km) per hour or more may be experienced once or twice a year during a severe thunderstorm.

Winter precipitation is generally light to moderate and occurs in the form of rain or snow or a mixture of both with an occasional, though infrequent, thunderstorm. Occasional heavy snowfall episodes do occur, but not often, and the accumulation does not last for any significant duration. Surface temperature conditions sometimes produce freezing rain or drizzle, although normally not more than twice per season.

A summary of the precipitation totals for the City of Columbia and the three other primary reporting stations in Missouri are presented in Table 2-4. The two months in which the areas receive the highest amounts of annual rainfall are given.

The following is a climatic summary of the historical seasonal and/or annual frequencies of severe weather phenomena in the central Missouri area.

- Hurricanes

  Given the fact that hurricanes receive their energy from warm tropical ocean water, and that upon landfall that energy supply is cut off and the storm's intensity rapidly dissipates, the influence of any given hurricane upon central Missouri climatology is normally insignificant. However, there have been instances in which the low-pressure remnant of a hurricane has brought rain to the area.

TABLE 2-4
PRECIPITATION DATA FOR THE STATE OF MISSOURI

Precipitation (inches)			
Normal		Maximum	
Station	Yearly Total	Monthly Total	24-Hour Total
Columbia	39.05	5.01 - May 4.32 - June	5.95 - July 1989
Kansas City	37.62	5.04 - May 4.86 - Sept	8.82 - Sept 1977
St. Louis	37.51	3.97 - May 3.85 - July	6.55 - May 1995
Springfield	43.04	5.09 - June 4.62 - Sept	6.85 - July 1958

Data published by the National Climate Data Center (NCDC), Asheville, North Carolina.

- Tornadoes and Waterspouts

The City of Columbia is located east of the region of maximum worldwide tornado occurrence, an area that is generally accepted to include parts of the states of Kansas, Oklahoma, and Texas. A total of 1,779 tornadoes were reported in Missouri from 1950 to 2005, and around two-thirds of these occurred during the months of April, May, and June. In Missouri, the average annual number of tornado days - days on which one or more tornadoes were reported - is 12. Boone County itself has experienced 32 reported tornadoes within the recording period 1950 to 2005. Structural damage has generally been limited to frame/lumber and mobile home residential units.

Waterspouts are not a significant meteorological concern in central Missouri. The term waterspout refers to a rotating column of wind which forms, or passes, over a large body of water. This vortex may or may not be of tornadic intensity. While a given tornado in Missouri may pass over a significant lake or river, there are no bodies of water within the state of sufficient size to support the formation of the typically defined waterspout.

- Thunderstorms

Thunderstorms in Missouri occur most frequently in the spring, summer, and early fall. The City of Columbia's average number of thunderstorm days - days on which one or more thunderstorms occur - is shown in Table 2-5. The months of April through September are shown to have the greatest thunderstorm frequency, averaging 41.2 thunderstorm days, while May through July is the most active three-month period, during which roughly 46% of the annual thunderstorm activity occurs.

- Winds

Extreme wind speeds are uncommon in central Missouri. When they do occur, they are usually caused by pressure gradients and temperature contrasts present in the mid-latitude cyclones that pass through the state. These cyclones may spawn storms which produce high winds from gust fronts, microbursts, and tornadoes. Non-storm-related extreme winds are rare. Occasionally, cold high-pressure air filling in behind a front will cause high wind, especially in the winter when temperature contrasts are large.

2.3.2 Site Meteorology

The meteorological conditions at the reactor facility site are represented by the recorded data for the City of Columbia and are similar to the climatic data collected at the state's other primary reporting stations located at Kansas City, St. Louis, and Springfield (Figure 2.4). The official weather station for Columbia is located at the Columbia Regional Airport (Section 2.2.2.4), elevation 880 feet (268 m) above sea level (ASL), approximately 10 miles (16 km) southeast of the reactor facility. This automated reporting station has wind instruments situated 20 feet (6 m) above ground level and temperature instruments 5 feet (1.5 m) above ground level.

The reactor site occasionally experiences valley fog associated with its location in the Hinkson Creek valley. The prevailing surface wind direction at the reactor is southerly.

TABLE 2-5
AVERAGE THUNDERSTORM DAYS FOR COLUMBIA, MISSOURI
(Period of Record 1969 - 1998)

Month	Thunderstorm Days	Month	Thunderstorm Days
January	0.7	July	7.8
February	0.9	August	6.9
March	3.1	September	5.0
April	5.3	October	3.1
May	8.2	November	1.9
June	8.0	December	0.9
Annual			51.8

Data published by the National Climate Data Center (NCDC), Asheville, North Carolina.

FIGURE 2.4
CLIMATOLOGICAL STATIONS IN THE STATE OF MISSOURI

The general meteorological conditions for the City of Columbia are summarized below. The included tables delineate the seasonal distribution of temperature, precipitation, and other climatic characteristics during a 30-year period of record from 1969 to 1998.

2.3.2.1 Temperature

The normal and extreme temperatures for the City of Columbia are shown in Table 2-6. The coldest daily low temperatures occur in January [18.5 °F (-7.5 °C)] while the warmest daily high temperatures occur in July [88.6 °F (31.4 °C)]. During the 30-year reporting period, the highest recorded temperature was 111 °F (44 °C), occurring in July 1980. The lowest recorded temperature was -20 °F (-29 °C), occurring in December 1989.

2.3.2.2 Precipitation

The normal and extreme precipitation totals for the City of Columbia are shown in Table 2-7. The average annual precipitation is 39.05 inches (99.19 cm) per year. The highest monthly averages occur in May [5.01 inches (12.73 cm)] and June [4.32 inches (10.97 cm)]. The maximum twenty-four hour total for the 30-year period was 5.95 inches (15.11 cm) which occurred in July 1989. The normal and extreme snowfall totals for Columbia are shown in Table 2-8.

2.3.2.3 Humidity

The mean relative humidity for the Columbia area is 70%. The lowest value occurs in April (64%) while the highest occurs in December (74%). The monthly and average annual relative humidities during the 30-year reporting period for 6:00 a.m., noon, 6:00 p.m., and midnight are shown in Table 2-9.

2.3.2.4 Winds

The annual surface wind rose (Figure 2.5) compiled by the National Weather Service for the period 1960 to 1969 shows a southerly prevailing wind direction for the City of Columbia. However, during periods when Columbia experiences winds in excess of 15 knots, the winds generally blow from a north or northwesterly direction. Wind data for Columbia is given in Table 2-10. The maximum 2-minute and 5-second winds are based on a three-year period of record.

2.3.2.5 Severe Weather

The meteorological conditions at the reactor facility site are reflective of that of the state as a whole. The severe weather probability will be the same as that discussed in Section 2.3.1, General and Local Climate. However, on a meso scale, the reactor is located in the Hinkson Creek valley with a high bluff directly to the west. This location should help protect the reactor from severe weather phenomena, such as high winds and tornadoes.

TABLE 2-6
NORMAL AND EXTREME TEMPERATURES FOR
COLUMBIA, MISSOURI

Temperature (°F)							
Normal				Extreme			
Month	Daily Max.	Daily Min.	Monthly	Record High	Year	Record Low	Year
Jan	36.6	18.5	27.6	74	1989	-19	1982
Feb	41.4	22.8	32.1	82	1972	-15	1979
Mar	53.3	33.0	43.2	85	1986	-5	1978
Apr	65.7	43.7	54.7	90	1987	19	1975
May	74.1	53.1	63.6	92	1998	29	1976
June	82.8	61.2	72.1	103	1988	40	1993
July	88.6	66.2	77.4	111	1980	48	1975
Aug	86.7	63.8	75.2	110	1984	42	1986
Sept	78.8	57.0	67.9	101	1971	32	1984
Oct	67.6	45.5	56.5	93	1981	22	1993
Nov	53.6	34.6	44.1	83	1978	0	1991
Dec	40.3	23.2	31.8	76	1991	-20	1989
Year	64.1	43.5	53.8	111	July 1980	-20	Dec 1989

Data published by the National Climate Data Center (NCDC), Asheville, North Carolina.

TABLE 2-7
NORMAL AND EXTREME PRECIPITATION TOTALS FOR
COLUMBIA, MISSOURI

Precipitation (inches)							
Normal		Extreme					
Month	Monthly	Record Max.	Year	Record Min.	Year	24-hour Max.	Year
Jan	1.45	4.78	1995	0.05	1986	1.91	1995
Feb	1.84	6.18	1985	0.11	1991	2.88	1985
Mar	3.17	10.09	1973	0.78	1971	4.03	1990
Apr	3.83	11.69	1994	1.38	1971	4.50	1996
May	5.01	12.31	1995	1.43	1998	5.73	1995
June	4.32	10.28	1985	0.35	1980	3.87	1995
July	3.67	12.14	1981	0.24	1976	5.95	1989
Aug	3.28	8.96	1982	0.21	1976	5.43	1993
Sept	3.86	12.06	1993	0.45	1979	3.81	1986
Oct	3.22	7.28	1998	1.03	1992	4.94	1998
Nov	2.93	10.42	1985	0.42	1989	2.97	1983
Dec	2.47	6.96	1982	0.48	1996	2.88	1982
Year	39.05	12.31	May 1995	0.05	Jan 1986	5.95	July 1989

Data published by the National Climate Data Center (NCDC), Asheville, North Carolina.

TABLE 2-8
NORMAL AND EXTREME SNOWFALL TOTALS FOR COLUMBIA, MISSOURI

Snowfall (inches)							
Normal		Extreme					
Month	Monthly	Record Max.	Year	24-hour Max.	Year	Depth Max.	Year
Jan	6.6	23.5	1979	19.7	1995	13	1995
Feb	7.6	20.2	1993	12.7	1993	14	1993
Mar	4.1	18.0	1978	8.8	1990	12	1978
Apr	1.0	7.1	1980	7.1	1980	5	1980
May	0.0	Trace	1990	Trace	1990	0	N/A
June	0.0	Trace	1990	Trace	1990	0	N/A
July	0.0	Trace	1995	Trace	1995	0	N/A
Aug	0.0	0.0	N/A	0.0	N/A	0	N/A
Sept	0.0	0.0	N/A	0.0	N/A	0	N/A
Oct	0.0	0.1	1976	0.1	1976	0	N/A
Nov	2.2	8.3	1971	8.1	1975	8	1975
Dec	5.2	17.8	1973	12.7	1987	13	1987
Year	26.7	23.5	Jan 1979	19.7	Jan 1995	14	Feb 1993

Data published by the National Climate Data Center (NCDC), Asheville, North Carolina.

TABLE 2-9
RELATIVE HUMIDITIES FOR COLUMBIA, MISSOURI

Relative Humidity (%)													
	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Yr
Normal	71	72	67	64	71	71	70	71	72	69	72	74	70
Hr 06 LST	74	76	72	69	79	81	81	81	80	76	76	77	77
Hr 12 LST	78	80	79	78	85	85	86	87	87	84	82	80	83
Hr 18 LST	65	65	59	55	60	59	56	56	58	57	62	68	60
Hr 24 LST	67	65	58	54	60	60	56	59	63	62	67	71	62

Data published by the National Climate Data Center (NCDC), Asheville, North Carolina.
LST = Local Sidereal Time.

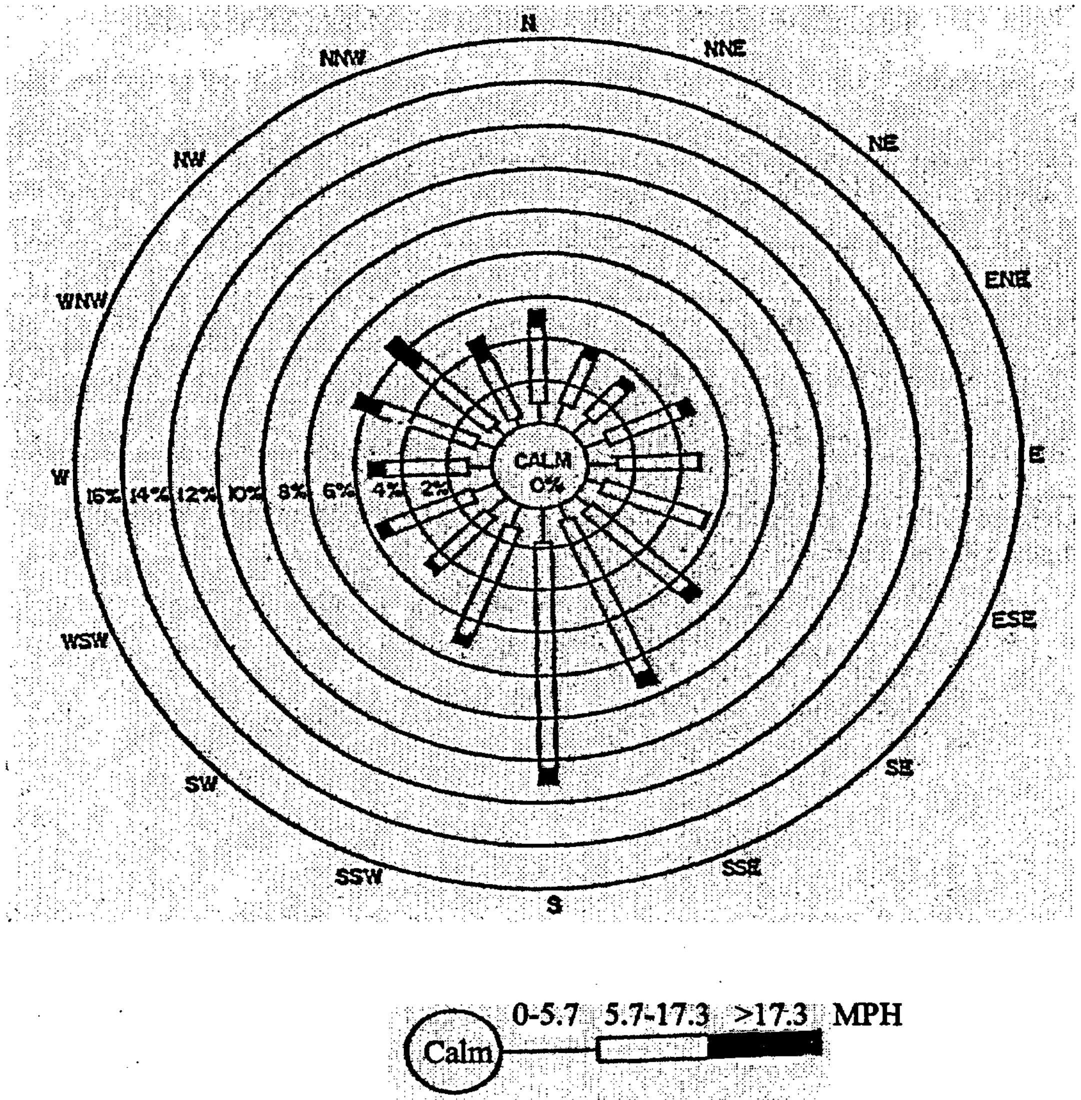


FIGURE 2.5
SURFACE WIND ROSE AT COLUMBIA, MISSOURI

TABLE 2-10
NORMAL AND EXTREME WINDS FOR COLUMBIA, MISSOURI

Wind								
			Maximum 2-Minute			Maximum 5-Second		
Month	Mean Speed (mph)	Prevail Dir (degs)	Speed (mph)	Dir (degs)	Year	Speed (mph)	Dir (degs)	Year
Jan	10.9	300	41	20	1996	49	20	1996
Feb	10.6	300	34	20	1996	41	30	1996
Mar	11.9	310	41	230	1998	49	210	1998
Apr	11.4	180	43	30	1996	52	30	1996
May	9.2	180	33	10	1998	45	280	1998
June	8.6	180	44	310	1998	52	300	1998
July	8.2	180	29	250	1997	38	70	1997
Aug	7.8	180	36	300	1997	43	310	1997
Sept	8.5	180	31	190	1998	37	190	1998
Oct	9.5	180	47	250	1996	59	250	1996
Nov	10.5	180	45	190	1998	53	240	1998
Dec	10.6	300	37	60	1997	45	60	1997
Year	9.8	180	47	250	Oct 1996	59	250	Oct 1996

Data published by the National Climate Data Center (NCDC), Asheville, North Carolina

2.4 Hydrology

2.4.1 Surface Water

The principal stream of the Boone County drainage basin is Perche Creek[1] (Figure 2.6). This creek enters the county from the northwest and then flows south and then southeasterly before entering the Missouri River approximately 8.5 miles (13.7 km) from the reactor facility. Hinkson Creek, which drains the MURR site, is a major tributary of Perche Creek. It originates in the Centralia uplands near the town of Hallsville approximately 15 miles (24 km) from the reactor. Hinkson Creek is a sinuous, rocky stream with an average grade of 25 feet (7.6 m) per mile. Near its origin, the creek drains level Putnam silt loam prairies prior to flowing into the rocky soils of the Columbia dissected uplands. By the time the Hinkson has approached the site of the reactor facility, it has drained approximately 50 square miles (129.5 km^2) of Boone County. Few of the tributaries of Hinkson Creek are named and all are short with steep gradients (Ref. 2.5). Grindstone Creek and Flat Branch are the two major named tributaries of the Hinkson in the City of Columbia area. The Grindstone enters Hinkson Creek approximately 1 mile (1.6 km) east of the reactor facility and ½ mile (0.8 km) south of Stadium Boulevard. Flat Branch enters the Hinkson approximately ¾ of a mile (1.2 km) southwest of the reactor facility site.

Hinkson Creek was formed after the regression of the Kansanian glacier. The stream ran into a preglacial valley filled with glacial drift, taking advantage of any irregularities. The downward cutting was controlled by the water level of the Missouri River, which was considerably higher than today due to the overland debris from the glacier. The area of the reactor facility site offers an excellent example of "incised meanders" caused by the downward lateral cutting of Hinkson Creek as the Missouri River was lowered. During this incising, the Hinkson entrenched itself from 80 to 120 feet (24.4 to 36.6 m) below the surrounding uplands (Ref. 2.6).

Hinkson Creek is a relatively shallow stream with frequent riffles, many pools, and rocky and/or dirt banks (Ref. 2.5). The stream bottom is variable; loose rock and gravel make up a good percentage, while silt and sand coverings of pool bottoms are frequent. Its greatest flow period normally occurs between April and July. The maximum recorded flow for the stream is 10,000 ft^3/sec (283,168 lps) which occurred on April 11, 1979. The average flow over a 15-year period (1967 to 1981) was 51.9 ft^3/sec (196.5 lps) with no flow typically recorded several times within a year.

Table 2-11 is a summary of Hinkson Creek flow-rates at Columbia, Missouri for the period stated (Ref. 2.10).

[1] In older records of the State of Missouri, Perche Creek was identified as Roche Perche Creek.

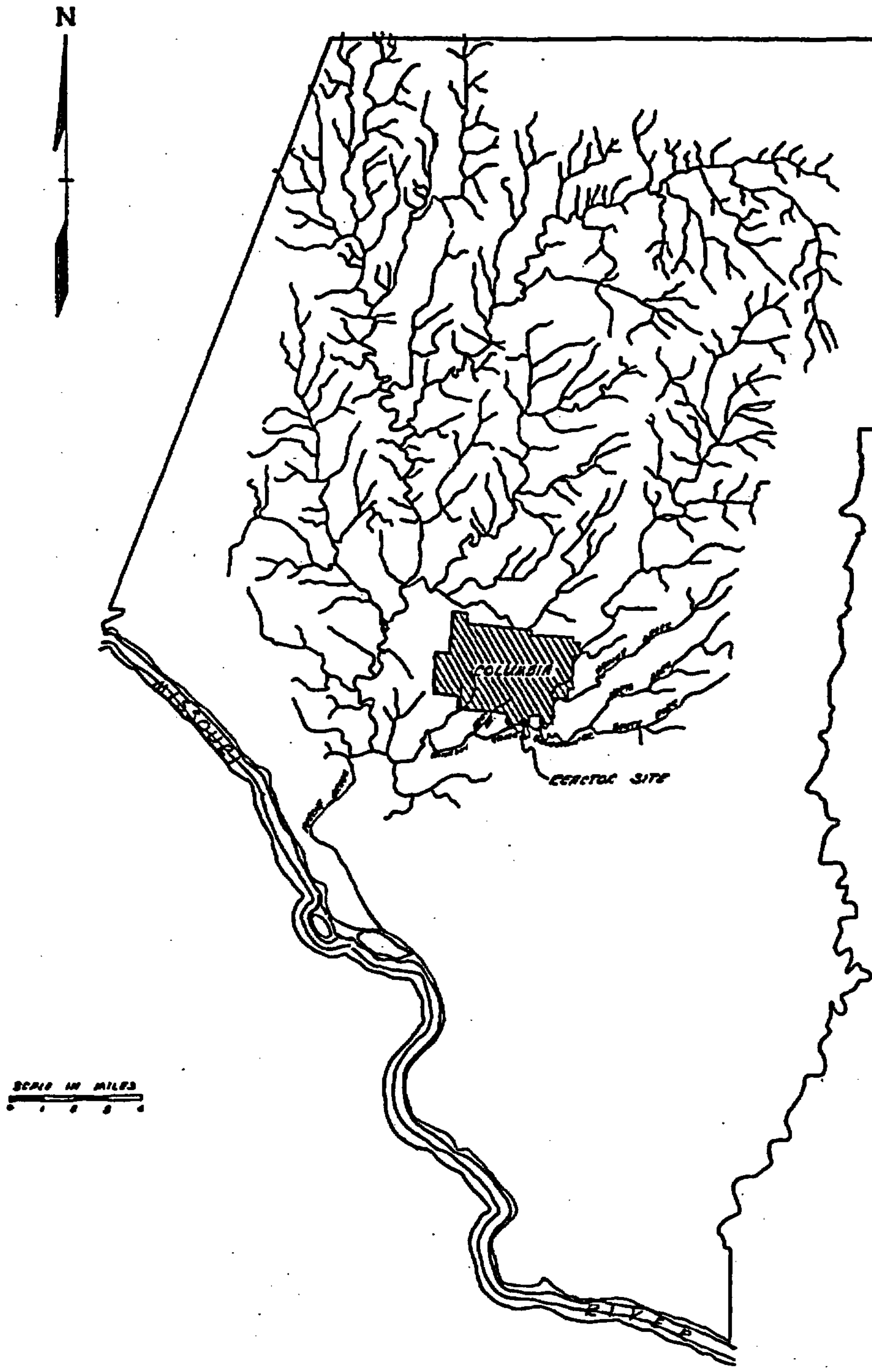


FIGURE 2.6
PERCHE CREEK DRAINAGE IN BOONE COUNTY, MISSOURI

TABLE 2-11
SUMMARY OF HINKSON CREEK FLOW-RATE
AT COLUMBIA, MISSOURI (1980-1981)

Month and Year	Average Flow (ft^3/s)
October - 1980	3.67
November - 1980	0.58
December - 1980	2.26
January - 1981	1.34
February - 1981	3.20
March - 1981	1.81
April - 1981	34.50
May - 1981	198.00
June - 1981	261.00
July - 1981	301.00
August - 1981	6.62
September - 1981	0.82
Grand Average for Period	67.90

Information based on Water Resources Data for Missouri, 1981; United States Geological Survey [data taken by gauging station No. 6910230 located on the left bank 400 feet (122 m) downstream from the bridge on State Highway 163 (Providence Road)].

2.4.2 Stormwater Drainage System

The City of Columbia and the adjacent areas tributary to the municipal stormwater drainage system are divided into eleven (11) main watersheds (Figure 2.7). These watersheds and their respective waterways include (Ref. 2.8):

1. Watershed No. 01 - Flat Branch of Hinkson Creek;
2. Watershed No. 02 - County House Branch of Hinkson Creek;
3. Watershed No. 03 - Harmony Creek;
4. Watershed No. 04 - Merideth and Goodwin Branches of Hinkson Creek;
5. Watershed No. 05 - Perche Creek;
6. Watershed No. 06 - Mill Creek;
7. Watershed No. 07 - Clear Creek;
8. Watershed No. 08 - Hominy Branch of Hinkson Creek;

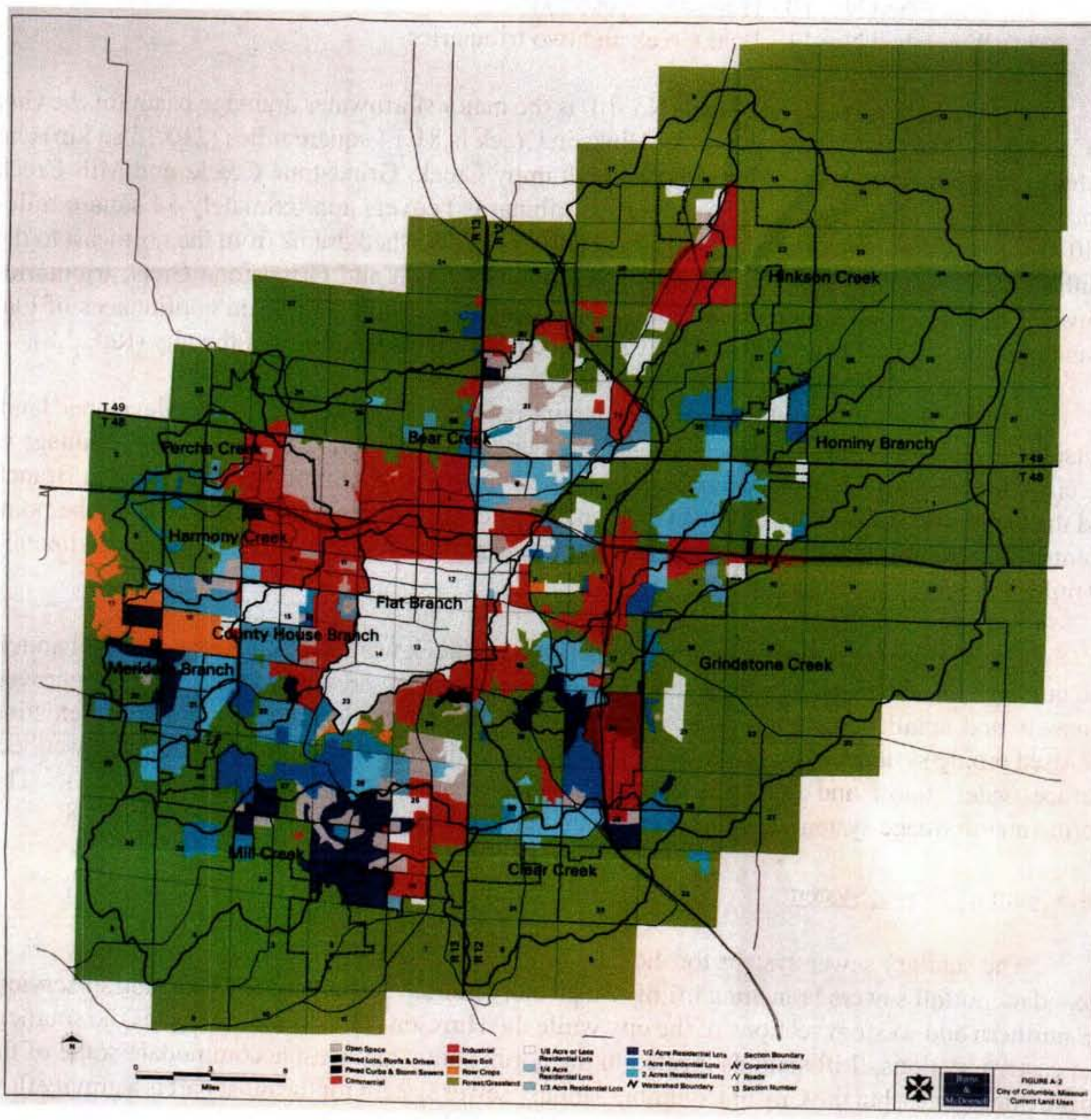


FIGURE 2.7
WATERSHEDS FOR THE COLUMBIA AREA

C-01

9. Watershed No. 09 - Grindstone Creek;
10. Watershed No. 10 - Hinkson Creek; and
11. Watershed No. 11 - Bear Creek and two tributaries.

The Hinkson Creek Watershed (No. 10) is the major stormwater drainage basin for the City of Columbia. The total tributary area for Hinkson Creek is 81.14 square miles (210.15 sq km) and includes Flat Branch, County House Branch, Hominy Creek, Grindstone Creek, and Mill Creek tributaries. The watershed (not including the tributaries) covers approximately 54 square miles (140 sq km) and lies mainly to the north of the city. The watershed drains from the northeast to the southwest and virtually bisects the city. The Hominy Branch and Grindstone Creek tributaries converge with Hinkson Creek on the southeast side of the city. The Hinkson confluences of Flat Branch, County House Branch, and Mill Creek occurs in the western part of the city (Ref. 2.8).

Within the city limits, the watershed contains residential, commercial, and undeveloped land. Outside the city limits, the watershed is considered undeveloped (Ref. 2.8). All surface drainage in the city from property identified as belonging to the University of Missouri goes to the Flat Branch and the Grindstone or directly into the Hinkson. The actual (water course) distance from the point of entry of any reactor facility site runoff water to the mouth of Perche Creek is approximately 19 miles (31 km).

The stormwater drainage system for the Hinkson Creek watershed consists of open channels and culverts. No enclosed piping systems of substantial size are presently installed. Unimproved channels and small culverts (pipe size less than 48 inches in diameter) comprise the existing enclosed piping systems. The enclosed systems are sporadic and noncontiguous and mainly collect surface water runoff and convey the flows to natural streams and channels (Ref. 2.8). The stormwater drainage system in the vicinity of the reactor facility site is shown in Figure 2.8.

2.4.3 <u>Sanitary Sewer System</u>

The sanitary sewer system for the City of Columbia (Figure 2.9) consists of a network of secondary outfall sewers branching off of two primary outfalls. The Perche Creek outfall services the northern and western sections of the city while the Hinkson Creek outfall services the southern and eastern sections. Lift stations located to the north, south, and west accommodate some of the flow in these areas, but the City of Columbia sanitary sewer system for the most part is a gravity flow system.

The Columbia Regional Wastewater Treatment Plant is located in the southwest section of the city, approximately 3.7 miles (6 km) west of the reactor facility. This is an activated sludge/anaerobic digester treatment facility with a design flow capacity of 20.6 million gallons (78 million liters) per day (design population equivalent of 174,058 people). After processing, the effluent from the treatment facility is diverted to a series of constructed wetlands which serve as polishing units to further purify the discharge. The effluent is then directed from these "polishing" wetlands to a final wetlands area maintained by the Missouri Department of Conservation known

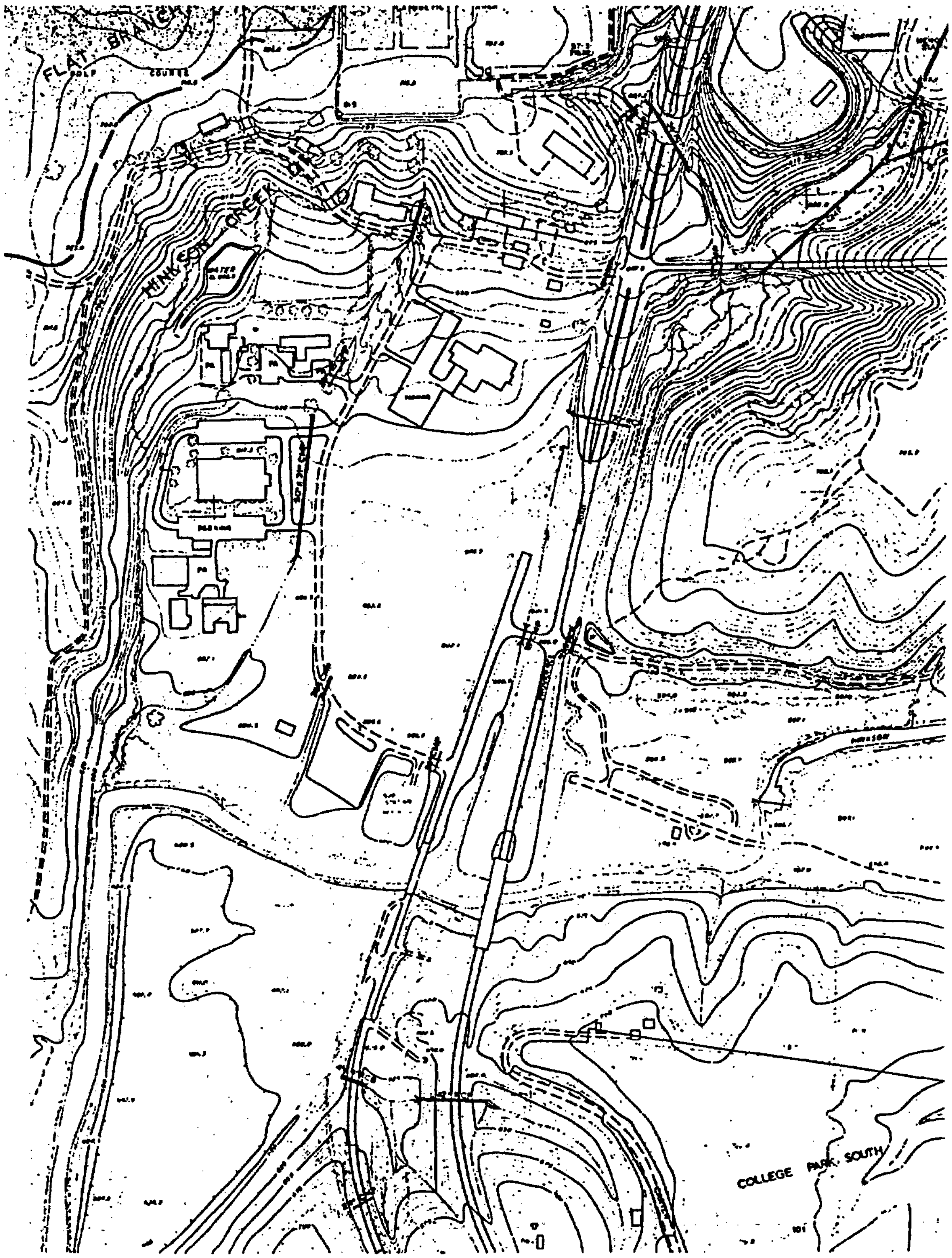


FIGURE 2.8
STORMWATER DRAINAGE SYSTEM FOR THE REACTOR FACILITY SITE

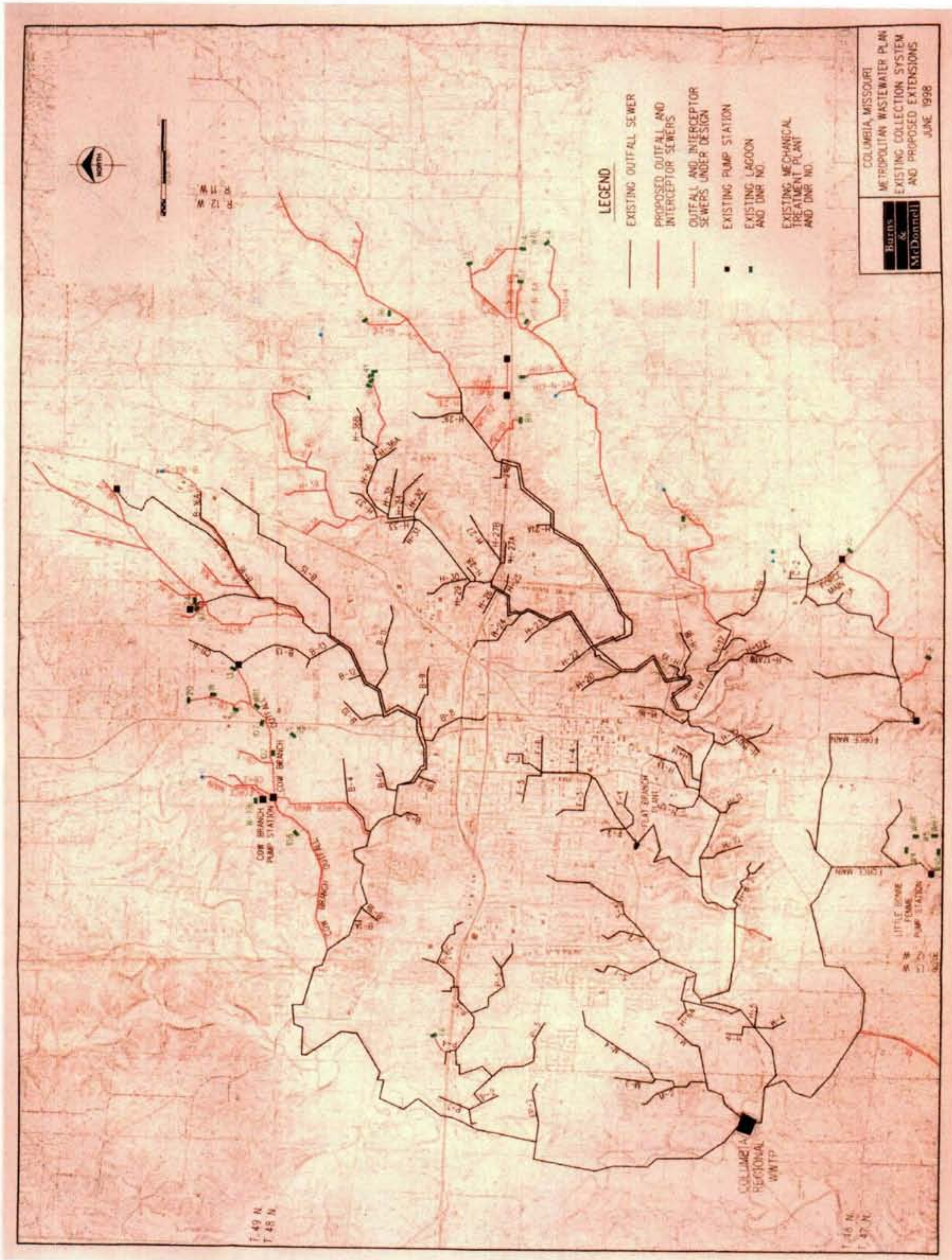


FIGURE 2.9
CITY OF COLUMBIA SANITARY SEWER SYSTEM

C-02

as Eagle Bluffs, located adjacent to the Missouri River, approximately 5 miles (8 km) south of the wastewater treatment plant.

The university's sanitary sewer system discharges directly into the City of Columbia's sanitary sewer system at several locations throughout the campus. Sewage from the reactor facility flows southward through a 12-inch line (H-12) which runs adjacent to Research Park Drive, eventually discharging into the 24-inch Hinkson Creek outfall sewer near Old Providence Road (Figure 2.10).

2.4.4 Ground Water

In addition to its numerous rivers, lakes, and streams, Missouri has a vast supply of groundwater within its underlying sediments and bedrock. The largest quantities are located in the alluvial sediments adjacent to major streams and rivers. Small to moderate quantities are found in the bedrock aquifers in northern and southern Missouri (Ref. 2.9).

The City of Columbia originally obtained its water from deep bedrock wells located around the metropolitan area. Declining well output and limited expansion sites prompted the city to seek an alternative water supply source. In 1972, a series of wells were installed in the McBaine Bottom of the Missouri River flood plain approximately 12 miles (19.3 km) southwest of Columbia. This wellfield consists of 12 alluvial wells approximately 100 feet (30.5 m) deep with a total design capacity of 24 million gallons (91 million liters) per day. Presently, four of the original metropolitan deep wells are maintained in a "standby state," providing an emergency backup supply to the McBaine alluvial wellfield (Ref. 2.9). The remainder of the city's deep wells are inactive and are no longer considered operational. The deep wells are located throughout the city, typically near storage tanks or reservoirs, with the closest standby well approximately 2.3 miles (3.7 km) northwest of the reactor facility. The average depth of the standby wells is 1,320 feet (402 m).

The university maintains five deep wells, each with varying flow rates, servicing the entire Columbia campus (Table 2-12). The wells are located on property owned and controlled by the university. Seven water ties connecting the City of Columbia's water supply system with the campus' water supply system are installed at various locations on campus. These water ties provide an emergency source of water from the city to the university in the event a section of the system is secured for maintenance or a pipe leak.

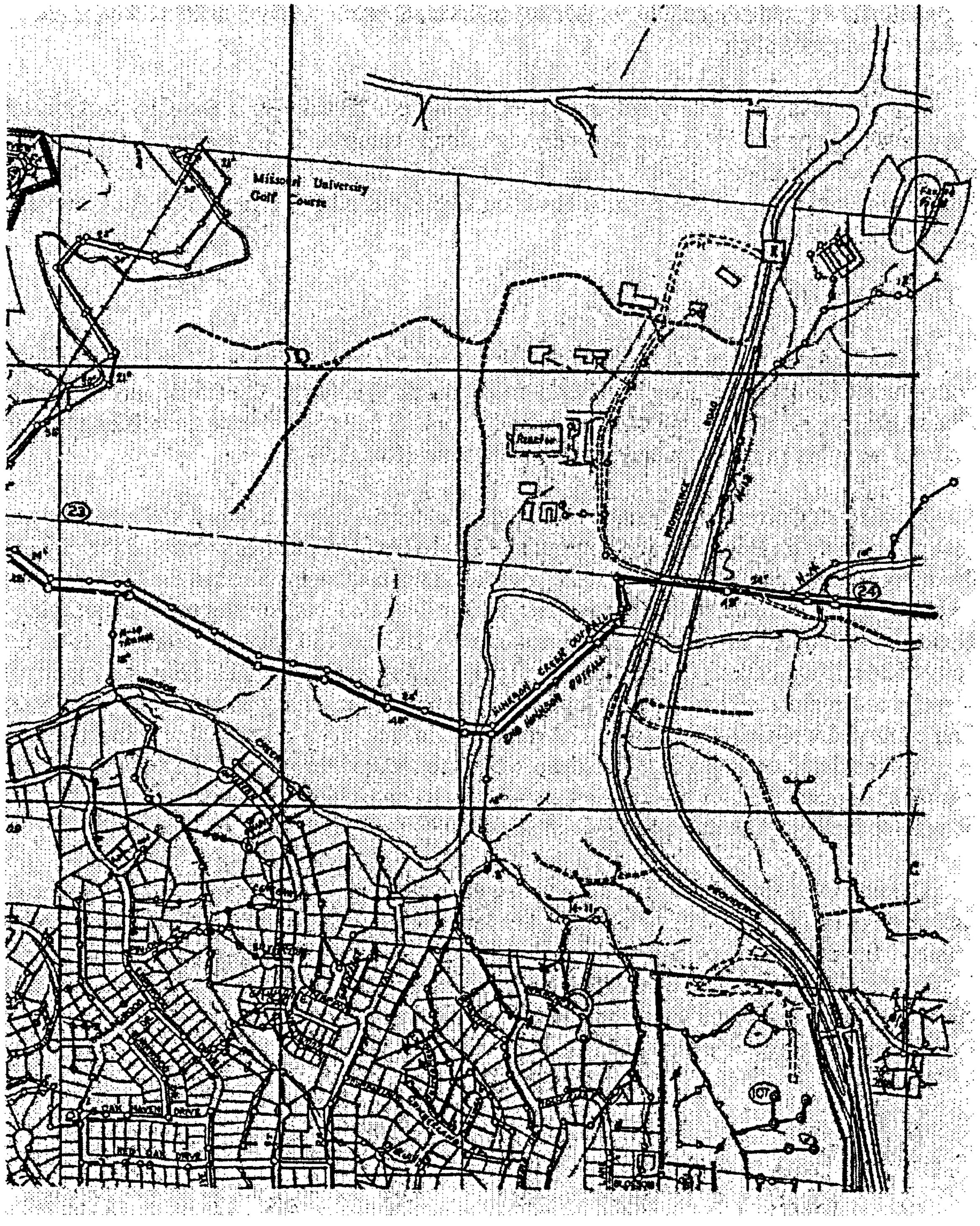


FIGURE 2.10
UNIVERSITY SANITARY SEWER SYSTEM

TABLE 2-12
UNIVERSITY OF MISSOURI DEEP WELLS

Well Designation	Year	Depth (feet)	Supply Rate (gpm)	Reservoir Capacity (gallons)
West Well (No. 1)	1938	1,200	1,000	600,000
East Well (No. 2)	1938	1,237	700	600,000
South Well (No. 3)	1959	1,237	1,000	950,000
Southwest Well (No. 4)	1968	1,440	1,000	1,500,000
North Well (No. 5)	1992	1,415	1,000	250,000

In addition to those maintained by the city and the university, there are (in 1993) 21 other deep wells supplying major water users for industrial and/or commercial purposes (Missouri Department of Natural Resources, 1995) in the Columbia metropolitan area. A major water user is defined by the state as a water user capable of pumping or diverting 100,000 gallons (378, 541 liters) per day or more from surface or groundwater sources.

The City of Columbia's (current usage) wellfield is located on the Missouri River alluvial flood plain near the town of McBaine (Ref. 2.9). At this site, the flood plain is approximately 2 miles (3 km) wide and bounded by limestone bluffs. The alluvium is a fining-upward sequence of cobbles, gravel, sand, silt, and clay which parallels and underlies the Missouri River. The majority of the alluvium was deposited by the river (and to a lesser extent by glacial processes) and continues to be deposited today by mechanisms such as fluctuations in river stage, meandering (sand bars), and flooding (overbank deposits). The granular nature of this alluvium, in its deeper portions, renders it excellent for storing and transmitting water. This alluvial aquifer is replenished by the Missouri River and from direct rainfall infiltration. It is semi-confined to confined by the silt and clay deposits present at the surface. The Missouri River carries roughly 45 billion gallons (170 billion liters) of water per day through the reach near McBaine [United States Geological Survey (USGS), 1990]. It has low discharges in the winter months and average to high discharges the remainder of the year. During the winter months, the river acts as an influent stream (gaining water from the aquifer); during the summer months, the river acts as an effluent stream (losing water to the aquifer).

The underground formations which supply water to the deep wells in the Columbia metropolitan area are generally referred to as the Ozark Aquifer. The Ozark Aquifer is a sequence of dolomite and sandstone of Upper Cambrian through Ordovician age and is sometimes referred to as the Cambrian-Ordovician Aquifer. It has a thickness of approximately 1,300 feet (396 m) in the Columbia vicinity and is encountered at depths ranging from 150 to 400 feet (46 to 122 m) below the ground surface (Ref. 2.10). Pennsylvanian, Mississippian, and Devonian-aged strata overlay the Ozark Aquifer in this area. The generalized geological column for the area is discussed in greater detail in Section 2.5.1, Regional and Site Geology. The deepest formations in the aquifer that are hydrologically important are the Eminence and Potosi dolomites. These formations have good permeability and can give yields of 400 to 1,100 gpm (1,514 to 4,164 lpm) (Ref. 2.10). The four city wells that are maintained in emergency standby have yielded from 700 to 1,200 gpm (2,650 to 4,542 lpm) (Ref. 2.9).

2.4.5 Flooding

During a recent evaluation (1998) of the City of Columbia Stormwater Management Plan, a computer model developed by the U.S. Army Corps of Engineers Hydrological Engineering Center was used for modeling the existing and future stormwater drainage systems for the city (Ref. 2.8). The HEC-1 "Flood Hydrograph Package" generates stream flow or runoff hydrographs at desired locations for specific storm events. The model simulates a defined area's surface runoff and its subsequent routing through various drainage system components including pipes, channels, and reservoirs. Rating curves were used to determine new water surface elevations for the flow rates calculated by the hydrologic models. The design storm used in the analysis of the drainage systems is the pattern of rainfall over a specific period, or duration, for a given return period. A storm duration of six hours (360 minutes) was used to ensure that modeling reflected peak flow when all areas of the watershed were contributing to runoff. Experience in modeling with HEC-1 has shown that a 6-hour storm duration produces results which correlate well to other accepted synthetic hydrological models (Ref. 2.8).

Rainfall data for the City of Columbia is based on records obtained from the Missouri Department of Transportation (MoDOT). The following mathematical relationship between storm duration, return period and rainfall intensity was used to calculate rainfall depth and intensity (Ref. 2.8);

$$I = A/(B + t_t)^m;$$

where:

I = rainfall intensity (inches/hour);
t_t = storm duration (minutes);
A = curve fitting parameter;
B = curve fitting parameter; and
m = curve fitting parameter.

Table 2-13 shows the rainfall intensity during the design storm period of six hours. Table 2-14 shows the rainfall depth within the same storm period.

TABLE 2-13
RAINFALL INTENSITY (inches/hour)

Return Period (years)	Storm Duration (minutes)							
	5	15	20	30	60	120	180	360
2	5.46	3.68	3.19	2.54	1.62	0.98	0.72	0.42
5	6.41	4.44	3.87	3.10	1.99	1.20	0.88	0.51
10	7.21	5.04	4.41	3.56	2.32	1.42	1.05	0.62
25	8.37	5.91	5.19	4.20	2.75	1.70	1.26	0.74
50	9.24	6.55	5.76	4.68	3.09	1.93	1.44	0.85
100	9.97	7.10	6.24	5.07	3.33	2.06	1.53	0.90

TABLE 2-14
RAINFALL DEPTH (inches)

Return Period (years)	Storm Duration (minutes)							
	5	15	20	30	60	120	180	360
2	0.45	0.92	1.06	1.27	1.62	1.97	2.17	2.52
5	0.53	1.11	1.29	1.55	1.99	2.41	2.64	3.04
10	0.60	1.26	1.47	1.78	2.32	2.85	3.16	3.70
25	0.70	1.48	1.73	2.10	2.75	3.39	3.77	4.42
50	0.77	1.64	1.92	2.34	3.09	3.86	4.31	5.13
100	0.83	1.77	2.08	2.53	3.33	4.12	4.58	5.39

A hydrologic model of the Hinkson Creek watershed was prepared using the HEC-1 hydrologic package and the above hydrological data to estimate peak runoffs for various points in the watershed. The calculated flows were compared with those published by the Federal Emergency Management Agency (FEMA). The calculated flows for the City of Columbia Stormwater Management Plan closely approximate the values accepted by FEMA. Table 2-15 is a comparison of the calculated flows with those of FEMA (Ref. 2.8).

There are no credible hydrological events which can lead to flooding or other water-induced damage to the reactor facility. The most recent assessment of the Hinkson Creek watershed shows no significant increases in the flood stages from the established FEMA values. Any physical alterations, man-made or natural, have not significantly altered the hydraulics of the watershed. The closest point to the Hinkson Creek 100-year flood plain is about 200 feet (61 m) south of the facility (Figure 2.11).

TABLE 2-15
HINKSON CREEK WATERSHED FLOWS (ft^3/sec)

Location	Drainage Area (mi^2)		Return Period 10 year	50 year	100 year
MKT Railroad Bed	76.08	HEC-1	8,446	17,880	20,566
		FEMA	10,700	17,800	24,000
Providence Road (Route K)	67.19	HEC-1	8,560	17,293	19,673
		FEMA	9,900	16,100	21,800
Ashland Gravel Road	50.41	HEC-1	5,390	10,486	11,841
		FEMA	8,000	12,800	17,600
East Broadway Street	43.19	HEC-1	4,298	8,030	9,015
		FEMA	7,100	11,500	15,700
Interstate Highway 70 (I-70)	41.02	HEC-1	4,063	7,518	10,430
		FEMA	6,850	11,100	15,000

2.4.6 Accidental Release of Liquid Effluents

The probability of an accidental release of radioactive effluents from the MURR to the surrounding surface waters is extremely low. Three (3) MURR systems may contain contaminated liquid: the primary coolant, the pool coolant and the radioactive liquid waste retention and disposal system. The radioactive liquid waste retention and disposal system collects potentially contaminated liquids from the laboratory sinks, containment and laboratory building below grade level floor drains, and any location where radioactive liquid waste is routinely generated. The primary and pool coolant systems are, by design, contained to the maximum extent possible. However, certain reactor maintenance operations do result in small amounts of primary or pool coolant being drained from the system, but this liquid is easily collected at the point of origin and normally reused in the system. All components for these systems (heat exchangers, pumps,

FIGURE 2.11
REACTOR FACILITY SITE - 100 YEAR FLOOD PLAIN

filters, valves, tanks, piping, etc.) are located within the MURR containment and laboratory buildings. Any contaminated water leakage from this equipment will be wiped up and disposed of as discussed in Chapter 11. Liquid that is collected in the radioactive liquid waste retention and disposal system is retained or chemically treated until an assay indicates that activity levels are less than the limits specified in 10 CFR 20.2003 for disposal by release into sanitary sewerage.

Safe operation and shutdown of the reactor would not be precluded by a hydrological event. The MURR facility design gives reasonable assurance that contamination of ground and surface waters at the site from inadvertent release by leakage of primary coolant, neutron activation, or airborne releases would not exceed applicable limits.

2.5 Geology, Seismology, and Geotechnical Engineering

2.5.1 Regional and Site Geology

The site of the reactor facility and its immediate vicinity is representative of central Boone County in that most of its topography is sloped along streams running through the area. The elevation varies from 650 feet (198 m) at the flood plains of Hinkson Creek where the University Research Park is located, to 700 feet (213 m) at the peak of the slopes. The site is at a point just south (approximately 5 miles) of a change in the geology of central Boone County. The site is located on the edge of the Osagean series which is characterized by its crinoidal, very cherty, crystalline, and fossiliferous limestones. To the north, the Desmoinesian series of the Pennsylvanian system is represented by the cabaniss subgroup. This consists of sandstone, siltstone, shale, underclay, limestone, and coal beds (Ref. 2.11). Coal, formerly an important mineral of Columbia, was at one time mined a few miles north of the city from this Pennsylvanian system (Ref. 2.12). The Kansasian glacial stage of the Pleistocene laid a glacial till (clay, sand, and pebbles) in Boone County which was later covered by loess deposition following glaciation (Figure 2.12).

The soils in the Columbia area are generally fine-grained with moderately pervious surface soils and less pervious subsoils. The soils are typically classified by the Soil Conservation Service (SCS) system with the hydrological Groups C and D being the most prevalent (Ref. 2.9). Table 2-16 provides a description of the soils shown in Figure 2.13.

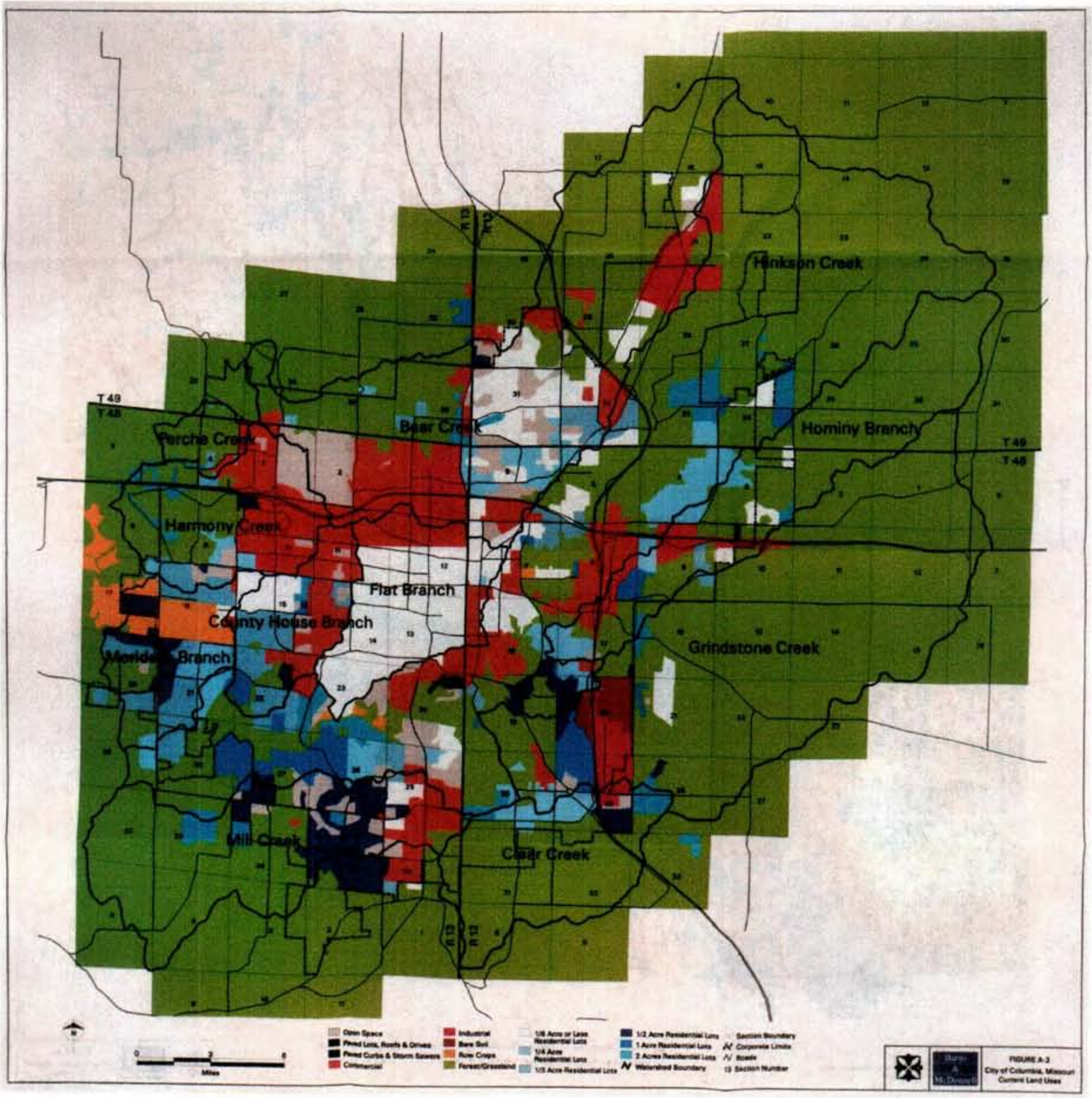


FIGURE 2.12
SOILS OF BOONE COUNTY, MISSOURI

C-03

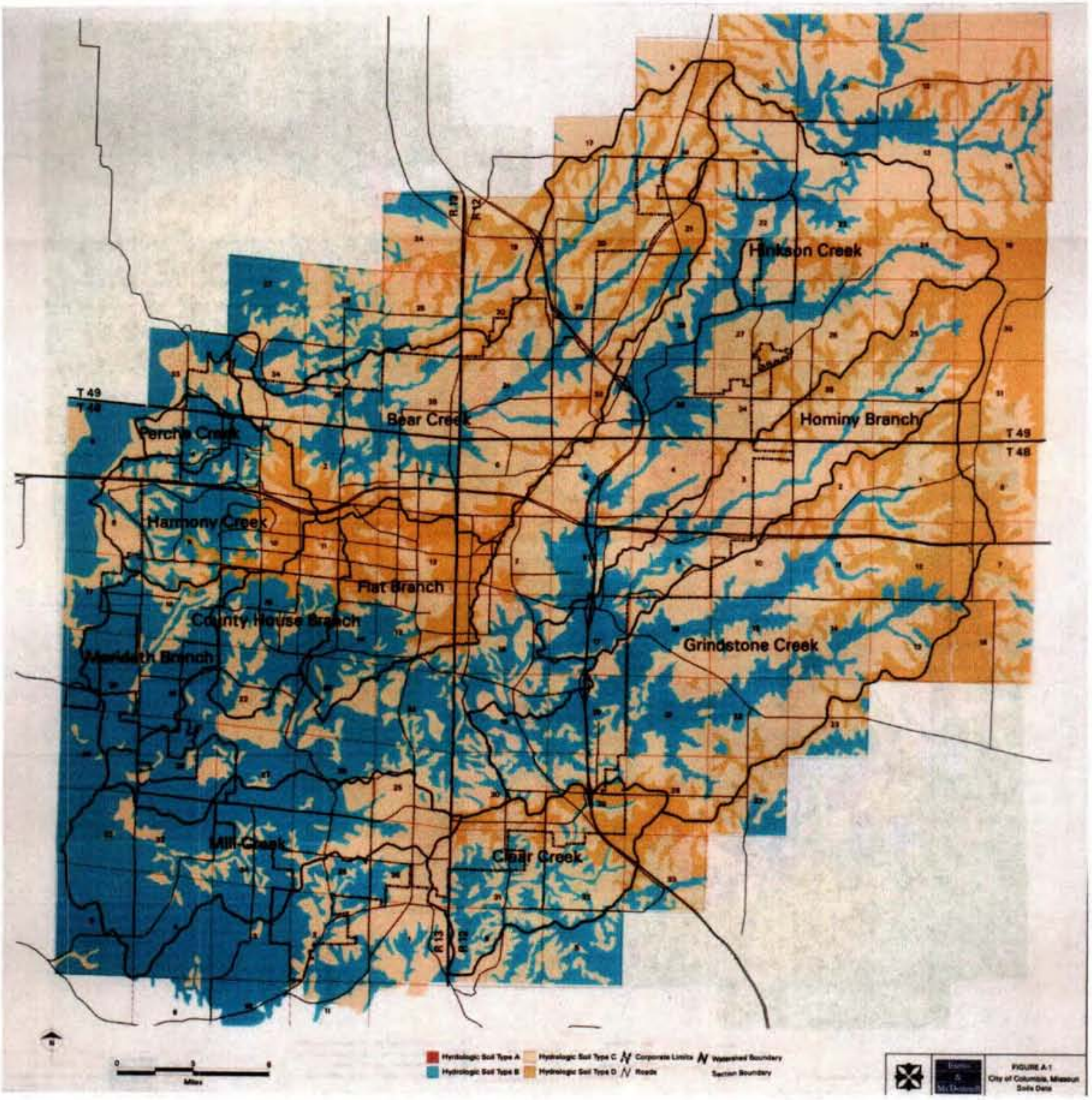


FIGURE 2.13
SOILS DATA FOR THE COLUMBIA AREA

C-04

TABLE 2-16
HYDROLOGICAL SOILS GROUPS

Soils Code	Hydrological Soils Group	Description
1	A	Sand, Loamy Sand or Sandy Loam
2	B	Silt Loam or Loam
3	C	Sandy Clay Loam
4	D	Clay Loam, Silty Clay Loam, Sandy Clay, Silty Clay or Clay

The soils near the reactor facility site are alluvial and tend to be stratified along the slopes. A golf course to the west, the University Research Park, and fields adjacent to the park that have been cleared once represented a continuation of the woodland which is now restricted to the slopes with steep, stony terrains. The University Research Park and nearby areas are covered with Shirley silt loam of high fertility and favorable depth representative of Hinkson Creek alluvium. The golf course to the west and the residential area to the north of the reactor facility are underlain by a thin silty loam. The sloped woodlands are covered by loose, thin silty loam of low fertility (Ref. 2.13).

The area throughout the site is underlaid by a considerable thickness of sedimentary rocks, primarily carbonates, limestone, and dolomite. These rock formations are essentially horizontal, dipping only slightly to the north. The immediate vicinity of Columbia is relatively free of faults, folds, and other major structural features. The generalized geologic column for this area is shown in Figure 2.14. Figure 2.15 shows the outcrop pattern of the formations in the Columbia area. The general features of the separate units and their water bearing characteristics are described below (Ref. 2.2).

- Glacial Deposits - These deposits mantle the upland areas and consist of a heterogenous mixture of clay, sand, and pebbles of diverse rock types. They vary greatly in thickness and are as much as 140 feet (42.7 m) thick in the northern part of Boone County. This material is relatively impermeable and supplies very little water to wells.

- Pennsylvanian - This part of the column consists largely of clay and shale with minor amounts of coal and thin, impure limestone. The total thickness may be as much as 100 feet (30.5 m). These beds produce only small quantities of water and are not used in this area as a source of supply. The water in them is usually high in iron and sulphur.

- Burlington Limestone - This formation is a coarse-grained limestone with many beds and nodules of chert. It is the principal limestone exposed in quarries, creek banks, and road cuts around Columbia. It may also contain minor amounts of pyrite and limonite. It is about 160 feet (48.8 m) thick in the Columbia area, but the thickness is variable. The Burlington

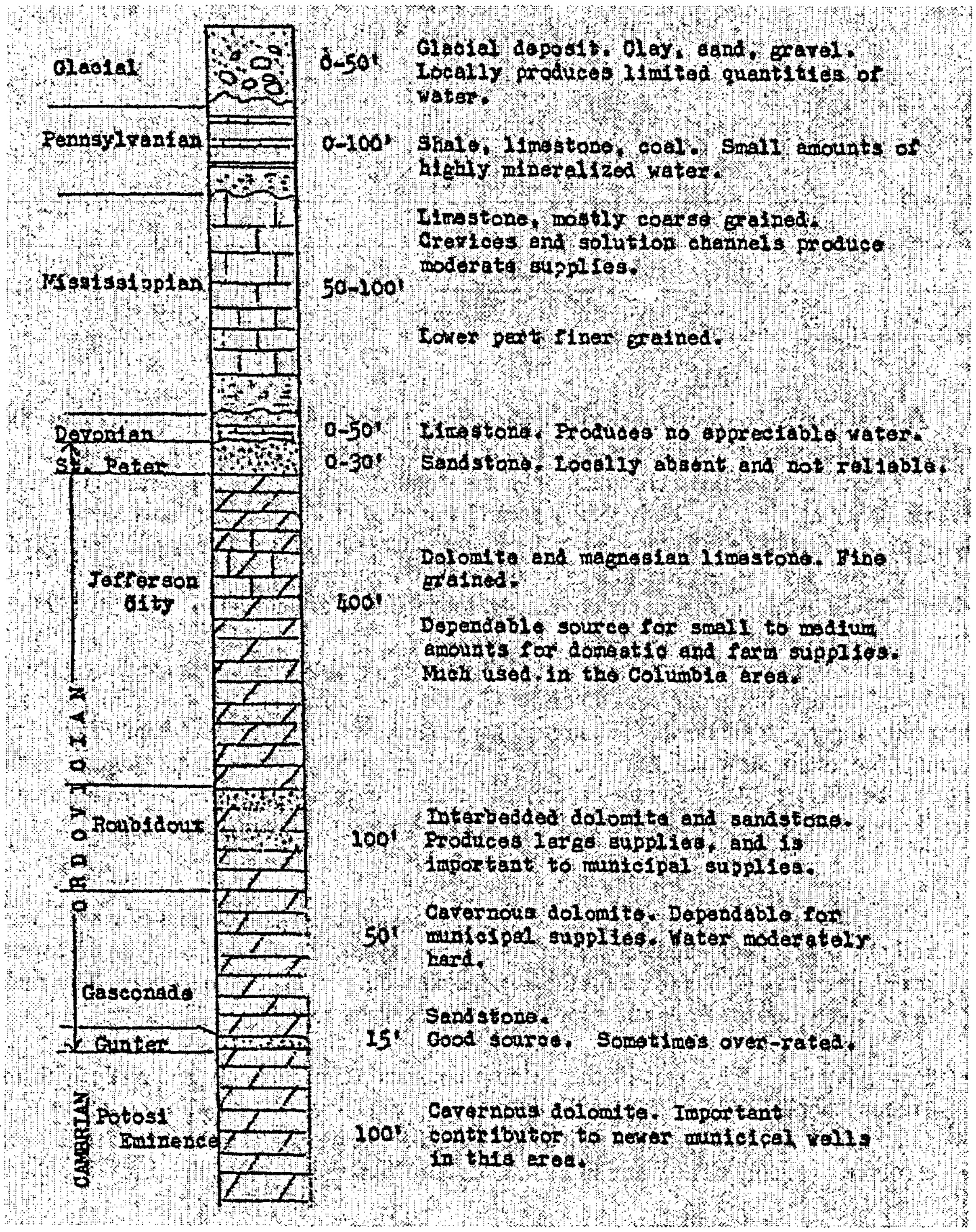
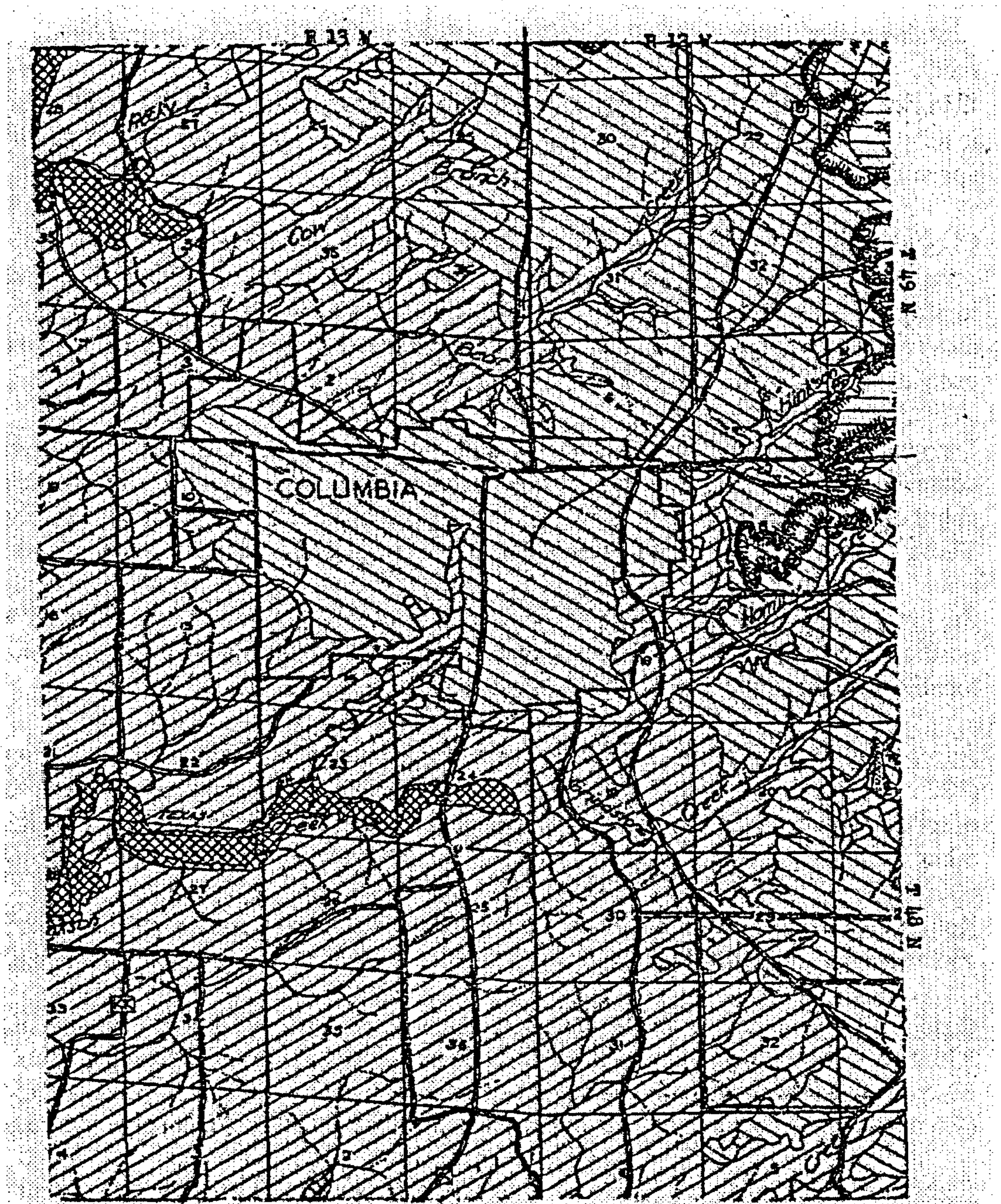


FIGURE 2.14
GEOLOGIC COLUMN AT COLUMBIA, MISSOURI



GEOLOGIC MAP OF COLUMBIA AREA
Scale 1" to 1 mile

FIGURE 2.15
GEOLOGIC MAP OF THE COLUMBIA AREA

contains many relatively shallow drilled wells and yields sufficient quantities of relatively hard water for rural domestic supplies. The limestone is relatively soluble and the Burlington contains many caverns and solution passages. Where it is a surface formation, it forms areas of karst topography; the closest area of karst to the reactor facility site is approximately 5 miles (8 km) southeast near Pierpont.

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

- Devonian - These limestones are variable in lithology and range from very fine-grained to coarsely-textured beds. Some are slightly sandy. In the Columbia area, these beds are only about 30 feet (9 m) thick and in some of the Columbia wells they seem completely absent. These Devonian beds are not valuable as water producers.

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

- Jefferson City Formation - This predominantly dolomite formation averages about 400 feet (122 m) in thickness in the Columbia area and wells drilled in it produce moderate quantities of relatively hard water. It probably has more rural domestic wells terminating in it than any other formation in this area.

- Roubidoux Formation - This formation consists of alternating sandstone and dolomite beds and averages about 100 feet (30.5 m) in thickness. It is a very dependable water producer.

- Gasconade Formation - This unit consists mostly of light gray dolomite with a sandstone (Gunter) at the base. The thickness is about 280 feet (85.3 m). This dolomite unit is very cavernous and contains many interconnected solution passages. The sandstone, about 15 feet (4.6 m) thick, is very permeable and has a wide areal extent and is a good source of water.

- Cambrian Dolomites - The lower 400 to 450 feet (122 to 137 m) of the column consists largely of dolomite which is somewhat cavernous and which contains a few thin beds of sandstone. This thickness contributes considerable quantities of water to the deeper wells.

All of the aquifers from the Upper Cambrian through Ordovician age (inclusive) are now generally referred to as the Ozark Aquifer (Section 2.4.4).

2.5.2 Seismology

2.5.2.1 Introduction

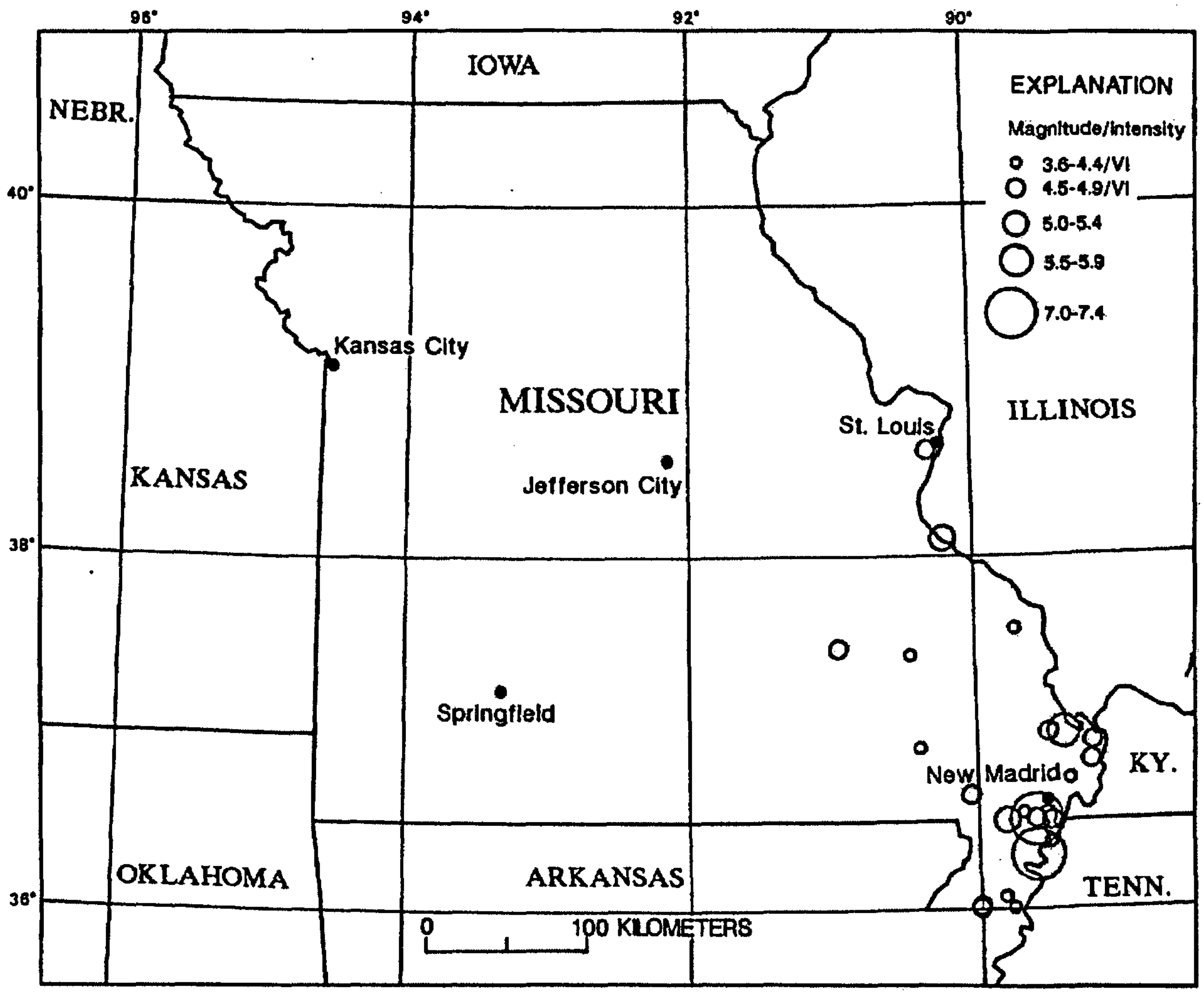
The most significant seismological feature in Missouri is the New Madrid Seismic Zone, located in the southeastern corner of the state near the "bootheel" and extending into parts of the contiguous states of Arkansas, Tennessee, Kentucky, and Illinois. In 1811 and 1812, this zone was the location of some of the highest intensity seismic events ever noted in the United States (Ref. 2.29). This area contains a deep, structurally complex fault system which is still very active. Numerous earthquakes originate in this area yearly, the great majority of which are of small magnitude. This zone can be fairly well defined by the registered location of seismic events (Figure 2.16). The linear distance from this zone to the reactor facility site is approximately 200 miles (322 km). While the potential for a significant earthquake originating from this zone exists, the distance to the MURR site would mitigate its effects.

The western section of the state has, in the past, experienced some low intensity seismic activity assumed to be associated with the Nemaha Anticline, a northeast-southwest trending subsurface structure extending along a general line from Omaha, Nebraska, through eastern Kansas, and terminating near Oklahoma City, Oklahoma (Ref. 2.17).

The remainder of the state, including the MURR location, is typical of the stable midcontinent United States. While this area is not immune to seismic activity, the intensities of midcontinent earthquakes are generally mild, although they are usually felt over wide areas. This area of the state does contain some localized faulting perhaps associated with some minor seismic activity, mainly in the eastern areas near Ste. Genevieve and Crystal City (Ref. 2.14). Boone County, however, shows considerable stability and has little evidence of past seismic activity.

2.5.2.2 Earthquake Districts

Based upon the number and location of recorded earthquakes, six more or less distinct zones of seismicity have previously been described as existing within or affecting the State of Missouri. These zones, or districts, are named for their locations, and are shown in Figure 2.17. They are: New Madrid; St. Marys; St. Louis; Hannibal; Springfield; and Northwestern Missouri. Of these, the New Madrid District is by far the most active, followed by the St. Marys District (Ref. 2.16).



Earthquakes in Missouri with magnitudes ≥ 4.5 or intensity ≥ VI.

FIGURE 2.16
NEW MADRID SEISMIC ZONE

- The New Madrid District

The New Madrid District, the epicenter of the well-known earthquakes of 1811 and 1812, is confined to the north end of the Mississippi Embayment region (Refs. 2.12, 2.15). The borders of the district extend southwestward from Cape Girardeau to the vicinity of Popular Bluff, Missouri, and from Cape Girardeau south-southeast along the eastern margin of the Mississippi River flood plain. Little is known about the basement configuration of the faults, which cause the earthquakes, except that most of the epicenters lie below the towns of New Madrid and Caruthersville, Missouri.

The term, "New Madrid earthquake" is applied to a series of shocks beginning in 1811 and continuing into the early part of 1812. The most violent shocks occurred on December 16, 1811, and January 23 and February 7, 1812 and caused several topographic changes in the area, the most notable being the formation of Reelfoot Lake, Tennessee. These shocks probably generated an intensity as high as a Modified Mercalli (MM) XI to XII and reportedly were felt over an area of some 1,000,000 square miles (2.6 million square km) (Ref. 2.29). No subsequent earthquake in the United Sates has been felt over a wider area (Ref. 2.16). Structural damage from these earthquakes was difficult to evaluate due to the sparse population and related absence of dwellings and other structures.

Since 1812, numerous shocks have been felt in the New Madrid District. Earthquakes in 1843, 1865, 1883, 1895, 1903, 1905, 1920, and 1934 have caused structural damage, consisting mostly of cracked walls and damage to unsupported masonry such as chimneys. None has ever approached the intensity of the earthquakes of 1811 and 1812.

- The St. Marys District

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

- Other Districts

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

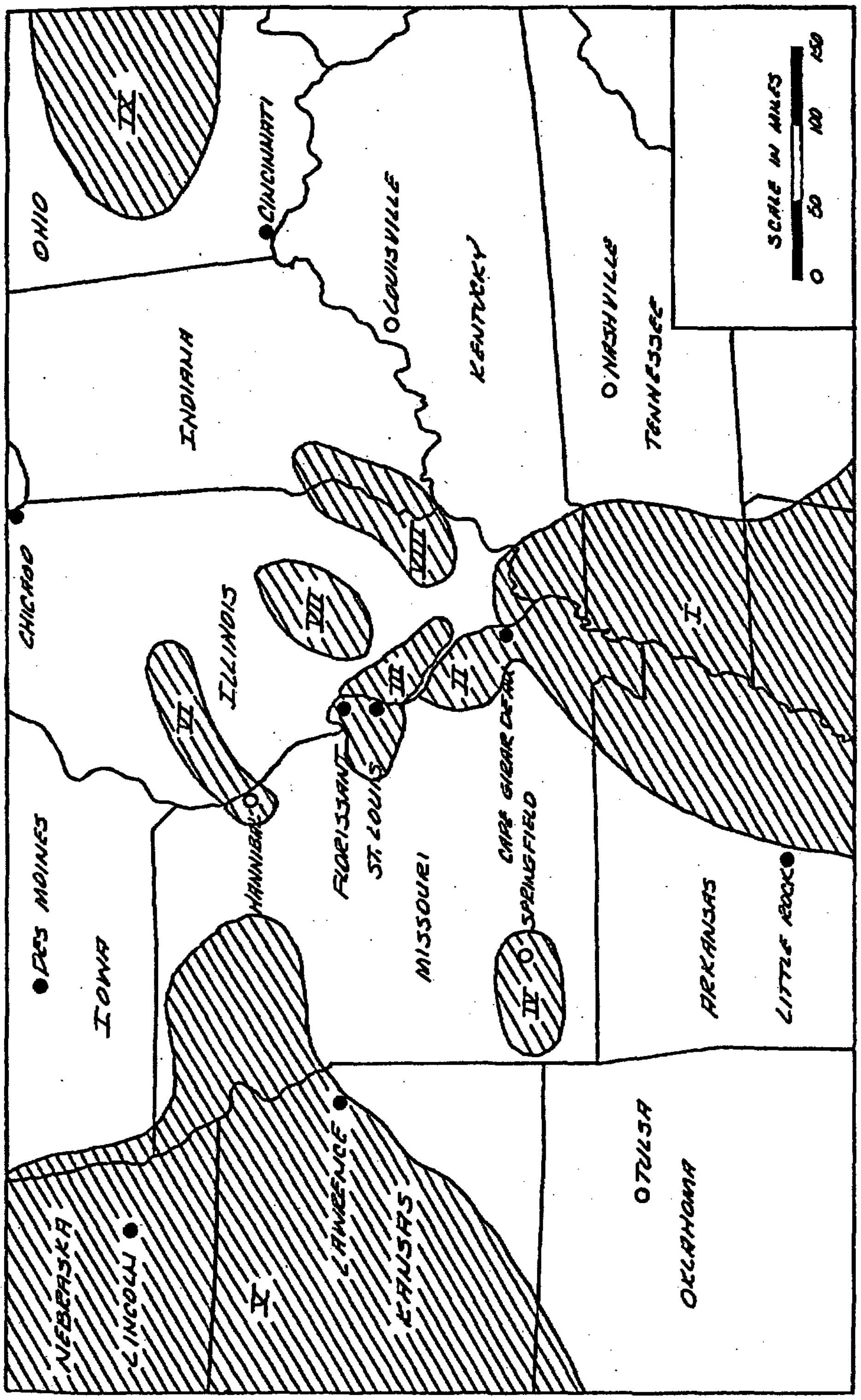


FIGURE 2.17
EARTHQUAKE DISTRICTS IN THE STATE OF MISSOURI

The following table records the number of earthquakes occurring by district from 1920 to 1939.

TABLE 2-17
EARTHQUAKES OF MISSOURI ORIGIN AS CLASSIFIED BY SEISMIC AREAS
1920 to 1939

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

About 85% of all earthquakes in Missouri were of slight to moderate intensity (Table 2-18). Only 10% of these were destructive to property and dangerous to human life and 2.5% of this value was due to the New Madrid earthquakes of 1811-1812 alone.

2.5.2.3 Local Seismology

Although a few local faults exist, no earthquakes are known to have originated in Boone County. The largest fault is exposed in Fox Hollow in the Southeast ¼ of Section 12, Township 46N, Range 13W [approximately 10.5 miles (17 km) south of the reactor facility], where it has a throw of about 120 feet (36.6 m). There are other smaller faults associated with the Browns Station anticline in the northern part of the county. These structures are early Paleozoic in age, they are not active, and they do not constitute an earthquake hazard (Refs. 2.2 and 2.18).

2.5.2.4 History of Earthquakes

The following is a chronology of the history of earthquakes felt in Columbia, Missouri.

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

- 1843, January 4, New Madrid, Missouri
  A strong earthquake occurring in the New Madrid District was felt from the seacoast of Georgia to beyond the western frontier posts. It was probably felt in Columbia, but the intensity of the shock is not known.

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

- 1886, August 31, Charleston, South Carolina
  The well-known earthquake occurring on this date had a MM intensity of II at St. Louis. It was felt as far westward as Columbia, however, there was no report of structural damage.

- 1895, October 31, Charleston, Missouri
  An earthquake that damaged every building in the commercial block of Charleston was felt slightly in Columbia; no damage was reported. In the New Madrid District this shock probably ranks second in intensity to the New Madrid series of 1811-1812.

- 1917, April 9, Ste. Genevieve, Missouri
  A sharp disturbance at Ste. Genevieve and St. Marys, Missouri had a Rossi-Forel intensity at Columbia of IV and a MM intensity of IV. At the epicenter it had a Rossi-Forel intensity of VI. According to the Daily Missourian, Number 187, April 9, 1917, the earthquake was not felt in Columbia. However, on the following day several people reported feeling the shock and attributed it to an explosion. No damage was reported.

- 1920, May 1, St. Louis, Missouri
  An earthquake on this date shook buildings in the entire St. Louis area. Two shocks were felt in Mt. Vernon, Illinois, and three were felt in Centralia, Illinois. An erroneous entry in the Earthquake History of the United States, as compiled by the U.S. Coast and Geodetic Survey, gives Columbia, Missouri, as the epicenter. The actual epicenter of this earthquake is unknown. It is thought to have originated well to the east of Columbia, probably in Illinois. In the Evening Missourian, Number 207, May 1, 1920, the U.S. Weather Bureau reported that the shock was not felt in Columbia. In a later investigation, however, a few people reported feeling a slight tremor.

TABLE 2-18
INTENSITIES OF EARTHQUAKES FELT IN MISSOURI

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

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

Additionally, the National Geophysical Data Center (NGDC) lists the following three earthquakes that may have also been felt in the City of Columbia.

- <u>1939, November 23, Western Illinois</u>
  An earthquake occurred near Red Bud, Illinois that had a MM intensity of III in Columbia. The approximate distance from the epicenter to Columbia was 213 km (132 miles).

- <u>1963, March 3, South-East Missouri</u>
  An earthquake occurred near Menorkanut, Missouri that had a MM intensity of III in Columbia. The approximate distance from the epicenter to Columbia was 317 km (197 miles).

- <u>1965, October 21, Eastern Missouri</u>
  An earthquake occurred near Brazil, Missouri that had a MM intensity of V in Columbia. The approximate distance from the epicenter to Columbia was 163 km (101 miles).

2.5.2.5 Seismic Assessment

A seismic assessment of the MURR reactor containment building was performed in June 2000 by the engineering firm of Sargent & Lundy[LLC] in order to determine the containment structure's resistance to a seismic event. Seismic response spectra, provided by U.S. NRC Regulatory Guide 1.60 and adjusted to reflect the ground acceleration response consistent with the criteria applicable to the Callaway Nuclear Plant (Ref. 2.27), located near Fulton, Missouri, were used for this assessment.

The methodology used to determine the containment building structure seismic resistance is based on the following three conservative assumptions:

1. Structural assessment of reinforced concrete items (walls and slabs) is performed in accordance with the criteria provided in Reference 2.23 [Ultimate Strength Design (USD) Method];

2. Material Strength is defined by the following parameters:
   a. Concrete compressive strength: 3,000 psi; and
   b. Steel rebar yield strength: 40,000 psi;

   Based on nuclear industry experience, these are reasonable, conservative values; and

3. Load combinations considered in this assessment are:

   1.5 x DL + 1.8 x LL,

   1.25 x DL + 1.25 x OBE + 1.25 x LL,

   1.0 x DL + 1.0 x SSE + 1.0 x LL,

   where:

DL	=	Dead load (self-weight of the structure being analyzed),
LL	=	Live load (15-ton capacity overhead crane, personnel),
OBE	=	Operating Basis Earthquake (normal operation of the reactor is maintained during and after this event), and
SSE	=	Safe Shutdown Earthquake (no damage to the structures, systems, and components required to safely shut down the reactor).

Dead weight for various components of the reactor containment building structure was estimated from the information provided in Reference 2.25. The total weight of the 15-ton capacity overhead crane is 17.2 kips.

Since the subject response spectra are normalized for a ground acceleration of 1 x g, the seismic acceleration values obtained from the normalized spectra are adjusted for a seismic acceleration of 0.2 x g to estimate the seismic response of the containment building structure to the SSE event. A ground acceleration of 0.2 x g is selected for this assessment to verify the structural adequacy of the shear walls when exposed to the seismic ground motion comparable to the one used at the Callaway Nuclear Plant (Ref. 2.27). The logic for selecting the Callaway Nuclear Plant is that it is located in relative proximity to the MURR site; approximately 30 miles (48 km) east of the reactor facility.

The seismic response assessment is based on the following simplified, but conservative methodology. This conservative methodology utilizes simple equivalent static techniques to account for the dynamic seismic effects.

- The Equivalent Static Method is used to account for the dynamic seismic effects. This method determines seismic accelerations based on the fundamental frequency mode multiplied by an amplification factor of 1.5 to account for higher mode participation in non-rigid response type behavior. Static force equivalent to the dynamic effect of the applied seismic loading is then calculated by multiplying the mass of the structure by the acceleration adjusted by an amplification factor of 1.5. In order to obtain the most critical response of the structure, it is necessary to estimate the lowest possible frequency of the system that would result in the highest seismic acceleration values.

  Since the concrete piers between the bottom of the structure and the rigid rock layer are relatively flexible compared to shear wall in-plain response, that "soft" response below ground level results in a lower frequency (close to the response spectra peak response accelerations). Therefore, the stiffness of the piers is used in this assessment to estimate the fundamental system frequency and corresponding seismic acceleration values. Properties of the soil were determined from References 2.20 and 2.21 and the dead weight of structural elements is obtained from Reference 2.22.

  The combined effect of three orthogonal components of seismic motion was evaluated for the purpose of this assessment, as 100% of the effects of one particular direction and 40% of the effects corresponding to two other directions of motion at right angles to the principal motion considered. This approach is recommended in Reference 2.24 for general use, especially for nuclear power plant design and evaluation.

- Based on the inspection of the configuration of the containment building structure, it is reasonable to state that at least six shear walls may be active in resisting seismic lateral loads proportionally to their stiffness. The Biological Shield is also capable of resisting a considerable portion of the seismic lateral loads. This fact was conservatively ignored in this assessment.

  Weight of the 15-ton capacity overhead crane is conservatively applied to only one shear wall for a seismic load in the east-west direction and to two shear walls for a seismic load in the north-south direction. In both cases, more shear walls will be active in resisting the seismically excited weight of the overhead crane.

- Seismic Live Load of 50 psf is applied to each floor elevation, which is generally an acceptable value used in the nuclear power industry.

Capacity of the reactor containment building walls between column rows 4 and 7, and C and F were evaluated to assess their capacity to carry the loads from the load combinations described above. Two wall sections were considered for the purpose of the assessment:

1. 12-inch thick section above grade level, reinforced with No. 4 at 8-inch vertical reinforcement, and
2. 16-inch thick section below grade level, reinforced with No. 4 at 8-inch vertical reinforcement.

For a 12-inch thick shear wall, the governing seismic load is the SSE Load Case with lateral loads applied in the north-south direction. The Interaction Ratio between the maximum applied bending moment and the 12-inch wall bending moment capacity is 0.864 < 1.0. For a 16-inch thick shear wall, the governing seismic load is the SSE Load Case with lateral loads applied in the north-south direction, combined with lateral water and soil pressure. The Interaction Ratio between the maximum applied bending moment and the 16-inch wall bending moment capacity is 1.0. Additional design margins for stresses in shear walls can be obtained if a refined analysis is performed eliminating some of the conservative assumptions described in the seismic response assessment methodology.

Bending stresses in the concrete piers caused by the transfer of seismic motion were not considered based on the arguments provided in Reference 2.26.

2.5.2.6 Summary and Conclusions

Undoubtedly, seismic activity will continue in the New Madrid and St. Marys Districts, and occasional strong shocks in these areas may be perceptible in Columbia, but they should not have damaging effects. The possibility of an earthquake as severe as the 1811-1812 New Madrid quakes, however, cannot be eliminated. The records which are available do suggest that it was a rare and unusual occurrence.

Although seismic activity occurs in the Ozark region, it is relatively mild when compared to earthquakes outside the midcontinent area. Columbia's location within the central stable area of Missouri, along with the seismic history of the region, indicates that the probability of seismic damage to the area is extremely low. Additionally, based on the seismic assessment approach described in Section 2.5.2.5, the MURR reactor containment building has been determined to be structurally adequate to resist the OBE and SSE seismic events.

The distance from the closest fault to the MURR far exceeds the siting requirements of the American National Standards Institute/American Nuclear Society, ANSI/ANS 15.7, "Research Reactor Site Evaluation," Section 3.2 (1), Fault Proximity, which states "No proposed facility shall be located closer than 400 meters from the surface location of a known capable fault (Ref. 2.4)."

CHAPTER 3

DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

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REFERENCES

3.1 Title 10, "Energy," Code of Federal Regulations, Part 50, "Domestic Licensing of Production and Utilization Facilities," Washington, D.C.

3.2 Institute of Electrical and Electronics Engineers Standard 279 (IEEE-279), "Criteria for Protection Systems for Nuclear Power Generating Stations," 1971.

3.3 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.7, "Research Reactor Site Evaluation," 1977.

3.4 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.15, "Criteria for the Reactor Safety Systems of Research Reactors," 1986.

3.5 American Association Standards, "Specifications - Covering Reactor System Work," Volume 4 of 4, Cornelius L. T. Gabler and Associates, Detroit, Michigan, February 1963.

3.0 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

This chapter identifies and describes the principal architectural and engineering design criteria for the structures, systems, and components that are required to ensure reactor facility safety and protection of the public. The material presented emphasizes the safety and protective functions and related design features that help provide defense-in-depth against the uncontrolled release of radioactive material to the environment. The bases for the design criteria for some of the systems discussed in this chapter are developed in other chapters and are appropriately cross-referenced, where required.

3.1 Conformance with NRC General Design Criteria

3.1.1 Introduction

This section discusses the "General Design Criteria for Nuclear Power Plants" as set forth in Title 10, Chapter I of the Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, as they apply to the Missouri University Research Reactor (MURR). The principal design criteria for a nuclear power plant establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety, that is, those that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the general public.

The General Design Criteria were formulated for the purpose of establishing minimum requirements for the principal design criteria for water-cooled nuclear power plants. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for other such units. As the title of 10 CFR 50, Appendix A indicates, it is intended for application to nuclear power plants. The MURR is a research facility and its systems are not designed for, and thus cannot be logically categorized according to power-plant application. The discussions here are therefore oriented toward the individual criterion rather than toward identification of areas of noncompliance with power plant criteria.

The MURR is designed and licensed for operation at a maximum steady-state power level of 10 Megawatts thermal. Thus, its fission-product inventory is substantially less than that of conventional nuclear power plants for which the General Design Criteria were primarily developed. In addition, the MURR operates at relatively low temperatures and pressures. No active engineered safety features are required for reactor core protection after shutdown. A conservative upper limit of energy released for an entire year of operation would be about 3,270 Megawatt-Days (average of 151 hours per week at 10 MW). These comparisons illustrate why the MURR is in a much lower risk category when reviewing for compliance with the General Design Criteria.

The General Design Criteria are divided into the following six categories: Overall Requirements; Protection by Multiple Fission Product Barriers; Protection and Reactivity Control

Systems; Fluid Systems; Reactor Containment; and Fuel and Radioactivity Control. Table 3-1 presents a synopsis of the conclusions regarding application of the General Design Criteria to the MURR.

3.1.2 Overall Requirements (Criteria 1-5)

- Criterion 1: Quality Standards and Records

  Structures, systems, and components important to safety were designed, fabricated, constructed, and tested to the original design specifications and associated codes and standards. This required major contributions from six contractors: two for design and construction and four for construction and testing. All design and construction work was monitored by the contractors to assure that the specifications incorporated appropriate standards, and that the design and construction were in accordance with these specifications. Modifications to the facility have been made in accordance with existing standards and requirements. The contractors and their responsibilities are discussed in Section 1.7.2.

- Criterion 2: Design Bases for Protection Against Natural Phenomena

  Tsunamis and seiches do not occur in central Missouri. Hurricanes typically develop over tropical ocean waters and dissipate rapidly when passing over land masses and regions of cooler temperatures. Hence, the influence of hurricanes on the climatology of the site is normally insignificant. Flooding in the area could be caused by run-off from local rainstorm activity. However, the reactor facility is situated about 200 feet (61 m) from the nearest 100-year floodplain.

  A significant climatic feature of the region is the occurrence of severe thunderstorms and tornadoes. In Missouri, the average annual number of tornado days - days on which one or more tornadoes were reported - is 12. Boone County has experienced 32 reported tornadoes within the recording period of 1950 to 2005. Structural damage has generally been limited to frame/lumber and mobile home residential units. Based on the probability of occurrence, postulated intensity, historical data, and low fission-product inventory, no criteria for tornadoes have been established for the MURR structure. Thunderstorms may be observed during any month of the year, but are most frequent during the summer, when they may occur weekly. The most damaging thunderstorms are usually those associated with the passage of a cold front or a squall line. The MURR is designed to withstand the extreme wind speeds associated with thunderstorm activity and, less frequently, extratropical cyclones, which usually produce their fastest wind speeds in the winter due to temperature contrasts between air masses.

  MURR's location within the central stable area of Missouri, along with the seismic history of the region, indicates that the probability of seismic damage to the area is extremely low. Additionally, based on the seismic assessment approach described in Section 2.5.2.5, the

MURR reactor containment building has been determined to be structurally adequate to resist the Operating Basis Earthquake (OBE), where normal operation of the reactor is maintained during and after this event, and the Safe Shutdown Earthquake (SSE), where no damage to the structures, systems, and components required to safely shut down the reactor has occurred.

Actions initiated by reports or observations of severe natural phenomena, which may create a previously nonexistent hazard, are discussed in the MURR Emergency Plan Implementing Procedures.

- Criterion 3: Fire Protection

The reactor containment building and its internal structures, constructed of steel, concrete, and concrete block, are highly fire resistant. However, material inventories inside the structure could include various flammable materials (paper, wood, etc.), and these, coupled with potential ignition sources, require that fires be considered.

Systems, structures and components important to safety have been designed and located to minimize the probability and effect of fires and explosions. Additionally, several features help reduce both the likelihood and consequences of a fire. First, periodic in-house inspections are made for the explicit purpose of reducing nonessential combustible material inventory. Second, a fire detection/suppression system is installed in the facility. Third, fire extinguishers are strategically located throughout the facility. Fourth, established emergency procedures will be used in the event of a fire. Fifth, the large volume of water in the reactor pool would protect the core from any conceivable fire. Sixth, the reactor is failsafe and will shut down should fire damage the instrumentation and control systems. Fire protection is not required to accomplish a safe shut down of the reactor or to maintain a safe shutdown condition.

- Criterion 4: Environmental and Dynamic Effects Design Bases

The design of the MURR accommodates the effects of, and is compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The design and construction of the facility preclude the catastrophic rupturing of the reactor pool and reactor pressure vessels. There is no postulated accident which could occur in the reactor containment building which would generate a positive pressure differential sufficient enough to compromise containment building integrity.

Operating procedures control the use or exclusion of corrosive, flammable, and toxic materials in any experimental facility or in the reactor containment building. The amount of explosive materials irradiated or allowed to generate in any experiment has been limited to reduce the likelihood of damage to the reactor or pool should they detonate. The reactor core is protected from missiles by being located [redacted]

[redacted] which is surrounded by and anchored to a reinforced concrete edifice. Dynamic effects of such conditions as whipping pipes are not a concern because there are no high pressure systems within the facility. The primary and pool coolant systems operate at relatively low pressures (Criterion 15) with all piping and components suitably supported and anchored. The supply and return lines for the primary and pool coolant systems enter and exit the containment structure through piping sleeves welded to the bottom of the reactor pool liner. The sleeves are cast into the biological shield and welded to the primary and pool piping, forming a seal which is functionally an integral part of the reactor pool liner. The probability of an event or condition resulting from the dynamic effects of missiles, aircraft, etc., causing a reactor accident, is, therefore, very small.

- Criterion 5: Sharing of Structures, Systems, and Components

  Normal electrical power and domestic cold water are supplied to the reactor facility from the campus' electrical power and water supply systems. These systems constitute the only systems "shared" by the MURR. The design of the reactor does not require electrical power or domestic cold water to safely shut down or to maintain an acceptable shutdown condition. An electrical power failure or loss of domestic cold water at any point within the distribution network of either system will not compromise the safety of the reactor.

3.1.3 Protection by Multiple Fission Product Barriers (Criteria 10-19)

- Criterion 10: Reactor Design

  The Safety Limit Analysis as described in Section 4.6.3 presents three (3) parametric curves which together define a four-dimensional safety limit envelope relating reactor inlet water temperature, pressurizer pressure, and core flow rate to the reactor power level corresponding to a departure from nucleate boiling ratio (DNBR) of 1.2. This is based on burnout heat flux data experimentally verified for Advanced Test Reactor (ATR) type fuel elements. Operation within this safety envelope will prevent fuel plate meltdown or cladding damage. Limiting Safety System Settings (LSSSs) are settings for automatic protection devices which prevent the safety limits from being exceeded. The LSSSs for the MURR have been chosen such that any true value of the four safety-related variables will not exceed a safety limit even under the most severe anticipated transient.

  Accident analyses presented in Chapter 13 show that, under credible accident conditions, the MURR safety limits will not be exceeded. Hence, there would be no fission product release that would exceed the allowable radiation limits as stated in Title 10, Chapter I of the Code of Federal Regulations, Part 10 (10 CFR 20).

- Criterion 11: Reactor Inherent Protection

The MURR has many of the inherent safety features characteristic of a heterogenous, water-moderated, small compact core reactor: a negative void coefficient and a negative temperature coefficient of reactivity.

Even if a sudden large insertion of positive reactivity causes reactor power to rise quickly on a short period, the large negative reactivity feedback produced by the voids caused by nucleate boiling in the core would help mitigate the power excursion before a safety limit is approached. A reduction in moderator density due to an increase in temperature also causes a decrease in reactivity due to the negative moderator temperature coefficient. However, a reduction in moderator density in the center test hole region causes a relatively small but positive reactivity change. Because of this, pool coolant flow through the center test hole is from the 20,000 gallons (75,708 l) of water in the bulk pool, which is maintained below 120 °F (49 °C) to prevent boiling. Additionally, any positive void effect in the center test hole would be more than offset by the significantly larger negative void effect in the reactor core. A negative void coefficient produces a decrease in core reactivity should the core heat flux rise to such an extent as to cause nucleate boiling in the primary coolant. The production of this void will tend to make the reactor subcritical because of the decrease in neutron moderation.

- Criterion 12: Suppression of Reactor Power Oscillations

Due to the small dimensions of the core and the low power levels at which the MURR operates, the reactor is tightly coupled and inherently stable with respect to space-time and xenon oscillations.

- Criterion 13: Instrumentation and Control

The instrumentation and control (I&C) systems which provide the information and means to safely control the reactor and avoid or mitigate potential accidents include the following: nuclear instrumentation system; rod control system; process instrumentation and control system; reactor safety system; engineered safety features actuation systems; and radiation monitoring systems.

The nuclear instrumentation (NI) system consists of three (3) neutron flux monitors, a wide range neutron flux monitor, and a multiscaler. The neutron flux monitors provide nine channels (only 6 are currently used) of neutron flux measurement from source level through 100% of full power operation. The system also provides input signals to the reactor safety and rod run-in systems upon sensing a high power level or a short reactor period condition and to the rod control system upon sensing low count rate. The wide range neutron flux monitor provides continuous neutron flux monitoring over a ten-decade range, in addition to providing an input signal to the rod control system for automatic control of reactor power level. The multiscaler is a timer/multiple counter which provides a digital readout of count rate to assist the reactor operator in reliably predicting criticality by the 1/M method during a reactor startup.

The rod control system is a relay and switch logic system which manages all control blade movements, interlocks and bypasses, based upon a choice of particular operating mode: manual or automatic. Manual control is used for reactor startup, changes in power level, and steady-state operation for short periods of time. Automatic control is selected only after a minimum power level has been attained and is used for long term steady-state operation. The controlling parameter for automatic operation is the neutron flux monitored by a compensated ion chamber (CIC). The pulse signal generated by the CIC is routed to the wide range neutron flux monitor drawer. The output signal from the wide range drawer is compared with the power schedule setting of a servo amplifier unit. The servo amplifier adjusts the position of the regulating blade, stepwise, in a direction to minimize the discrepancy between the power demand setting as set by the reactor operator and actual power level. The rod control system also has the capability of placing the reactor in a subcritical condition via rod run-in. Automatic insertion of the shim blades at a controlled rate is triggered by the rod control system when a monitored parameter exceeds a predetermined value or when manually initiated by the reactor operator. The rod run-in system is not part of the reactor safety system, however, it does provide a protective function by introducing shim blade insertion to terminate a transient prior to actuating a reactor safety system set point trip.

The process instrumentation and control system monitors, displays, and controls the following reactor plant parameters: temperature, pressure, flow, and pressurizer liquid level. The system also provides input signals to the reactor safety system as well as control interlocks for the primary and pool coolant system circulation pumps and automatic isolation valves. The reactor safety system is designed to place the reactor in a subcritical, safe, shutdown condition by a reactor scram, which initiates the instantaneous drop of all four shim blades by interrupting power to their electromagnets should a monitored parameter exceed a predetermined value. A reactor scram may also be initiated manually by the reactor operator.

The engineered safety features actuation systems receive input signals from monitoring instruments and initiate the operation of the engineered safety features (ESFs). The two systems which are identified as ESFs are the containment system and the anti-siphon system.

The containment actuation (reactor isolation) system is designed to completely isolate the reactor containment building, thereby preventing or mitigating any uncontrolled release of radioactive materials to the environment during an accident. Isolation of the reactor containment building can be automatically initiated by radiation detectors located at the reactor pool upper bridge and in the containment building ventilation exhaust plenum. Isolation can also be manually actuated by switches in the reactor control room or the facility lobby (Room 202). Manual or automatic actuation of the reactor isolation system causes the following actions to occur:

1. Reactor scram;
2. Closing of all normally-open reactor containment building penetrations with sealable closures;
3. Sounding of an audible alarm throughout the containment building; and
4. Illumination of a flashing light at the entrance to the containment building personnel airlock.

To ensure system reliability, the reactor isolation system is normally energized. Redundancy is incorporated into the actuation system to ensure that no single component or circuit failure will render any portion of the system inoperative.

The anti-siphon system functions as a backup system to the various safety instrumentation and equipment (e. g., pressure sensors, pump and valve interlocks, etc.) thus ensuring that the reactor core does not become uncovered during a loss of primary coolant accident. The system is designed to admit a fixed volume of air into the high point of the reactor outlet piping, or invert loop, instantaneously establishing the pressure in this area at equal to or greater than atmospheric. This prevents a siphon action from being created due to a rupture of the primary coolant piping. The anti-siphon system is automatically actuated upon detection of primary coolant system low pressure. Should primary coolant system pressure decrease below a predetermined value, pressure transmitters will cause the following actions to occur:

1. Reactor scram;
2. Stopping of primary coolant circulation pumps P501A and P501B;
3. Closing of primary coolant isolation valves V507A and V507B; and
4. Opening of anti-siphon system isolation valves V543A and V543B.

Redundancy is incorporated into the anti-siphon actuation system to ensure that no single component or circuit failure will render any portion of the system inoperable. System reliability is achieved through instrumentation and equipment which fail-safe when de-energized.

The permanently installed radiation monitoring systems include the area radiation monitoring system (ARMS), the fuel element failure monitoring system, the secondary coolant monitoring system, and the off-gas radiation monitoring system. These systems detect and quantify radiation and activity levels at various locations within the facility, within various reactor systems, and within the exhaust gases released to the uncontrolled environment. The ARMS provides input signals to the engineered safety features actuation system, which actuates the containment system, an engineered safety feature which provides a complete isolation of the reactor containment building.

The I&C systems are discussed in further detail in Chapter 7, Instrumentation and Control Systems.

- Criterion 14: Reactor Coolant Pressure Boundary

  The pressure boundaries for the primary and pool coolant systems have been designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. All metal surfaces in contact with primary or pool coolant are either aluminum or stainless steel, except where otherwise noted in the design specifications. The system components outside of the reactor pool have a low probability of serious leakage or gross failure. Further, the design of the primary and pool coolant systems is such that, even if a leak develops outside the reactor pool, it can be isolated from the remainder of the system by automatic isolation valves that are located as close as practicable to the biological shield. Rupture of the reactor pool liner is virtually impossible since it is supported on the bottom and sides by reinforced concrete. In addition, the primary and pool coolant systems operate at low pressures and temperatures.

- Criterion 15: Reactor Coolant System Design

  The reactor coolant systems are designed to permit full-power operation at any power level up to 10 MW by forced circulation. Natural convection operation is permitted at power levels up to 50 kW. The MURR utilizes three coolant systems: primary, pool, and secondary. Primary coolant system pressure is established and controlled during forced circulation operation by the pressurizer system. Reactor inlet pressure is maintained at 85 psia (0.59 MPa) by nitrogen gas that is automatically admitted to, or released from, the pressurizer in response to system instrumentation signals. Should system pressure increase to 115% above normal operating pressure, a high pressure scram will shut down the reactor. The system is further protected from overpressure by relief valves installed on the pressurizer and primary coolant system. The relief valves are set lower than the Technical Specification limit of 110 psig (0.76 MPa above atmosphere), thus providing a sufficient margin to assure that the primary coolant system design pressure of 125 psig (0.86 MPa above atmosphere) is not exceeded. The pool coolant system uses forced circulation to remove heat generated within the reflector region during normal 10-MW operation. Pool water is drawn from the reactor pool through the reflector elements and returned to the pool through a diffuser. The reactor pool is an open system and the maximum pressure at the suction of the pool coolant circulation pumps is the static head [about 29 feet (8.8 m)] of the water in the reactor pool. The remainder of the pool coolant system is pressurized by the circulation pumps. The primary and pool coolant purification systems are pressurized by small capacity pumps. Piping and valves in the primary and pool coolant systems are stainless steel or aluminum, except where otherwise noted in the design specifications, and of such size to provide adequate operating margins. The secondary coolant system components are generally carbon steel.

The reactor coolant systems are discussed in further detail in Chapter 5, Reactor Coolant Systems.

- Criterion 16: Containment Design

The reactor containment building provides an essentially leak-tight barrier against the accidental release of radioactivity to the environment. The reinforced concrete walls of the containment structure have been designed to withstand a peak internal pressure of 2.0 psig (13.8 kPa above atmosphere). Penetrations through the containment structure are limited to minimize potential leakage points. The design pressure was not predicated on any one accident scenario, rather, pressure was selected to envelope all postulated accident scenarios. Based on the discussion in Section 6.2.2 (Reactor Containment Building), the reactor containment building design pressure of 2.0 psig (13.8 kPa above atmosphere) is completely adequate for all postulated accident scenarios.

- Criterion 17: Electric Power Systems

Normal electrical power is supplied by the University of Missouri at Columbia Power Plant and/or the City of Columbia through an electrical distribution system which serves the entire campus. Should the reactor facility experience a loss of normal electrical power, a 275-kW diesel generator will provide emergency electrical power to essential reactor components in order to maintain the ability to monitor systems and to assure personnel safety.

An Uninterruptible Power Supply (UPS) provides precise regulated and transient free electrical power to the reactor I&C system. The UPS also ensures that the reactor control console and instrument panel indications are maintained during the transition period from the time of a loss of normal electrical power to the time the emergency electrical load is powered by the diesel generator. This allows the shutdown condition of the reactor to be monitored continuously by the reactor operator. If there is a complete loss of electrical power to the facility, the UPS will provide electrical power to the I&C system until its battery bank has reached its discharge limit, a period lasting about 2 hours at a normal load current of approximately 60 amps and a rated bank life of 120 amp-hours.

The design does not require electrical power to safely shut down the reactor or to maintain an acceptable shutdown condition. Should a failure of the UPS occur, a reactor shutdown may be affirmed upon visual observation that the shim blades are fully inserted and that reactor power has been reduced by noting a reduction or decrease in the intensity of the Cerenkov radiation in the region of the reactor core. In addition, the emergency electrical power system is not required for protection of the integrity of the fuel elements. [redacted]

A loss of normal electrical power and a subsequent failure of the emergency diesel generator to start (i.e., complete loss of electrical power to the facility) are analyzed in Chapter 13, Accident Analyses.

- Criterion 18: Inspection and Testing of Electric Power Systems

  The distribution system which supplies normal electrical power to the reactor facility is maintained by MU electrical maintenance crews. Preventive and corrective maintenance performed on the facility's motor control centers is accomplished by the MURR staff and campus electricians.

- Criterion 19: Control Room

  The reactor control room, located on the third level of the reactor containment building, contains the essential displays and control equipment which enable a reactor operator to observe and control the operation of the reactor. In the event of an accident which calls for shutting down the reactor, continuous or even partial occupancy of the control room is not required since the reactor has been shut down and experiments in progress have been terminated. Given that continuous occupancy of the control room is not required for any postulated accident scenario, adequate radiation protection is not provided for the MURR control room that would prevent an individual from receiving in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Therefore, compliance with Criterion 19 is conditional. Should a radiological accident require complete evacuation of the reactor containment building, radiation levels within the containment building can be displayed on the ARMS remote indicator located in the electronic shop. Equipment at appropriate locations outside the containment building are provided to safely shut down the reactor, if required.

3.1.4 Protection and Reactivity Control Systems (Criteria 20-29)

- Criterion 20: Protection System Functions

  The reactor protective system has been designed (1) to initiate automatic actions to assure that reactor safety limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety. The reactor protective system (reactor safety system) includes all of the sensing devices, electronic circuits and equipment, signal-conditioning equipment, and electro-mechanical devices that serve to affect either a reactor shutdown by the removal of the holding current from the four control rod drive mechanism electromagnets, or to activate the engineered safety features.

- Criterion 21: Protection System Reliability and Testability

The reactor protective system satisfies the applicable criteria of IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations." As the title of IEEE-279 implies, it is intended for application, especially during the design phase, to nuclear power generating station protection systems. Although the MURR is not a nuclear power generating station, much of the criteria of IEEE-279 can be applied to evaluate the adequacy of its reactor protective system with respect to present-day standards for functional performance and reliability. The control rod drive mechanisms are not within the scope of IEEE-279; however, they are an integral part of the reactor protective system and therefore were considered in the analysis of protective system reliability.

The MURR has been in operation since 1966 and its reactor protective system has functioned adequately and reliably. Any components replaced have been by components of a quality that is consistent with minimum maintenance requirements and low failure rates.

The reactor protective system complies with the single failure criterion of IEEE-279. A malfunction in one of the reactor safety system trip actuator amplifiers (TAAs) could result in, at most, the failure to interrupt the current to two control blade electromagnets, in which case the other two control blades would drop and successfully shut down the reactor. The reactor safety system non-coincidence logic units (NCLUs) are designed with 1/N logic: any one of N signal inputs to either logic unit will cause the TAAs to trip and initiate a reactor scram. Manual initiation of a scram by switch 1S10 opens an input to each NCLU as well as interrupting power to the TAAs. The short reactor period scram channels [Nuclear Instrumentation (NI) Channels 2 and 3] and the high power level scram channels (NI Channels 4, 5, and 6) are redundant, with separate detectors and electronic chassis. This arrangement satisfies the single failure criterion, although the relative physical location of the signal cables and electronics leaves them vulnerable to an external event such as a fire. However, resulting damage would cause a reactor scram. Further, an operator is always stationed in the control room during normal operation of the reactor and, therefore, is in the immediate vicinity of the cables and electronics. The instrument channels to the reactor safety system from the process instrumentation and control system are redundant. Reactor plant parameters which initiate a reactor scram (i.e., temperature, pressure, flow) are measured by multiple, independent channels. The only exceptions are the instrument channels which initiate a pressurizer high pressure scram and a pressurizer low water level scram. The pressurizer low water level scram ensures the reactor is shut down during a loss of primary coolant accident before water level in the pressurizer falls sufficiently to introduce nitrogen gas into the primary coolant system. This instrument channel merely provides redundancy for the four channels that provide a primary coolant system low pressure scram. The pressurizer high pressure scram ensures that the reactor will be shut down during a high pressure transient prior to reaching the primary relief valve set point pressure or exceeding the primary coolant system design pressure. However, this protective function is not required to ensure that the safety limits are not exceeded.

A failure of one instrument safety channel does not affect the operation of another. The signal inputs are arranged in series such that action by any one interrupts current to the control blade electromagnets resulting in a shutting down of the reactor. Additional redundancy is provided by two process instrument strings ("yellow" and "green" legs), the loss of either of which, due to an open circuit or a loss of power, will result in a scram.

Redundancy is incorporated into the ESFs and their actuation systems. No single component or circuit failure will render any portion of the ESF or its actuation system inoperative. A complete description of the redundant and reliable characteristics of the ESFs and their actuation systems are contained in Section 7.8 and Chapter 6, Engineered Safety Features.

As described in Section 5.3.5, the reflector plenum natural convection valve satisfies the single failure criterion. As described in Section 5.8, the decay heat removal system satisfies the single failure criterion. The control rod drive mechanisms also satisfy the single failure criterion.

Testing and calibration of all channels and devices used to derive the final safety system output may be accomplished with the reactor shutdown. While the operating schedule does not call for long periods of continuous operation, experience has shown that testing more frequently than weekly is not necessary to insure reliability. The reactor protective system, therefore, satisfies the intent of IEEE-279 with regard to the capability for sensor checks, and testing and calibration.

The instrument channels which provide protective actions are discussed in further detail in Chapter 7, Instrumentation and Control Systems.

- <u>Criterion 22: Protection System Independence</u>

The channels used in the reactor protective system are redundant. The circuitry for the nuclear instruments and the protective equipment located in the reactor control room, including the signal cables, are not physically separated, however, these channels are accessible to the reactor operators and hence, under continuous surveillance. Signal lines from the process sensors, transmitters, controllers which provide redundant protective functions are separated and clearly identified. Therefore, the reactor protective system satisfies the intent of IEEE-279 with regard to channel independence.

- <u>Criterion 23: Protection System Failure Modes</u>

The MURR protection systems are designed to fail into a safe state or into a state demonstrated to be acceptable under loss of energy or adverse environments. The reactor protective system is designed such that the failure of one instrument safety channel will not affect the operation of another channel. The Non-Coincidence Logic Circuit is designed for fail-safe operation. A component failure or malfunction will cause the output voltage to drop to less than 2 volts (safe condition) and actuate the trip actuator amplifiers (TAAs). A malfunction in one of the TAAs could result in, at most, the failure to interrupt the current to two control blade electromagnets, in which case the other two control blades would drop and successfully shut down the reactor. Tests of extremes of power supply voltage and ambient operating temperature have shown no significant effects on the operation of the reactor protective system. Very extreme conditions such as fire, flood, and earthquake have not been experienced; however, these occurrences would be such that more than sufficient time would exist for prompt operator action. Furthermore, the reactor protective system is fail-safe upon a loss of power. The reactor protective system contains no reactivity control functions. Therefore, a loss of an instrument channel could not initiate an uncontrolled addition of reactivity.

- Criterion 24: Separation of Protection and Control Systems

The reactor protective system satisfies the intent of IEEE-279 with regard to control and protection system interaction. No instrument channel provides both safety (scram) and control functions. Protective system separation ensures that failure or removal from service of any single control system component or channel leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system.

- Criterion 25: Protection System Requirements for Reactivity Control Malfunctions

The reactor protective system is designed to ensure that the fuel design limits are not exceeded for any single malfunction of the reactivity control system. The continuous withdrawal of all four shim blades, as discussed in Chapter 13, Accident Analyses, presents the worst case failure scenario of the rod control system. The reactor protective system will terminate the resulting power excursion by a short period or high power reactor scram with no resulting fuel damage.

- Criterion 26: Reactivity Control System Redundancy and Capability

The MURR has five independent reactivity control rods (blades): four shim rods and one regulating rod. Each of the control rods has its own drive mechanism and may be operated individually. The four shim rods may also be operated simultaneously, "in gang." The regulating blade is positioned by a control drive mechanism similar to the ones used for the shim blades, but with a few differences. The major difference is that a scram signal will not insert the regulating blade. The regulating rod (blade) is used to control power either manually or by automatic control. Upon receipt of a scram signal, the four shim blades are released from their drive mechanisms, allowing them to drop and shut down the reactor.

No emergency (backup) shutdown system is required. The reliability of the control blades and their support and guiding mechanisms (offset mechanisms) has been demonstrated by over 40 years of operation. A comprehensive preventive maintenance program which includes periodic disassembly of the offset mechanisms and inspection of the control blades ensures continued reliability.

The reactivity control system has been configured to control the excess reactivity needed for 10-MW operation 24 hours per day (including xenon override) and will provide a shutdown margin of at least 0.02 Δk. This shutdown margin ensures that the reactor can be shut down from any operating condition even if the most reactive control blade should remain stuck in the fully-withdrawn position. Additionally, the requirement that core excess reactivity above cold clean critical shall not exceed 0.098 Δk/k provides assurance that the previous specification is satisfied.

- <u>Criterion 27: Combined Reactivity Control Systems Capability</u>

  This criterion is not applicable.

- <u>Criterion 28: Reactivity Limits</u>

  No conceivable malfunction of the reactivity control system will result in core damage. The continuous withdrawal of all four shim blades, as discussed in Chapter 13, Accident Analyses, presents the worst case failure scenario of the rod control system. The reactor protective system will terminate the resulting power excursion by a short period or high power reactor scram with no resulting fuel damage. The total reactivity worth of the regulating blade is limited such that any condition resulting in a step insertion of the maximum worth of 0.006 Δk will not result in fuel plate damage.

  The control blades operate vertically within an annular gap between the outer reactor pressure vessel and the inner beryllium reflector. Since the reactor pool is at atmospheric pressure, control-rod ejection is not a credible event. Downward travel of the control blades is limited by the design of the offset mechanisms and the core support structure. The shim blades cannot drop out of the reactor core region because in the fully inserted position the shim blades are approximately 11 inches (27.9 cm) above the pressure vessel tube flange. Each shim blade is mechanically fastened to its offset mechanism by six (6) metal screws. In the non-credible event of all six metal screws shearing, travel out of the core region in the downward direction would be prevented by the vessel tube flange. Approximately 96% of the fuel region would still be shrouded by the neutron absorbing material of the detached shim blade.

  Reactivity limits placed on the MURR experiments ensure (1) that the rate of change of any movable experiment is such that, when the experiment is intentionally set in motion, the capacity of the reactivity control system to provide compensation is not exceeded and (2) that the magnitude of the potential reactivity worth of each unsecured experiment is less than the value of reactivity which would cause a violation of a safety limit. As shown in Chapter 13, Accident Analyses, the MURR can withstand a positive step reactivity insertion of 0.006 Δk with no core damage. Reactivity limits placed on the center test hole and each secured experiment ensure that this value is not exceeded.

- Criterion 29: Protection Against Anticipated Operational Occurrences

  Multiple, independent channels provide redundancy in reactor scram capability. The reactor protective and reactivity control systems conform to all existing design standards. Periodic checks (i.e., startup, shutdown, and maintenance procedures) of all the reactor protective system channels and the reactivity control system demonstrate that they perform their intended function. Tests of extremes of power supply voltage and ambient operating temperature have shown no significant effects on the operation of either system.

  Should the reactor facility experience a loss of normal electrical power, a 275-kW diesel generator will provide emergency electrical power to essential reactor components. If there is a complete loss of electrical power to the facility (i.e., failure of the diesel generator to start), the UPS will provide electrical power to the I&C systems until its battery bank has reached its discharge limit (about 2 hours at a normal load current of approximately 60 amps and a rated bank life of 120 amp-hours). The design does not require electrical power to safely shut down the reactor or to maintain an acceptable shutdown condition. In view of the aforementioned characteristics, the protection and reactivity control systems assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

3.1.5 Fluid Systems (Criteria 30-46)

- Criterion 30: Quality of Reactor Coolant Pressure Boundary

  The pressure boundaries for the primary and pool coolant systems have been designed, fabricated, assembled, and tested to the original design specifications and associated standards for quality control. Fittings and flange dimensions conform to the American Association Standards. All metal surfaces in contact with primary or pool coolant are either aluminum or stainless steel, except where otherwise noted in the design specifications. In addition, the reactor coolant cleanup system maintains a level of water quality which limits chemical corrosion of the pressure boundaries of the primary and pool coolant systems.

  A lowering of liquid level in the pressurizer would indicate a leak in the primary coolant system. A coinciding increase in pool water level would indicate the leak is in a section of the primary coolant system located within the reactor pool. If a leak should develop in a section of the primary or pool coolant system external to the reactor pool, the location would easily be determined since all piping, valves, meters, etc. are located in accessible spaces. Regardless, should a significant leak develop in either the primary or pool coolant system that could not be immediately isolated, the reactor would be shut down.

- <u>Criterion 31: Fracture Prevention of Reactor Coolant Pressure Boundary</u>

The MURR operates at relatively low powers and temperatures. The fluence levels and low temperatures produce no significant change in material properties. Stresses due to temperature and pressure transients are minimal. The reactor coolant pressure boundary normally experiences an increase of only 30 °F (17 °C) and 70 psig (0.48 MPa above atmosphere) from startup (cold core) to steady-state power operation. The primary coolant system is protected from overpressure by relief valves installed on the pressurizer and the primary coolant piping. The relief valves are set lower than the Technical Specification limit of 110 psig (0.76 MPa above atmosphere), thus providing a sufficient margin of assurance that the primary coolant system design pressure of 125 psig (0.86 MPa above atmosphere) will not be exceeded.

- <u>Criterion 32: Inspection of Reactor Coolant Pressure Boundary</u>

The reactor pool liner is surrounded by, and anchored to, a reinforced concrete edifice which prevents external forces from being directly transmitted to the liner, thus precluding any movement of the liner. The lower levels and bottom of the pool liner can only be inspected by visual observation through the water inside the pool. Nearly all primary coolant system components located within the reactor pool are accessible with the pool water level lowered.

All piping, valves, pumps, etc. associated with the primary and pool coolant systems which are external to the reactor pool are located in open spaces and are readily accessible for inspection.

The MURR operates at relatively low powers and temperatures. The fluence levels and low temperatures produce no significant change in material properties.

- <u>Criterion 33: Reactor Coolant Make-Up</u>

The primary coolant make-up water system provides a means for the addition of primary grade water lost during normal operation of the reactor through primary coolant sampling, pump leakage, etc. The system is not designed to provide make-up water for protection against any significant leaks developing in the primary coolant system. If a significant leak should develop that could not be immediately isolated, the reactor would be shut down.

Make-up water is added to the pool coolant system from the reactor plant make-up water storage tanks. Two steel tanks, each with a capacity of 7,000 gallons (26,498 l), provide storage of demineralized make-up water for the reactor pool during normal operation. The emergency pool fill system provides an emergency raw water supply deliverable at a rate in excess of 1,000 gpm (3,785 lpm) to the reactor pool in the event of a leak in the pool or pool coolant system. This rate of water addition is adequate to maintain greater than 3 feet (0.9 m) of water above a completely severed 6-inch beamport with no impediments in the port. This is a manually-operated system with a virtually unlimited supply of raw water from the MU water supply system. The emergency pool fill system is discussed in greater detail in Chapter 9, Auxiliary Systems and Section 13.2.9.2.2.

- Criterion 34: Residual Heat Removal

Operating procedures require that the primary and pool coolant systems remain in operation for a specified period of time after a reactor shutdown to remove any residual heat from the reactor core and reflector region.

The decay heat removal system is designed to transfer core decay heat to the reactor pool following an emergency shutdown accompanied by a primary loop isolation, or, in the event of a loss of primary coolant flow. The decay heat removal system complies with the single failure criterion of IEEE-279. Furthermore, as discussed in Chapter 13, analyses have shown that in the non-credible event of both automatic isolation valves V546A and V546B failing to open following a loss of primary coolant flow or pressure from 10-MW operation, thereby prohibiting the dissipation of reactor core decay heat through the in-pool heat exchanger, core decay heat can adequately be dissipated to the reactor pool through the pressure vessels and primary coolant piping, assuring that the integrity of the fuel element cladding can be maintained even without an engineered decay heat removal system

- Criterion 35: Emergency Core Cooling

No emergency core cooling system is required for the MURR. As discussed in Chapter 13, Accident Analyses, the reactor possesses sufficient redundant safety features to prevent core damage as a result of a double-ended rupture of the largest diameter primary coolant system pipe and requires no additional emergency core cooling system for core protection in the event of a loss of primary coolant accident.

- Criterion 36: Inspection of Emergency Core Cooling System

This criterion is not applicable.

- Criterion 37: Testing of Emergency Core Cooling System

This criterion is not applicable.

- Criterion 38: Containment Heat Removal

There are no systems, components, equipment, experiments, etc. with sufficient stored energy to require a heat removal system for the reactor containment building.

- Criterion 39: Inspection of Containment Heat Removal System

This criterion is not applicable.

- Criterion 40: Testing of Containment Heat Removal System

This criterion is not applicable.

- Criterion 41: Containment Atmosphere Cleanup

Post-accident activities are not contingent upon reducing the concentration and quality of fission products released into the reactor containment building following the Maximum Hypothetical Accident (MHA). As discussed in Chapter 13, Accident Analyses, the MHA will not cause an increase in pressure within the reactor containment building. Any leakage from the containment building would occur as a result of normal changes in atmospheric pressure and the pressure equilibrium between the containment building and atmosphere. Although pressure within the containment building would not lag the change outside the building, it is conceivable that eventually a 0.7 inches of Hg (25.4 mm of Hg at 60 °C) pressure differential could be produced. However, as calculated, the release of any radioiodine or gaseous activity from the reactor containment building as a result of the MHA would not produce a hazard to the health and safety of the public.

The ventilation system provides the necessary air exchange capacity to ensure that concentrations of radioactive gases in the laboratory and reactor containment buildings are maintained at levels which are below 10 CFR 20 limits during normal operation of the reactor.

- Criterion 42: Inspection of Containment Atmosphere Cleanup Systems

This criterion is not applicable.

- Criterion 43: Testing of Containment Atmosphere Cleanup Systems

This criterion is not applicable.

- Criterion 44: Cooling Water

Heat from the primary and pool coolant systems is transferred to a secondary coolant system by means of heat exchangers. The heat is then dissipated to the atmosphere through a cooling tower. The MURR does not require an auxiliary cooling system for the primary and pool coolant systems when the reactor is in a shutdown condition. The decay heat removal system is designed to transfer core decay heat to the reactor pool following an emergency shutdown accompanied by a primary loop isolation, or, in the event of a loss of primary coolant flow. The decay heat removal system complies with the single failure criterion of IEEE-279.

- Criterion 45: Inspection of Cooling Water System

  Cooling equipment (e. g., heat exchangers, pumps, piping, and valves, etc.) used during normal operation of the reactor is located in either the mechanical equipment room (Room 114), cooling tower tunnel, or cooling tower building with adequate space provided to permit periodic inspection to ensure the integrity and capability of the equipment.

- Criterion 46: Testing of Cooling Water System

  The secondary coolant system is designed to permit periodic pressure and functional testing. This system is routinely checked, tested, and maintained.

  The secondary coolant system is not required for reactor safety.

3.1.6 Reactor Containment (Criteria 50-57)

- Criterion 50: Containment Design Basis

  The reactor containment building is designed to withstand a peak internal pressure of 2.0 psig (13.8 kPa above atmosphere) with a leakage rate over a 24-hour period not to exceed 10% of the contained volume. This design pressure was not predicated on any one accident scenario, rather, the pressure was selected to envelope all postulated accident scenarios. However, as shown in Chapter 13, Accident Analyses, no credible accident has been identified which can result in a significant overpressure condition within the containment structure.

- Criterion 51: Fracture Prevention of Containment Pressure Boundary

  The containment structure is a five-level poured concrete building with 12-inch (0.3-m) thick (minimum) reinforced exterior walls. The outside surface of the concrete walls are finished with removable sheet siding. The outside structure is subjected to natural environmental conditions while the inside or internal environment is maintained at regulated conditions.

  The structure is not subjected to any significant internal pressures during normal operations. As discussed in Chapter 13, no credible accident has been identified which can cause an increase in pressure within the containment building. Nevertheless, the containment structure is protected from overpressure by a water-filled trap (seal trench). The seal trench forms a water seal between the reactor containment building and the adjacent laboratory structure. In the event of an overpressure condition within the containment building exceeding 2.0 psig (13.8 kPa above atmosphere), water would be driven out of the seal trench, creating a path for pressure within the containment building to be relieved into the laboratory building. Once the pressure was relieved, the water level would drop and reseal the reactor containment building.

- Criterion 52: Capability for Containment Leakage Rate Testing

The reactor containment building structure has been designed to accommodate periodic integrated leakage rate testing. The leakage rate test ensures that sufficient integrity exists to prevent the leakage of greater than 10% of the contained volume at a 2.0 psig (13.8 kPa above atmosphere) overpressure condition over a 24-hour period. The leakage rate test utilizes three lines which penetrate the containment structure: one line is used for pressurizing the containment building; a second line is used for attaching the instruments necessary for measuring internal pressure and leakage rate; a third line allows for depressurizing the containment building upon completion of the leakage rate test.

- Criterion 53: Provisions for Containment Testing and Inspections

The reactor containment building leakage rate test is performed annually to ensure that the containment building design specifications are not exceeded (Criterion 50). The actuation system which initiates automatic isolation of the reactor containment building, and its associated radiation monitors, are periodically tested for operability.

All penetrations through the containment structure are located in areas with adequate space provided to permit appropriate surveillance and inspection, if required.

- Criterion 54: Piping Systems Penetrating Containment

Building services (e. g., domestic hot and cold water, demineralized water, compressed air, etc.) enter and exit the reactor containment building through the seal trench. These piping systems have no effect on the safety of operation. Therefore, isolation, redundancy, and secondary containment of these systems is not a consideration and only conditional compliance is met. Piping systems which penetrate the containment structure, connect directly to the containment atmosphere, and are normally maintained open during reactor operation, have redundant isolation capabilities (Criterion 56).

All electrical connections from equipment external to the reactor containment building, as well as electrical power supplies which must enter the containment building, pass through two steel penetration plates located in the containment structure walls. Each penetration plate contains sealed connectors which allow electrical lines to enter or exit the containment building with minimal air leakage. There are no through-the-containment-wall electrical conduits.

A detailed description of the reactor containment building and the penetrations through the containment structure are contained in Chapter 6, Engineered Safety Features.

- Criterion 55: Reactor Coolant Pressure Boundary Penetrating Containment

The primary coolant system inlet and outlet lines enter and exit the containment structure through piping sleeves welded to the bottom of the reactor pool liner. The sleeves are cast into the biological shield and welded to the primary piping, forming a seal which is functionally an integral part of the reactor pool liner. A quick-acting automatic isolation valve is installed on each primary coolant line immediately outside the containment structure, as close as practicable to the biological shield.

Item (4) of Criterion 55 states: "One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment." There are no primary coolant system automatic isolation valves located within containment; therefore, compliance with this criterion is conditional.

- Criterion 56: Primary Containment Isolation

Penetrations through the containment building concrete structure which connect directly to the containment atmosphere are limited in order to minimize potential leakage points. Existing penetrations include the seal trench, a pedestrian entry, an entry for heavy equipment, and building ventilation supply and exhaust air ducts.

The pedestrian entry (airlock) consists of two electric-motor-driven doors which seal against inflatable gaskets when in the closed position. The entry control system is designed and interlocked such that one door is always closed and sealed, ensuring containment integrity is maintained.

The heavy equipment entry consists of an electric-motor-driven door which seals against an inflatable gasket when in the closed position. This door is maintained shut at all times during reactor operation. A pressure switch monitors gasket seal pressure and initiates an annunciator alarm and rod run-in upon depressurization of the gasket.

The containment building supply and exhaust air ducts contain two electric-motor-driven doors (Doors 504 and 505) which seal against inflatable gaskets when in the closed position. These doors are normally kept in the open position during reactor operation. A closure signal from the containment actuation system will cause the motors to drive the isolation doors to the closed position, inflate the gaskets, and seal the doors against the containment structure. A second set of isolation doors, which seal against a solid rubber gasket, provide redundancy to Doors 504 and 505. The backup doors are located in the containment building supply and exhaust plenums, immediately adjacent to Doors 504 and 505. These doors also shut on a closure signal from the containment actuation system. A separate ventilation duct discharges potentially contaminated gases from the containment building. Exhaust air from areas which produce radioactive gases or airborne contamination is ducted to a 16-inch line which penetrates the west wall of the containment structure and discharges to the facility exhaust plenum. A 12-inch long steel collar surrounding the 16-inch pipe is cast into the containment building wall. The 16-inch pipe is welded to this collar. Two quick-closing isolation valves are located on this line within the containment building. A closure signal from the containment actuation system will cause both isolation valves to close. These isolation valves are installed in series to provide redundancy.

Item (4) of Criterion 56 states: "One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment." Since both automatic isolation valves are located within containment, compliance with this criterion is conditional.

- Criterion 57: Closed System Isolation Valves

Piping systems which enter the reactor containment building that are not part of the primary coolant system pressure boundary nor connected directly to the containment atmosphere have no effect on the safety of reactor operation. Therefore, automatic or manual isolation of these systems is not a concern. However, most building service lines which enter the containment structure through the seal trench have manual isolation valves located immediately outside containment. The 6-inch fire protection water line and the 6-inch emergency pool fill line have manual isolation valves in close proximity to the laboratory building.

The containment building leakage rate test is based on a volumetric leakage rate and not on individual component leak rates (e. g., valve leakage, inflatable gasket leak-by, etc.), therefore the capability to identify or measure individual component leakage is not performed.

3.1.7 Fuel and Radioactivity Control (Criteria 60-64)

- Criterion 60: Control of Release of Radioactive Materials to the Environment

Exhaust air from both the laboratory and reactor containment buildings is combined in the facility ventilation exhaust plenum just prior to being discharged to the atmosphere. Air from the two buildings is never mixed until this point. Potentially contaminated air is thus diluted by mixing with uncontaminated air, minimizing the concentration of radioactive gases released to the environment.

There is no readily available path for radioactive liquid waste to be discharged directly into the environment. All potentially radioactive liquid waste is directed to the liquid waste retention and disposal system where it is analyzed for radioactivity, and disposed of appropriately (Criterion 64). Sufficient storage capacity exists if unfavorable conditions impose unusual operational limitations upon the release of such liquid effluent to the environment. Gaseous releases do not require retention because they are well within regulatory limits.

Handling of solid radioactive waste is discussed in Section 11.2.2.1. Normally, solid waste is transferred to an authorized waste broker or brokerage service. However, the facility may opt to ship it directly to a waste processing site without the use of a broker.

- Criterion 61: Fuel Storage and Handling and Radioactivity Control

The major design concern regarding the storage, handling, and control of the radioactivity of irradiated reactor fuel is shielding. [redacted] Supplemental cooling is not required due to the sufficiently large heat sink and the absence of a large decay heat source in the fuel. Irradiated fuel elements are handled either under water or with a shipping cask. The elements are transferred one at a time to ensure a criticality-safe configuration.

[redacted]

- Criterion 62: Prevention of Criticality in Fuel Storage and Handling

[redacted]. The storage locations are designed such that the calculated K_{eff} is less than 0.9 under all conditions of moderation and irrespective of the number of fuel elements stored or the amount of burnup per element. New, unirradiated reactor fuel is kept in a vault equipped with special racks that allow storage of the fuel in a criticality-safe geometry such that the calculated K_{eff} is also less than 0.9.

Only one fuel element can be moved at a time, therefore handling does not present a criticality problem.

Handling and storage of reactor fuel is discussed in detail in Chapter 9, Auxiliary Systems.

- Criterion 63: Monitoring Fuel and Waste Storage

No residual heat removal or temperature monitoring capability is required for irradiated fuel elements. Natural convective flow of pool coolant around a fuel element is sufficient to remove any residual heat.

The reactor pool surface and the fuel storage vault radiation levels are measured by the ARMS.

- Criterion 64: Monitoring Radioactivity Releases

Most of the reactor containment building air is recirculated with only a small amount of air being exhausted through a 16-inch line. The recirculated containment air is monitored by the ARMS. A radiation level greater than the set point of either one of the two radiation monitors in the exhaust plenum will provide an input signal to the engineered safety features actuation system, which initiates the containment system, an engineered safety feature which provides a complete isolation of the reactor containment building.

Exhaust air from both the laboratory building and the reactor containment 16-inch line is combined in the facility ventilation exhaust plenum just prior to being discharged to the unrestricted environment through the exhaust stack. The off-gas radiation monitoring system monitors the exhaust air for radioactive iodine, beta/gamma particulate, and noble gases. Actions initiated to reduce the release of radioactivity if predetermined limits are exceeded are discussed in the MURR Emergency Plan Implementing Procedures. Continuous Air Monitors (CAMs) are placed at locations within the facility that have a higher probability of producing radioactive gases or airborne contamination. These mobile units can be moved anywhere in the facility as needed.

All potentially radioactive liquid wastes are directed to the liquid waste retention and disposal system. The liquid waste is retained until an assay indicates that the specific activity of all radioactive isotopes is less than the limit specified in 10 CFR 20 indicating it is acceptable for disposal by release into the sanitary sewage.

TABLE 3-1
APPLICABILITY OF COMPLIANCE WITH GENERAL DESIGN CRITERIA

OVERALL REQUIREMENTS				
Criterion Number and Title	Compliance	Compliance Not Required	Conditional Compliance	Conditional Noncompliance
1. Quality Standards and Records	X			
2. Design Bases for Protection Against Natural Phenomena	X			
3. Fire Protection	X			
4. Environmental and Dynamic Effects Design Bases	X			
5. Sharing of Structures, Systems, and Components	X			

PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS				
Criterion Number and Title	Compliance	Compliance Not Required	Conditional Compliance	Conditional Noncompliance
10. Reactor Design	X			
11. Reactor Inherent Protection	X			
12. Suppressions of Reactor Power Oscillations	X			
13. Instrumentation and Control	X			
14. Reactor Coolant Pressure Boundary	X			
15. Reactor Coolant System Design	X			
16. Containment Design	X			
17. Electrical Power Systems	X			
18. Inspection and Testing of Electrical Power Systems	X			
19. Control Room			X	

PROTECTION AND REACTIVITY CONTROL SYSTEMS				
Criterion Number and Title	Compliance	Compliance Not Required	Conditional Compliance	Conditional Noncompliance
20. Protection System Functions	X			
21. Protection System Reliability and Testability	X			
22. Protection System Independence	X			
23. Protection System Failure Modes	X			
24. Separation of Protection and Control Systems	X			
25. Protection System Requirements for Reactivity Control Malfunctions	X			
26. Reactivity Control System Redundancy and Capability	X			
27. Combined Reactivity Control Systems Capability		X		
28. Reactivity Limits	X			
29. Protection Against Anticipated Operational Occurrences	X			

FLUID SYSTEMS				
Criterion Number and Title	Compliance	Compliance Not Required	Conditional Compliance	Conditional Noncompliance
30. Quality of Reactor Coolant Pressure Boundary	X			
31. Fracture Prevention of Reactor Coolant Pressure Boundary	X			
32. Inspection of Reactor Coolant Pressure Boundary	X			
33. Reactor Coolant Make-Up	X			
34. Residual Heat Removal	X			
35. Emergency Core Cooling		X		
36. Inspection of Emergency Core Cooling System		X		
37. Testing of Emergency Core Cooling System		X		
38. Containment Heat Removal		X		
39. Inspection of Containment Heat Removal System		X		

FLUID SYSTEMS (cont.)				
Criterion Number and Title	Compliance	Compliance Not Required	Conditional Compliance	Conditional Noncompliance
40. Testing of Containment Heat Removal System		X		
41. Containment Atmosphere Cleanup		X		
42. Inspection of Containment Atmosphere Cleanup Systems		X		
43. Testing of Containment Atmosphere Cleanup Systems		X		
44. Cooling Water	X			
45. Inspection of Cooling Water System	X			
46. Testing of Cooling Water System	X			

REACTOR CONTAINMENT				
Criterion Number and Title	Compliance	Compliance Not Required	Conditional Compliance	Conditional Noncompliance
50. Containment Design Basis	X			
51. Fracture Prevention of Containment Pressure Boundary	X			
52. Capability for Containment Leakage Rate Testing	X			
53. Provisions for Containment Testing and Inspection	X			
54. Piping Systems Penetrating Containment			X	
55. Reactor Coolant Pressure Boundary Penetrating Containment			X	
56. Primary Containment Isolation			X	
57. Closed System Isolation Valves			X	

FUEL AND RADIOACTIVITY CONTROL				
Criterion Number and Title	Compliance	Compliance Not Required	Conditional Compliance	Conditional Noncompliance
60. Control of Releases of Radioactive Materials to the Environment	X			
61. Fuel Storage and Handling and Radioactivity Control	X			
62. Prevention of Criticality in Fuel Storage and Handling	X			
63. Monitoring Fuel and Waste Storage	X			
64. Monitoring Radioactivity Releases	X			

3.2 Meteorological Damage

A significant climatic feature of the region is the occurrence of severe thunderstorms and tornadoes. Boone County, Missouri, has experienced 32 reported tornadoes within the recording period of 1950 to 2005. Structural damage has generally been limited to frame/lumber and mobile home residential units. Thunderstorms may be observed during any month of the year, but are most frequent during the summer, when they may occur weekly. The MURR is located in the Hinkson Creek valley with a high bluff directly to the west. This location helps protect the facility from severe weather phenomena, such as the high winds associated with thunderstorms and tornadoes. The reactor itself is further protected from damage by virtue of its below grade location and the thick, reinforced concrete edifice surrounding the reactor pool liner. The containment structure has been designed for area wind loads.

A complete description of the site meteorology is contained in Chapter 2, Site Characteristics.

3.3 Water Damage

The reactor facility is situated about 200 feet (61 m) from the nearest 100-year floodplain. As discussed in Chapter 2 (Section 2.4.5), there are no credible hydrological events which can lead to flooding or other water-induced damage to the reactor facility. However, even if flooding occurred, reactor safety would not be a concern since the reactor core is located in a water pool.

3.4 Seismic Damage

MURR's location within the central stable area of Missouri, along with the seismic history of the region, indicates that the probability of seismic damage to the area is extremely low. Additionally, based on the seismic assessment approach described in Section 2.5.2.5, the MURR reactor containment building has been determined to be structurally adequate to resist the Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) seismic events.

3.5 Mechanical System and Components

The MURR does not have structures, systems, or components that are important to safety in the same context as water-cooled nuclear power plants. For the MURR, a LOCA, failure of the electrical power system, or any other credible accident does not have the potential for causing off-site exposures comparable to those listed for accident exposures in the guideline of ANSI/ANS 15.7-1977, "Research Reactor Site Evaluation" (Ref. 3.3). The design bases of MURR's electrical and mechanical components give reasonable assurance that the facility systems and components will function as designed to ensure safe operation and safe shutdown of the reactor.

The design of the MURR does not require electrical power to safely shut down the reactor or to maintain an acceptable shutdown condition. In addition, the emergency electrical power system is not required for protection of the integrity of the fuel elements. However, certain systems and components are required to function as designed to ensure safe operation and safe shutting down of the reactor. The following subsections describe the design bases of these systems and components. Surveillances ensure that these systems and components will operate and the health and safety of the public and workers will be protected.

3.5.1 Fuel System

One type of fuel assembly is designed for use in the reactor core. This design has proven reliable with over forty years of continuous operation at the MURR, and at other test and research reactors with no significant failures. Each fuel assembly is longitudinally-symmetrical with 24 fuel bearing plates. The fuel plates are segments of concentric circles fabricated using the "picture frame" method, which consists of the fuel material encased in a one piece window-shaped aluminum frame and clad on both sides with aluminum cover plates. The fuel system is a uranium aluminide dispersion UAl_x matrix with a maximum loading of [redacted] grams of uranium-235 (^{235}U). The UAl_x dispersion fuel system was developed at the Idaho National Engineering Laboratory (INEL) for the high flux, high power Advanced Test Reactor (ATR) and subsequently used at the Materials Test Reactor (MTR) and the Engineering Test Reactor (ETR) prior to use at the MURR. Several features of the UAl_x dispersion fuel system contribute to its extended performance capability in the high flux reactors.

(1) The powder dispersion allows voidage to be fabricated into the fuel matrix for the accommodation of the increased volume of fission products.

(2) Burnable poisons can be readily dispersed in the fuel matrix.

(3) The structure has an exceptional tolerance for fission product gas, with an attendant high blister temperature.

The fuel material for the UAl_x dispersion fuel system consists of a uranium powder fully-enriched in the isotope ^{235}U (93.0 ± 1.0%) dispersed in an aluminum powder. The uranium content of the metal used in the fuel material shall be a minimum of 99.85% by weight as determined by the difference between 100% and the total weight percentage of all the impurities.

To ensure that the validity of the Safety Limit Curves is maintained, the reactor core will consist of eight fuel assemblies. However, operation to 100 watts above shutdown power on less than eight assemblies is permitted for the purposes of reactor calibration or multiplication studies. Reactor core loadings are normally comprised of a mixture of fuel assemblies with varying stages of burnup. Generally, this mixture consists of two relatively new fuel assemblies, four fuel assemblies [redacted]. A mixed core allows the fuel assemblies to be used until their licensed burnup limit, as specified in the Technical Specifications, is reached.

The fuel system is described in greater detail in Chapter 4, Reactor Description.

3.5.2 Control Rod Drive Mechanisms

The four shim blades are actuated by electromechanical control rod drive mechanisms that position, hold, and scram each shim blade. The control rod drive mechanisms are of the same type used on a number of research reactors designed by General Electric. The drive mechanisms are supported by guide housings which are fastened to the tread plate of the upper reactor bridge, and are joined to the housings by flanged, bolted connections. Each drive mechanism consists of a single-phase motor connected to a lead screw assembly through a reduction gear box and overload clutch. A ball nut coupled directly to a drive tube is driven inward and outward by the lead screw. Limit switches mounted on each drive mechanism stop the drive motor at the top and bottom of travel. The nominal speed of the shim blades is two inches (5.1 cm) per minute in the inward direction and one inch (2.5 cm) per minute in the outward direction. The speed of the shim blades cannot be adjusted without physically altering the system. Connected to the bottom of the drive tube is an electromagnet which engages a steel anvil that is attached above the pool water level to the end of a lift-rod assembly. The shim blade can be withdrawn when the electromagnet is energized. When the reactor is scrammed, the electromagnet is de-energized, releasing the anvil, and allowing the shim blade and lift-rod assembly to drop. A dash pot assembly cushions the fall of the shim blade during the final 20% of travel. The control rod drive mechanisms satisfy the single failure criterion of IEEE-279.

The regulating blade is positioned by a control rod drive mechanism similar in basic configuration to the ones used for the shim blades with a few exceptions. A scram signal does not insert the regulating blade, therefore, an electromagnet and an anvil are not required. The lift-rod assembly is pinned directly to the drive tube. Also, the regulating blade is driven at 40 inches (102 cm) per minute in both the inward and outward directions, thereby requiring a different reduction gear box.

Position indication for the shim blades and the regulating blade is provided by an encoder transducer mounted to each control rod drive mechanism. Analog signals from the transducers are converted into digital form by a microcomputer controlled instrument mounted into the control room instrument panel. Control rod height is displayed in inches on the instrument panel and the control console.

3.5.3 Reactor Core Assembly Support Structure

The reactor core assembly support structure consists of the reactor pressure vessel, the island tube, and the reflector tank. The entire structure is approximately 12 feet (3.7 m) in overall length and fabricated of Aluminum-Alloy 6061-T6 with stainless steel hardware. The design of the support structure ensures that the top of the reactor pressure vessel will not deflect greater than 0.015 inches (0.381 mm) from its vertical axis or distort in any manner which would interfere with travel of the control blades. Piping connections to the reactor pressure vessel are suitably braced and supported such that reactions to the pressure vessel are kept to a minimum.

The reactor core assembly support structure is described in detail in Section 4.2.5, Reactor Core Assembly Support Structure.

3.5.4 Reactor Safety System

The reactor safety system is designed to prevent operation of the reactor in regions in which fuel damage may occur. This is accomplished by promptly placing the reactor in a subcritical, safe, shutdown condition by a reactor scram, which initiates the instantaneous drop of the control blades by interrupting power to their electromagnets should a monitored parameter exceed a predetermined value. A reactor scram may also be initiated manually by the reactor operator.

The reactor safety system is discussed in Chapter 7, Instrumentation and Control Systems.

3.5.5 Containment System

The containment system is designed to completely isolate the reactor containment building, thereby preventing or mitigating an uncontrolled release of radioactive materials to the environment during an accident. Redundancy is incorporated into the system to ensure no single component or circuit failure will render any portion of the containment system inoperative. Isolation of the reactor containment building can be automatically initiated by radiation detectors located at the reactor pool upper bridge and in the containment building exhaust plenum. Isolation can be manually actuated by switches in the reactor control room or the facility lobby (Room 202).

The reactor containment building is designed to hold a peak internal overpressure of 2.0 psig (13.7 kPa above atmosphere) with a leakage rate over a 24-hour period not exceeding 10% of the contained volume. However, no credible accident has been identified which can result in a significant overpressure condition within the containment building.

The containment system is described in greater detail in Chapter 6, Engineered Safety Features. The instrumentation that actuates this engineered safety system is discussed in Chapter 7, Instrumentation and Control Systems.

3.5.6 Anti-Siphon System

The anti-siphon system functions as a backup system to the various safety instrumentation and equipment (e. g., pressure sensors, pump and valve interlocks, etc.) thus ensuring that the reactor core does not become uncovered during a LOCA. A rupture of the primary coolant system followed by a loss of pressure causes the anti-siphon system to admit a fixed volume of air to the high point of the reactor outlet piping, thus breaking any potential siphon which may have been created by the pipe rupture. Redundancy is incorporated into the system to ensure no single component or circuit failure will render any portion of the anti-siphon system inoperative.

The anti-siphon system is described in greater detail in Chapter 6, Engineered Safety Features. The instrumentation that actuates this engineered safety system is discussed in Chapter 7, Instrumentation and Control Systems.

CHAPTER 4

REACTOR DESCRIPTION

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4.0 REACTOR DESCRIPTION

This chapter discusses and describes the principal design features, operating characteristics, and parameters of the reactor. The analysis in this chapter supports the conclusion that the reactor is conservatively designed for safe operation and can be shut down under all credible operating conditions. The information in this chapter provides the design bases for many systems, subsystems, and functions discussed elsewhere in the SAR and for many of the Technical Specifications.

4.1 Introduction

The Missouri University Research Reactor (MURR) is a pressurized, reflected, heterogeneous, open pool-type, which is light-water moderated and cooled. The reactor is designed and licensed to operate at a maximum thermal power level of 10 MW with forced cooling, or up to 50 kW in the natural convection mode. A unique design feature provides an experimental position (flux trap) through the center of the core. A flux trap-type reactor is characterized by a thin fuel region adjacent to a good moderator which thermalizes the neutrons and causes the thermal neutron flux to peak in a region (center test hole) accessible for experiments. A relatively high neutron flux is provided for beamport experiments as well.

The reactor core assembly is located eccentrically within a cylindrically-shaped, aluminum-lined pool, approximately 10 feet (3.0 m) in diameter and 30 feet (9.1 m) deep. The reactor core consists of three major regions: fuel, control blade, and reflector. The fuel region has a fixed geometry consisting of eight (8) fuel elements having identical physical dimensions placed vertically around an annulus in between two cylindrical reactor pressure vessels. Each fuel assembly is comprised of 24 circumferential plates containing uranium enriched to approximately 93% in the isotope uranium-235 (^{235}U) as the fuel material. The control blade region is an annular gap between the outer pressure vessel and the beryllium reflector, so that no penetration of the pressure vessels is required. Five control blades operate vertically within this gap, controlling reactor power by varying neutron reflection. The control blades are coupled to drive mechanisms by means of a support and guide extension. The drive mechanisms, mounted on the upper bridge over the reactor pool surface, consist of a motor and a reduction gear drive assembly. The reflector region consists of two concentric right circular annuluses surrounding the control blade region. The inner reflector annulus is a 2.71-inch (6.9-cm) thick solid ring of beryllium metal. The outer reflector annulus consists of vertical elements of graphite canned in aluminum, having a total thickness of 8.89 inches (22.6 cm). The graphite elements are designed to accept the source ends of four (4) radial and two (2) radial-tangential neutron beam tubes.

The fuel region is cooled by a pressurized primary coolant system which, at 10 MW, circulates a nominal 3,750 gpm (14,195 lpm) of light-water coolant through the reactor pressure vessels. The reflector region, the control blade region, and the center test hole are cooled by pool water which is drawn through these regions and circulated through the pool coolant system at a nominal flow rate of 1,100 gpm (4,164 lpm). The heat from the pool and primary coolant systems is transferred to a secondary coolant system by means of separate pool and primary heat exchangers. The heat is then dissipated to the atmosphere through a cooling tower.

The primary design features of the MURR provide the maximum neutron flux to the greatest number of users. The reactor offers the following experimental facilities:

- Center Test Hole (Flux Trap);
- Six Beamports;
- Pneumatic Tube System;
- Graphite Reflector Region Irradiation Positions;
- Bulk Pool Region Irradiation Positions; and
- Thermal Column.

The experimental facilities are described in greater detail in Chapter 10, Experimental Facilities and Utilization.

4.2 Reactor Core

The reactor core consists of the following components: the reactor fuel, the control blades, reflectors, and the reactor core support structure. The reactor core assembly, including the pressure vessels, the fuel element support matrix, and the reflector support assembly are supported from the bottom of the reactor pool in circular sections which also serve as part of the primary and pool coolant piping. Cross-sectional views of the reactor core assembly are shown in Figures 4.1 and 4.2.

Each of the reactor core components are described in greater detail in the subsequent sections.

4.2.1 Reactor Fuel

4.2.1.1 Reactor Fuel System

The fuel material at time of initial startup was a uranium-aluminum alloy with each fuel assembly loaded to a maximum of [redacted] grams of ^{235}U. This type of fuel system had performed very reliably in the Materials Test Reactor (MTR) and the Engineering Test Reactor (ETR) at the Idaho National Engineering Laboratory (INEL), as well as in other reactors throughout the world. However, in order to reduce the fuel cycle cost and the amount of ^{235}U needed per MWD of energy produced at the MURR, a conversion was performed in 1971 to switch to a uranium-aluminide dispersion UAl_x fuel material with a maximum loading of [redacted] grams of ^{235}U per assembly.

[redacted]

FIGURE 4.1
REACTOR ASSEMBLY SECTION IN ELEVATION VIEW

[redacted]

FIGURE 4.2
REACTOR ASSEMBLY SECTIONS IN PLAN VIEW

The MURR [redacted] fuel element is a product of the UAl_x dispersion fuel system development. The UAl_x dispersion fuel system was developed at INEL for the high flux, high power Advanced Test Reactor (ATR) and subsequently used at the MTR and ETR prior to its use at the MURR (Refs. 4.1, 4.2). Several features of the UAl_x dispersion fuel system increase the fuel performance capability in high flux reactors (Refs. 4.3, 4.4, 4.5, 4.6). One of these features is powder dispersion in the fuel matrix causing voidage which will accommodate fission products. The UAl_x structure has exceptional tolerance for fission gas retention and burnable poisons can be readily dispersed in the fuel matrix. Due to the compact core size, the fuel uses high enriched uranium (HEU) to obtain sufficient excess reactivity to be able to operate at 10 MW.

In 1986, the University of Missouri requested that a determination be made by the U.S. Nuclear Regulatory Commission that the MURR has a *Unique Purpose*, as defined by 10 CFR 50.2, and is therefore exempt from the conversion from highly-enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel (Ref. 4.9). The MURR core has a compact design that presently cannot perform its intended function without the use of HEU fuel. However, MURR is actively collaborating with the Department of Energy (DOE) and other U.S. research reactor facilities that use HEU fuel to find a suitable LEU fuel replacement. A more detailed discussion of MURR's conversion status is included in Chapter 18, Highly-Enriched to Low-Enriched Uranium Conversions.

The MURR core was reanalyzed in the mid-1980s to also operate with fuel assemblies which contain a maximum ^{235}U loading of 1,270 grams per assembly (Extended Life Aluminide Fuel or ELAF). The ELAF Program, conducted by the EG&G Idaho for the DOE, the MURR, and the Massachusetts Institute of Technology Reactor (MITR), had an objective of determining whether fuel loading and burnup limits for fuel elements used in University Research Reactors could safely be increased beyond the limits previously allowed by reactor licensing restrictions (Refs. 4.7, 4.8).

The excellent performance of aluminide UAl_x fuels has been consistently demonstrated over the past thirty-five years in test and research reactors such as the ATR and the MURR. MURR has used more than 700 UAl_x fuel elements since 1971 with no failures. ATR has used more than 3,950 UAl_x fuel elements since 1972 with 24 possible leaking fuel elements. All ATR fuel leakage has been caused by pitting corrosion. ATR has had no other type of UAl_x fuel element failure and no failures have occurred during the past thirteen years. There also have been no aluminide fuel element failures at the MURR due to pitting corrosion (Ref. 4.46), but one fuel element was retired early after it had been used for 126 MWDs of its planned 150-MWD usage due to a slight increase in Iodine-131 level in the primary coolant (Ref. 4.47). A corrosion pit develops slowly and penetration of the cladding is easily detected by the installed on-line fission product monitor.

The aluminide fuels loaded to 50 vol% UAl_x are capable of burnup levels greater than 2.3×10^{21} fissions/cm^3. This has been demonstrated by the more than 2,000 fuel elements used to this level in the ATR core. Burnups to this level result in less than 10% swelling of the fuel plates and it has been found that fuel plate swelling of less than 10% has no detrimental effects on fuel plate performance (Refs. 4.7, 4.11, 4.14). Therefore, based on the many years of experience that

TABLE 4-1

[redacted]

TABLE 4-2
SUMMARY SHEET FOR [redacted] FUEL ELEMENTS

	6.2-Kg Core
Fuel Content (grams ^{235}U per element)	[redacted]
Type of Fuel	Aluminide-UAl_x mostly UAl_3 Phase
Fuel Density (grams of ^{235}U loaded per cubic centimeter)	1.5 - 1.6
Boron Content (natural boron per element)	Trace Impurities
K_{eff} (clean core - control blades full out)	1.079
Control Blade Worth (Δk)	0.165
Peak Burnup Density (fissions per cubic centimeter)	$< 2.3 \times 10^{21}$
Energy per Element at Peak Fission Density (MWD per element)	≈ 180*

*Currently limited to 150 MWD per element by the requirements on using the BMI-1 spent fuel shipping cask.

ATR has had using this burnup limit with no failures due to loss of fuel meat integrity, the limiting conditions of operation on peak burnup can safely be established at 2.3×10^{21} fissions/cm^3. MURR studies (Ref. 4.15) and ELAF irradiations (Refs. 4.10, 4.11) have also confirmed that fuel plate swelling, blistering and corrosion will not preclude operation of the MURR fuels to burnup levels of 2.3×10^{21} fissions/cm^3. Additionally, any pitting failure would not result in exposures to the general public in excess of those analyzed in the Maximum Hypothetical Accident (MHA) and the fuel clad failure accident as described in Sections 13.2.1 and 13.2.5, respectively.

4.2.1.2 Fuel Element Description

A drawing of the MURR fuel element is shown in Figure 4.3. Eight such fuel elements comprise the fixed MURR core. This design has proven to be reliable with more than thirty-five years of continuous operation at the MURR, and in other research reactors with no significant failures. Fuel element specifications are summarized in Table 4-3.

[redacted]. Each element is longitudinally-symmetrical with 24 fuel bearing plates. The fuel plates are segments of concentric circles 0.050 inches (1.27 mm) thick separated by a gap of 0.080 inches (2.03 mm). These gaps provide the coolant channels necessary for the removal of the heat generated during the fission process, and for neutron moderation. Additional coolant gaps exist between the innermost fuel plate (No. 1) and the island tube wall (inner pressure vessel), and the outermost fuel plate (No. 24) and the outer pressure vessel wall. These gaps are 0.095 inches (2.41 mm) and 0.075 inches (1.91 mm), respectively. The nominal radius to the innermost plate is 2.77 inches (7.04 cm), and 5.76 inches (14.63 cm) to the outermost plate.

Each fuel plate is fabricated using the "picture frame" method which consists of the fuel compact (fused UAl_x and aluminum powders) encased in a one-piece window-shaped aluminum frame with an aluminum cladding plate welded on both sides, sandwiching the fuel compact. This sandwich is then heated and repeatedly rolled until the fuel plate is 0.050 inches (1.27 mm) thick. This process ensures that a metallurgical bond develops between the fuel material, aluminum powders, and cladding at all interfaces. The fuel meat in each plate is 0.020 inches (0.51 mm) thick. The fuel plates are supported along their vertical edge by slotted aluminum side plates. The fuel plates are permanently fastened into the side plates using a mechanical binding procedure that provides a tensile strength of greater than 150 pounds per linear inch of the side plate joint. This ensures a rigid assembly fully capable of withstanding the hydraulic forces imposed by the primary coolant design velocity of 23 ft/sec (7.01 m/sec). In fact, fuel assemblies of similar construction have withstood severe hydraulic tests at flow velocities up to 50 fe[redacted] without distortion, thus indicating an adequate design margin in this regard [redacted]

[redacted]

FIGURE 4.3
FUEL ELEMENT - PICTORIAL VIEW

TABLE 4-3
SUMMARY OF FUEL ELEMENT SPECIFICATIONS

[redacted]

A handling and guide fixture (end fitting) at each end holds the fuel assembly together. The end fittings are attached to the side plates by rivets. Two rollers are affixed to each end fitting using a hexagon nut and socket head shoulder screw. The rollers allow the fuel assemblies to run on the inner surfaces of the reactor pressure vessels during refueling. Each fuel assembly weighs approximately [redacted]).

The fuel plates and assemblies are identified by individual markings. Numbers are scored onto every fuel plate identifying the specific alloy heats from which they were fabricated. The numbering is also sequential to identify the plates to the proper fuel assembly. Each complete fuel assembly has an identifying number engraved on both side plates using block letters and numbers 2 inches (5.08 cm) high. The depth of the engraving is from 0.004 to 0.006 inches (0.102 to 0.152 mm).

4.2.1.3 Fuel Element Materials of Construction

The fuel material for the UAl_x dispersion fuel system consists of a uranium powder enriched in the isotope ^{235}U (93.0 $\pm$1.0%) dispersed in an aluminum powder. The uranium content of the metal used in the fuel material shall be a minimum of 99.85% by weight as determined by the difference between 100% and the total weight percentage of all the impurities. The isotopic composition of the uranium metal is listed in Table 4-4.

TABLE 4-4
ISOTOPIC COMPOSITION OF URANIUM METAL FOR
UAL_X DISPERSION FUEL SYSTEM

[redacted]

The maximum individual impurities in the fissionable material are shown in Table 4-5. The aluminum for the fuel material is Alcoa 99.8% pure alloy melt stock. The reactivity effect of all impurities in this alloy is less than the equivalent of 15.0 ppm of boron. The fuel material is fabricated according to standard powder-metallurgical and roll-bonding techniques.

The fuel material frame and cover plates are aluminum-alloy 6061, ASTM[1] B 209. A layer of aluminum-alloy 1100 may also be clad to both sides of the frame and one side of a cover

[1]ASTM is the acronym for the American Society for Testing and Materials.

plate. If the cover plate is clad, the aluminum-alloy 1100 side must face the fuel material when the fuel plate is assembled. The reactivity effect of all impurities in these alloys is also limited to the equivalent of 15.0 ppm of boron.

The handling and guide fixtures (end fittings) are aluminum-alloy 6061-T6 or 6061-T651, ASTM B 209, or Alloy 356-T6, ASTM 221 if fabricated from solid bar stock, or Alloy 356-T71, ASTM B 618 or ASTM B 26 if a casting was used. The side plates are aluminum-alloy 6061-T6 or 6061-T651, ASTM B 209. The reactivity effect of all impurities in this alloy is less than the equivalent of 15.0 ppm of boron. The rollers on the end fittings are constructed of 304 Stainless Steel, ASTM A 276.

TABLE 4-5
MAXIMUM IMPURITIES IN URANIUM METAL FOR
UAL_X DISPERSION FUEL SYSTEM

Element	Parts Per Million (ppm)
Aluminum	100.0
Beryllium	15.0
Boron	5.0
Cadmium	20.0
Calcium	100.0
Carbon	500.0
Cobalt	10.0
Copper	100.0
Chromium	200.0
Iron & Nickel	400.0
Lithium	10.0
Magnesium	50.0
Manganese	30.0
Molybdenum	150.0
Lead	25.0
Silicon	300.0
Sodium	25.0

4.2.1.4 Fuel Element Acceptance Criteria

All fuel elements are inspected and tested after fabrication and prior to delivery to the reactor facility. The acceptable fuel plate tolerances are shown in Table 4-6.

TABLE 4-6
FUEL PLATE TOLERANCES

Fuel Plate Dimension	Tolerance
Cladding Thickness	0.015 inches ± 0.003 (0.381 mm ± 0.076)
Fuel Plate Thickness	0.050 inches ± 0.002 (1.270 mm ± 0.051)
Distance Between Fuel Plates	0.080 inches ± 0.008 (2.032 mm ± 0.203)
Fuel Material Thickness	0.020 inches (0.508 mm) - nominal

The acceptance criteria and quality control requirements which reasonably assure proper performance of the MURR fuel elements are described in References 4.17 and 4.18. The fuel elements are also inspected at the reactor facility prior to initial use in the reactor. Fuel elements measurements and alignments are verified using a straight and a curved plane and end fitting templates.

4.2.1.5 Core Loadings

To ensure that the validity of the safety limits is maintained, the reactor core will consist of eight (8) fuel elements (See Fig. 4.4). However, operation at power levels up to 100 watts above shutdown power using a core loading of less than eight (8) fuel elements is permitted for the purposes of reactor calibration or multiplication studies.

Reactor core loadings are normally comprised of a mixture of eight (8) fuel elements with varying stages of burnup (fissions/cm^3). Generally, this mixture consists of two relatively new fuel elements, four fuel elements that are approximately at the "halfway" point, and two fuel elements that are approaching their burnup limit. A mixed core allows the fuel elements to be used to a higher burnup while not exceeding their licensed burnup limit, as specified in the Technical Specifications.

4.2.1.6 Surveillance

One out of every eight (8) fuel elements that have reached their end-of-life will be inspected for anomalies. These inspections, coupled with continuous fission product monitoring and routine water analysis of the primary coolant, provide a means for detection of any possible defects resulting from reactor operation, thereby reducing the possibility of a fission product release to the primary

[redacted]

FIGURE 4.4
MURR CORE LOADING

coolant system. Inspection of the fuel elements at the end of their life allows for decay which results in a reduction in exposure to personnel.

4.2.2 Control Blades

4.2.2.1 Description

The reactivity of the reactor is controlled by five neutron-absorbing control blades, which are all located external to the pressure vessels. Each control blade is coupled to a control blade drive mechanism by means of a support and guide extension (offset mechanism). This offset mechanism is shown in Figure 4.5. Four of the control blades, referred to as the shim blades, are used for coarse adjustments to the neutron density within the reactor core. The fifth control blade is a regulating blade. The low reactivity worth of this blade allows for very fine adjustments in the neutron density in order to maintain the reactor at the desired power level.

The shim blades are constructed of formed boral plate which is, by weight, nominally 50% boron carbide and 50% aluminum. The boron carbide-aluminum mixture is clad with 0.0375 inches (0.9525 mm) of aluminum-alloy 1100 for a nominal blade thickness of 0.175 inches (4.445 mm). All four sides of the blades have a 0.25-inch (6.35-mm) aluminum frame where the blade thickness is 0.25 inches (6.35 mm). The active length of the neutron absorbing material is 34 inches (86.35 cm), and the overall blade length is approximately 40 inches (101.6 cm). The upper 6 inches (15.25 cm) of the shim blade is a 0.25-inch (6.35-mm) thick aluminum mounting plate that is curved to the shape of the blade. Each shim blade occupies approximately 72° of a circular arc around the pressure vessel.

The regulating blade is constructed of stainless steel. The blade has an overall length of approximately 30 inches (76.2 cm) and occupies approximately 18° of the circular arc.

The shim blades are positioned by four control blade drive mechanisms mounted on the upper bridge over the reactor pool surface. Each control blade drive mechanism consists of a 0.02-HP, 115-volt, one-amp, single-phase, 60-cycle motor connected to a lead screw assembly through a reduction gear box and overload clutch. The lead screw assembly converts the rotating motion of the drive motor to the linear motion of the control blades. The overload clutch allows slippage of the lead screw should a shim blade become bound within the control gap. A ball nut coupled directly to a drive tube is driven inward and outward by the lead screw. Connected to the bottom end of the drive tube is an electromagnet which engages a cadmium-plated carbon steel anvil attached above the water level to the end of a lift-rod assembly. The lift-rod assembly allows the positioning of a shim blade through a support and guiding mechanism (offset mechanism) mounted on a pedestal attached to the reflector tank. Limit switches mounted on each control blade drive mechanism stop the drive motor at the top and bottom of travel. These limit switches also provide indication on the reactor control console that the drive mechanism is either in the full-in or full-out position.

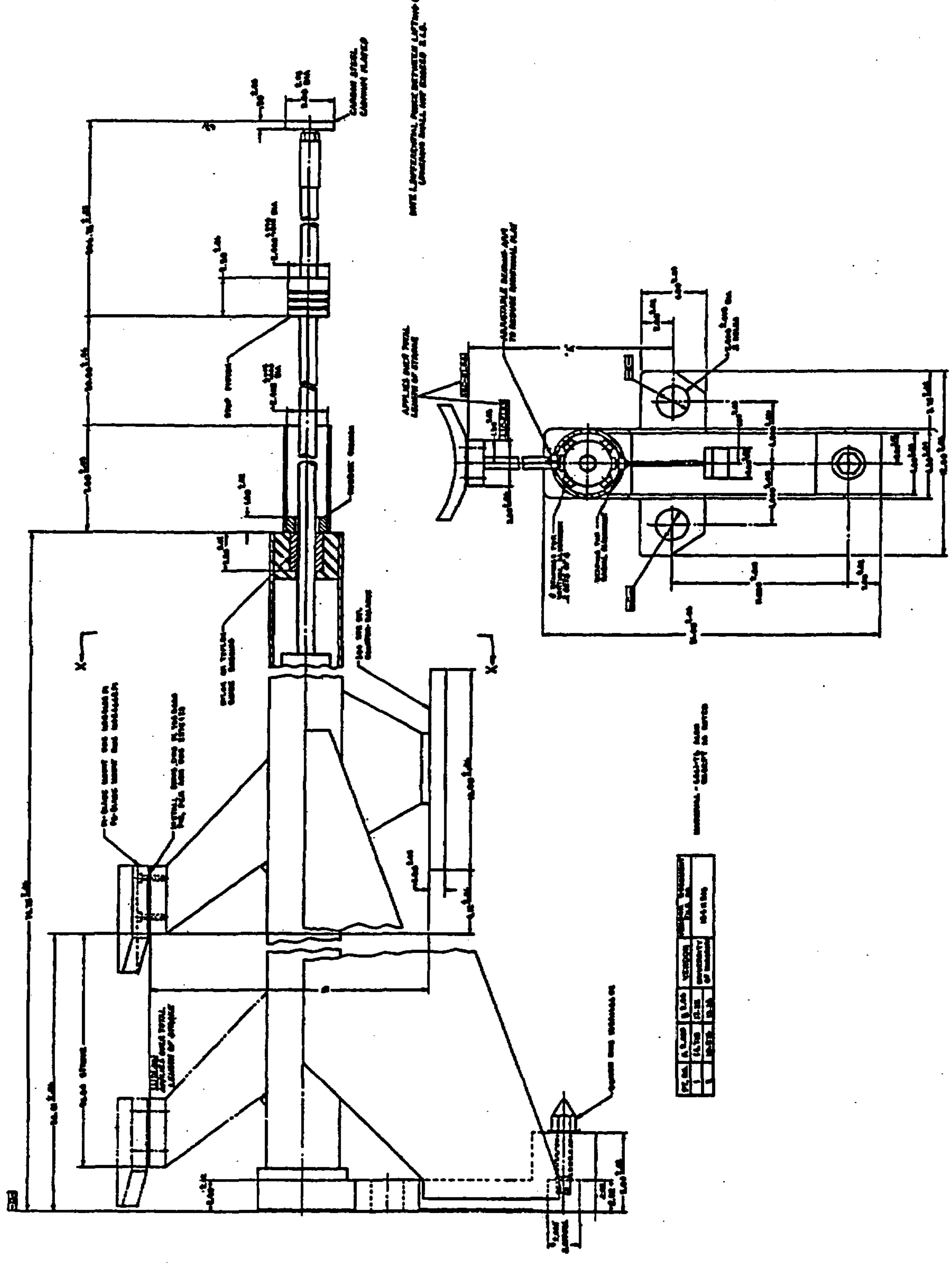

FIGURE 4.5
OFFSET MECHANISM

The shim blade can be withdrawn when the electromagnet is energized. When the reactor is scrammed, the electromagnet is de-energized, releasing the anvil, and allowing the shim blade and lift-rod assembly to drop. Loss of electrical power to the control blade system results in a reactor scram and safe shut down of the reactor. A dash pot assembly cushions the fall of the shim blade during the final 20% of travel.

The regulating blade is positioned by a control blade drive mechanism similar to the ones used for the shim blades with a few exceptions. A scram signal does not insert the regulating blade; therefore, an electromagnet and an anvil are not required. The lift-rod assembly is pinned directly to the drive tube. Also, the regulating blade is driven at a faster speed than the shim blades, requiring a different reduction gear box.

Position indication for the shim blades and the regulating blade is provided by an encoder transducer mounted on each control blade drive mechanism. Analog signals from the shaft encoder are converted into digital form by the Rod Position Indication (RPI) chassis mounted into the control room instrument panel. Digital position indication is displayed on the RPI chassis and an Operator Display Assembly (ODA) installed in the reactor control console. The RPI system is described in greater detail in Section 7.5.6, Control Rod Position Indication.

4.2.2.2 Evaluation of the Control Blades

The reactivity worth and speed of travel for the control blades are sufficient to allow complete control of the reactor system from a shutdown condition to full power operation. The insertion rate for the control blades is adequate to ensure prompt shutdown of the reactor in the event a scram signal is received.

The control blades have been configured to control the excess reactivity needed for 10-MW continuous operation (including xenon override) and will provide a shutdown margin of at least 0.02 Δk. This shutdown margin ensures that the reactor can be shut down from any operating condition even if the most reactive control blade and the regulating blade should remain in the fully-withdrawn position.

The nominal speed of the shim blades is one inch per minute (2.54 cm/min) in the outward direction and two inches per minute (5.08 cm/min) in the inward direction. Speed of the regulating blade is 40 inches per minute (101.6 cm/min) in both directions. Overall travel of the shim blades and the regulating blade is 26.0 inches (66.04 cm), centered around the core vertical centerline.

4.2.2.3 Evaluation of Control Blade Thermal Distortion

The control blades operate in a gap between the outside of the outer reactor pressure vessel and the inside of the beryllium reflector ring. The gap width is maintained by vertical spacer bars which are set into the beryllium reflector. The outside diameter (O.D.) of the outer reactor pressure vessel is 12.556 ±0.010 inches (31.892 ±0.025 cm) and the inside diameter (I.D.) of the beryllium reflector is 13.6895 ±0.0095 inches (34.771 ± 0.024 cm). Therefore, the minimum width of the control blade gap is:

$$(13.680 - 12.566) / 2 = 0.557 \text{ inches } (1.415 \text{ cm}).$$

The thickness of a control blade is 0.250 inches (6.35 mm), leaving a gap of approximately 0.15 inches (3.81 mm) on each side of the blade.

In order for control blade binding due to thermal distortion to occur, some point on a blade must be displaced 0.15 inches (3.81 mm) with respect to the constrained (upper) end. The most probable way that this amount of displacement could occur would be as a result of a differential expansion between the inner and outer longitudinal fibers accompanying a radial temperature gradient through the blade. Assuming the inner radius of a blade is at a higher temperature than the outer radius, it was calculated that a differential expansion of 2.89×10^{-3} inches (7.34×10^{-2} mm) was required to displace the unconstrained (bottom) end of the control blade 0.15 inches (3.81 mm) outward.

The coefficient of linear expansion of aluminum is 13.0×10^{-6} inches / inch °F for the temperature range 68 to 212 °F (20 to 100 °C) (Ref. 4.18). The unconstrained length of a control blade is approximately 26 inches (66.04 cm). Therefore, the differential expansion is:

$$\frac{2.89 \times 10^{-3} \text{ inches}}{26 \text{ inches}} = 1.11 \times 10^{-4} \text{ inches/inch.}$$

The corresponding temperature change is:

$$\frac{1.11 \times 10^{-4} \text{ inches/inch}}{13.0 \times 10^{-6} \text{ inches/inch }^\circ\text{F}} = 8.54\,^\circ\text{F } (4.7\,^\circ\text{C}).$$

The above result implies that the control blade must support a temperature gradient of 8.6 °F (4.8 °C). This temperature difference would require a heat flow radially through the blade of almost 860 Btu/ft^2/h, which is extremely high considering the major heat source in the blades is from the absorption of gamma rays and neutrons. Additionally, temperature gradients will be suppressed by cooling water which enters the blade gap at approximately 100 °F (37.8 °C) and passes on both sides of the blade.

In summary, the clearances provided in the gaps are adequate to accommodate, without significant interference, the maximum anticipated degree of control blade distortion due to thermal gradients impressed during full-power operation. Therefore, control blade binding due to thermal distortion of the control blades is a highly unlikely event, and has not occurred in the thirty-two years of operating at 10 MW.

4.2.2.4 Surveillance

The drop times of each of the four shim blades are measured at intervals as stated in the Technical Specifications. Insertion of the shim blades to the 20% withdrawn position in less than 0.7 seconds ensures that the reactor will be promptly shut down when required. Approximately 91% of the total shim blade worth is inserted at this level. This test provides a means for detecting degradation of the control blades which could affect their mechanical operability. In more than thirty-five years of conducting this surveillance, to date, the shim blades have never failed to meet this specification.

The control blades are inspected at intervals as stated in the Technical Specifications. Periodic inspection of the control blades provides detection of singular blade abnormalities and any potential generic blade deficiencies.

4.2.3 Neutron Moderator and Reflector

In addition to removing the heat generated from the fission process, coolants from the primary and pool systems also serve as the neutron moderator.

Neutron reflection is provided by two concentric right circular annuluses surrounding the reactor pressure vessels. The reflector materials have a rigid structure that retains size and shape, and supports all projected forces and weights. No changes would occur to the reflector materials that would interfere with safe reactor operation or safe shutdown of the reactor. The inner reflector annulus is a 2.71-inch (6.88-cm) thick solid ring of beryllium metal which forms the outer wall of the control blade gap. Additional reflection is provided by an outer reflector annulus consisting of canned vertical elements of graphite having a total thickness of 8.89 inches (22.58 cm). Each graphite element is protected from water penetration by a leak-tight welded aluminum can. The graphite elements are designed to allow horizontal penetration of the beam tubes perpendicular to the beryllium reflector ring. The overall height of the graphite reflector region is 36 inches (91.44 cm), and it is centered vertically with respect to the fueled region. The height of the beryllium reflector is 37 inches (93.98 cm) and it extends 1 inch (2.54 cm) below the graphite reflector. The beryllium reflector ring and the graphite elements are shown in Figure 4.6.

The reactor is designed to allow the removal and/or replacement of both neutron reflectors. Graphite elements are periodically replaced on an individual basis or if a breach of the aluminum cladding is suspected. Safe operation of the reactor may continue until a time when the suspect element can be replaced. Replacement of a graphite element which is not adjacent to a beam tube requires minimal disassembly of reactor equipment. Graphite elements which allow penetration of the beam tubes to the beryllium reflector require a more extensive maintenance procedure. All experimental equipment and associated shielding adjacent to the biological shield for the beamport contiguous to the damaged graphite element must be relocated. A special withdrawal tool is then attached to the face of the biological shield and the removable beamport liner (beam tube). The packing gland assembly which seals the beam tube to the fixed liner is then loosened, allowing the beam tube to be retracted out of the graphite element [approximately 10 inches (25.4 cm)]. The graphite element may then be removed and replaced. Any leakage from the graphite element will not interfere with safe operation of the reactor or prevent safe shutdown of the reactor. Replacement of the beryllium reflector is performed approximately every 26,000 MWD, which corresponds to eight years of operation at our current operating schedule of approximately 150 hours per week. The replacement interval is based on past operational experience with four beryllium reflector replacements after the original beryllium reflector cracked in service. With this replacement schedule, no additional in-service beryllium reflectors have failed. Replacement of the beryllium reflector requires the reactor to be shut down for approximately one week and includes the removal of the upper portion of the reactor pressure vessel and primary inlet piping.

4.2.4 Neutron Startup Source

The Neutron Startup Source was used in the initial startup program, however, (γ, n) and (n, 2n) reactions in the beryllium reflector as a result of activation products (i.e., structural material) from continued reactor operation have far surpassed the startup neutron population which could be produced by this source. Therefore, it is no longer required for reactor startup. The neutron source is presently used for subcritical multiplication measurements for spent fuel storage racks and shipping casks, and to response-check newly installed nuclear instrumentation (NI) detectors.

The neutron source consists of a mixture of 138.4 grams of antimony and 39.03 grams of beryllium powder doubly encapsulated in 304 stainless steel with the outer cylinder being [redacted]. The source was originally licensed for a maximum antimony-124 activity of two curies. The neutron emission rate was found to be too low for the accurate counting statistics needed for subcritical measurements, therefore, the license was amended during the upgrade to 10-MW operation. The neutron startup source is now licensed to a maximum source strength of [redacted].

Periodic irradiation of the neutron source in the reflector region ensures that the resultant activity is sufficient to provide the source strength necessary to perform subcritical measurements and nuclear instrument detector response-checks. A handling line allows the neutron source to be lowered into an irradiation position located in the graphite reflector region. Administrative controls applied to the irradiation time prevent the activity level from exceeding the operating license limit.

[redacted]

4.2.5 Reactor Core Assembly Support Structure

The reactor core assembly support structure consists of the outer reactor pressure vessel, the island tube (inner reactor pressure vessel), and the reflector tank. The entire structure is approximately 13 feet (4 m) in overall length and fabricated of aluminum-alloy 6061-T6 with stainless steel hardware. The design of the support structure ensures that the tops of the reactor

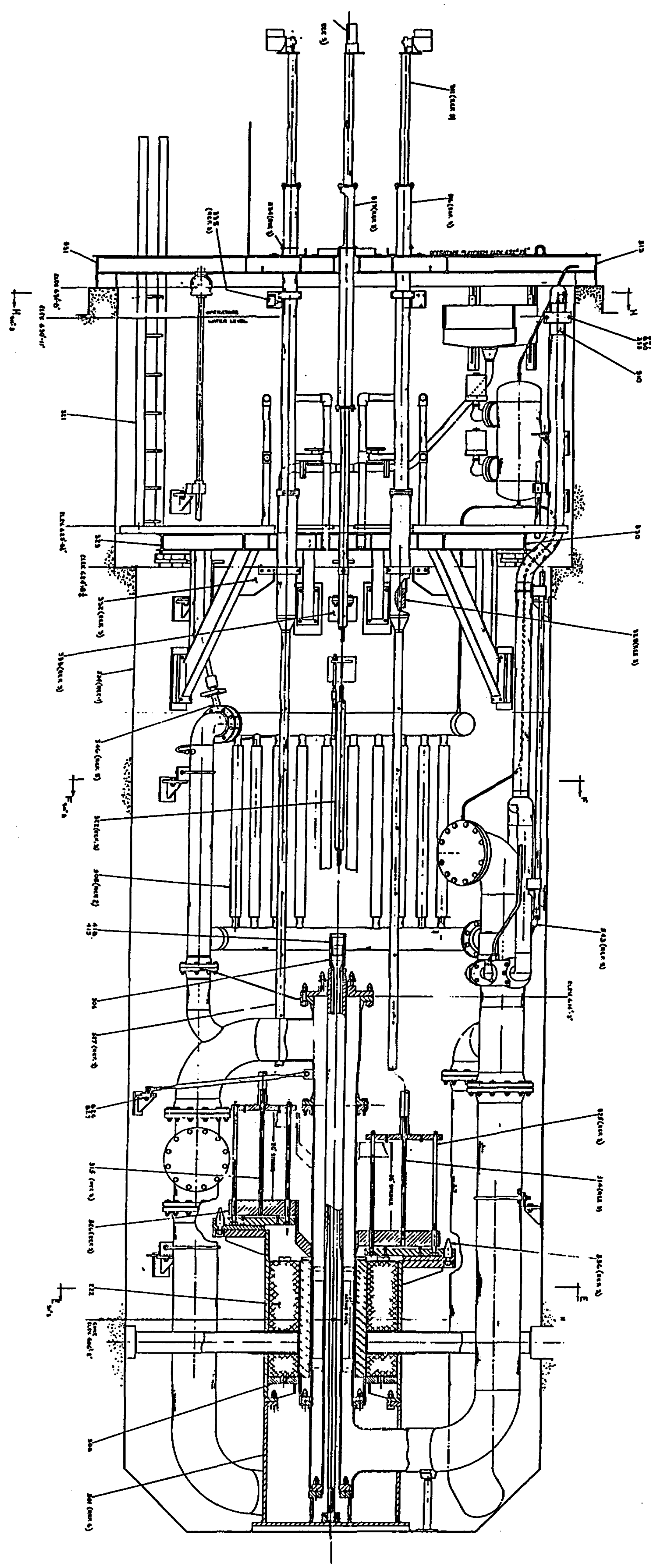

FIGURE 4.6
REACTOR ASSEMBLY AND ARRANGEMENT

pressure vessels will not deflect greater than 0.015 inches (0.381 mm) from their vertical axes or distort in any manner which would interfere with travel of the control blades. Piping connections to the reactor pressure vessels are suitably braced and supported such that piping reactions to the pressure vessels are kept to a minimum.

The inner and outer reactor pressure vessels were designed and fabricated in conformance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section VIII, and applicable ASME Code cases. The outer pressure vessel is comprised of three sections: the concentric reducer, the vessel tube, and the lower tee. These components are mechanically-fastened together by flanged, bolted connections in order to establish a pressure boundary with the primary coolant system. The entire assembly is supported by a vessel support ring welded to the bottom of the lower tee. Primary coolant enters the concentric reducer through a 12-inch nozzle and flows downward through the vessel tube and the fuel region and out the lower tee through an identical nozzle. The fuel elements are positioned within the reactor pressure vessels by a support matrix (spider). Lateral support is provided by concave and convex rollers attached to the fuel element handling and guide fixtures (end fittings). The rollers allow the fuel elements to run on the inner surface of the pressure vessel and the outer surface of the island tube. Vertical alignment is provided by slots in the outside wall of the island tube which engage the top and bottom guide assemblies of the fuel elements. Gravity and downward flow of the primary coolant ensures that the fuel elements are properly seated. Consideration of the manner in which the fuel elements are positioned and supported, combined with the design clearances of the fuel elements, ensures that thermal expansion can be readily accommodated and that no fuel assembly motion of sufficient magnitude to affect reactor behavior will occur.

The island tube is designed to allow experiments to be conducted in the center of the reactor core. The lower end of the island tube terminates in a flange which is bolted to the island tube support flange at the bottom of the outer reactor pressure vessel. These flanges are welded to the vessel support ring so that pool water may enter the top of the island tube to cool the experimental area of the center test hole and discharge through the vessel support ring. The upper end of the island tube is sealed in the pressure vessel by means of a slip flange and packing arrangement (vessel cover). The vessel cover is bolted to the closure flange in the vessel head assembly and is designed to be removed remotely without removal of the island tube.

The reflector tank is comprised of two cylinders, approximately 37 inches (93.98 cm) in diameter, joined together by mated flanges. The beryllium and graphite reflectors occupy the upper cylinder. The graphite reflector elements rest on a reflector support ring which is connected to the upper tank cylinder inner wall just above the mated flanges by means of gussets. The beryllium reflector rests on the lower flange plate of the outer reactor pressure vessel. The upper cylinder wall is penetrated by three 6-inch (15.24-cm) beamport sleeves and three 4-inch (10.16-cm) beamport sleeves. The lower cylinder serves as a discharge plenum for water leaving the reactor pool. Pool water flows downward through the reflector region, control blade gaps, center test hole, and reflector plenum natural convection valve V547 and collects in the lower cylinder prior to discharging to the pool loop coolant system through a penetration in the tank wall. The upper tank cylinder is approximately 4 feet 7 inches (1.4 m) in height and the lower tank cylinder is approximately 3 feet 3.25 inches (1 m) including the base flange.

The vessel support ring which supports the outer reactor pressure vessel and the lower reflector cylinder are welded to a base flange. The base flange is bolted to a base support plate which is welded to the bottom of the pool liner. The base support plate thereby becomes a permanent part of the reactor pool liner supporting and aligning the entire reactor core assembly.

Additional vertical alignment is provided and maintained by three (3) vessel braces (tie-rods) which are attached to the reactor pool liner by pin and plate connections. Brackets welded to the concentric reducer allow the attachment of the tie-rods to the outer pressure vessel. Each tie-rod is loaded as required to hold the reactor pressure vessel in a fixed location relative to the reactor pool.

Five horizontal brackets are installed around the upper periphery of the upper tank cylinder outer wall. Four of these are pedestals for the support and guide extensions (offset mechanisms) of the four control blades while the fifth is for the regulating blade. The reactor core assembly support structure is shown in Figures 4.1 and 4.2.

4.3 <u>Reactor Pool</u>

The reactor core assembly is located near the bottom of an aluminum pool liner which is surrounded by and anchored to a reinforced concrete edifice (biological shield). The pool liner is a defined integral structure consisting of a cylindrical tank, a rectangular fuel transfer canal, and a semi-cylindrical spent fuel storage tank. The liner provides a barrier between the pool water and the concrete surface. The cylindrical tank, which supports the reactor core assembly, is approximately 10 feet (3.0 m) in diameter and 30 feet (9.1 m) deep. The volume of the reactor pool, including the pool coolant system, is approximately 28,000 gallons (105,991 l). Water at a depth of 23 feet 9 inches (7.24 m) covers the reactor during reactor operation, reducing the direct gamma dose rate at the pool surface to approximately one millirem/h at the maximum power level of 10 MW. The liner is constructed of ¼-inch (6.35-mm) thick aluminum-alloy 5086-0 plates welded together and supported by structural members fabricated of aluminum-alloy 6061-T6. The exterior surface of the pool liner is coated with an epoxy-tar compound (Carbomastic #12) to prevent any chemical interactions between the aluminum and concrete surfaces.

All penetrations, regardless of size, are reinforced with strengthening plates and all interior appurtenances are attached to reinforcement pads which are welded to the liner and reinforced into the biological shield concrete to withstand the designed load. A doubling plate is welded to the outside of the spent fuel storage tank to provide additional reinforcing. To distribute the load imposed by a spent fuel shipping cask, the rectangular fuel transfer canal is reinforced by a transverse beam and a doubling plate.

The integrity of the fluid boundary welds and exterior weld seams was verified by a liquid penetrant inspection and a leak test at the time of construction. A field hydro test was also performed after installation but prior to the application of the liner coating. The test consisted of filling the liner with water and visually checking for leaks.

Water in the reactor pool is cooled by the pool coolant system. Pool water is drawn through the reflector region, control blade gaps, and center test hole and circulated through a heat exchanger. The heat is transferred from the heat exchanger to the secondary coolant system and dissipated to the atmosphere through a cooling tower.

Two bridge assemblies provide working surfaces over the reactor pool for inspection, maintenance, and fuel handling. The lower bridge assembly consists of three removable sections, each with two welded, structural-aluminum, cantilevered supports. The supports are bolted to the aluminum bridge grating so that the supports and grating may be removed as a unit. Two vertical studs on the pool ledge locate each support. The cantilevered load of each support is carried by a vertical pad bearing on the aluminum pool liner. Each lower bridge removable section may be lifted out of the pool by the overhead crane after unlocking the section from the vertical studs. The upper bridge assembly consists of two independently-removable sections installed over the reactor pool. The upper bridge has a welded structural-aluminum substructure to which a tread plate is bolted. A hinged cover is provided in the tread plate through which a portable ladder is placed for access to the lower bridge. The ladder fits into sockets installed on both bridge assemblies.

4.4 Biological Shield

The biological shield is a reinforced concrete edifice approximately 31 feet (9.45 m) in height which surrounds the reactor pool liner. The purpose of the biological shield is to provide radiation shielding for personnel working in and around the reactor facility. The following dose rate schedule, based on 10-MW operation, was used in the analysis and design of the biological shield:

(1) At one foot (0.3 m) from the biological shield at the reactor core centerline midway between the beamports, the radiation level shall not exceed 2.5 millirem/h;

(2) At one foot (0.3 m) from the biological shield and three feet (0.9 m) from any experimental facility opening in the shield, the radiation level shall not exceed 2.5 millirem/h;

(3) At three feet (0.9 m) from any experimental facility opening in the biological shield and on the centerline of the opening, the radiation level shall not exceed 2.5 millirem/h; and

(4) At any location at the top surface of the reactor pool, the radiation level shall not exceed 20 millirem/h.

The shielding requirements and analysis of the biological shield are discussed in greater detail in Section 11.1.5.1, Radiation Shielding.

The biological shield is a massive bulk shield structure created by the placement of three lifts of concrete, one on top of the other. Each lift varies in size and thickness. The lower lift is approximately 7.5 feet (2.3 m) thick at its widest point (i.e., adjacent to the thermal column) while the middle and upper lifts vary in thickness from 5.5 to 3 feet (1.7 to 0.9 m), with the smaller dimension at the top. Heavy concrete, using magnetite as the aggregate, is used to construct the lower and middle lifts. The upper lift is constructed of regular concrete. The external surface of the biological shield is protected by ¼-inch (6.35-mm) thick steel plating fastened to the structure by tie-rods. The biological shield is shown in Figure 4.7.

The biological shield is supported by a 3.5-foot (1.1-m) thick concrete pad poured directly onto a 12-foot (3.7-m) caisson. The caisson, constructed of concrete, extends horizontally one foot (0.3 m) out beyond the biological shield in all directions and extends downward to a minimum depth of 6 inches (15 cm) below "sound" bedrock at the lowest point around the edge of the caisson.

The immense size of the biological shield also provides excellent protection against natural phenomena that could result in damage to the reactor core assembly.

4.5 Nuclear Design

The MURR was originally licensed to operate at a thermal power of 5 MW even though the reactor design, except for the coolant and instrumentation systems, allowed for 10-MW operation. The major objectives during the initial nuclear design phase were as follows:

1. To provide enough cold-clean reactivity to enable 400 MWD of core life with either continuous operation at design power or intermittent operation of eight-hour on, sixteen-hour off, and weekend shutdown operating cycle;

2. To provide a reasonable amount of reactivity available for experiments (approximately 2.5%) at the end of life; and

3. To provide enough control rod worth to enable a complete shutdown with one "stuck rod" at any time during operation and to provide reasonable assurance that the rods may be withdrawn far enough to allow continuous operation of all beamtubes without significant local flux depression by the rods.

After the conversion from a U-Al alloy fuel to an aluminide fuel, and after gaining considerable operating experience with the aluminide fuel at the MURR and elsewhere, the core life limit specified in condition (1) above was subsequently extended to 1,200 MWD.

Historically, most of the MURR nuclear design was performed using one- and two-dimensional diffusion theory methods employing four energy groups. The original nuclear design of the core was performed using the one-dimensional diffusion theory programs AIM-6 and WANDA-4,

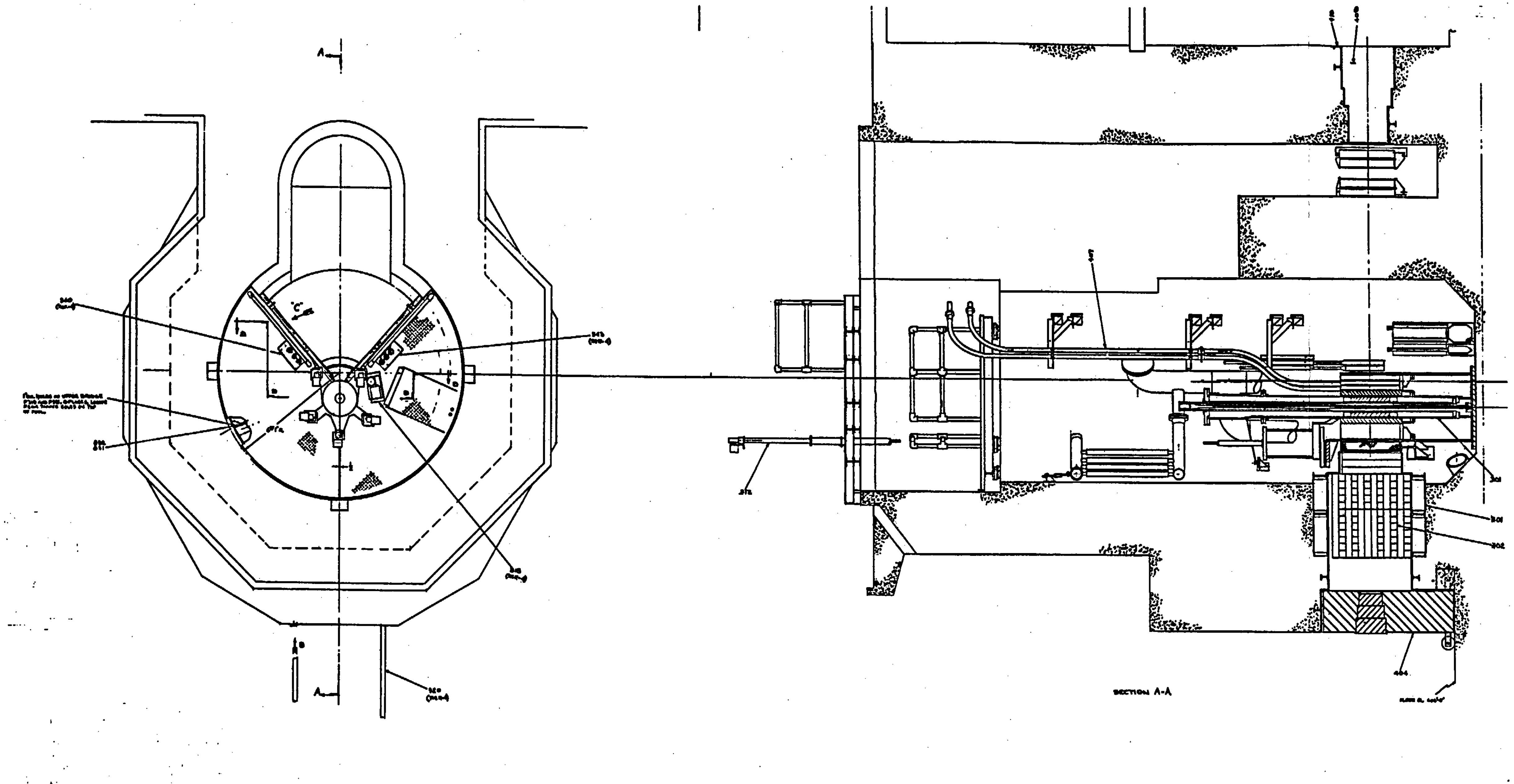

FIGURE 4.7
REACTOR AND BIOLOGICAL SHIELD - PLAN AND SECTION VIEW

and using the two-dimensional diffusion theory program PDQ. These historical calculational methodologies are described in detail in MURR Design Data, Volume II (Ref. 4.48).

In July 1974, Amendment No. 2 to Facility License R-103 authorized reactor operations at an upgraded steady-state power level of 10 MW upon the completion of coolant system upgrades and associated equipment and instrumentation. Analysis and expertise in support of license Amendment No. 2 were provided under contract by the NUS Corporation of Rockville, Maryland. The work provided nucleonics and thermal-hydraulic analysis of the MURR core for operations up to a maximum power level of 10 MW.

The primary tool employed for reactor physics studies in support of the 10-MW power upgrade was the two-dimensional multi-group neutron diffusion theory code EXTERMINATOR-II in conjunction with a MURR four-group macroscopic cross section set. EXTERMINATOR-II is capable of modeling complex reactor systems in X-Y, R-Z or R-Theta geometries. This code calculates the spatial and energy-dependent neutron flux, effective multiplication factor (K_{eff}) and several other relevant reactor parameters.

The results of the analysis are provided in Addendum 4 (February 1973) and Addendum 5 (January 1974) to the MURR Hazards Summary Report. Addendum 4 also provides the changes and bases for the Technical Specifications associated with the upgrades. The safety limits and limiting safety system settings for 6.2-Kg cores [redacted] fuel elements) were derived based on a nuclear peaking factor of 3.676, which corresponds to a peak power density of 1,114 watts/cm^3 of core region. This does not include the engineering peaking factors. This peaking factor was based on a mixed core loading of fresh and depleted [redacted] fuel elements. The peaking factor and other nuclear parameters for the 6.2-Kg cores were determined by employing two-dimensional neutron diffusion code models using the EXTERMINATOR-II code (Ref. 4.20).

In 1986, an additional nucleonics analysis for the MURR core was performed in support of a submittal to the NRC for approval of Amendment No. 20 to the Facility License R-103 (Ref. 4.21). This analysis was primarily done in support of an application for MURR to use a heavier uranium loaded fuel containing 1,270 grams of ^{235}U per element (Extended Life Aluminide Fuel or ELAF). However, the ELAF elements were never fabricated on a production scale, and hence, were never used in MURR cores to date. During the ELAF analyses, newer computer codes were used to verify and confirm many of the nucleonics analyses done earlier for the 6.2-Kg MURR core employing the [redacted] elements and reported in the Hazards Summary Report and its addenda. The following section details the methodology used for that analysis and the results obtained for just the 6.2-Kg core containing [redacted] fuel elements.

To analyze the MURR core with various fuel loadings, the BOLD VENTURE-IV (Refs. 4.22, 4.23) and AMPX-II (Ref. 4.24) code systems were obtained from the Radiation Shielding Information Center at Oak Ridge National Laboratory (ORNL). The BOLD VENTURE-IV code system can perform three-dimensional nucleonics analysis, including depletion, using diffusion theory methods. The code systems and the MURR computer models were benchmarked against the results of a destructive analysis performed on a [redacted] fuel element with an 82.5-MWD power history. The results of the benchmark indicated an excellent agreement between the two. The code analyses methodology of the MURR fuel upgrade is documented in the MURR internal report "MURR Upgrade Neutronics Analysis Using AMPX-II/BOLD VENTURE IV Computation System Benchmarked to the Destructive Analysis of Fuel Element 775F3" (Ref. 4.21). The report describes the use of AMPX-II to generate the cross-section sets for the various nuclides in the MURR components, core models and benchmark calculations performed using BOLD VENTURE, analytical results for R-Z, and θ-R-Z calculations, and conclusions. Appendices of the report provide more detailed information regarding the input to the selected modules of AMPX-II and to BOLD VENTURE. The benchmark work resulted in the development of core models utilizing an advanced and powerful computing tool, AMPX-II/BOLD VENTURE IV, to perform the core analysis and other necessary nucleonics studies for the MURR. Four group cross-section sets for various materials in the MURR system were generated in the form of ISOTXS data files using modules of the AMPX-II system. The files were then used in BOLD VENTURE to determine K_{eff}, spatial flux and power distributions.

The BOLD VENTURE IV computational system is a modular system in which individual modules can be accessed in a desired sequence by a CONTROL module. The major components of the computational system are shown in Figure 4.8. A brief description of the function of each module is also provided. BOLD VENTURE has the capability to model up to a three-dimensional reactor system and up to 18 different types of reactor geometries. However, since the fuel element in the MURR core has curved fuel plates, only the R-Z, θ-R, and θ-R-Z geometric types were considered in the analysis. Figure 4.9 shows a diagram of the R-Z model of the MURR reactor used extensively in the BOLD VENTURE calculations. The symmetry axis corresponds to the actual symmetry axis of the MURR. The numbers are the MURR homogenized zones representing the flux trap, the core section, the control blades, the beryllium and graphite reflectors, etc. The location of the zone boundaries are given in centimeters relative to the top of the model and center line of the reactor. Presented in Table 4-7 is the material specification for each zone in the R-Z model.

Also used in the analysis is a θ-R model to better describe the curved fuel plates and control blades. Figure 4.10 shows the top view of the MURR core and beryllium reflector modeled in the BOLD VENTURE calculations. There are a total of five movable blades used in the reactor. Four shim control blades each occupy approximately 72° of a circular arc around the outer pressure vessel. The regulating control blade covers 17.8° of the circular arc and is made of stainless steel. The code user can easily combine the above R-Z and θ-R models to set up a three-dimensional θ-R-Z MURR reactor model. An R-Z MURR model was set up and depleted up to 650 MWD using a typical reactivity worth of a flux trap sample loading. A comparison of the calculated and measured peak burnup in the fuel plates of element 775-F3 is presented in Fig. 4.11. The "v"symbol curve depicts the measured peak burnup using a gamma scan correlated to fission product sample punchings, whereas the "□" symbol curve depicts the calculated peak burnup using BOLD VENTURE.

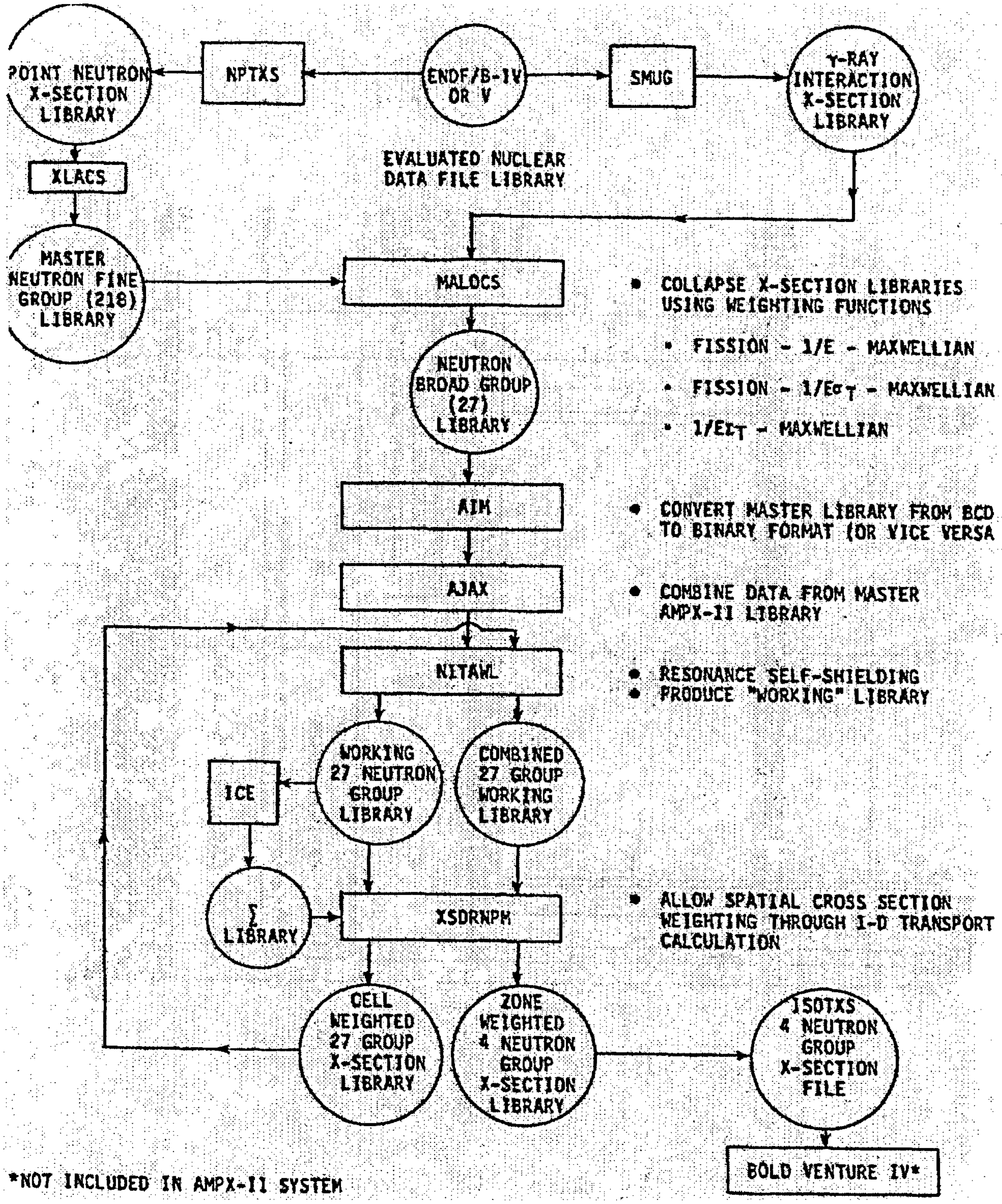

FIGURE 4.8
CALCULATIONAL SEQUENCE OF AMPX-II

[redacted]

FIGURE 4.9
R-Z MODEL OF THE MURR

TABLE 4-7
MURR R-Z ZONE SPECIFICATIONS

Zone No.	Description
1	Flux Trap
2	Inner Aluminum Pressure Vessel
3	Inner Coolant Channel
4	Reactor Core
5	Fuel Extension
6	Fuel Hanger
7	Water
8	Outer Coolant Channel
9	Outer Aluminum Pressure Vessel
10	Water
11	Aluminum Cladding of Control Blade
12	Control Blade Meat
13	Beryllium Reflector
14	Water
15	Upper Graphite Reflector
16	Lower Graphite Reflector
17	Water
18	Aluminum
19	Water

[redacted]

FIGURE 4.10
θ-R MODEL OF THE MURR CORE AND BERYLLIUM REFLECTOR

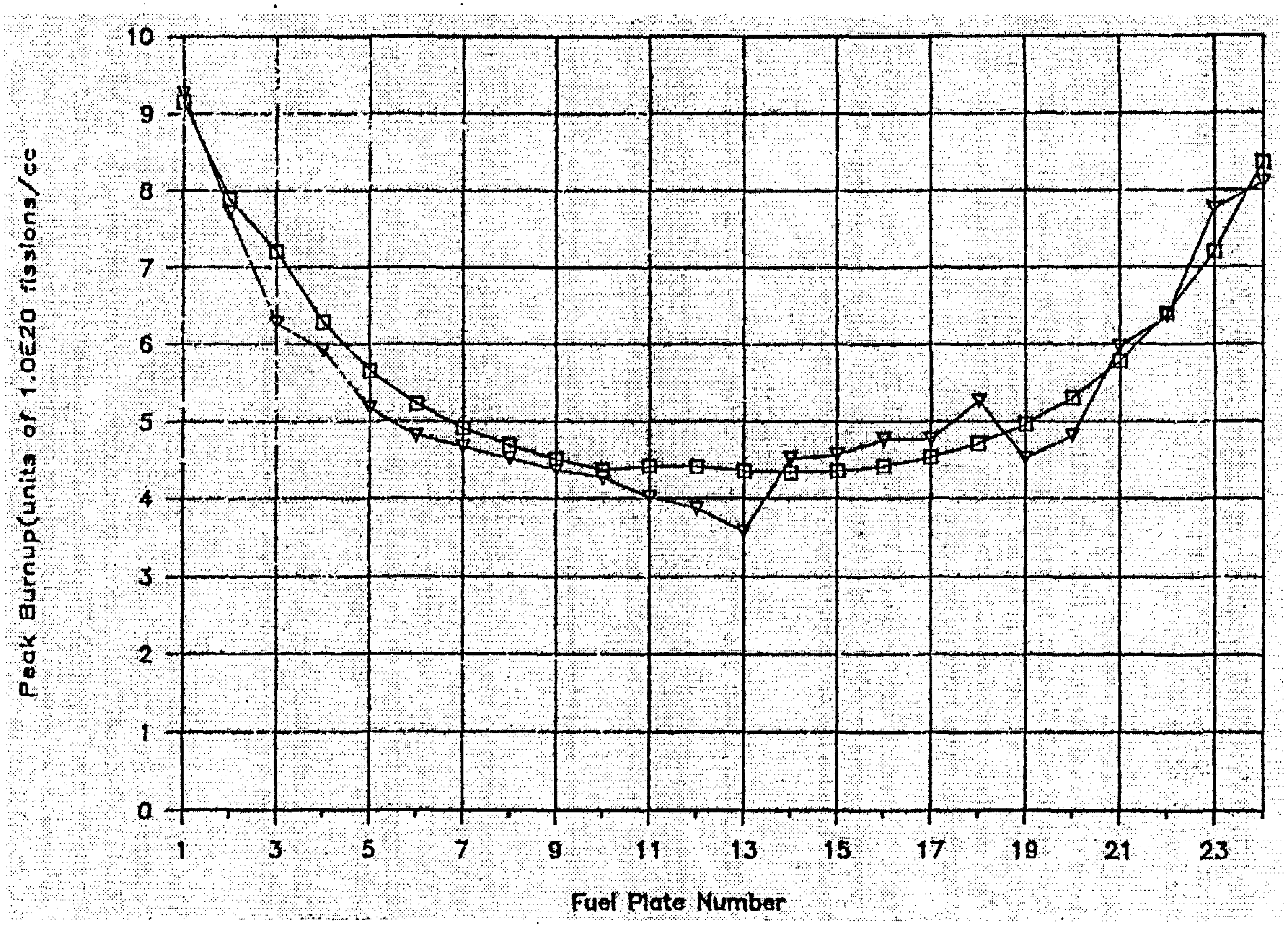

FIGURE 4.11
COMPARISON OF PEAK BURNUP IN FUEL PLATES OF ELEMENT 775F3

The BOLD VENTURE calculations agreed very closely with the measured values of depleted fuel element 775-F3, clearly establishing the validity of AMPX-II/BOLD VENTURE for conducting a neutronic analysis of the MURR core.

BOLD VENTURE runs using R-Z and θ-R-Z models were performed. The purpose of the R-Z runs was (1) to evaluate the core reactivity and the power peaks at the cold clean core condition, and (2) to deplete the 6.2-Kg core. It should be noted that the flux trap model was used only in the benchmark study. Other BOLD VENTURE calculations assumed that only water is in the flux trap region. This results in more thermal neutrons reaching the inside fuel plates which is conservative since the inside fuel plates have the highest power density.

Table 4-8 summarizes the BOLD VENTURE R-Z results for the cold clean 6.2-Kg core at 10-MW power. Excess reactivity was evaluated by performing R-Z runs of the MURR core with the four control blades fully withdrawn and fully inserted into the core.

TABLE 4-8
BOLD VENTURE R-Z RESULTS AT COLD CLEAN CORE CONDITION

6.2-Kg Core	Control Blades Full-Out	Control Blades Full-In
K_{eff}	1.1124	0.9315

The total worth of the four control blades for the 6.2-Kg core was calculated to be negative 0.1809 Δk/k. The total worth of the four blades as reported in the Hazards Summary Report is negative 0.1655 Δk/k. The over prediction of blade worth is due to the fact that the control blade constants being used are based on the actual partial 288-degree control blade coverage, but the R-Z model applies them to a full 360-degree coverage.

The R-Z model was then depleted up to 1,300 MWD. Table 4-9 displays the BOLD VENTURE results for the five burnup steps. The decrease in reactivity due to the burnup of fuel was 4.6%.

The 3D calculations were used to verify the results of the scoping θ-R 2D calculations. When the more detailed 3D information is necessary, the 3D runs are indispensable. Using BOLD VENTURE, the 6.2-Kg core at cold clean core condition was analyzed using the 3D MURR model. The results are given in Table 4-10. These results were similar to the 2D R-Z calculations, indicating that the 2D models are simpler but still provide consistent values. For the "all-blades-full-out" condition, the 2D K_{eff} is in close agreement with the 3D value within the maximum error of 0.43%.

TABLE 4-9
BOLD VENTURE R-Z DEPLETION RESULTS

Cumulative Burnup (MWD)	Control Blade Position (inches withdrawn from full-in)	6.2-Kg Core K_{eff}
0	16.38	1.06697
325	17.61	1.02537
650	19.00	1.02588
970	20.50	1.02605
1,300	20.50	1.0173

In the 3D model of the 6.2-Kg core, two beamport holes and the impurities in the beryllium reflector were not taken into account. According to a previous study (Ref. 4.25), the reactivity for these effects is a negative 0.0034 Δk. When considering this, the analytical K_{eff} at the cold clean core is 1.109, which is higher than the measured value of 1.095 by approximately 1.3%. This difference may be attributed to other details that are not included in the model that cause negative reactivity, such as the thermal column, and impurities in the aluminum of the fuel elements and the reactor pressure vessels.

TABLE 4-10
MURR 3D MODEL RESULTS

6.2-Kg Core		Control Blades Full-Out	Control Blades Full-In
K_{eff}		1.11725	0.95340
Peak Power Density	W/cc	702.9	901.3
	at	Fuel Plate Number-1	Fuel Plate Number-1

The total control blade worths calculated from the R-Z and the θ-R-Z runs are summarized in Table 4-11. The 3D model gives better agreement (See Table 4-10) due to the fact that the control blade constants being used are based on the actual 288-degree coverage which can only be modeled in the θ-R-Z model.

TABLE 4-11
TOTAL CONTROL BLADE WORTH

6.2-Kg Core	R-Z Model	θ-R-Z Model	Experimentally Measured
Reactivity (Δk/k)	0.1813	0.1638	0.1655

Core loadings of fresh and depleted [redacted] fuel elements were modeled to determine which would have the highest power density. The highest power peaking factor was 3.146 for four fresh and four depleted [redacted] fuel elements, corresponding to a maximum power density of 953 watts/cm^2. This is significantly less than the assumed safety limit peaking factor of 3.676, which does not include the engineering hot channel factors on flux. Therefore, fresh and depleted [redacted] fuel elements can be used together at burnups up to 2.3 x 10^{21} fissions/cm^3 and operated with greater margins from DNB and flow instability than those on which the safety limits are based.

4.5.1 <u>Normal Operating Conditions</u>

As determined in the NUS analysis supporting Amendment No. 2 to Facility License R-103 and also the MURR BOLD VENTURE analysis supporting Amendment No. 20 to the license, the operation of MURR cores have the following normal operating conditions.

The reactor shall be operated in a safe manner to assure that the integrity of the fuel element cladding is maintained. Limits have been placed upon the variables of reactor power level, primary coolant flow, reactor inlet water temperature, and pressurizer pressure to reasonably protect the integrity of the fuel and guard against an uncontrolled release of radioactivity.

A complete safety limit analysis for the MURR is presented in Section 4.6.3. A family of curves is presented which relates the reactor inlet water temperature and primary coolant flow to the reactor power level corresponding to a DNBR ratio of 1.2. This is based on the burnout heat flux data experimentally-verified for ATR-type fuel elements. Curves are presented for pressurizer pressures of 60, 75, and 85 psia (413.7, 517.1, and 586.1 kPa). The safety limits were chosen from the results of this analysis for Modes I and II operation, i.e., forced convection operation with greater than 400-gpm (1,514-lpm) flow.

Steady-state reactor operation is prohibited for core flow rates less than 3,200 gpm (12,113 lpm) for Mode I or 1,575 gpm (5,962 lpm) for Mode II by the low flow scram settings of the safety system. The region below 1,575 gpm (5,962 lpm) will only be entered following a reactor shutdown when the primary coolant circulation pumps are secured or during a loss of flow transient in which the reactor scrams, the flow coasts to zero, reverses, and natural convective cooling is established through the decay heat removal system. The RELAP5 analysis of a loss of primary coolant flow transient from the ultraconservative conditions of 11 MW of power, a core flow rate of 3,800 gpm (14,384 lpm), and a reactor inlet coolant temperature of 155 °F (68.3 °C) indicated a maximum coolant channel temperature of 237.5 °F (114.2 °C), which is well below the saturation temperature of 277 °F (136 °C). The analysis of natural convection cooling of the core resulting in an operational limit of 150 kW is presented in Section 4.6.1.

Safe operation under the required safety limits is provided by the application of the Limiting Safety System Settings (LSSSs). The LSSSs are the bases for the scram set points for the reactor safety channels monitoring reactor power level, primary coolant flow, reactor inlet water temperature, and pressurizer pressure. These scrams provide automatic protective actions to prevent a safety limit from being exceeded. Typically scram set points are set more conservatively than the LSSSs.

The LSSSs were chosen such that the true value of any of the four safety-related variables (i.e., reactor power level, primary coolant flow, reactor inlet water temperature, and pressurizer pressure) will not exceed a safety limit under the most severe anticipated transient. Section 4.6.4.1 presents analyses to show that the LSSSs for operating in Modes I, II and III meet this criterion.

4.5.2 Reactor Core Physics Parameters

Calculated and measured reactor core physics parameters for typical 6.2-Kg core loadings are presented in Table 4-12.

4.5.3 Operating Limits

Reactivity limitations assure that the reactor can be controlled and shut down at all times and that the safety limits will not be exceeded. The following are the reactivity limitations.

1. A limit on the core temperature coefficient being more negative than -6.0×10^{5} Δk/k/°F assures that core damage will not occur following any credible step reactivity insertion (See Section 13.2.2).

2. The average core void coefficient of reactivity being more negative than -2.0×10^{-3} Δk/k/% void limits the step reactivity insertion accident as analyzed in Section 13.2.2.

3. The regulating blade total reactivity worth is limited such that any condition resulting in the step insertion of the maximum worth of 6×10^{-3} Δk/k will not result in fuel plate damage (See Section 13.2.2). The limit on the rate of reactivity addition provides for reasonable control response from the reactor operator.

4. Reactivity insertion of 3.0×10^{-4} Δk/k/sec maximum value for the four shim blades operating simultaneously assures that power increases caused by control blade motion

will be safely terminated by the reactor safety system. The reactor startup accident is analyzed in Section 13.2.2.

5. The limit on the shutdown margin of 0.02 Δk with any one shim blade and the regulating blade fully-withdrawn assures that the reactor can be safely shut down and remain subcritical without further operator action. The limit of 0.098 Δk/k on core excess reactivity for MURR operational cores provides further assurance that the required shutdown margin can always be satisfied.

6. A limit on the reactivity worth of each secured removable experiment of 0.006 Δk/k provides assurance that any inadvertent insertion/removal or credible malfunction of a secured removable experiment would not introduce positive reactivity whose consequences would lead to radiation exposures in excess of the 10 CFR 20 limits. The step reactivity insertion accident is analyzed in Section 13.2.2.

7. The absolute value of the reactivity worth of all experiments in the center test hole is limited to 0.006 Δk/k such that the introduction of the maximum reactivity worth of all experiments would not result in damage to the fuel plates (See Section 13.2.2).

8. A limit of a maximum reactivity worth of 0.001 Δk/k for each movable experiment or the movable parts of any individual experiment provides assurance that the movement of movable experiments or movable parts of any experiment will not introduce reactivity transients more severe than one that can be controlled without initiating a reactor safety system action (See Section 13.2.2).

9. A limit of a maximum reactivity worth of 0.0025 Δk/k for each unsecured experiment and a value of 0.006 Δk/k for all unsecured experiments provide assurance that the installation of unsecured experiments resulting in a positive step change of reactivity sufficient to place the reactor on a transient would not cause a violation of a safety limit (See Section 13.2.2).

In order to provide adequate control of reactivity during reactor operation and shutdown modes, operation requirements and limits are placed on the control blades. Operable control blade systems provide the normal method of reactivity control.

10. The control blades shall be maintained so that the maximum distance between the highest and lowest shim blade shall not exceed one inch (2.54 cm) when the reactor power is above 100 kilowatts. This is to restrict maximum neutron flux tilting and ensure the validity of the power peaking factors described in the introduction of Section 4.5.

TABLE 4-12
SUMMARY OF TYPICAL REACTOR AND MAJOR NEUTRON PHYSICS DATA FOR 6.2-Kg CORES HAVING [redacted] FUEL ELEMENTS

Parameter	Description
Reactor Type	Flux Trap; Pressurized; Open Pool-Type; Fully Enriched Uranium; Light-Water Moderated and Cooled; Reflected by Beryllium, Graphite, and Water
Nominal Power	10 MW Maximum
Active Core Volume	33.0 Liters
Average Power Density	0.151 MW/liter at 5 MW 0.303 MW/liter at 10 MW
Average Specific Power	1.613 MW/Kg at 10 MW
Current Operating Cycle	6½ days per week on, ½ day shutdown
Core Lifetime - UAl_X	1,200 MWD
Uniform Critical Mass	Approximately 2.6 Kg
Cold Clean Core K_{eff}; 6.2-Kg Core	1.0795
Power Density; Max/Avg: a. Uniform Loading b. Non-uniform Loading	 3.306 3.676
Control Blade Characteristics: a. Total Worth of Four Shim Blades b. Total Worth of the Regulating Blade c. Total Worth of One Stuck Blade d. Max Worth Per Blade (One Blade In) e. Max K_{eff} With One Stuck Blade - 6.2-Kg Core	 0.1655 Δk/k 0.0023 Δk/k 0.10923 Δk/k 0.0425 Δk/k 0.938 Δk/k
Xenon Worth: a. Equilibrium b. After 16-Hour Shutdown c. Peak to Equilibrium Xenon Ratio	 - 0.0273 Δk/k - 0.0607 Δk/k 2.22

TABLE 4-12
SUMMARY OF TYPICAL REACTOR AND MAJOR NEUTRON PHYSICS DATA
FOR 6.2-Kg CORES HAVING [redacted] FUEL ELEMENTS (cont'd)

Parameter	Description
Samarium Worth: a. Equilibrium b. Weekday Maximum 1. 5-MW Operation 2. 10-MW Operation	- 0.008275 Δk/k - 0.0079 Δk/k - 0.0099 Δk/k
Fuel Burnup; 1,200-MWD Operation	1.51 Kg of ^{235}U
Worth of Burned Fuel and Fission Products	- 0.0324 Δk/k
Maximum Worth of Experiments	- 0.025 Δk/k
Total Reactivity Change From Cold to Hot Operating (Temperature)	- 0.0048 Δk/k
Prompt Neutron Lifetime	5.7×10^{-5} sec
Effective Delayed Neutron Fraction (Assumes Number Fraction = 0.006544)	0.00738
Thermal Fission Fraction	0.874
Mass Coefficient of Reactivity: 6.2-Kg Core	0.0736 (Δk/k)/(Δm/m)
Temperature Coefficients: a. Primary: 85 to 145 °F (29.4 to 62.8 °C) b. Pool: 85 to 115 °F (29.4 to 46.1 °C)	-7.0×10^{-5} Δk/k/°F $+1.34 \times 10^{-4}$ Δk/k/°F
Void Coefficients: a. Primary b. Flux Trap	-2.51×10^{-3} Δk/k/% void $+0.865 \times 10^{-5}$ Δk/k/cc void
Reactivity Increase - Flooding One Beamtube	~ 0.001 (Δk/k)/beamtube

11. The requirement to promptly shut down the reactor in the event of a scram signal (Section 13.2.2) requires the shim blade to be capable of insertion to 20% of the withdrawn position in less than 0.7 seconds. Approximately 91% of the shim blade total worth is inserted at the 20% level. Shim blade worths for 6.2-Kg cores are presented in the introduction to Section 4.5 and in the tables of Section 4.5.2.

12. Studies on reactor fuels (See Section 4.2.1) and their use in core loadings (See introduction to Section 4.5) resulted in limitations on peak burnup of the UAl_x dispersion fuel not to exceed 2.3×10^{21} fissions per cubic centimeter. This burnup limit is correlated to a less than 10% swelling of the fuel plates at end-of-life. Operation of the reactor at significant power levels limits the reactor core loadings at eight (8) fuel elements.

13. The safety limits, as analyzed in Section 4.6.3, are for eight (8) fuel element core loadings. The reactor may be operated up to 100 watts above shutdown power on less than eight elements for purposes of reactor calibration or multiplication studies.

14. Fuel elements that have been inspected and found to be defective are removed from service to assure proper heat removal requirements and to prevent the release of radioactivity into the primary coolant system.

15. The safe storage of irradiated fuel elements requires that the storage system, either by geometry or by use of neutron absorbing materials, maintains the fuel storage at K_{eff} less than 0.9 under all conditions of moderation and reflection.

16. Irradiated fuel elements shall be stored in an array which will permit sufficient natural convection cooling such that the fuel element temperature will not exceed design values.

4.6 Thermal-Hydraulic Design

The MURR is a pressurized, closed-loop, water-cooled system that is situated at the bottom of an open pool. Because of this unique design, there are separate primary and pool coolant loops. A single secondary coolant system removes the heat from the primary and pool coolant loops through heat exchangers and dissipates the heat to the atmosphere via a cooling tower. The components of the MURR coolant systems are described in detail in Chapter 5, Reactor Coolant Systems.

During the design phase of MURR, three separate operating regimes were defined: low-power mode with natural convection cooling; 5-MW operation with forced cooling; and 10-MW operation, also with forced cooling. MURR initially started operating with a single primary coolant pump and heat exchanger with a maximum licensed power level of 5 MW. After upgrading the primary, pool and the secondary coolant systems and associated instrumentation, a request was submitted to the NRC in 1972 to allow for 10-MW operation.

System performance analyses of all three modes of operation were done prior to initial operation. Although most of the earlier analyses depended heavily on analytical calculations to obtain steady-state operating parameters such as pressure drops and flow velocities, the analyses performed in support of the 10-MW power upgrade used more modern computer programs such as RELAP. In the following sections the analyses for all three modes of operation are described. The following worst-case assumptions were used during the forced convection cooling analysis to determine the conditions required to suppress boiling within the core:

1. Most unfavorable power distributions for ganged control rods, i.e., shim rods half-withdrawn from hot clean core having the most unfavorable anticipated fuel loading;

2. Simultaneous occurrence of hot channel and hot spot conditions; and

3. 10% overload relative to nominal power level.

4.6.1 Natural Convective Cooling Analysis

Analysis of natural convective cooling of the core is provided in the Hazards Summary Report, Section 5.5.3 (Ref. 4.27). The reactor core is cooled by natural convection during low-power operation. To accomplish this, a flanged opening is provided in the invert loop. By removing this flange and the pressure vessel head, an open path is provided between the pool and the core thereby allowing natural circulation to take place. The object of the analysis was to determine the natural convection flow rate and the corresponding maximum fuel plate temperature for the initial low-power operation.

During natural convection operation, the reactor core is cooled by pool water flowing in through the open flange, down through the 12-inch (30.48-cm) pipe, up through the pressure vessel and core, and out again into the pool. The flow rate is determined by equating the total system pressure loss to the driving head resulting from the heating of the water in the core. The total pressure loss consists of turbulent friction loss in the piping and pressure vessel, laminar friction loss in the core, and expansion and contraction losses at changes in cross sections throughout the loop.

Calculations for operating at 150 kW resulted in a maximum fuel surface temperature of 230.2 °F (110.1 °C) and a natural convection flow rate of 11.96 lb/sec (5.42 Kg/sec). The core temperature rise is large compared to the rise which occurs coincidentally in the island tube. Consequently, no problems should be encountered with respect to the temperature coefficient of reactivity during cooling of both the core and the island tube by natural convection. The 17 feet (5.2 m) of water required for shielding at low powers is more than sufficient to prevent local boiling in the core. The results of the analysis indicate that a reactor power level of 150 kW can safely be attained with natural convection cooling of the core. This leads to Technical Specification 2.1.c having a 150-kW safety limit for natural convection cooling of the core and Technical Specification 2.2.c having a LSSS of 125% for a maximum power level of 50 kW. The LSSS scram set point occurs at 62.5 kW, thus, a margin of 85.7 kW exists between the LSSS and the safety limit of 150 kW.

4.6.2 Steady-State Forced Cooling Analysis

The steady-state forced cooling analysis for MURR cores operating at the maximum licensed power of 10 MW was first presented in Addendum 4, Appendix F, and Addendum 5, Section 6, of the Hazards Summary Report. This analysis replaced Section 5.5.2, "Analysis of Forced Convection Cooling High Power Operation," of the Hazards Summary Report. The study was conducted by the NUS Corporation in the development of safety limit curves for MURR (Ref. 4.26). All data used in the determination of the MURR safety limits was obtained from the MURR Hazards Summary Report (Refs. 4.27, 4.28, 4.29), the MURR Design Data report (Ref. 4.30), and the MURR hydraulic analysis (Ref. 4.31). A summary of heat transfer data is provided in Table 4-13.

Safe reactor operation is achieved by limiting reactor power to a level which avoids either (1) subcooled boiling burnout (or DNB) or (2) flow instabilities which can lead to premature burnout. Operation above this power limit can cause unpredictably high fuel and clad temperatures and consequentially permanent fuel damage and fission product release to the primary coolant. This condition must be avoided for every core region and for every reactor operating condition.

To avoid DNB, the heat flux at each local section in the core is maintained at a value less than the locally-evaluated DNB heat flux. It is also necessary to avoid any core operating conditions (such as hydraulic instability) that could prematurely reduce the DNB heat flux. The following discussion presents the basis for specifying criteria to include both possibilities.

The MURR fuel assembly geometry is similar to the ATR fuel element so that ATR experience can be applied to the MURR (Refs. 4.32, 4.33). Since the MURR fuel channel length [redacted] about one half that of ATR, the use of ATR test results can, in fact, provide conservatism for the MURR because investigators (Ref. 4.34) have shown higher or equal burnout heat flux levels for shorter channel lengths. Similarly, the shorter channel lengths are less susceptible to the hydraulic instabilities related to incipient bulk boiling.

Preliminary ATR testing indicated that both subcooled boiling burnout and bulk boiling burnout can occur for the range of channel thicknesses then under design consideration (Ref. 4.32). Tests were performed at Argonne in 1963 on three channel thicknesses (0.054, 0.072, and 0.094 inches), and it was found that, for the two thinnest channels (0.054 and 0.072 inches), the burnouts were due to hydraulic instability (or autocatalytic vapor binding) when the coolant reached saturation at the channel exit. Presumably, the hydraulic instabilities led to subnormal flow conditions and a lower burnout heat flux. Subcooled burnout occurred for the 0.094-inch channel before the coolant reached saturation conditions at the channel exit.

The subcooled burnout heat flux data obtained in these tests were 0.6 of the burnout heat flux predicted by the Bernath correlation (Ref. 4.35):

$$\Phi_{DNB} = h_{bo}\left(T_{bo} - T_{sat} + \Delta T_{sub}\right),$$

where:

$$h_{bo} = 10{,}890\left(\frac{D_e}{D_e + D_i}\right) + \frac{48}{D_e^{0.6}} \times V;$$

$$T_{bo} = 1.8\left[57\ln P - 54\left(\frac{P}{P+15}\right) - \frac{V}{4}\right] + 32;$$

T_{sat} = saturation temperature at P, in °F;
ΔT_{sub} = bulk water temperature, degrees subcooling, in °F;
D_e = wetted hydraulic diameter, in ft;
D_i = heated hydraulic diameter, in ft;
V = coolant velocity, in fps; and
P = system pressure in psia.

Subsequent full-scale ATR testing (Ref. 4.33) at Battelle Northwest with a channel thickness of 0.070 inches (0.178 cm) confirmed the earlier test results; namely, that burnout induced by hydraulic instability was the limiting factor for ATR. In addition, it was established that the hydraulic instability condition did not correspond to the initiation of local boiling, but to the beginning of bulk boiling at the channel exit in the region where the coolant enthalpy was highest. Test results also indicated that lateral mixing (in the channel) was quite small.

In view of the ATR experience, and in absence of burnout test results for MURR fuel and at MURR operating conditions, the following safety limit criteria were adopted for the study:

- The coolant exit temperature from the hot channel shall be less than the saturation temperature at the core exit pressure; and

- The local heat flux at any point in the core shall be less than 0.5 of the burnout heat flux as given by the Bernath correlation at that point.

The bulk boiling limitation is adopted to exclude occurrence of the in-core hydraulic instabilities related to incipient bulk boiling. The above burnout heat flux limitation is adopted to provide some additional design safety margin by a reduction of the correlated ATR test data by the factor 0.5/0.6 relative to the original Bernath correlation. The above criteria are sufficient to preclude the possibility of fuel failure and attendant fission product release due to excessive temperatures.

TABLE 4-13
SUMMARY OF HEAT TRANSFER DATA

Parameter	5-MW Operation	10-MW Operation	Units
Average Power Density in Core	151	303	kW/liter
Average Specific Power	806.5	1613	kW/Kg ^{235}U
Reactor Core Coolant			
Total Flow Rate	1,800 (6,813)	3,600 (13,627)	gpm (lpm)
Design Inlet Temperature	140 (60)	140 (60)	°F (°C)
Estimated Pressure Drop Across the Fuel at 150°F (65.6 °C) Ave.	3.1 (21.4)	12.44 (85.8)	psi (kPa)
Min. Pressure at Vessel Inlet Nozzle	70.5 (486.1)	67.4 (464.7)	psia (kPa)
Min. Pressure at the Pressurizer	75.0 (517.1)	75.0 (517.1)	psia (kPa)
Heat Transfer Area	184.3 (17.1)		ft^2 (m^2)
Flow Area - Fuel Elements	0.3231 (0.0300)		ft^2 (m^2)
Total Flow Area	0.3505 (0.0326)		ft^2 (m^2)
Heat Fraction Released in Core	0.93		
Coolant Velocity in Core	11.55 (3.52)	23.10 (7.04)	ft/sec (m/sec)
Average Heat Flux	0.86 x 10^5	1.72 x 10^5	Btu/ft^2-h
Fuel Assembly Average Outlet Temp.	159.3 (70.7)	159.3 (70.7)	°F (°C)
Ave. Power Density in Hot Channel / Ave. Core Power Density	2.469[a]	2.469[a]	
Max. Power Density in Hot Channel / Ave. Power Density in Hot Channel	1.489[a]	1.489[a]	
Hot Spot Position on Fuel (Blades Half Out)	18 (45.7)	18 (45.7)	inches (cm)
Fractional Bulk Rise at Hot Spot	0.701	0.701	

[a]This value assumes worst-case - nonuniform core loading.

TABLE 4-13
SUMMARY OF HEAT TRANSFER DATA (cont'd)

Parameter	5-MW Operation	10-MW Operation	Units
Pool Coolant			
Total Flow Rate	600 (2,271)	1,200 (4,542)	gpm (lpm)
Design Inlet Temperature	100 (37.8)	100 (37.8)	°F (°C)
Estimated Heat Input	600	1,100	kW
Maximum Pool Inlet Velocity	2.5 (0.76)	5.0 (1.52)	ft/sec (m/sec)
Nominal Pressure Drop Across Reflector Region	1.25 (8.62)	5.0 (34.47)	psi (kPa)
Design Velocities: a. Island Region b. Unobstructed Test Hole c. Control Blade Gaps d. Reflectors	 2.0 (0.6) 4.7 (1.4) 2.2 to 2.9 (0.7 to 0.9) variable	 4.0 (1.2) 9.4 (2.9) 4.5 to 5.9 (1.4 to 1.8) variable	 ft/sec (m/sec) ft/sec (m/sec) ft/sec (m/sec)
Design Temperature Rise in Island	5.8	5	°F

The BOLERO program was used to perform the calculations which determine local conditions of enthalpy, heat flux, and DNB heat flux for the core hot channel (Ref. 4.36). Table 4-14 presents a summary of hot channel factors used in the analysis. Since the Bernath burnout heat flux depends on absolute pressure, it was necessary to calculate the absolute pressure at the core exit for each set of inlet water conditions and core power. Since most BOLERO input is dependent on absolute pressure and on either flow rate or power, a special computer program, MURRPGM, was written to generate consistent input for all the cases needed for the study (Ref. 4.37).

The MURRPGM program was developed to calculate the absolute pressure at the core outlet for every combination of operating conditions in the study. Since the core outlet pressure calculation required the same data as BOLERO, the program was expanded further to generate input for the BOLERO program.

The pressure drop from the pressurizer to the core outlet was calculated by correcting individual Δp components as given in Reference 4.31 to new flow, temperature, and core power conditions (See Table 4-15). The new Δp components were then totaled and the result was subtracted from the desired pressurizer operating pressure (60, 75, or 85 psia) to obtain the absolute pressure at the core outlet.

MURRPGM includes an iterative scheme to determine the core power level that would cause incipient bulk boiling at the hot channel exit and interpolation routines to evaluate intermediate fluid

TABLE 4-14
SUMMARY OF MURR HOT CHANNEL FACTORS

	Safety Limits Basis
On Enthalpy Rise	
Power-related Factors	
Nuclear Peaking Factors	
Radial	2.220
Non-uniform Burnup	1.112
Local (Circumferential)	1.040
Axial	1.000
Engineering Hot Channel Factors	
Fuel Content Variation	1.030
Fuel Thickness/Width Variation	1.030
Overall Product	2.72
Flow-related Factors	
Core/Loop Flow Fraction	1.000
Assembly Minimum/Average Flow Fraction	1.000
Channel Minimum/Average Flow Fraction	
Inlet Variation	1.000
Width Variation	1.000
Thickness Variation	1./1.080
Within Channel Minimum/Average Flow Fraction	
Thickness Variation	1./1.050
Effective Flow Area	0.3231/0.3505
Overall Product	0.81
On Heat Flux	
Power-related Factors	
Nuclear Peaking Factors	
Radial	2.220
Non-uniform Burnup	1.112
Local (Circumferential)	1.040
Axial	1.432
Engineering Hot Channel Factors on Flux	
Fuel Content Variation	1.030
Fuel Thickness/Width Variation	1.150
Overall Product	4.35
Energy Fraction Generated in Fuel Plate	0.930

TABLE 4-15
REFERENCE PRESSURE DROP DATA*

Component**	ΔP_0 (psi)	Q_0 (gpm)	T_0 (°F)	Frictional	In Core
1, 2, 3	3.259	1,800	155	yes	no
4	0.2689	1,800	155	no	no
5, 6, 7, 8, 9, 10	4.08	3,600	155	yes	no
11	0.1977	3,600	155	yes	no
12	0.8980	3,600	155	no	no
13	12.35	3,600	165	yes	yes

* Data from Reference 4.31

** Component description using notation of Reference 4.31

1. Across pressurizer surge line to pressurizer outlet
2. Across 5 feet of 8-inch pipe
3. Across 8-inch Y-strainer
4. Across 8-inch/12-inch expansion
5. Across 80 feet of 12-inch pipe
6. Across four 12-inch 90-degree elbows
7. Across three 12-inch 45-degree elbows
8. Across one 12-inch butterfly valve (507B)
9. Across one 12-inch swing check valve (502)
10. Across entrance to annular pressure vessel
11. Across 6 feet of annular pressure vessel
12. Across entrance to fuel element plates
13. Across Core ...25.5 inches of fuel element plates ... to core exit

property values from tabulated input values using absolute pressure as the independent variable. MURRPGM also provides transformations to generate BOLERO input from non-standard BOLERO flow and power units.

The BOLERO program performs all the necessary thermal-hydraulic calculations required to establish the minimum ratio of the local burnout heat flux to the local surface heat flux (DNBR) for a single coolant channel. BOLERO input specifies the single channel dimensions, operating conditions, and the Bernath DNB correlation and its parameters.

The single channel analyzed in BOLERO is a representation of the thermally limiting channel (or hot channel). The channel power is 2.72 times average channel power, and the channel flow rate is 0.81 times average channel flow rate. The bases for these data and for the local heat flux multipliers are given in Table 4-14. Safety limits for reactor operation implicitly depend on these power and flow-related factors. The normalized axial power distribution used for the channel is given in Figure 1 of TM-WRP-62-10 (Ref. 4.30). This power distribution occurs at the beginning core life when the control blades are partially inserted and represents the most limiting condition during core life due to the high flux level at the channel exit. Channel dimensions such as flow area [0.3505 ft^2 (0.0326 m^2)], heat transfer surface area [184.28 ft^2 (17.12 m^2)] and core length [2.0 ft (0.6 m)] are developed from nominal core dimensions. The effects of worst-case dimensions are included in the corresponding hot channel factors.

BOLERO input data for the Bernath DNB correlation include a DNB heat flux multiplier (0.5), a heated-to-wetted perimeter ratio (0.924) and a saturation temperature corresponding to the absolute pressure at the core exit (available from the MURRPGM program results) for each core power, pressurizer pressure, and core inlet condition. This approach ensures the correct Bernath DNB heat flux when the minimum DNBR occurs at the channel exit, and produces a conservative result when the minimum DNBR occurs elsewhere in the channel.

4.6.3 Safety Limit Analysis

The NUS study as described in Section 4.6.2 resulted in the generation of reactor power limits shown in Table 4-16 and three safety limit curves corresponding to pressurizer pressures of 60, 75, and 85 psia (413.7, 517.1, and 586.1 kPa) shown in Figures 4.12, 4.13, and 4.14. Established by the NUS study in 1973-1974, these results are the bases for current safety limits and maximum power levels. Verification of the NUS safety limit analysis was provided in 1988 in a MURR fuel upgrade utilizing a COBRA-3C/MURR reactor code version as discussed later in this subsection.

The underscored entries in Table 4-16 are the power limits as established by the criterion of avoiding any bulk boiling of the coolant, whereas the remaining entries reflect the thermal limits established by the subcooled burnout criterion. The safety limit criterion on incipient bulk boiling of the coolant is associated with experimentally-observed premature burnout caused by hydraulic instabilities. The power limits for coolant flow rates greater than 2,800 gpm (10,599 lpm) are dictated by the burnout criterion, while for flow rates less than 800 gpm (3,028 lpm), the incipient bulk boiling criterion dictates the safe power level. Those values limited by a bulk boiling criterion dictate the safe power level. Those values limited by bulk boiling (underscored values) were immediately evident because BOLERO results indicated that

$$\mathrm{DNBR} = \frac{\Phi\ \mathrm{DNB}}{\Phi\ \mathrm{LOCAL}} > 1.0$$

for the initial core power estimate evaluated by the MURRPGM program at the threshold of bulk boiling. No further iterative procedure was required because any core power increase to reach the DNB flux limit would also violate the bulk boiling criterion.

TABLE 4-16
SAFETY LIMITS FOR MURR OPERATIONS

Maximum Allowable Core Power Level (MW) with Pressurizer at 60 PSIA										
INLET WATER CONDITIONS										
Temp	Flow Rate (GPM)									
°F	400	800	1,200	1,600	2,000	2,400	2,800	3,200	3,600	4,000
120	<u>3.011</u>	5.870	7.980	9.843	11.574	13.099	14.426	15.450	16.217	16.654
140	<u>2.650</u>	<u>5.262</u>	7.299	9.035	10.582	11.960	13.155	14.071	14.729	15.075
160	<u>2.292</u>	<u>4.546</u>	6.675	8.202	9.600	10.822	11.877	12.669	13.228	13.501
180	<u>1.935</u>	<u>3.834</u>	<u>5.667</u>	7.409	8.612	9.685	10.603	11.267	11.715	11.906
200	<u>1.583</u>	<u>3.131</u>	<u>4.615</u>	<u>6.009</u>	<u>7.282</u>	<u>8.400</u>	9.301	9.863	10.204	10.267

Maximum Allowable Core Power Level (MW) with Pressurizer at 75 PSIA										
INLET WATER CONDITIONS										
Temp	Flow Rate (GPM)									
°F	400	800	1,200	1,600	2,000	2,400	2,800	3,200	3,600	4,000
120	<u>3.278</u>	6.334	8.647	10.742	12.668	14.435	16.050	17.394	18.532	19.438
140	<u>2.916</u>	<u>5.798</u>	7.939	9.906	11.667	13.282	14.746	15.967	16.993	17.787
160	<u>2.556</u>	<u>5.080</u>	7.317	9.067	10.676	12.138	13.458	14.534	15.437	16.139
180	<u>2.197</u>	<u>4.363</u>	<u>6.474</u>	8.236	9.680	10.988	12.152	13.104	13.892	14.467
200	<u>1.843</u>	<u>3.656</u>	<u>5.415</u>	<u>7.099</u>	<u>8.686</u>	9.845	10.868	11.689	12.339	12.810

Maximum Allowable Core Power Level (MW) with Pressurizer at 85 PSIA										
INLET WATER CONDITIONS										
Temp	Flow Rate (GPM)									
°F	400	800	1,200	1,600	2,000	2,400	2,800	3,200	3,600	4,000
120	<u>3.292</u>	<u>6.584</u>	9.097	11.299	13.336	15.227	16.878	18.356	19.662	20.776
140	<u>2.930</u>	<u>5.860</u>	8.421	10.452	12.326	14.062	15.568	16.920	18.108	19.102
160	<u>2.570</u>	<u>5.139</u>	<u>7.709</u>	9.607	11.326	12.908	14.271	15.488	16.549	17.434
180	<u>2.211</u>	<u>4.421</u>	<u>6.632</u>	8.766	10.319	11.749	12.969	14.051	14.980	15.741
200	<u>1.856</u>	<u>3.712</u>	<u>5.568</u>	<u>7.424</u>	<u>9.280</u>	10.593	11.673	12.617	13.418	14.069

Note: Underlined power levels are limited by bulk boiling.

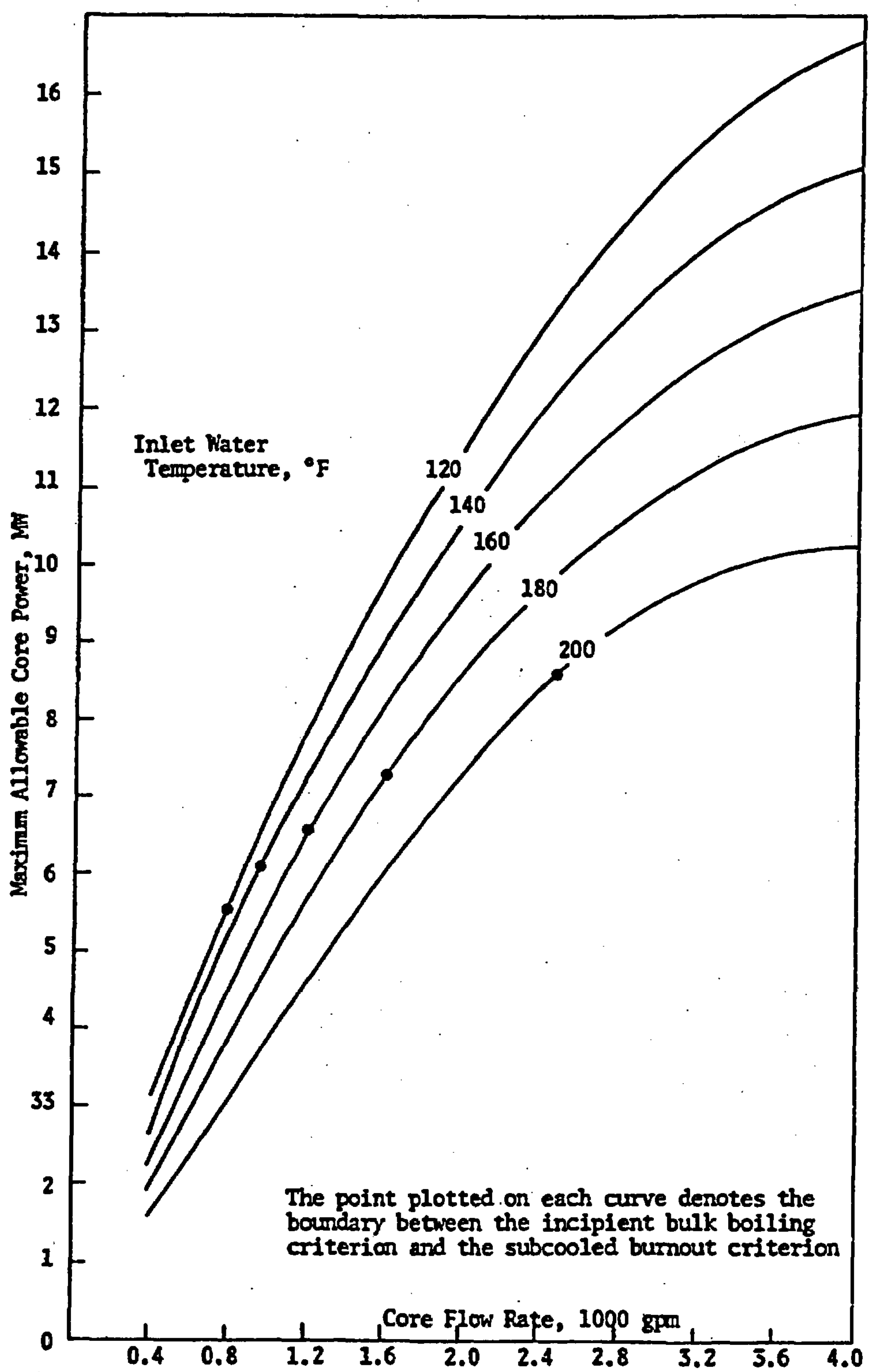

FIGURE 4.12
MURR SAFETY LIMIT CURVES (PRESSURIZER AT 60 PSIA)

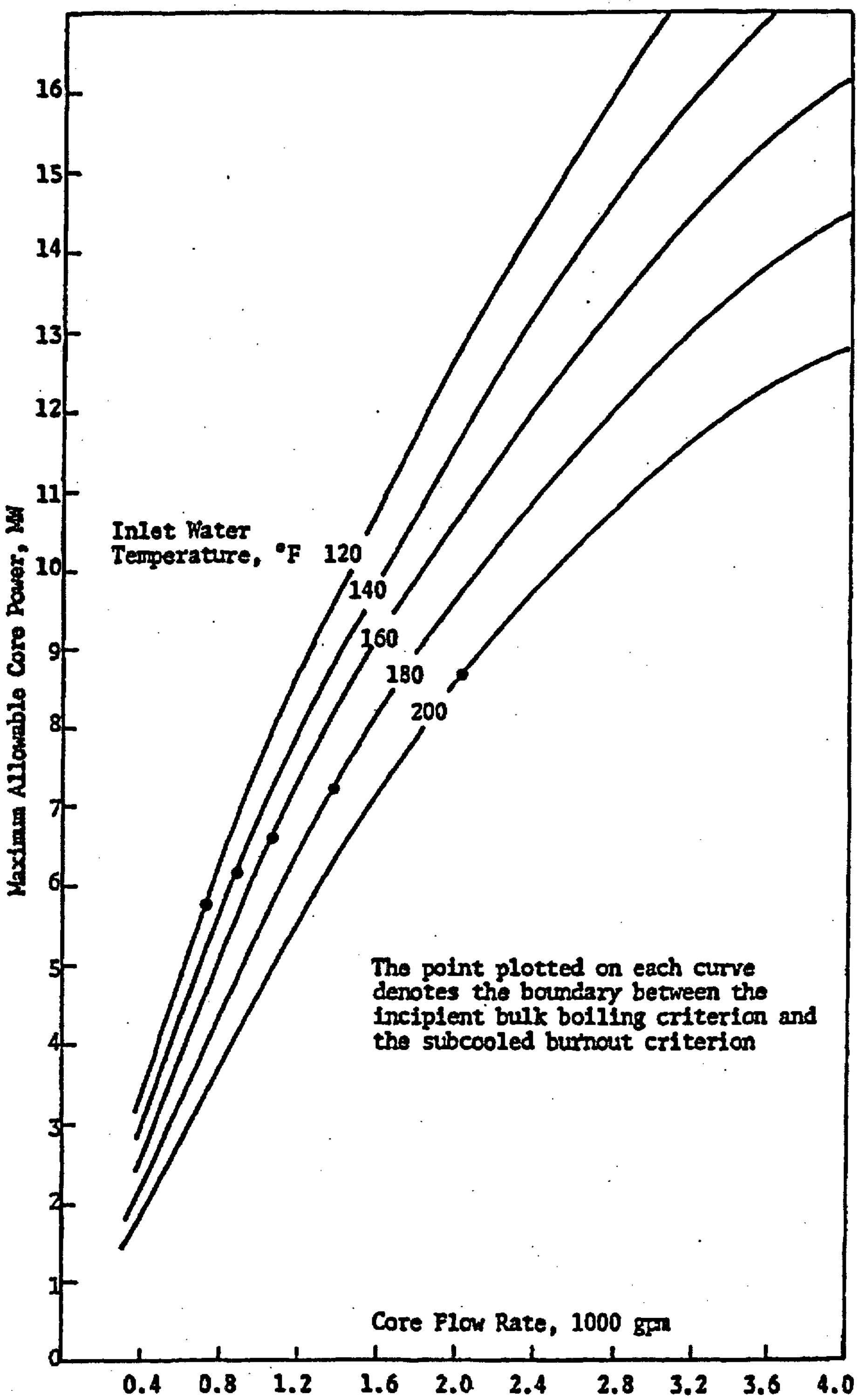

FIGURE 4.13
MURR SAFETY LIMIT CURVES (PRESSURIZER AT 75 PSIA)

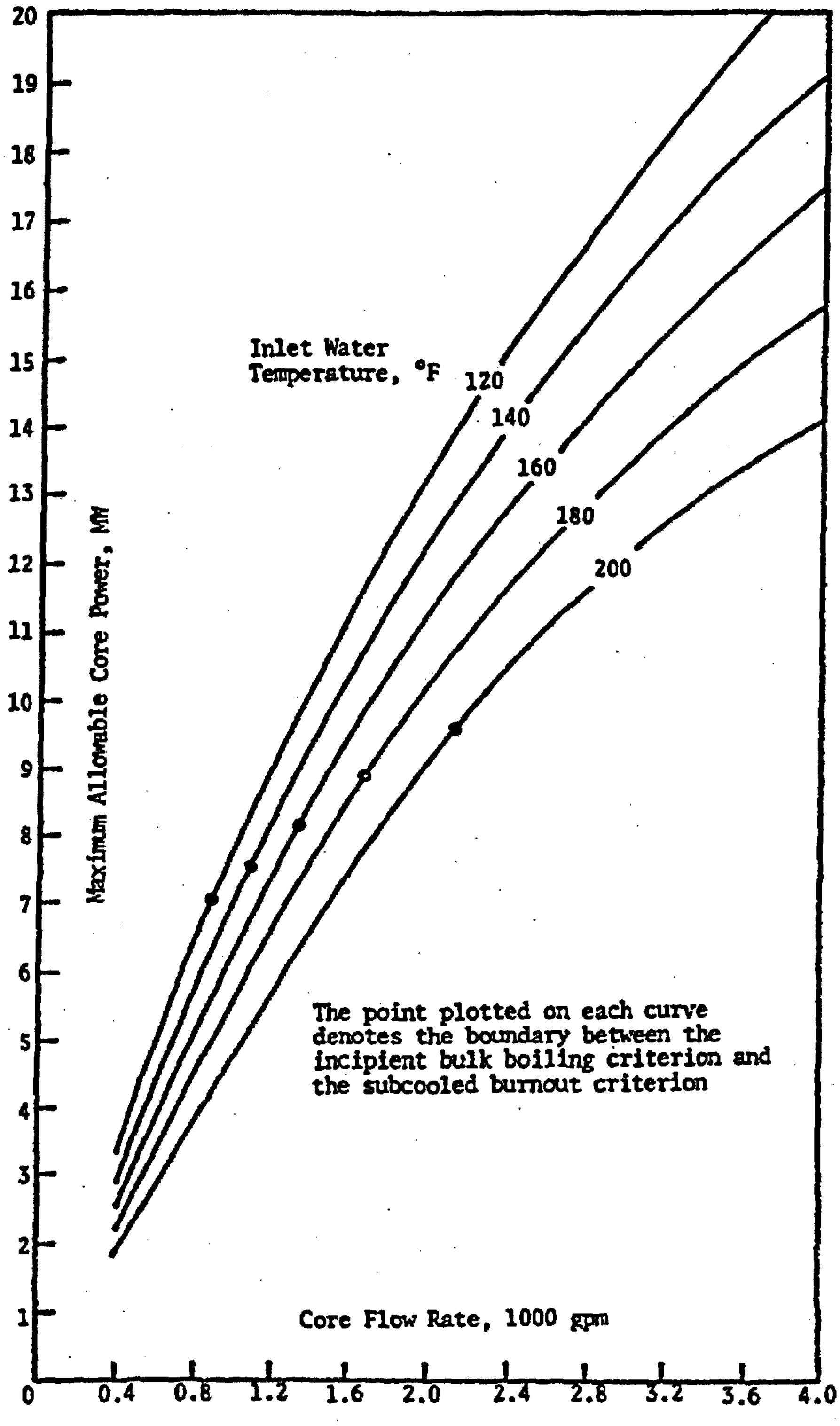

FIGURE 4.14
MURR SAFETY LIMIT CURVES (PRESSURIZER AT 85 PSIA)

The core power levels limited by the DNB criterion were the result of an iterative procedure. The procedure included the sequential use of the MURRPGM program to calculate the absolute pressure at the core exit and the BOLERO program to calculate the DNBR. The DNB-limited power levels in Table 4-16 were determined by terminating the iteration procedure when the DNBR = 1.0000+0.01, where DNB heat flux is defined as 0.5 of the burnout heat flux as given by the Bernath correlation.

As described above, the BOLERO (Ref. 4.36) and MURRPGM (Ref. 4.37) computer codes were used to model the steady-state thermodynamic and hydrodynamic operation of the MURR. The singularity of design makes the experimental determination of an exact DNB correlation for every individual research reactor prohibitively expensive. A literature review demonstrates that the ATR closely compares to the MURR in fuel design, although the ATR core is twice as long as that of the MURR. Extensive experimental tests (Refs. 4.32, 4.33) were made on mockups of the ATR coolant channels to determine the most accurate of the numerous DNB burnout correlations available. It was observed that the limiting conditions for ATR operation were set by subcooled burnout due to hydraulic instabilities in the hot channel. This was found to occur at 60% of the DNB heat flux predicted by the widely used Bernath correlation (Ref. 4.35). To provide a reasonable margin between predicted DNB and the MURR safety limits, the safety limit criterion was established such that the local heat flux at any point in the core shall be less than 50% of the burnout heat flux given by the Bernath correlation at that point.

A parameter of safety significance for nuclear reactors is the DNB ratio (DNBR) defined as the ratio of the anticipated DNB heat flux to the actual peak reactor core heat flux. Thus, DNB and associated fuel damage will not occur as long as the DNBR is greater than 1.0. For the study in question, safety limits were derived based on a DNBR of 2.0 relative to the Bernath correlation, i.e., conditions were limited to 50% of the predicted burnout heat flux. However, in relation to the experimentally-observed burnout at 60% of the Bernath prediction, one can say with assurance that the DNBR for the MURR safety limit curves is not less than 60%/50% or 1.2. Thus, the derived curves allow sufficient margin between the safety limits and actual predicted DNB. The usual conservatism of worst-case power peaking and non-uniform fuel loading and appropriate hot channel factors were used, lending greater assurance that the MURR will not approach DNB under the most severe anticipated transient from a power level of 10 MW.

Figures 4.12, 4.13, and 4.14 illustrate the effects of core operating conditions on the maximum allowable power for safe operation of the MURR. The trends noted here generally represent the behavior of the two design criteria for various core operating conditions. These three curves together define a four-dimensional safety limit envelope prescribing limiting combinations of values for reactor power, pressurized pressure, reactor inlet water temperature and primary coolant flow.

The variable most strongly affecting safe core operation is core flow rate. The higher the core flow rate, the higher the maximum allowable power level. The effect is essentially linear at low core flow rates where the bulk boiling criterion is controlling and becomes non-linear as the flow rate is increased into the DNB controlled regions. The non-linearity in the safety limit is more pronounced for higher inlet water temperatures. Two competitive coolant flow related phenomena are responsible for this observed behavior. An increase in the coolant flow rate results in (1) lower absolute pressures at the core exit which, in turn, decreases the water saturation temperature and, thereby, decreases the Bernath burnout heat flux limit; and (2) higher predictions of the Bernath burnout heat flux limit with increasing coolant velocity.

The allowable power limit is inversely related to the reactor inlet water temperatures. This is readily understood in terms of a higher permissible core power level for an increased inlet subcooling; that is, the channel power to achieve incipient bulk boiling or local burnout increases as the inlet subcooling increases (coolant inlet temperature decreases) with all other variables held constant.

The effect of pressurizer pressure is available from a comparison of corresponding curves on Figures 4.12, 4.13 and 4.14. Clearly, higher pressurizer pressure results in an increase in the safety limits on core power due to the increase in the coolant saturation temperature and the pronounced absolute pressure dependence of Bernath correlation at low absolute pressure. As already noted, the influence of the coolant flow rate on the channel exit pressure and the dependence of the Bernath correlation on absolute pressure are responsible for the slope change observed in the safety limit curves of Figures 4.12, 4.13, and 4.14.

Operation of the MURR within this safety envelope will prohibit fuel meltdown or cladding damage as a result of DNB. To evaluate safety limits for pressurizer pressures intermediate to the three pressures considered, interpolation will be used. For example, the true values of core flow and inlet temperature in a particular case may be applied to the three curves to obtain a three point relationship between pressurizer pressure and the limiting reactor power. The safety limit on reactor power level will then be fixed by interpolation. For pressurizer pressures below 60 psia (413.7 kPa), extrapolation will be used to determine the safety limits. For pressurizer pressures greater than 85 psia (586.1 kPa), the 85 psia (586.1 kPa) curves shall be used and no pressure extrapolation shall be permitted.

Verification of the NUS Safety Limit Analysis, as previously discussed, was provided in 1988 in response to a request from the NRC for additional information for License Amendment No. 20 (Ref. 4.38). BOLERO was used by NUS for analysis of the safety limits; however, the code is proprietary and a rather old program. To verify the safety limit analysis, COBRA-3C/RERTR (Ref. 4.39) was purchased from ANL for the MURR fuel upgrade safety limit analysis. This is a modified version of COBRA-3C/MIT (Ref. 4.40); a thermal-hydraulic subchannel analysis code. The purpose for modifying the MIT version was to make the code more suitable for research and test reactors which are operated at low pressures and temperatures, and which may use plate-type fuel elements like those in the MURR core. It should be noted that in the Reduced Enrichment Research and Test Reactor (RERTR) program (Ref. 4.41), COBRA-3C/RERTR has been extensively used for steady-state thermal hydraulic analysis for the various research reactors.

4.6.4 Limiting Safety System Settings

4.6.4.1 Introduction

Limiting Safety System Settings (LSSS) were established from the results of the safety limit analysis as presented in Section 4.6.3. The LSSSs are settings for automatic protection devices which prevent the MURR safety limits from being exceeded. The settings for Mode I and Mode II operation are given in Tables 4-17 and 4-18, respectively. For reactor power, the LSSS is 125% of full power for both modes of operation. Thus, the highest powers obtainable before a reactor scram occurs would be 12.5 MW [1.25 x 10 MW] for Mode I and 6.25 MW [1.25 x 5 MW] for Mode II. For both modes, the LSSS on primary pressure is a minimum of 75 psia (517 kPa) in the pressurizer, and the LSSS on reactor inlet water temperature is a maximum of 155 °F (68.3 °C). The LSSS on primary coolant flow for Mode I operation is a minimum of 1,625 gpm (6,151 lpm) in either of the parallel primary coolant loops [total of 3,250 gpm (12,302 lpm)]. The same LSSS of 1,625 gpm (6,151 lpm) applies for the single operating loop in Mode II operation. Since 50 gpm (189 lpm) of the primary coolant flow is diverted to the reactor coolant cleanup system, the actual reactor core flow rates at the LSSSs are 3,200 gpm (12,113 lpm) and 1,575 gpm (5,962 lpm) in Modes I and II, respectively. The core differential pressure scrams that have set point trips corresponding to flow values of 3,200 gpm (12,113 lpm) and 1,600 gpm (6,057 lpm) provide a backup to the primary coolant low flow scrams. The setting for Mode III operation is given in Table 4-19. The LSSS for reactor power in this mode is 125% of full power. Thus, the highest power obtainable before a reactor scram occurs would be 62.5 kW [1.25 x 50 kW].

TABLE 4-17
MODE I LIMITING SAFETY SYSTEM SETTINGS

Limited Parameter	Limiting Safety System Setting
Reactor Power	125% Full Power [10 MW] (Maximum)
Primary Coolant Flow	1,625 gpm (6,151 lpm) each loop (Minimum)
Reactor Inlet Water Temperature	155 °F (68.3 °C) (Maximum)
Primary Coolant Pressurizer Pressure	75 psia (517 kPa) (Minimum)

TABLE 4-18
MODE II LIMITING SAFETY SYSTEM SETTINGS

Limited Parameter	Limiting Safety System Setting
Reactor Power	125% Full Power [5 MW] (Maximum)
Primary Coolant Flow	1,625 gpm (6,151 lpm) [Single Loop Operation] (Minimum)
Reactor Inlet Water Temperature	155 °F (68.3 °C) (Maximum)
Primary Coolant Pressurizer Pressure	75 psia (517 kPa) (Minimum)

TABLE 4-19
MODE III LIMITING SAFETY SYSTEM SETTINGS

Limited Parameter	Limiting Safety System Setting
Reactor Power	125% Full Power [50 kW] (Maximum)

4.6.4.2 Bases

The safety limit analysis as described in Section 4.6.3 presents three (3) parametric curves which together define a four-dimensional safety limit envelope prescribing limiting combinations of values for reactor power, primary coolant flow, reactor inlet water temperature, and pressurizer pressure. Operation within this safety envelope will prevent fuel plate meltdown or cladding damage as a result of DNB. The LSSSs were chosen such that the true value of any of the four safety-related variables will not exceed a safety limit under the most severe anticipated transient. Figure 4.13 depicts the DNB conditions for the LSSS for a pressurizer pressure of 75 psia (517 kPa). From this curve, the safety margin for three (3) anticipated transients may be predicted.

- Case One
  This case involves a severe power transient with primary coolant flow and pressure already reduced to their LSSS values in Mode I operation. Figure 4.13 predicts that the temperature LSSS of 155 °F (68.3 °C) could not be reached until reactor power has increased to about 14.75 MW, or 2.25 MW above the reactor high power scram set point. Thus, an ample safety margin exists for the reaction time required to prevent reaching the DNB threshold as follows:

$$P(t) = P_0 e^{-\frac{t}{\tau}},$$

where:

P(t)	=	final power;
P_0	=	initial power;
t	=	time; and
τ	=	reactor period.

- Case Two
  This case involves steady-state Mode I operation with primary coolant flow and pressure again reduced to their LSSSs and reactor power at the LSSS of 12.5 MW. Figure 4-13 predicts that DNB would not occur until a reactor inlet water temperature of approximately 185 °F (85 °C) was reached. The safety margin is thus 30 °F (16.7 °C) above the LSSS of 155 °F (68.3 °C). Primary coolant temperature increase would be slow, thus little or no margin is required for safety system reaction time. Periodic compliance checks and past operating history provide confidence that the primary coolant temperature measurement error is no greater than +5 °F (+2.8 °C). Therefore, an excess safety margin for a temperature transient of this type exists.

- Case Three
  This case involves Mode I operation with pressurizer pressure reduced to the LSSS of 75 psia (517 kPa) with reactor power and reactor inlet water temperature raised to their LSSSs of 12.5 MW and 155 °F (68.3 °C), respectively. Figure 4.13 predicts that the coolant flow rate through the core could be reduced to approximately 2,400 gpm (9,085 lpm) before DNB would occur, implying a safety margin of 800 gpm (3,028 lpm) below the actual LSSS core flow of 3,200 gpm (12,113 lpm). Operating history has shown that the true value of primary coolant flow does not deviate from the measured value by more than +50 gpm (+189 lpm). Thus, there is sufficient primary coolant flow to prevent reaching the DNB threshold.

Consideration of the same transients for Mode II operation yields even greater safety margins. Case One predicts a DNB at 9.25 MW (i.e., 3 MW above the LSSS of 6.25 MW). Case Two results indicate that the reactor could be operated with a reactor inlet water temperature in excess of 200 °F (93.3 °C) without reaching DNB. Case Three shows the DNB occurring only with primary coolant flow through the core reduced to 1,000 gpm (3,785 lpm) or 575 gpm (2,177 lpm) below the actual LSSS core flow of 1,575 gpm (5,962 lpm). Thus, the safety margins for Modes I and II are similar and sufficient to ensure safe reactor operation.

For Mode III operation, the reactor high power scram of 125% of full power will occur at 62.5 kW, thus, there is a margin of 87.5 kW between the LSSS and the safety limit of 150 kW.

The LSSS for pressurizer pressure is 75 psia (517 kPa) and past operating experience has shown the pressurizer pressure sensors to be accurate to within ±2 psi (13.8 kPa). Additionally, there are four independent pressure sensors capable of causing a reactor scram in the event of a loss of pressure accident. Thus, a sufficient margin exists to ensure that the low pressure safety limit will not be violated.

4.6.4.3 Conclusions

The LSSSs on the four important parameters of reactor power, primary coolant flow, reactor inlet water temperature, and primary coolant pressurizer pressure provide sufficient margins for the automatic devices to scram the reactor and prevent a violation of the safety limit envelope.

4.6.5 Decay Heat Removal Analysis

The MURR design consists of a reactor core situated between an inner and outer aluminum pressure vessel located in an open pool with a water temperature of less than 120 °F (49 °C). The pool water provides a heat sink for the removal of core decay heat that is being conducted through the pressure vessel walls, thus the need for an additional heat removal system to protect the integrity of the fuel is not required. However, for redundancy, and to provide a greater safety margin, a reactor decay heat removal system is installed to remove decay heat following an emergency shutdown accompanied by reactor loop isolation or in the event of a loss of normal primary coolant flow.

The reactor decay heat removal system functions as a convective coolant loop consisting of two redundant parallel automatic isolation valves, an in-pool heat exchanger loop, and associated piping. The in-pool heat exchanger consists of 10 vertical finned tubes approximately 5 feet (1.5 m) long with an overall finned length of about 4 ½ feet (1.4 m). Each finned tube has an internal diameter of approximately 1.71 inches (4.34 cm) with 14 internal and 28 external fins.

A loss of primary coolant flow accident was analyzed using the thermal-hydraulic transient code RELAP5 with the following initiating assumptions: 30 days continuous operation at 11 MW, a reactor inlet temperature of 155 °F (68 °C), and a bulk pool water temperature of 120 °F (49 °C). The analysis indicated a maximum fuel plate centerline temperature of 280.3 °F (137.9 °C) and a maximum coolant channel temperature of 237.5 °F (114.2 °C), which is well below the saturation temperature of 277 °F (136 °C).

The results of studies under these conditions, and with loss of primary coolant pressure, show that the decay heat removal system will transfer the decay heat from the core to the pool with no damage to the reactor core. Nucleate boiling will occur in the high heat flux region of two fuel plates during the first second of the transient, while the rest of the heat transfer is in single-phase liquid convection mode. The reactor core will be completely protected by the reversed flow through the in-pool heat exchanger. A failure of in-pool heat exchanger isolation valves V546A and V546B to open during the loss of flow transient was also analyzed and the results predict that no fuel cladding failure will occur.

A detailed analysis of the most severe loss of flow accident is described in Section 13.2.4.

CHAPTER 5

REACTOR COOLANT SYSTEMS

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5.0 REACTOR COOLANT SYSTEMS

This chapter provides the design bases, descriptions, and functional analyses of the reactor coolant systems. Included in this chapter are all the significant heat sources in the reactor and how the heat is safely removed and transferred to the environment.

5.1 Introduction

The Missouri University Research Reactor (MURR) utilizes three coolant systems: Primary, Pool, and Secondary. The primary and pool coolant systems are designed for three modes of operation:

(1) Mode I - At power levels of up to 10 MW with the primary coolant system pressurized and at a flow rate of approximately 3,750 gpm (14,195 lpm), and a pool coolant flow rate of approximately 1,100 gpm (4,164 lpm); used when all heat exchange and pumping capacity is available;

(2) Mode II - At power levels of up to 5 MW with the primary coolant system pressurized and at a flow rate of approximately 1,875 gpm (7,098 lpm), and a pool coolant flow rate of approximately 600 gpm (2,271 lpm); utilizing only half the design heat exchange and pumping capacity available; being the maximum operating power level at the time of initial startup; and

(3) Mode III - At power levels of up to 50 kW with the primary coolant system open to the reactor pool, the reactor pressure vessel head removed, the flanged port open, and the pool water level at the elevation of either the upper or lower reactor bridge; used for core flux calibrations following the loading of a new core, or after a fuel rearrangement.

The reactor coolant systems and the instrumentation that accommodates these three operating modes are shown in Figure 5.1.

Heat from the primary and pool coolant systems is transferred to the secondary coolant system by means of heat exchangers. This heat is then dissipated to the atmosphere through a cooling tower. Auxiliary systems which support the three principal reactor coolant systems are also discussed in this chapter.

5.2 Primary Coolant System

5.2.1 Introduction

The primary coolant system consists of the reactor pressure vessels, two main circulating pumps, two heat exchangers, two automatic isolation valves, a pressurizer, a closed in-pool convective cooling system (decay heat removal system), an in-pool invert loop and anti-siphon system, a fuel element failure monitoring system, a bypass loop for water clean-up, and associated piping and valves. All metal surfaces in contact with the primary or pool coolant are either aluminum or stainless steel, except where noted.

The primary coolant system is operated and monitored from the reactor control room. Instrumentation used to monitor the system's operation, pressures, water temperatures, and flows are described in Chapter 7, Instrumentation and Control Systems. The reactor pressure vessel is described in detail in Chapter 4, Reactor Description. The design bases of the primary coolant system provide reasonable assurance that the environment and health and safety of the public will be protected.

5.2.2 General Operating Conditions

When the reactor is operated in the open reactor pool mode, the maximum power is limited to 50 kW and core cooling is by natural convection. At power levels up to a maximum of 5 MW, the primary coolant system supplies water at 1,875 gpm (7,098 lpm) to the reactor with a minimum inlet pressure of 75 psia (0.52 MPa) and 120 °F (49 °C) temperature. At a thermal power level of 10 MW, the primary coolant system will supply water at 3,750 gpm (14,195 lpm) with a pressurizer pressure of 85 psia (0.59 MPa) and 120 °F (49 °C) inlet temperature to the reactor.

5.2.3 Circulation Pumps

The primary coolant circulation pumps are horizontal, centrifugal, single-stage pumps that are direct-connected to 125-HP drive units through flexible couplings. The pump and driving unit are mounted on a common base. One pump will provide approximately 1,875 gpm (7,098 lpm) to the reactor, and two pumps will supply 3,750 gpm (14,195 lpm) with sufficient discharge head to overcome system pressure drop losses. The primary coolant circulation pumps are designated as P501A and P501B.

5.2.4 Heat Exchangers

The primary coolant heat exchangers are tube type, water-to-water shell, with removable tube bundles. The tubes, and all materials in contact with the primary coolant, are made of stainless steel. The primary coolant flow makes two passes through the tube side of the heat exchanger with a velocity of no greater than 7 ft/sec (2.1 m/sec). At a maximum of 1,600 gpm (6,057 lpm) of secondary water flow and an inlet water temperature at 87 °F (31 °C), one heat exchanger is capable of removing 17 x 10^6 BTU/h of heat from 1,800 gpm (6,814 lpm) of primary coolant water and returning it at 140 °F (60 °C). Two heat exchangers are installed for design power operation. The primary coolant heat exchangers are designated as HX503A and HX503B.

[redacted]

FIGURE 5.1
PIPING AND INSTRUMENT DIAGRAM

5.2.5 Isolation Valves

Two 12-inch, air-operated-to-open, spring-to-close, quick acting automatic isolation valves are located on the inlet and outlet primary coolant lines as close as practicable to the biological shield in the pipe tunnel located beneath the reactor containment building beamport floor (below grade level). If primary pressure decreases to a predetermined value indicating low primary pressure, the primary coolant isolation valves close, isolating the in-pool portions of the primary coolant system from the remainder of the system. The same signal that actuates these isolation valves also initiates a reactor scram, de-energizes the primary coolant circulation pumps, and initiates the opening of the anti-siphon system isolation valves. The primary coolant isolation valves are designated as V507A and V507B.

5.2.6 Pressurizer System

The pressurizer system ensures that primary coolant system pressure is maintained within the LSSSs for both 5- and 10-MW operation. The pressurizer system also provides a path for the addition of primary grade water which is lost during normal operational evolutions such as primary coolant sampling. Water level and pressure are maintained automatically in the pressurizer in response to system instrumentation signals. By nitrogen gas admitted to, or released from, the pressurizer, pressure is maintained at 85 psia (0.59 MPa) for both 1,875- and 3,750-gpm (7,098- and 14,195-lpm) operation. Water level is maintained by a positive displacement pump on system demand for additional water. Excess water is discharged to the drain collection tank on high level conditions.

The reactor pressurizer and water make-up systems are discussed in greater detail in Section 5.6, Primary Coolant Make-Up Water System.

5.2.7 Reactor Convective Cooling Loop

The reactor convective cooling loop serves to remove core decay heat following an emergency shutdown accompanied by a primary loop isolation, or in the event of a loss of primary coolant flow.

The reactor convective cooling loop is discussed in greater detail in Section 5.8, Decay Heat Removal System.

5.2.8 Anti-Siphon System

The function of the anti-siphon system is to prevent the loss of water from the reactor core in the event of a rupture in the largest diameter primary coolant system piping external to the reactor pool.

The anti-siphon system is discussed in greater detail in Chapter 6, Engineered Safety Features.

5.2.9 Fuel Element Failure Monitoring System

The fuel element failure monitoring system continuously monitors the primary coolant system for fission product activity buildup that may indicate a fuel element failure. The monitoring system consists of a water sampling unit and a scintillation detector. The detector output is fed to the Area Radiation Monitoring System (ARMS) where the signal is processed and displayed on a meter located in the reactor control room. The meter is equipped with an adjustable set point trip that initiates an annunciator alarm on detection of a high radiation level.

The fuel element failure monitoring system is discussed in greater detail in Chapter 7, Instrumentation and Control Systems.

5.2.10 Primary Coolant Demineralizer Loop

The main purpose of the primary coolant demineralizer loop is to reduce the inventory of radionuclides in the primary coolant through demineralization. The system consists of a pump, filters, a hold-up tank, a demineralizer bed, and associated piping and valves.

This system is discussed in greater detail in Section 5.5, Reactor Coolant Cleanup Systems.

5.2.11 Surveillance

Sampling frequency of the primary coolant system and normal operational checks on the system will be performed as required by the Technical Specifications. The objective of this specification is to reasonably ensure proper operation of the system components.

5.3 Pool Coolant System

5.3.1 Introduction

The pool coolant system consists of two main circulating pumps, a heat exchanger, an automatic isolation valve, a reflector plenum natural convection valve, a hold-up tank, a return diffuser, a bypass loop for water clean-up, and associated valves and piping. The system is designed to transfer 3.6 x 10^6 BTU/h at a flow rate of 1,200 gpm (4,542 lpm). This allows for reactor operation at a maximum thermal power of 10 MW.

The pool coolant system is operated and monitored from the reactor control room. Instrumentation used to monitor the system's operation, pressures, water temperatures, and flows are described in Chapter 7, Instrumentation and Control Systems.

5.3.2 Circulation Pumps

The pool coolant is circulated by horizontal, centrifugal, single-stage pumps that are direct-connected to 60-HP drive units through flexible couplings. The pumps are designed to overcome system pressure drops at variable flow rates to a maximum flow of 1,075 gpm (4,069 lpm) with one pump operating and 1,325 gpm (5,016 lpm) when both pumps are operating. The pool coolant circulation pumps are designated as P508A and P508B.

5.3.3 Heat Exchanger

The pool coolant heat exchanger is a water-to-water plate type with all surfaces in contact with pool water constructed of stainless steel. This heat exchanger is designed to remove 3.6×10^6 BTU/h of heat from 1,200 gpm (4,542 lpm) of pool coolant and return it at 99 °F (37 °C) with a secondary flow rate and inlet temperature of 500 gpm (1,893 lpm) and 87 °F (31 °C), respectively. One heat exchanger is installed for operating at a maximum thermal power level of 10 MW. The pool coolant system heat exchanger is designated as HX521.

5.3.4 Isolation Valve

The pool coolant system outlet line contains a 6-inch, air-operated-to-open, spring-to-close, quick acting automatic isolation valve which, in conjunction with a check valve on the inlet line, prevents a loss of pool water in the event of a break in a system line. These valves are located in the pipe tunnel as close as practicable to the biological shield. The pool loop automatic isolation valve and check valve are designated as V509 and V519A, respectively.

5.3.5 Reflector Plenum Natural Convection Valve

The reflector plenum natural convection valve, under the original design, opened to allow natural circulation of pool coolant into the lower plenum (lower reflector tank cylinder) and up through the reflector elements, the control blade gaps, and the center test hole (flux trap) upon loss of forced pool loop flow. To insure compliance with the Institute of Electrical and Electronic Engineers Standard 279 (IEEE-279), Single Failure Criterion, this valve is presently left open. Operational experience has shown that the reflector plenum natural convection valve can be left in the open position while increasing normal pool flow to provide sufficient cooling to these regions. The reflector plenum natural convection valve is designated as V547.

5.3.6 Pool Water Hold-Up Tank

The pool water hold-up tank provides a sufficient time delay for oxygen and nitrogen activity in the pool coolant to decay before being returned to the reactor pool.

The pool water hold-up tank is described in greater detail in Section 5.7, Nitrogen-16 Control System.

5.3.7 Pool Water Return Diffuser

Pool water that has been cooled is returned to the reactor pool at a maximum velocity of 2 ft/sec (0.6 m/sec) at 1,200 gpm (4,542 lpm) through a diffuser spool which provides good mixing and prevents flow directly to the pool surface. The diffuser spool is a circular pipe with 252 ⅝-inch (1.59-cm) diameter holes (36 rows of seven holes per row), pointing in a downward direction which allows the return water to discharge to the reactor pool at a minimal velocity, yet achieve mixing by dispersing the pool water over a large area.

5.3.8 Pool Coolant Demineralizer Loop

The primary function of the pool coolant demineralizer loop is to reduce the inventory of radionuclides in the pool coolant through demineralization. The system consists of a pump, filters, a demineralizer bed, and associated piping and valves.

This system is discussed in greater detail in Section 5.5, Reactor Coolant Cleanup Systems.

5.3.9 Surveillance

Pool coolant is sampled on a weekly basis to determine the concentrations of gamma-emitting isotopes.

5.4 Secondary Coolant System

5.4.1 Introduction

The heat from the primary and pool coolant systems is transferred to the secondary coolant system by means of heat exchangers. The heat is then dissipated to the atmosphere through a cooling tower. The secondary coolant system also provides a heat sink for the laboratory building air-conditioning system. The secondary system consists of four circulation pumps, a pool coolant heat exchanger, two primary coolant heat exchangers, two automatic temperature control valves, a water treatment system, temperature and flow instrumentation, a cooling tower, radiation monitoring instrumentation, and associated piping and valves. The secondary coolant system is designed such that a failure or malfunction of any system component will not lead to reactor damage, fuel failure, or the uncontrolled release of radioactivity to the environment.

5.4.2 Circulation Pumps

The secondary coolant circulation pumps are horizontal, centrifugal, single-stage pumps that are direct-connected to 150-HP variable speed drive units through flexible couplings. Three pumps are installed in a parallel configuration with each pump capable of supplying 2,200 gpm (8,328 lpm) to the secondary coolant system. One pump is typically required for 5-MW operation and two pumps are normally required for 10-MW operation. A fourth pump, with a capacity of 700 gpm

(2,650 lpm), is dedicated to the laboratory building air-conditioning system. The secondary coolant circulating pumps are designated SP-1, SP-2, SP-3, and SP-4.

5.4.3 Heat Exchangers

The secondary water flow makes a single pass on the shell side of the primary coolant system heat exchangers and on the opposite plate side of the pool coolant in the pool coolant system heat exchanger.

The primary and pool coolant system heat exchangers are further described in Sections 5.2.4 and 5.3.3, Heat Exchangers.

5.4.4 Automatic Temperature Control Valves

The amount of water passing through the primary and pool heat exchangers is controlled by automatic temperature control valves located in bypass lines adjacent to each heat exchanger system. The valves are electro-hydraulic butterfly valves which respond to the respective primary or pool coolant outlet temperature from the heat exchangers, and maintain a constant cold leg temperature in both the primary and pool coolant systems. The automatic temperature control valves for the primary and pool heat exchangers are designated as S-1 and S-2, respectively.

5.4.5 Water Treatment System

The pH and conductivity of the secondary coolant system are maintained automatically by a water quality control system located in the cooling tower building. A solenoid-controlled acid addition valve gravity-feeds concentrated sulfuric acid into the sump of the secondary system to maintain pH within specification. Conductivity is maintained by the removal of water from the secondary system through a blowdown valve and the subsequent addition of make-up water. Other chemicals are added manually and automatically to help control hardness and microbiological growth.

5.4.6 Instrumentation

The secondary water inlet and outlet temperatures to the primary and pool coolant heat exchangers, and the total flow in the secondary system, are displayed and recorded in the reactor control room.

5.4.7 Cooling Tower

The cooling tower is a wood-framed, induced-draft, cross-flow type, with three cells and two-speed fan assemblies for each cell. The tower is designed to cool 4,688 gallons (17,746 l) of water per minute to a temperature of 87 °F (31 °C) from an initial temperature of 104 °F (40 °C) at a maximum wet bulb temperature of 77 °F (25 °C). Vibration cutout switches are installed on each fan assembly to secure the associated fan motor to prevent damage to the fan or cooling tower structure should an imbalance develop. The number of fans and fan speed is configured as required to provide sufficient cooling for 10-MW operation.

5.4.8 Secondary Coolant Monitoring System

The secondary coolant monitor continuously monitors the secondary water for the presence of radioactive isotopes which could indicate a leak from the primary or pool coolant systems through their respective heat exchangers. The monitoring system consists of a scintillation detector that measures the gross activity of the secondary water. The output from the detector is fed to the ARMS where the signal is processed and displayed on an analog meter located in the reactor control room. The meter is equipped with an adjustable set point trip that initiates an annunciator alarm on detection of a high radiation level. The secondary coolant monitor is located in the return leg of the secondary piping, downstream of the pool and primary heat exchangers. This location ensures a faster response in the event of a leak from either system heat exchanger.

The secondary coolant monitoring system is discussed in greater detail in Chapter 7, Instrumentation and Control Systems.

5.4.9 Surveillance

In addition to being continuously monitored by the secondary coolant monitoring system, the secondary system is sampled on a monthly basis for gamma-emitting isotopes and tritium level, and daily for water quality.

5.5 Reactor Coolant Cleanup System

5.5.1 Introduction

There are two demineralizer loops associated with the reactor: one serving the primary coolant system and one serving the pool coolant system. Each loop is independent of the other with the exception of the demineralizer tanks, which are interchangeable from one loop to another by means of a piping and valving arrangement. This arrangement allows a depleted demineralizer bed to be removed from service and a new bed placed on-line without an interruption in reactor operation. The two main purposes of the cleanup system are (1) to reduce the inventory of radioactive nuclides present in the coolant and (2) to help maintain a primary-grade level of water quality which limits chemical corrosion to essential components.

5.5.2 Demineralizers

The water purification system consists of three rubber-lined carbon steel demineralizer tanks sized to hold 12 cubic feet (0.34 m^3) of mixed bed resin. Each resin bed is capable of removing 1,500 grains of hardness per day from water at a design flowrate of 50 gpm (189 lpm) and a temperature not exceeding 140 °F (60 °C). The effluent from a resin bed contains no greater than 0.05 ppm sodium and 0.05 ppm silicon. The three demineralizer tanks are located in a shielded area and are connected to a common manifold distribution system of piping. Any two of the three units are in service simultaneously; one connected to the pool loop and the other connected to the primary loop. The third unit is in standby.

5.5.3 Filters

Primary and pool coolant is filtered both before and after passing through a demineralizer bed. The filters are replaceable cartridge type with a stainless steel shell, base, and cover. The filter cartridges in use are typically sized to remove 1 micron or larger particles from a continuous 50 gpm (189 lpm) influent. Installed isolation valves allow filter cartridge replacement when necessary.

5.5.4 Primary Coolant Demineralizer Loop

The primary coolant demineralizer loop supply line is connected to the primary coolant system immediately downstream of the primary heat exchangers. A portion of the primary coolant is diverted to the demineralizer loop where it flows through a 100-gallon (379-l) hold-up tank. This tank provides a 2-minute delay to allow oxygen and nitrogen activity to decay. Coolant then flows to the suction of a single-stage centrifugal pump, which overcomes system headlosses, and circulates the coolant through the filters and demineralizer tank and back to the primary coolant system.

5.5.5 Pool Coolant Demineralizer Loop

Operation of the pool coolant demineralizer loop is similar to that of the primary coolant demineralizer loop with the exception of a hold-up tank. The pool system demineralizer loop supply line is connected to the pool coolant system upstream of the pool heat exchanger, but after the pool water hold-up tank. Pool coolant entering the demineralizer loop has thus already undergone a five-minute delay, allowing oxygen and nitrogen activity to decay.

Demineralized pool coolant, which is approximately 5 °F (2.8 °C) warmer than pool coolant leaving the heat exchanger, is returned to the reactor pool approximately 2 feet (0.6 m) below the surface. This creates a blanket of warmer water at the pool surface, helping to reduce the mixing of bulk pool water near the surface. Therefore, pool coolant with a lower activity remains near the surface and hence reduces reactor pool surface dose rates.

5.5.6 General Description

The entire water purification system for both the primary and pool coolant systems is contained behind concrete shield walls in areas referred to as cells. These cells are arranged so as to shield the demineralizer tanks from a limited access passageway by the use of stub wall shadow shields. System operations are performed remotely by the use of reach rods that connect the valve handles to the valve bodies through a two-foot (0.6-m) thick shield wall. In addition to this two-foot

(0.6-m) thick wall, there is an additional foot (0.3 m) of shielding separating the valve bodies from the demineralizer tanks, assuring a minimum of three feet (0.9 m) of shielding separating an operator from the demineralizers. Any malfunctions or leaks in the reactor coolant cleanup system will not lead to radiation exposure to personnel or releases to the environment that exceed the regulatory requirements or facility ALARA Program guidelines.

The reactor coolant cleanup system is also equipped with a resin storage tank, enabling the use of an additional (fourth) bed of resin in the system. This significantly increases the decay time from resin depletion to resin replacement, thus reducing the activity levels sent to the radioactive liquid waste system during resin replacement.

The regeneration station (R-200) and demineralizer filters are also located in separate cells with valving for remote operation. The regeneration station is designed as a central transfer point, allowing resin to be transferred to or from any of the demineralizer tanks, including the resin storage tank. A single regenerator tank is used for both dumping depleted resin and loading new, pre-regenerated resin.

Access to the demineralizer cells is restricted by a number of wire gates which are locked and remotely alarmed to the control room. The physical arrangement of this system is shown in Figure 1.1, Below Grade Level Plan (Basement).

5.5.7 Description of Operation

Primary or pool coolant flow may be remotely diverted to any of the three mixed bed demineralizer tanks by the use of reach rods. When a resin bed is depleted, the standby bed is placed on service and the depleted resin bed is removed from service. A previously depleted resin bed that has undergone decay is then transferred by the use of a water carrier from the resin storage tank to the regenerator. The depleted bed is then dumped into resin drying barrels. The depleted bed removed from service is then transferred to the resin storage tank for decay. A new, pre-regenerated resin bed is loaded into the regenerator and transferred to the empty demineralizer tank and placed in standby. All transfer water, including effluent (wastes) from R-200, is directed to either the radioactive liquid waste retention disposal system or the drain collection tank.

Resin beds are removed from service when conductivity can no longer be maintained at less than 3.0 μmho. Operational history has shown that resin beds for the pool coolant system generally last five months. Primary coolant system resin beds generally last 5 years.

5.5.8 Instrumentation

Demineralizer loop flow for both the primary and pool coolant systems is displayed and recorded in the reactor control room. Conductivity of the coolant entering and exiting a demineralizer bed is measured by conductivity probes, with the output displayed on digital meters in the control room. The meters are equipped with an adjustable set point trip that initiates a "Reactor or Pool Loop Hi Cond" annunciator alarm on detection of high conductivity.

5.5.9 Surveillance

The conductivity of the coolant is continuously monitored. The pH of the primary and pool coolant systems is checked on a weekly basis.

5.6 Primary Coolant Make-Up Water System

5.6.1 Introduction

The primary coolant make-up water system consists of a pressurizer tank, a positive displacement pump, automatic control valves, a nitrogen supply system, a primary grade water supply, and associated piping and valves.

The purpose of the primary coolant make-up water system is to ensure that primary coolant pressure is maintained within the Limiting Safety System Settings for both 5- and 10-MW operation. The system also provides a path for the addition of primary grade water lost during normal operational evolutions such as primary coolant sampling.

5.6.2 Pressurizer Tank

The pressurizer tank is a 300-gallon (1,136-l), ¾-inch (1.91-cm) thick aluminum tank with a design pressure of 200 psig (1.4 MPa above atmosphere). The tank is sized to provide sufficient volume for primary water expansion from a cold plant temperature of 70 °F (21 °C) to a hot plant temperature of 160 °F (71 °C). This heatup would cause 43 gallons (163 l) of primary coolant to expand into the pressurizer. The pressurizer tank is designated as T520.

5.6.3 Primary Coolant Charging Pump

The primary coolant charging pump is a horizontal, positive displacement, triplex pump that is belt-connected to the driving unit. The pump is designed to provide sufficient discharge head to overcome primary coolant system pressure at a capacity of 50 gpm (189 lpm). The primary coolant charging pump is designated as P533.

5.6.4 Automatic Control Valves

All automatic control valves associated with the primary coolant make-up water system are air-operated-to-open, spring-to-close (fail-safe). Actuation of these valves is initiated by control signals from system instrumentation which maintains pressurizer pressure and water level within normal operating limits. The designations for the automatic control valves are as follows:

a.	Nitrogen Addition Valve:	V526;
b.	Nitrogen Vent Valve:	V545;
c.	Water Addition Valve:	V527B;
d.	Water Drain Valve:	V527A;
e.	Primary System Drain Valve:	V527D; and
f.	Surge Line Isolation Valve:	V527C.

5.6.5 Nitrogen Supply System

The nitrogen supply system supplies nitrogen gas to the pressurizer to maintain it at 85 psia (0.59 MPa). The system consists of two (three-cylinder) banks of nitrogen, one in service with the other in standby.

This system is discussed in greater detail in Chapter 9, Auxiliary Systems.

5.6.6 Primary Grade Water Supply System

The primary grade water supply system consists of two 7,000-gallon (26,498-l) demineralized water storage tanks that provide sufficient volume for reactor plant make-up and pool water level lowering operations. Demineralized water at a conductivity of less than 2.0 µmho is automatically added to the primary coolant system by the coolant charging pump when required by system demand.

This system is discussed in greater detail in Chapter 9, Auxiliary Systems.

5.6.7 Description of Operation

Primary coolant system pressure is maintained by a nitrogen bubble on top of the water volume in the pressurizer tank. If pressure decreases below normal, pressure switch 941 initiates a signal to open valve V526 to emit nitrogen into the pressurizer, increasing pressure back to normal operating limits. If pressure increases above normal, pressure switch 940 initiates a signal to open valve V545 to vent the pressurizer to the exhaust system and thus lower system pressure.

Water level is maintained in the pressurizer by the primary coolant charging pump, P533. If water level decreases sufficiently, level controller 936 initiates a signal to open valve V527B and start P533, increasing water level back to normal operating limits. If water level increases above normal, level controller 936 initiates a signal to open valve V527A to drain water from the pressurizer to the drain collection tank.

The instrumentation which provides pressure and liquid level control is described in greater detail in Chapter 7, Instrumentation and Control Systems.

5.7 Nitrogen-16 Control System

5.7.1 Introduction

Radiation levels due to nitrogen-16 (^{16}N) activity are reduced by the use of hold-up tanks in both the primary coolant demineralizer loop and the pool coolant system. ^{16}N, a high-energy beta and gamma emitter with a half-life of seven seconds, is produced when oxygen in the primary and pool coolant is irradiated with neutrons of sufficient energy. These internally-baffled tanks hold up, or delay, the primary and pool coolant within a restricted, shielded space for a sufficient amount of time to allow short-lived activity, primarily ^{16}N, to decay, thereby reducing radiation levels and exposure to personnel. The design of the nitrogen-16 control system provides reasonable assurance that the system will not interfere with reactor cooling, cause an uncontrolled loss or release of primary coolant, or prevent safe reactor shutdown.

5.7.2 Primary Coolant Demineralizer Loop Hold-Up Tank

The primary coolant demineralizer loop hold-up tank has a volume of 100 gallons (379 l) and is constructed of 5/16-inch (7.94-mm) thick aluminum with a design pressure of 150 psig (1.03 MPa above atmosphere). The tank contains five internal 3/16-inch (4.76-mm) thick aluminum baffles in an alternating pattern from bottom to top. This internal baffling promotes slug flow of coolant that provides a two-minute delay at an inlet flowrate of 50 gpm (189 lpm). The primary coolant demineralizer loop hold-up tank is designated as T506.

5.7.3 Pool Coolant Hold-Up Tank

The pool coolant hold-up tank has a volume of 6,000 gallons (22,712 l) and is constructed of ¼-inch (6.35-mm) thick aluminum with a design pressure of 50 psig (0.34 MPa above atmosphere). The tank contains seven internal ¼-inch (6.35-mm) thick aluminum baffles in an alternating pattern from bottom to top. This internal baffling promotes slug flow of coolant that provides a five-minute delay at an inlet flowrate of 1,200 gpm (4,542 lpm). The pool coolant hold-up tank is designated as T504.

5.7.4 Description of Operation

The pool coolant hold-up tank and the primary coolant demineralizer loop hold-up tank are located in the mechanical equipment room (Room 114) external to the reactor containment building. Stackable concrete blocks create a shield wall between this space and a limited use passageway. This wall is a minimum of three feet (0.9 m) thick at any given location. During reactor operation, Room 114 is controlled as a high radiation area with access restricted by a wire gate which is locked and remotely alarmed in the reactor control room.

5.8 Decay Heat Removal System

5.8.1 Introduction

The reactor has an emergency decay heat removal system located within the reactor pool consisting of two parallel automatic isolation valves, an in-pool heat exchanger loop, and associated piping.

The function of this system is to ensure adequate decay heat removal from the reactor core following an emergency shutdown accompanied by a primary loop isolation, or, in the event of a loss of coolant flow. Based on 30 days of continuous operation at 10 MW, an inlet coolant temperature of 140 °F (60 °C) and a pool temperature of 100 °F (38 °C), the decay removal system will transfer decay heat from the core to the reactor pool with no net formation of steam in the loop.

The decay heat removal system complies with IEEE-279, Single Failure Criterion.

5.8.2 In-Pool Heat Exchanger

The in-pool heat exchanger consists of 10 (finned) aluminum tubes approximately 5 feet (1.5 m) long with an overall finned length of about 4.5 feet (1.4 m). Each tube has an internal diameter of 1.71 inches (4.34 cm) with 14 internal and 28 external fins. The tubes are vertically aligned in parallel and encompass an arc of 90 degrees. The in-pool heat exchanger is designated as HX505.

5.8.3 Automatic Isolation Valves

Two 6-inch, air-operated-to-close, spring-to-open, quick acting parallel isolation valves are located on the inlet piping of the in-pool heat exchanger. If primary coolant flow or pressure decreases to a predetermined value, the isolation valves open, allowing a flow path for primary coolant through the in-pool heat exchanger. The same signal that operates these isolation valves also initiates a reactor shutdown. Operation of either valve will perform the intended function of the decay heat removal system. The valves are designed to be manually operated from the reactor pool upper bridge if required. The in-pool heat exchanger automatic isolation valves are designated as V546A and V546B.

5.8.4 Description of Operation

In the event of a loss of normal primary coolant flow, a primary loop isolation or a loss of electrical power, the automatic isolation valves open, allowing a flow path for primary coolant through the in-pool heat exchanger. Primary coolant flow, which is normally downward through the core, will stop, then a major portion will reverse, flowing upward through the core and through the in-pool heat exchanger. The driving force will be created by the different water densities in the hot and cold legs of the loop. Heat from the reactor core will be transferred to the in-pool heat exchanger and dissipated to the pool. As discussed in Section 13.2.9.3.1, the failure of both in-pool heat exchanger automatic isolation valves to open when required is not considered credible. Therefore, there are no credible or postulated malfunctions of the Decay Heat Removal System that will lead to an uncontrolled loss of primary coolant, radiation exposures, or releases of radioactivity to the unrestricted environment that exceed the requirements of 10 CFR 20 and the facility ALARA Program guidelines.

5.9 Limiting Conditions for Operation

The limiting conditions for operation of the reactor coolant systems and their associated support systems and their bases are discussed in the Technical Specifications. The objective of these conditions is to protect the fuel integrity and prevent the release of fission product radioisotopes.

5.10 Design Features

The specifications and bases for the design features of the reactor coolant systems and their associated support systems are discussed in the Technical Specifications. The objective of these specifications is to ensure proper cooling of the reactor for safe operation.

CHAPTER 6

ENGINEERED SAFETY FEATURES

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6.1 Preliminary Hazards Report, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, March 1961.

6.2 Hazards Summary Report, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, July 1965.

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6.7 Hazards Summary Report, Addendum 5, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, January 1974.

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6.0 ENGINEERED SAFETY FEATURES

This chapter discusses and describes the Engineered Safety Features (ESFs) for the reactor facility. The ESFs are active or passive features designed to mitigate the consequences of accidents and to keep radiological exposures to the public, the facility staff, and the environment within federal limits. The concept for ESFs evolved from the defense-in-depth philosophy of multiple layers of design features to prevent or mitigate the release of radioactive materials to the environment during accident conditions. The need for ESFs is determined by analyzing accidents that could occur, even though prudent and conservative design of the reactor facility has made the incidence of an accident very unlikely.

6.1 Introduction

During the design and the subsequent safety evaluation of the Missouri University Research Reactor (MURR),[1] two systems were identified as required ESFs: the containment system and the anti-siphon system. These systems are designed to mitigate the consequences of certain identifiable accidents and to keep radiological exposures to the operating staff and the general public within the limits of Title 10, Chapter I, of the Code of Federal Regulations, Part 20 (10 CFR 20). The two accident scenarios which require an engineered safety feature are the Maximum Hypothetical Accident (MHA) and the Loss of Coolant Accident (LOCA).

The MHA is an accident condition which postulates the melting of four fuel plates in the reactor core and the subsequent release of fission products to the primary coolant system.[1] This accident is selected to postulate conditions which lead to consequences worse than those resulting from any other anticipated accident. The circumstances that lead to the MHA are immaterial to the analysis. It is assumed that the first barrier of protection against the release of fission products to the atmosphere, the fuel plate cladding, has failed. The second barrier of protection is the primary coolant system. Should this barrier also fail, the containment system will perform a complete isolation of the reactor containment building, thus providing a third barrier against an uncontrolled release of radioactive materials to the environment.

The anti-siphon system functions as a backup system to the various safety instrumentation and equipment (e. g., pressure sensors, pump and valve interlocks, etc.) all of which ensures that the reactor core does not become uncovered during a LOCA. A rupture of the primary coolant system, followed by a loss of pressure, causes the anti-siphon system to admit a fixed volume of air to the high point of the reactor outlet piping, thus breaking any potential siphon which may have been created by the pipe rupture.

The MHA and LOCA are discussed in detail in Chapter 13, Accident Analyses.

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[1]The original safety evaluation of the MURR is documented in the Preliminary Hazards Report (Ref. 6.1), the Hazards Summary Report (Ref. 6.2), and Hazards Summary Report, Addenda 1-5 (Ref. 6.3-6.7).

6.2 Containment System

6.2.1 Introduction

The containment system is designed to completely isolate the reactor containment building, thereby preventing or mitigating any uncontrolled release of radioactive materials to the environment during an accident.[1] Redundancy is incorporated into the system to ensure that no single component or circuit failure will render any portion of the containment system inoperative. Isolation of the reactor containment building can be automatically initiated by radiation detectors located at the reactor pool upper bridge and in the containment building exhaust plenum. Isolation can be manually activated by switches in the reactor control room or the facility lobby (Room 202). Actuation of the Room 202 switch will also cause a facility evacuation.

The instrumentation necessary for the containment system to perform its intended function is described in greater detail in Section 7.8, Engineered Safety Features Actuation Systems.

6.2.2 Reactor Containment Building

6.2.2.1 Description

The reactor containment building is a five-level poured concrete building with 12-inch (0.3-m) thick reinforced exterior walls configured to form the shape of a cube, with each side being approximately 60 feet (18.3 m) long. All horizontal and vertical seams created in the pouring are sealed with a polyvinylchloride (PVC) plastic waterstop of the dumbell type. The waterstops were manufactured to meet the Corp. of Engineers Specification, CRD-C572.61. The outside surfaces of the concrete walls are finished with removable aluminum sheet siding 16 feet (4.9 m) long by 4 feet (1.2 m) wide. The siding is attached to angle standoffs on the walls. Centered on the exterior of each wall are two pilaster support columns closed at the back to form a tower. Located within each tower is support equipment necessary for the operation of the reactor facility. The pilaster support columns, comprising the side walls of the towers, are built into the main building structure to achieve the requisite structural strength. Below grade, within the containment structure, is a space (Room 101B) extending to the north that is 15 feet (4.6 m) high by 37 feet (11.3 m) deep by 40 feet (12.2 m) wide that contains various experimental equipment (Fig. 6.1). The walls and ceiling of this space are also of poured concrete with the pour joints sealed with the same type of PVC plastic waterstop.

6.2.2.2 Design Basis

The reinforced concrete walls of the reactor containment building have been designed to withstand a peak internal pressure of 2.0 psig (13.8 kPa above atmosphere). This design pressure was not predicated on any one accident scenario, rather the pressure was selected to envelope all

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[1]See Footnote on Page 6-1.

[redacted]

FIGURE 6.1
BASEMENT LEVEL FLOOR PLAN

postulated accident scenarios. It is difficult to envision a pressure buildup within the containment structure to 2.0 psig (13.8 kPa above atmosphere) because the maximum design temperature of the bulk water in the reactor is only 160 °F (71 °C) (Ref. 6.8). In the event of a rupture in the primary coolant system piping, the coolant in the reactor core region would remain subcooled relative to atmospheric pressure and the water would not flash to steam as in the case of a highly pressurized water reactor. In addition, the one foot (0.3 m) diameter column of water immediately above the one foot (0.3 m) diameter reactor pressure vessel is calculated to absorb 108 MW-sec of energy before reaching 212 °F (100 °C).[1] Therefore, any large, positive reactivity insertion leading to an uncontrolled power excursion which results in the formation of steam in the reactor pressure vessel and a rupture in the primary coolant system within the reactor pool volume would be quickly quenched by the pool water.

The energy release necessary to cause a 2.0-psig (13.8-kPa above atmosphere) equilibrium loading within the reactor containment building has been calculated.[1] The calculation includes the following assumptions:

1. No heat sinks;
2. Initial containment building air temperature: 75 °F (24 °C) with 50% relative humidity;
3. Initial pool water temperature: 100 °F (38 °C);
4. Containment building volume: 240,000 ft^3 (6,796 m^3); and
5. Final condition of 100% saturated air in the containment building.

The amount of steam that must be released from the pool to cause a 2.0-psig (13.8-kPa above atmosphere) increase within the containment structure corresponds to 1,040 MW-sec of energy release. Under the above conditions, the final equilibrium temperature within the containment building would be 113 °F (45 °C). This amount of energy release is considerably greater than would be expected as a result of any accident to this type of reactor.[1]

However, the possibility that a severe fuel meltdown causing a molten aluminum water (Al-H_2O) reaction which releases a considerable amount of energy does exist. The specific conditions under which such a reaction might occur are widely discussed in literature pertaining to reactor hazards.[1] It is currently believed that the occurrence of an Al-H_2O reaction requires first, that the aluminum be present in a finely divided form such as droplets or particles of less than 500 microns, and second, that the temperature of the aluminum be greater than its melting point.

The amount of aluminum in the fuel plates of eight fuel assemblies (complete core loading) is 29.4 kg, more than 75% of which is cladding. If this quantity of aluminum were to react completely with water, it would produce 441 MW-sec of energy and release approximately 1,340 ft^3 (38 m^3) of hydrogen at standard conditions. If the hydrogen were to completely recombine, an additional 523 MW-sec of energy would be released. This total release of 964 MW-sec is less than the 1,040 MW-sec of corresponding energy in steam which would have to be released to cause

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[1]See Footnote on Page 6-1.

a 2.0-psig (13.8-kPa above atmosphere) equilibrium overpressure in the reactor containment building.

To postulate that the total reaction of the aluminum and the recombination of the evolved hydrogen would occur is an excessively conservative and unrealistic assumption. A more realistic, but still exceedingly conservative, approach would be to assume that only 1.3% of the aluminum would react. This assumption would be consistent with the MHA which postulates the melting of four number-1 fuel plates in the reactor core. The four number-1 fuel plates contain 78.58 grams of uranium-235 (^{235}U). Considering the total reactor core inventory of 6.2 kg of ^{235}U, during the MHA, approximately 1.3% of the core would melt. Therefore, the amount of aluminum which would react with the coolant would be 0.3822 kg (1.3% of 29.4 kg). This would equate to a total energy release of only 5.7 MW-sec.

In conclusion, the reactor containment building design pressure of 2.0 psig (13.8 kPa above atmosphere), based on the previous calculations, is completely adequate for all postulated accident scenarios. The design of the containment engineered safety feature gives reasonable assurance that it will not interfere with reactor operation or shutdown.

6.2.3 Penetrations and Closures

Penetrations through the reactor containment building concrete structure are limited in order to minimize potential leakage points. Existing penetrations include a utility entry water seal, a pedestrian entry, an entry for heavy equipment, supply and exhaust air ducts, a hot exhaust line, and two steel penetration plates which allow electrical lines and the pneumatic tube system to enter and exit the containment building.

Three lines used for the reactor containment building leakage rate test also enter the containment structure. Two of these lines enter through one of the penetration plates while the third line penetrates the containment building wall.

The pedestrian and heavy equipment entries and the supply and exhaust air ducts include sealable closures which ensure that sufficient integrity can be maintained to prevent the accidental release of radioactivity to the environment.

Table 6-1 lists all of the penetrations through the reactor containment building concrete structure and their corresponding sealing methods.

6.2.3.1 Utility Entry Water Seal

The utility entry water seal (seal trench) is a water-filled trap which provides 2.0-psig (13.8-kPa above atmosphere) overpressure relief protection for the reactor containment building. The seal trench also provides a path for site utilities to enter and exit the containment building. The seal trench forms a water seal between the reactor containment building and the adjacent laboratory structure. In the event that an overpressure condition within the containment building exceeds 2.0 psig (13.8 kPa above atmosphere), the water is elevated out of the seal trench, overflowing onto the laboratory basement floor. This creates a path for pressure within the containment building to be released into the laboratory building. Once the pressure is relieved, the water level will drop and reseal the reactor containment building. The water in the seal trench also serves as an absorbent for radioactive particulate or gas.

Water from the domestic cold water system is manually added to the seal trench as required to make up for losses due to evaporation. Level is maintained at approximately 65 inches (1.7 m) for all operating conditions. The seal trench is also equipped with a float assembly that initiates a "Utility Trench Seal Water Lo Level" annunciator alarm on low water level.

TABLE 6-1
REACTOR CONTAINMENT BUILDING PENETRATIONS

Penetration	Sealing Method	No.
Utility Entry[a]	Water	1
Pedestrian Entry (Airlock)	Door and Inflatable Gasket	2
Heavy Equipment Entry	Door and Inflatable Gasket	1
Supply Air Duct[b]	Door and Inflatable Gasket	1
Exhaust Air Duct[b]	Door and Inflatable Gasket	1
Pneumatic Tube System	None Required[c]	N/A
Electrical Lines	Sealed Connectors	N/A
Leakage Rate Test Connections 1. Instrument Line 2. Pressurization Line 3. Depressurization Line	 Manual Valve Threaded Plug and Sealant Bolted Flange and Gasket	 1 1 1
Hot Exhaust Line	Automatic Valve	2

[a]All site utilities which enter or exit the reactor containment building through the utility entry water seal are listed in Section 6.2.3.1.
[b]Redundancy for the ventilation plenums is provided by backup doors. Operation of the backup doors, including the sealing method, is described in Section 6.2.4.
[c]Entry for the pneumatic tube system is described in Section 6.2.3.6.

The following services or lines enter or exit the reactor containment building through the seal trench:

a. 6-inch fire protection water line;
b. 6-inch emergency pool fill line;
c. 1½-inch gas line (purged and blank flanged - no longer in use);
d. 2-inch compressed air line;
e. 2-inch domestic cold water line;
f. 2-inch domestic hot water line;
g. ¾-inch domestic hot water return line;
h. 1 ½-inch containment sump discharge;
i. 1 ½-inch sanitary sump discharge;
j. ½-inch demineralized water line;
k. ⅝-inch cooling water discharge line (from the experimental facilities);
l. 1¼-inch vacuum line;
m. ¾-inch PVC alternate air supply line to the supply and exhaust plenum backup doors from the emergency air system;
n. ¾-inch PVC (capped - no longer in use); and
o. ¾-inch PVC (capped - no longer in use).

6.2.3.2 Pedestrian Entry

The pedestrian entry for the reactor containment building (Fig. 6.2) consists of two electric-motor-driven horizontal sliding doors and an intervening vestibule (airlock). The outer door, designated as Door 276, and the inner door, designated as Door 277, allow the entrance of personnel from the laboratory building main corridor, through the airlock, to the second level of the containment building. The airlock is a portion of the containment system.

The doors are constructed of steel and designed to withstand a 2.0-psig (13.8-kPa above atmosphere) overpressure. Each door is suspended from an overhead rail by two adjustable one-ton trolleys. A 3-phase motor connected through a gear reducer to a chain drive assembly drives the door open and closed. When a door is in the fully closed position, a rotary limit switch actuates an air supply valve, inflating a gasket mounted in the door facing, sealing the door. A manually-operated throw-out clutch disengages the motor and gear reducer from the chain drive assembly allowing manual operation of the door at zero differential pressure.

The pedestrian entry control system is designed and interlocked such that one door is always closed and sealed, ensuring that containment integrity is maintained.

6.2.3.3 Heavy Equipment Entry

The heavy equipment entry consists of an electric-motor-driven horizontal sliding door which allows entrance from the laboratory building basement to the reactor containment building beamport

[redacted]

FIGURE 6.2
GRADE LEVEL FLOOR PLAN

floor. This entry, serviced by the laboratory building 15-ton (13,608-Kg) freight elevator, provides an access for heavy and large equipment to be moved into and out of the containment building at below-grade level (Fig. 6.1). The closure for this entry is designated as the truck entry door (Door 101).

Door 101 is constructed of steel and suspended from an overhead rail by two adjustable one-ton (907-Kg) trolleys. A 3-phase motor connected through a gear reducer to a chain drive assembly drives the door open and closed. When the door is in the fully-closed position, a rotary limit switch actuates an air supply valve, inflating a gasket mounted in the door facing, thus sealing the door. A track built into the beamport floor guides the door up against the inflatable gasket. A manually-operated throw-out clutch disengages the motor and gear reducer from the chain drive assembly allowing manual operation of the door at zero differential pressure. A pressure switch monitors gasket seal pressure and initiates a "Truck Entry Door Open Rod Run-In" annunciator alarm and rod run-in upon depressurization of the gasket.

Door 101 is maintained in the closed and sealed position at all times during reactor operation.

6.2.3.4 Supply and Exhaust Air Ducts

The supply and exhaust air ducts contain two electric-motor-driven horizontal sliding doors located in the ventilation plenums on the fifth level of the reactor containment building (Fig. 6.3). When these doors are in the open position, a path is created for containment air to be circulated through the east tower and then back into the containment building. The openings in the containment building wall for each plenum are approximately 4 feet by 4 feet (1.2 m x 1.2 m). The motorized isolation door for the supply plenum is designated as Door 504 and the motorized isolation door for the exhaust plenum is designated as Door 505.

The doors are constructed of ¼-inch (0.64-cm) thick metal plate welded to a steel frame. Each door is suspended from a 12½-foot (3.8-m) long, 6-inch (15.2-cm) I-beam by two adjustable ½-ton (453-Kg) trolleys. To insure proper alignment, the doors travel in a guided slot mounted along the bottom of the door opening. A 3-phase, 440-volt, 1-HP motor connected through a gear reducer to a chain drive assembly drives the door open and closed. When a door is in the fully-closed position, a rotary limit switch energizes a solenoid-operated three-way valve, inflating a gasket mounted in the door facing, thus sealing the door. A manually-operated throw-out clutch disengages the motor and gear reducer from the chain drive assembly allowing manual operation of the door at zero differential pressure.

The supply and exhaust air doors are normally kept in the open position during reactor operation. Actuation of the reactor isolation or facility evacuation switches located in the reactor control room, or the facility evacuation switch located in the facility lobby (Room 202), will close Door 504 and Door 505. A radiation level greater than the set point of either of the reactor bridge radiation monitors, or either of the exhaust plenum radiation monitors, will also close both doors automatically. A closure signal from any of the above causes two parallel relays to de-energize,

[redacted]

FIGURE 6.3
FIFTH LEVEL FLOOR PLAN

closing a contact associated with each relay, and completing the circuit to the closing coil of each motor. The motors will drive the isolation doors to the closed position, actuating their rotary limit switches and energizing the solenoid-operated three-way valves, thus sealing the doors. It takes approximately 7 seconds from the initiation of the isolation signal until the isolation doors are closed and sealed.

6.2.3.5 Electrical Entry

All electrical connections from equipment external to the reactor containment building, as well as electrical power supplies which must enter the containment building, pass through two steel penetration plates located in the containment structure walls. One penetration plate is located in the east wall, and the other is located in the south wall. Each penetration plate contains sealed connectors which allow electrical lines to enter the containment building with minimal air leakage. There are no through-the-containment-wall electrical conduits.

6.2.3.6 Pneumatic Tube System Entry

The pneumatic tube system enters the reactor containment building through a penetration plate located in the east containment structure wall. Eight (8) lines [four 1½-inch (3.8-cm) sample carrier tubes and four 1½-inch (3.8-cm) air-vacuum driver tubes] pass through the penetration plate into the containment building and converge adjacent to the biological shield. The four sample carrier tubes (two on the north side and two on the south side) enter the reactor pool through the biological shield and terminate within the graphite reflector elements surrounding the reactor core.

The pneumatic tube facility is a closed system that extends from the outer building laboratories (which contain the sending-receiving stations) into the containment building, terminating near the reactor core. Since there is no leakage path into the pneumatic tubes after penetrating the containment wall, the pneumatic tube system is considered an extension of the laboratories into the reactor containment building. The pneumatic tubes are designed to withstand 2.0-psig (13.8-kPa above atmosphere) overpressure with zero leakage.

6.2.3.7 Building Leak Rate Test Penetrations

The reactor containment building leak rate test uses three lines which penetrate the containment structure. Two lines consist of pipes welded to the penetration plate located in the containment building east wall. One line is used for pressurizing the containment building and the second line is used for attaching the instruments necessary for measuring internal pressure and leakage rate. A third line runs through a steel collar positioned in the concrete of the containment building east wall and allows for depressurizing the containment structure upon completion of the leak rate test. The line consists of a 4-inch pipe welded to the collar. This line is blank-flanged within the reactor containment building when not in use.

6.2.3.8 Hot Exhaust Line

The hot exhaust line is a 16-inch pipe which discharges potentially contaminated gases from the reactor containment building. Exhaust air from areas which produce radioactive gases or airborne contamination is ducted to this 16-inch line which penetrates the west wall of the containment building just below the ceiling level and discharges to the facility exhaust plenum located in the west tower. A 12-inch (0.3-m) long steel collar surrounding the 16-inch pipe is cast into the containment building wall. The 16-inch pipe is welded to this collar. Two quick-closing 16-inch isolation valves, designated valve 16A and valve 16B, are located on this line within the containment building.

Isolation valve 16A is an air-operated-to-open, spring-to-close, butterfly valve located approximately 12 feet (3.7 m) from the east containment building wall. Isolation valve 16B is an air-operated-to-open, air-operated-to-close, butterfly valve located adjacent to the west containment building wall. Air is supplied to the valve operators from the facility main air compressors and the emergency air compressor. An additional emergency air compressor and accumulator tank dedicated to valve 16B ensures that three possible air sources are available to operate this valve. Closure time of both valves is approximately 3 seconds.

The hot exhaust line isolation valves are normally kept open during reactor operation. Actuation of the reactor isolation or facility evacuation switches located in the reactor control room, or the facility evacuation switch located in the facility lobby (Room 202) will close valves 16A and 16B. Detection of a radiation level greater than the set point of either of the reactor bridge radiation monitors, or either of the exhaust plenum radiation monitors, will also close both valves automatically. Two solenoid-operated valves, installed in series, control the air supply to valve 16A. A closure signal will de-energize both solenoid valves, causing air to be vented from the actuator, and allowing the spring to close the valve. Actuation of either solenoid-operated valve will close valve 16A. The air supply to valve 16B is controlled by four solenoid operated valves. Two valves, installed in series, control the air supply to the "open side" of the actuator, and two valves, installed in parallel, control the air supply to the "close side" of the actuator. A closure signal will de-energize all four solenoid operated valves, venting the "open side" and allowing air to be supplied to the "close side" of the actuator. Should one of the parallel solenoid-operated valves fail to operate, flow control valves installed on each valve's vent port ensure that air is supplied to the "close side" of valve 16B's actuator at a faster rate than is vented from the failed solenoid valve.

6.2.4 Backup Doors

A second set of isolation doors, designated the backup doors, are located in the reactor containment building supply and exhaust plenums thereby providing redundancy for containment building isolation. When the backup doors are shut, the steel plenum chamber above the door becomes part of the containment system.

The backup doors are constructed of ¼-inch (0.64-cm) thick metal plate and reinforced to withstand 2.0-psig overpressure (13.8-kPa above atmosphere) (Fig. 6.4). Each door is held open against gravity by a double acting pneumatic cylinder. Air is supplied to the pneumatic cylinders from the facility main air compressors and the emergency air compressor. A ⅜-inch (0.95-cm) rubber gasket installed in the door facing creates a seal for the backup doors when in the closed position.

The backup doors are normally kept open during reactor operation. A radiation level greater than the set point of either of the reactor bridge radiation monitors, or either of the exhaust plenum radiation monitors, will close both isolation doors automatically. Two solenoid-operated valves, installed in series, control the air supply to each pneumatic cylinder. A closure signal will de-energize both solenoid valves causing air to be vented from the pneumatic cylinder, and allowing gravity to close the isolation door. Actuation of either of the solenoid-operated valves will close the backup door.

6.2.5 Compressed Air and Inflatable Gaskets

To ensure proper operation, all reactor containment building closures with inflatable gaskets are dependent upon a continuous supply of compressed air. Two main air compressors located in the mechanical area (Room 278) of the laboratory building provide the normal supply of compressed air to the inflatable gaskets. This supply of compressed air enters the containment structure through the seal trench. Should the main air system become inoperative, an alternate supply of compressed air is provided by an emergency air compressor located on the fifth level of the containment building (Fig. 6.3). If supply air pressure decreases to less than 70 psig (482.6 kPa above atmosphere), the emergency air system will assume the compressed air load and cause a check valve in the air supply line near the seal trench to shut, thus isolating the main air system external to the containment building. The check valve ensures that the emergency air compressor will only supply compressed air to selected loads within the containment building. These loads include:

a. Personnel Airlock Door 276 Sealing Gasket;
b. Personnel Airlock Door 277 Sealing Gasket;
c. Motorized Ventilation Isolation Door 504 Sealing Gasket;
d. Motorized Ventilation Isolation Door 505 Sealing Gasket;
e. Truck Entry Door (Door 101) Sealing Gasket;
f. Ventilation Exhaust Valve 16A Actuator;
g. Ventilation Exhaust Valve 16B Actuator; and
h. Backup Doors.

The emergency air compressor is powered by the emergency electrical power system to ensure that containment integrity is maintained during a loss of normal electrical power to the reactor facility.

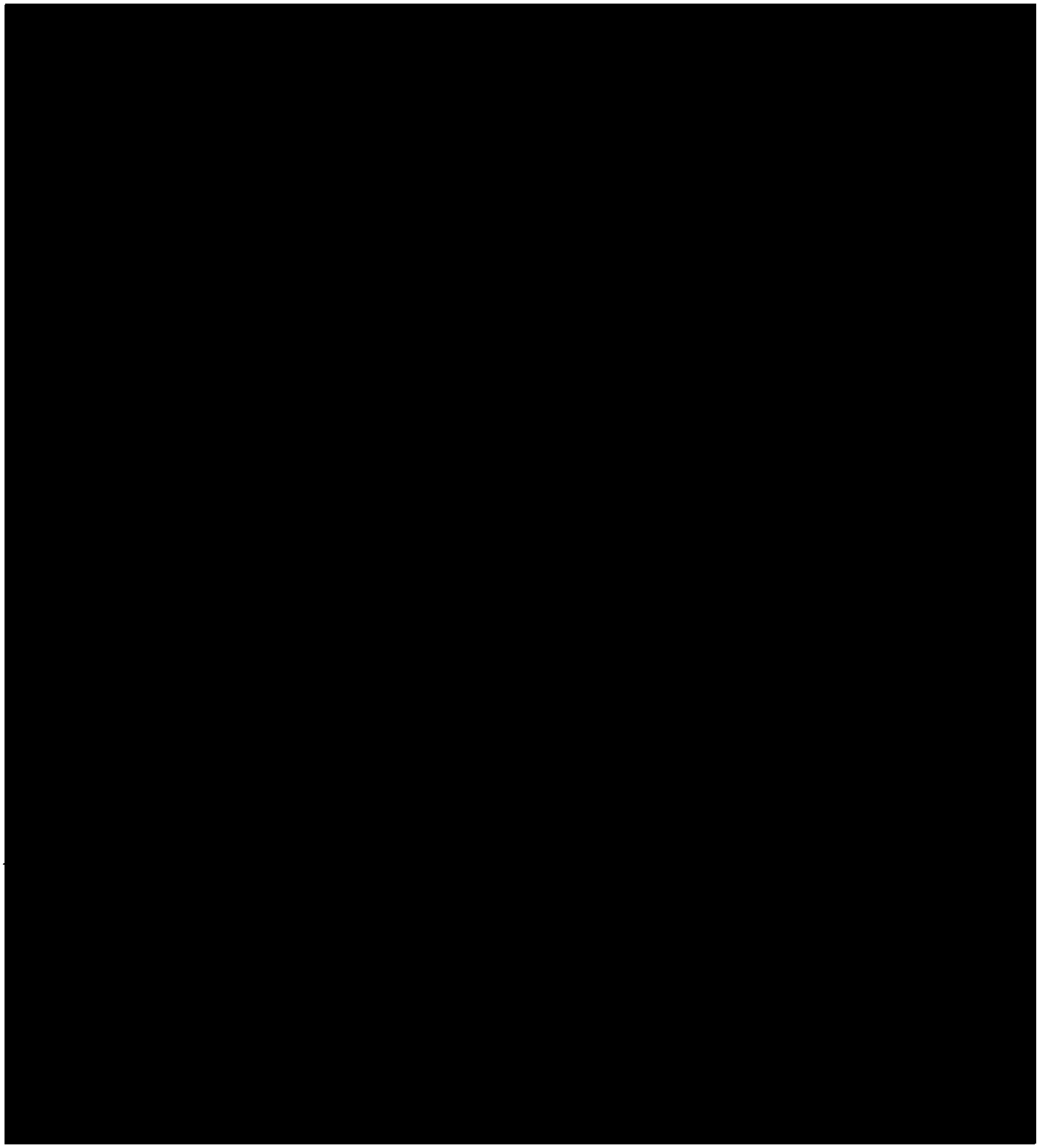

FIGURE 6.4
BACKUP DOOR

6.2.6 Description of Operation

During normal operation of the reactor, the following conditions exist: one pedestrian entry door is closed and sealed, the truck entry door is closed and sealed, the supply and exhaust air doors are open, the backup doors are open, the hot exhaust line valves are open, the utility entry water seal is filled to a level of approximately 65 inches, and pressure within the reactor containment building is slightly negative with respect to the surrounding laboratory building.

In the event of detection of a high radiation level at the reactor pool bridge or in the exhaust plenum, the containment system will provide complete isolation of the reactor containment building. The reactor operator may also initiate a containment building isolation from the control console in the reactor control room or a facility evacuation and subsequent containment building isolation from the facility lobby (Room 202).

6.2.7 System Redundancy and Reliability

To ensure compliance with Criterion 54 of Appendix A of Title 10, Chapter I of the Code of Federal Regulations, Part 50 (10 CFR 50) and IEEE-279-1971, Single Failure Criterion, redundancy is incorporated into the containment system. This ensures that no single component or circuit failure will render the system inoperable. System reliability is achieved through instrumentation and equipment which fail-safe when de-energized. Characteristics of the containment system which provide the redundancy and reliability are as follows:

(1) All relays which energize to perform their intended function are paralleled by a second relay such that, if one fails to energize, the unaffected relay will operate as required;

(2) All relays which de-energize to perform their intended function are assumed to be fail-safe;

(3) The solenoid-operated valves which control the compressed air supply to the hot exhaust line isolation valves (valve 16A and valve 16B) de-energize to shut the isolation valves, therefore, they are designed to be fail-safe;

(4) Hot exhaust line isolation valve 16A is spring-to-close which is designed fail-safe; isolation valve 16B is air-operated-to-close and supplied by three different compressed air sources; the isolation valves are installed in series to provide redundancy;

(5) Power to operate the supply and exhaust air doors (Door 504 and Door 505) is supplied by both the normal and emergency electrical power systems; therefore, the doors will still close in case of a simultaneous containment isolation and loss of normal electrical power; and

(6) The solenoid-operated valves which control the compressed air supply to the backup doors de-energize to shut the isolation doors, therefore, they are designed to be fail-safe; the backup doors provide redundancy for Door 504 and Door 505.

At least two major failures must occur before the containment system would be unable to perform its intended function. Containment building isolation is needed only in the event that fission products or other large sources of radioactivity have been released into the reactor containment building. This circumstance requires a major system failure; an event which itself is highly unlikely. A failure of the containment system must occur simultaneously with this event for the system to fail to perform its function.

6.2.8 <u>Limiting Conditions for Operation</u>

The limiting conditions for operation of the reactor containment building and their bases are discussed in the Technical Specifications. The objective of this specification is to reasonably ensure that the health and safety of the general public are not endangered as a result of reactor operation.

6.2.9 <u>Surveillance</u>

The reactor containment building is designed to hold a peak internal overpressure of 2.0 psig with a leakage rate over a 24-hour period not exceeding 10% of the contained volume or 16.3 ft^3/min (STP) with an overpressure of 1.0 psig. To ensure that these design limits are maintained, a containment building leakage rate test is performed as stated in the Technical Specifications.

The reactor isolation and facility evacuation system and its associated radiation monitors are tested for operability as stated in the Technical Specifications. The reactor containment building instrumentation and control systems, testing, surveillance provisions and intervals, and related Technical Specifications reasonably ensure that, if required, the containment engineered safety feature will be available and operable.

6.2.10 <u>Design Features</u>

The specifications and bases for the design features of the reactor containment building are discussed in the Technical Specifications. The objective of these specifications is to ensure adequate restriction in the event of an accidental release of radioactivity to the environment. The design and functional features of the reactor containment building ensure that exposures will be maintained below the limits of 10 CFR 20. Radiological consequences for postulated events are provided in Chapter 13, Accident Analyses.

6.3 <u>Anti-Siphon System</u>

6.3.1 <u>Introduction</u>

The anti-siphon system consists of a pressure tank, two automatic isolation valves, a level controller, and associated piping and valves. This system functions as a backup system to the various safety instrumentation and equipment (e. g., pressure sensors, pump and valve interlocks, etc.) all of which ensures that the reactor core does not become uncovered during a LOCA. The system is designed to admit a contained volume of air to the high point of the reactor outlet piping (invert loop), instantaneously establishing the pressure in this area at equal to, or greater than, atmosphere. This prevents a siphon action from being created due to a rupture of the primary coolant piping. Redundancy is incorporated into the system to ensure that no single component or circuit failure will render any portion of the anti-siphon system inoperative. The LOCA is analyzed in Chapter 13, Accident Analyses.

The instrumentation necessary for the anti-siphon system to perform its intended function is described in greater detail in Section 7.8, Engineered Safety Features Actuation Systems.

6.3.2 Design Criteria

The design and functional features of the anti-siphon system ensure that exposures will be maintained below the limits of 10 CFR 20. Radiological consequences for postulated events are provided in Chapter 13, Accident Analyses. Two main criteria were considered in the design of the anti-siphon system to prevent the reactor core from becoming uncovered during a rupture in the primary coolant system. The criteria are as follows:

(1) The entire anti-siphon system must contain a sufficient volume of air to break the siphon should a double-ended primary coolant pipe rupture occur; and

(2) The maximum pressure in the anti-siphon system will be maintained below the operating reactor core pressure to minimize the possibility of air introduction into the primary coolant system should anti-siphon isolation valve leak-by occur.

An analysis[1] based upon the above two criteria imposed the following operational limits on the anti-siphon system pressure:

- Maximum System Pressure - 45 psig (310 kPa above atmosphere); and
- Minimum System Pressure - 27 psig (186 kPa above atmosphere).

The minimum required anti-siphon system pressure is based upon ensuring that all the water can be displaced in the reactor outlet leg to the level of the primary coolant system outlet isolation valve (V507A) in the event of a double-ended primary coolant pipe rupture. If atmospheric pressure is then maintained at the junction of the riser leg (leading to the natural convection flange port) and the anti-siphon isolation valves, the siphon will be broken and the reactor core will remain covered with a minimum of five feet of water.

The thermal-hydraulic transient code RELAP5 was used to update and validate the original design analysis of the anti-siphon system.

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[1]See Footnote on Page 6-1.

6.3.3 Pressure Tank

The anti-siphon pressure tank is a 25-gallon (95-l), 0.105-inch (2.7-mm) thick stainless steel tank located on the lower bridge level of the reactor pool. The tank is cylindrically-shaped with nominal dimensions of 16 inches (41 cm) in diameter and 30 inches (76 cm) in length, and a design pressure of 150 psig (1.03 MPa above atmosphere). A ¾-inch (1.9-cm) aluminum pipe connects the pressure tank to the 4-inch (10.2-cm) anti-siphon vent line. A relief valve set at 100 psig (0.69 MPa above atmosphere) is connected to the pressure tank air supply line to protect the anti-siphon system from over-pressure.

6.3.4 Anti-Siphon Isolation Valves

Two 4-inch, air-operated-to-close, spring-to-open, quick acting automatic butterfly valves are located on the anti-siphon vent line in the reactor pool. If primary coolant system pressure decreases to a predetermined value, the isolation valves automatically open, allowing a flow path between the primary coolant system (invert loop) and the anti-siphon pressure tank. Operation of either valve will perform the intended function of the anti-siphon system. Manual operation of the anti-siphon isolation valves is also possible from the upper bridge, if required. The anti-siphon isolation valves are designated V543A and V543B.

6.3.5 Level Controller

A level controller, designated LC 965, initiates an "Anti-Siphon Line Hi Level Rod Run-In" annunciator alarm and rod run-in if water level within the anti-siphon system rises to greater than 6-inches (15.2-cm) above the anti-siphon isolation valves. The rod run-in ensures that the introduction of water will not prevent the anti-siphon system from performing its intended function. The level controller is mounted to the end of an aluminum drywell and actuated by a displacer (float) suspended from a stainless steel cable. This arrangement allows the switch housing to be positioned above the surface of the reactor pool, thus preventing the possibility of flooding the switch housing with pool water. The level controller drywell is located adjacent to the reactor pool liner and away from the reactor core and primary coolant system piping to prevent radiation streaming.

6.3.6 Description of Operation

During normal reactor plant operation, the anti-siphon system is maintained dry and pressurized to 36 psig (248 kPa above atmosphere). The system pressure is checked and recorded every four hours as part of the facility routine patrol. In the event of an "Anti-Siphon Vent Tank Hi-Lo Pressure" annunciator alarm, action is taken to vent or add air to clear the alarm. If an anti-siphon system leak or other malfunction prevents maintaining a minimum system pressure of 27 psig (186 kPa above atmosphere), the reactor is shut down until the malfunction can be corrected. A low point drain line and valve is used to blow out excess water during startup of the primary coolant system.

In the event of a rupture of the 12-inch primary coolant system piping, a loss of system pressure will be detected by four electronic pressure transmitters (PTs) or pressure sensors (PSs): PT 944A, PT 944B, PT 943, and PS 938. Each pressure transmitter will initiate a reactor scram. PT 944A and PT 944B will also cause the following to occur: the primary coolant circulation pumps 501A and 501B will stop; the primary coolant isolation valves V507A and V507B will close; and the anti-siphon system isolation valves V543A and V543B will open. Additionally, in pool heat exchanger isolation valves V546A and V546B will open when primary coolant isolation valves start to close due to a valve interlock. Upon opening, the anti-siphon isolation valves will admit a fixed volume of air to the high point of the reactor outlet piping, thus preventing the reactor core from becoming uncovered by breaking any potential siphon which may have been created by the pipe rupture.

6.3.7 System Redundancy and Reliability

To insure compliance with the IEEE 279-1971 Single Failure Criterion, redundancy is incorporated into the anti-siphon system. This ensures that no single component or circuit failure will render the system inoperable. System reliability is achieved through instrumentation and equipment which fail-safe when de-energized. Characteristics of the anti-siphon system which provide the redundancy and reliability are as follows:

(1) All relays which de-energize to perform their intended function are assumed to be fail-safe;

(2) Redundant primary coolant system low pressure signals (2), either of which is capable of initiating operation of the anti-siphon system, are provided;

(3) Two isolation valves (V543A and V543B) are installed in parallel; operation of either valve will perform the intended function of the anti-siphon system; and

(4) V543A and V543B are spring-to-open which is assumed fail-safe; the solenoid-operated valves which control the compressed air supply to V543A and V543B de-energize to vent air from the actuators, therefore, they are also designed fail-safe.

6.3.8 Design Analysis

RELAP5, which is a light and heavy water transient analysis code developed at the Idaho National Engineering and Environmental Laboratory (INEEL) for the U.S. Nuclear Regulatory Commission (NRC), was used to update the MURR LOCA and verify that the anti-siphon system can perform its intended function as designed. RELAP5 is capable of analyzing a wide variety of thermal-hydraulic transients in nuclear and non-nuclear systems involving mixtures of steam, water (light/heavy), non-condensables, and solute, and is one of the most widely used system codes for analyzing reactor accidents/transients.

A simplified diagram of the primary coolant and anti-siphon systems is shown in Figure 6.5. The RELAP5 code analysis assumes that the anti-siphon isolation valves will open 85 msec after the primary coolant piping rupture occurs, thus connecting the air volume in the anti-siphon pressure tank to the 12-inch primary coolant piping vertical riser and in pool heat exchanger. The primary coolant system valves and pumps will actuate as described in Section 6.3.6. The siphon-break caused by the introduction of air prevents the upstream air-water interface from moving through the piping and eventually uncovering the reactor core. Viscous and turbulent flow losses resist siphoning but are not sufficient enough to stop flow. The anti-siphon system provides a sufficient volume of air at the high point of the reactor outlet piping (invert loop) to break the siphon effect.

In the MURR RELAP5 model, with a total primary coolant flow rate of 4,000 gpm (15,142 lpm), the LOCA was analyzed at the following three different primary coolant temperatures: 80, 120 and 155 °F (27, 49 and 68 °C). For a primary coolant temperature of 80 °F (27 °C), it was determined that a minimum anti-siphon system pressure of 7 psig (48 kPa above atmosphere) was required to ensure that five feet (1.5 m) of water was maintained over the reactor core after a primary coolant piping rupture. Lower minimum anti-siphon system pressures could be used at the higher temperatures of 120 and 155 °F (49 and 68 °C). Regardless, the minimum anti-siphon system pressure will conservatively be kept at the current Technical Specification limit of 27 psig (186 kPa above atmosphere).

6.3.9 <u>Limiting Conditions for Operation</u>

The limiting conditions for operation of the anti-siphon system are discussed in the Technical Specifications. The objective of these conditions is to protect the fuel integrity and prevent a release of fission products.

6.3.10 <u>Surveillance</u>

The anti-siphon isolation valves are tested for operability as stated in the Technical Specifications. The objective of the specification is to reasonably ensure proper operation of the component. The anti-siphon instrument and control systems, testing surveillance provisions and intervals, and related Technical Specifications reasonably ensure that, if required, the anti-siphon system engineered safety feature will be available and operable.

[redacted]

FIGURE 6.5
SIMPLIFIED DIAGRAM OF THE PRIMARY COOLANT SYSTEM

CHAPTER 7

INSTRUMENTATION AND CONTROL SYSTEMS

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7.1 Preliminary Hazards Report, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, March 1961.

7.2 Hazards Summary Report, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, July 1965.

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7.0 INSTRUMENTATION AND CONTROL SYSTEMS

This chapter describes and discusses the operating characteristics of the MURR Instrumentation and Control (I&C) Systems. Included in this chapter are the design criteria and bases, and the functional and safety analyses of the I&C subsystems. These systems and their design criteria and bases assure that the MURR may be safely operated, monitored, and shut down as warranted. The system is sufficiently sensitive to ensure control and shutdown of the reactor.

7.1 Summary Description

The Instrumentation and Control Systems are comprised of the sensors, electronic circuitry, displays, and actuating devices that are available to provide the information and means to safely control the reactor and avoid or mitigate potential accidents. The I&C Systems include the following:

- Nuclear Instrumentation System;
- Rod Control System;
- Process Instrumentation and Control System;
- Reactor Safety System;
- Engineered Safety Features Actuation Systems; and
- Radiation Monitoring Systems.

The Nuclear Instrumentation (NI) System continuously monitors and displays the neutron flux from the subcritical source multiplication range, through the critical range, and through the intermediate flux range to full power while also providing reactor period information. In addition, the NI System provides input signals to the Reactor Safety and Rod Control Systems.

The Rod Control System enables manual control of reactor power from source to power range levels and automatic control after a minimum power level has been attained. This system also provides the capability of placing the reactor in a subcritical condition by a rod run-in, which initiates the automatic insertion of the control blades at a controlled rate should a monitored parameter exceed a predetermined value. Inputs which govern the rod run-in system are supplied from the neutron flux monitors, process transducers, and safety interlocks. A rod run-in may also be initiated manually by the reactor operator.

The Process Instrumentation and Control System monitors, displays, and controls the following reactor plant parameters: temperature, pressure, flow, and pressurizer liquid level. The system provides input signals to the Reactor Safety System in addition to control interlocks for the primary and pool coolant system circulation pumps and automatic isolation valves.

The Reactor Safety System is designed to prevent operation of the reactor in regions in which fuel damage may occur. This is accomplished through promptly placing the reactor in a subcritical, safe shutdown condition by a reactor scram, which initiates the instantaneous drop of the control blades by interrupting power to their electromagnets should a monitored parameter exceed a predetermined value. Inputs which govern the Reactor Safety System output are supplied from the neutron flux monitors, process transducers, and safety interlocks. A reactor scram may also be initiated manually by the reactor operator.

The Engineered Safety Features Actuation Systems receive input signals from monitoring instruments and initiate the operation of the engineered safety systems which are designed to mitigate the consequences of certain identifiable accidents, thereby keeping radiological exposures to the operating staff and the general public within the limits of 10 CFR 20.

Four Radiation Monitoring Systems detect and quantify radiation and activity levels at various locations within the facility, within various reactor systems, and within the exhaust gases released to the uncontrolled environment. One of these systems provides input signals to the Engineered Safety Features Actuation System, which initiates the Containment System - an engineered safety feature which provides a complete isolation of the reactor containment building.

7.2 Design of Instrumentation and Control Systems

7.2.1 Design Criteria

The Instrumentation and Control Systems are designed[1] to perform the following functions:

- Providing the reactor operator with information on the operating status of the reactor and the facility;
- Providing the means to manually insert and withdraw the control blades;
- Providing for automatic control of reactor power level;
- Providing the means to insert the control blades, by rod run-in or scram, should a monitored parameter exceed a predetermined value;
- Providing the means to detect and measure the radiation and activity levels at the reactor facility, including the release of radioactive gases from the facility;
- Providing a means to initiate the Engineered Safety Features; and
- Providing for the storage of operational data for later retrieval.

The principal purpose of the reactor facility is to provide neutrons to the experimental facilities, and the design effort has been directed toward accomplishing this end. An essential feature

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[1]The original design and safety evaluation of the MURR is documented in the Preliminary Hazards Report (Ref. 7.1), the Hazards Summary Report (Ref. 7.2), and Hazards Summary Report, Addenda 1-5 (Ref. 7.3-7.7).

of the reactor core design is the high neutron leakage. Control of the reactor, during all conditions of operation, is accomplished by surrounding the reactor core with a reflector region (beryllium metal followed by canned graphite) which increases the fraction of leakage neutrons that return to the core region, and by interposing a movable shroud of a material opaque to thermal neutrons (boron carbide-aluminum mixture) between the core and the beryllium reflector. No other reactor parameter, such as primary and/or pool coolant temperature and pressure, is used to change or alter reactor power other than through inherent reactivity feedback effects. In addition, the beryllium reflector effectively decouples the reactor core from reactivity effects caused by variations to the experimental facilities. The only experimental facility which is not decoupled from the reactor in this manner is the center test hole (flux trap). This decoupling design feature can be considered an inherent safety feature that effectively assists the control blades in controlling reactor power.

The Control System for the reactor is designed[1] such that cold clean criticality will be achieved with the control blades withdrawn 50%. This criticality position of the rods is based on the following three criteria:

1. At least 50% of the control blade worth being available at all times for the shutdown margin;

2. At least half of the core being exposed to the experimental facilities; and

3. Sufficient excess reactivity being provided in the core for a fuel loading lifetime of at least 400 Megawatt-days. (Note: This criterion is based on the assumption that the graphite region exhibits its maximum possible reflectivity, i.e., a solid ring.)

Since the purpose of the MURR is to provide the maximum flux, or current, of neutrons to the greatest number of experiment users, the reactor is operated continuously, with scheduled shutdowns for refueling, changing samples in the flux trap, and performing any corrective or preventive maintenance. Start-up of the reactor is performed manually until a desired power level is obtained. At this point, the reactor may be placed in automatic control. The controlling parameter for automatic operation is the neutron flux monitored by a wide range signal processor. The output signal from the wide range signal processor is compared with the power schedule setting of a servo amplifier unit. The servo amplifier adjusts the position of the regulating blade, stepwise, in a direction to minimize the discrepancy between the power demand setting and actual power level.

The only other parameter that will normally influence reactivity of the reactor is primary coolant temperature, but this coefficient is negative and its effect is relatively small. An increase in primary coolant temperature from 110 °F (43 °C) - cold core - to an operating temperature of 140 °F (60 °C) will add a negative 0.002 Δk to the reactor core.

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[1]See Footnote on Page 7-2.

7.2.2 Design-Basis Requirements

The design-basis requirements for the I&C Systems with respect to response time, accuracy, and continuity of operation are consistent with those of other open-pool type research reactors.[1]

The MURR Reactor Safety System includes all of the sensing devices, electronic circuits and equipment, signal conditioning equipment, and electro-mechanical devices that serve to effect a reactor shutdown by the removal of the holding current from the four control rod drive mechanism electromagnets, or to activate the engineered safety features (ESFs). Design modifications to the Reactor Safety System were performed during the upgrade in power to 10 MW to further reduce the probability of an accidental release of radioactivity from the reactor facility to the unrestricted environment. The function of the reactor scram circuit is to interrupt the magnet current in the event of an abnormal condition in an essential reactor system. Secondary actions which may be initiated by the reactor scram circuit are not considered part of the protective function. Based on an evaluation performed by NUS Corporation of Rockville, Maryland (Ref. 7.6), the Reactor Safety System was redesigned to meet all the applicable criteria of the Institute of Electrical and Electronics Engineers Standard 279-1971 (IEEE 279), "Criteria for Protection Systems for Nuclear Power Generating Stations" (Ref. 7.9). As the title of IEEE 279 implies, it is intended for application, especially during the design phase, to nuclear power generating station protection systems. Although the MURR is not a nuclear power generating station, much of the criteria of IEEE 279 can be applied to evaluate the adequacy of the Reactor Safety System with respect to present-day standards for functional performance and reliability. The control rod drive mechanisms are not within the scope of IEEE 279, however, they are an integral part of the Reactor Safety System and therefore were considered in the analysis of safety system reliability. The control rod drive mechanisms satisfy the single failure criterion.

In addition to the design modifications performed to the Reactor Safety System, particular attention was also paid to other plant safety systems to ensure compliance with the single failure criterion. The systems were analyzed and evaluated, and modified, if required, to meet this criterion. The Nuclear Instrumentation, Rod Control, and the Process Instrumentation and Control Systems all meet the single failure criterion of IEEE-279.

The channels which provide the input signals to the Reactor Safety System are redundant. The circuitry for the nuclear instruments and the protective equipment located in the reactor control room, including the signal cables, are not physically separated, however, these channels are accessible to the reactor operators and hence, under continuous surveillance. This lack of physical separation does not compromise the ability to safely operate or shut down the reactor, and the channels' accessibility is an added feature that ensures safe operation. Signal lines from the process sensors, transmitters, controllers, and switches located in the mechanical equipment room (Room 114) which provide redundant protective functions are separated and clearly identified. Redundancy is incorporated into the Engineered Safety Features Actuation Systems to ensure that

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[1]See Footnote on Page 7-2.

no single component or circuit failure will render any portion of the system inoperative. Characteristics of the actuation system which provide the redundancy and reliability are discussed in Section 7.8.

Bypass switches are used in the Reactor Safety System to change the protective system lineup to correspond to the three modes of operation: 50 kW, 5 MW, or 10 MW. All bypass switches are located on the reactor control console or the instrument panel and are in clear view of the reactor operator. The bypass switches are of a key lock design, preventing unauthorized or inadvertent actuation during reactor operation. The Power Level Selector Switch (1S8), which is also on the control console, is interlocked with the bypass switches to prevent reactor operation with a bypass function unless the reactor is lined up for the proper mode of operation. The reactor mode of operation is clearly indicated by a series of lights located on the instrument panel.

The prudent operational limits imposed upon the parameters monitored by the Reactor Safety System, the margin between those limits and the levels that mark the onset of unsafe conditions (i.e., safety limits), and the levels at which protective action should be initiated have effectively been set by the overall system design and by established reactor scram set points as discussed in Section 4.6.4, Limiting Safety System Settings.

7.2.3 Conclusion

Since 1966, operating experience with the I&C systems has demonstrated that the systems are safe and practical, and of proven design. Tests of extremes of power supply voltage and ambient operating temperature have shown no significant effects on the operation of the reactor protective system. Very extreme conditions such as fire, flood, and earthquake have not been experienced; however, at the MURR, these occurrences would be such that more than sufficient time would exist for prompt operator action.[1] All electronics with the exception of the remote sensors, shim blade electromagnets, and connecting wire are located in the reactor control room where a reactor operator is continuously present during operation of the reactor.

Periodic updating of the I&C systems has been performed to take advantage of technological improvements and state-of-the-art developments, while retaining the desirable design characteristics of the previous system. This has increased reliability and enhanced system performance.[1]

7.3 Control Console and Display Instruments

The essential displays and control equipment enabling a reactor operator to observe and control the operation of the reactor are located on two cabinets: the reactor control console and the instrument panel. The instrumentation on these cabinets is arranged in locations which incorporate human engineering factors in order to facilitate the safe and efficient operation of the reactor. The output instruments and the controls in the MURR control console have been designed for checking

[1]See Footnote on Page 7-2.

operability, inserting test signals, performing calibrations, and verifying trip settings. The availability and use of these features ensure that the control console devices and subsystems and display instruments will operate as designed. Control console locking devices reasonably ensure that the facility is only operated by authorized personnel. The control console and instrument panel are located within the reactor control room - a centralized operating station located on the third level of the reactor containment building.

The reactor control room is 28 feet 8 inches (8.7 m) wide by 17 feet (5.2 m) deep with an 8-foot (2.4-m) high suspended acoustical ceiling. Windows along the west wall provide the reactor operator with an unobstructed view of the reactor pool surface and operating (upper) bridge area. One-inch thick steel plating is positioned and supported against the lower half of the windows to provide shielding for personnel in the control room from reactor pool background radiation. A door in the north wall allows entry into the control room from the stairwell and the passenger elevator vestibule. Two openings in the floor provide cable access to the reactor control console and the instrument panel. Communication between the control room and other areas in the reactor facility is provided by a computerized telephone system and an intercommunication system which allows two-way communication between a master station and a staff station. A paging feature allowing several different telephones to originate a page is incorporated into the telephone system. The communication systems are discussed in greater detail in Chapter 9, Auxiliary Systems. The general layout of the reactor control room is shown in Figure 7.1.

The reactor control console is a straight desk type with a sloping face and a horizontal top of sufficient depth to permit utilization as a writing surface. The console is 93¾ inches (2.4 m) long and 44 inches (1.1 m) high (overall), consisting of steel panel and support frame construction sufficiently reinforced to minimize deflection as a result of the mounted instrumentation. All display instrumentation (Table 7-1) and control equipment (Table 7-2) mounted on the sloping panels are within easy reach of the reactor operator. The reactor control console layout is shown in Figure 7.2.

The instrument panel is 21 feet 7 inches (6.6 m) wide and 8 feet (2.4 m) high and positioned 5 feet 6 inches (1.7 m) from the east wall of the reactor control room and approximately 4 feet (1.2 m) in front of the control console. This arrangement creates a limited access cubicle where the connecting cables of the equipment mounted on the instrument panel can be reached. Access is provided by a small door near the north wall of the reactor control room. A 60-point annunciator, which provides the reactor operator with an audible and visual alarm of an abnormal condition, is mounted on the upper left side of the panel. Instrument panel display and control equipment locations are listed in Table 7-3 and shown in Figure 7.3.

[redacted]

FIGURE 7.1
REACTOR CONTROL ROOM LAYOUT

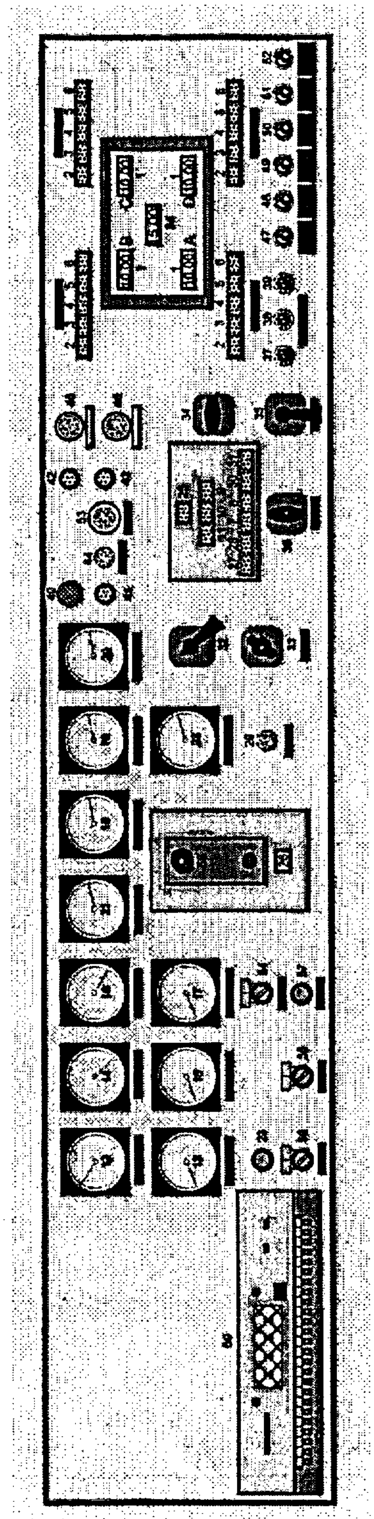


FIGURE 7.2
REACTOR CONTROL CONSOLE LAYOUT

TABLE 7-1
REACTOR CONTROL CONSOLE DISPLAY INSTRUMENTS

No.[a]	Function	Type
1	Control Rod Position Indication (4)	Digital
2	Control Rod Drive Mechanism "Power On" (4)	Light
3	Control Rod Drive Mechanism "Drive Full In" (4)	Light
4	Control Rod Drive Mechanism "Drive Full Out" (4)	Light
5	Control Rod Drive Mechanism "Magnet Engaged" (4)	Light
6	Control Rod Drive Mechanism "Blade Full In" (4)	Light
7	Regulating Rod "Full In"	Light
8	Regulating Rod "Full Out"	Light
9	Regulating Rod "10% Withdrawn"	Light
10	Regulating Rod "20% Withdrawn"	Light
11	Regulating Rod "60% Withdrawn"	Light
12	Source Range Level - Channel 1	Meter
13	Source Range Period - Channel 1	Meter
14	Intermediate Range Level - Channel 2	Meter
15	Intermediate Range Period - Channel 2	Meter
16	Intermediate Range Level - Channel 3	Meter
17	Intermediate Range Period - Channel 3	Meter
18	Power Range Level - Channel 4	Meter
19	Power Range Level - Channel 5	Meter
20	Power Range Level - Channel 6	Meter
21	Wide Range Level	Meter
22	Power Level Set	Meter
23	Pneumatic Tube System Blowers "On" Indication	Light
24	Regulating Rod Position Indication	Digital

[a]The number on this table refers to the position of the display instruments on the reactor control console (Figure 7.2).

TABLE 7-2
REACTOR CONTROL CONSOLE CONTROL EQUIPMENT

No.*	Function	Positions	Type of Switch
25	Rod Control "Auto Shim Engaged"	N/A	Light
26	High Power "Warning"	N/A	Light
27	Rod Control Mode	"Manual"	Push Button
28	Rod Control Mode	"Auto"	Push Button
29	Power Schedule Selector	"Lower-Off-Raise"	3 Pos. - Spring Ret.
30	Regulating Rod Operate	"Jog In"	Push Button
31	Regulating Rod Operate	"Jog Out"	Push Button
32	Master Control	"Off-Test-On"	3 Position - Keylock
33	Power Level Selector	"50kW-5MW-10MW"	3 Position
34	Control Rod Selector	"A-B-C-D-Gang"	5 Position
35	Control Rod Operate	"In-Normal-Out"	3 Pos. - Spring Ret.
36	Regulating Rod Operate	"In-Normal-Out"	3 Pos. - Spring Ret
37	Annunciator - Acknowledge	N/A	Push Button
38	Annunciator - Reset	N/A	Push Button
39	Annunciator - Test	N/A	Push Button
40	Scram	N/A	Push Button
41	Scram Reset	N/A	Push Button
42	Rod Run-In	N/A	Push Button
43	Rod Run-In Reset	N/A	Push Button
44	Magnet Current	"Off-On"	2 Position
45	Reactor Isolation	"Off-On"	2 Position
46	Facility Evacuation	"Off-On"	2 Position
47	Hi/Low Reflector ΔP	"Off-Bypass"	2 Position - Key Lock
48	Low Pressurizer Pressure	"Off-Bypass"	2 Position - Key Lock
49	Low Primary Pressure	"Off-Bypass"	2 Position - Key Lock
50	Vent Tank Low Level	"Off-Bypass"	2 Position - Key Lock
51	Rod Magnet Contact	"Off-Bypass"	2 Position - Key Lock

*The number on this table refers to the position of the display instruments on the reactor control console (Figure 7.2).

TABLE 7-2 (cont.)
REACTOR CONTROL CONSOLE CONTROL EQUIPMENT

No.*	Function	Positions	Type of Switch
52	Anti-Siphon High Level	"Off-Bypass"	2 Position - Key Lock
53	Intrusion Alarm	"Off-On"	2 Position
54	Airlock Door Security	"Off-On"	2 Position
55	Thermal Column Shutter	"Closed-Open"	2 Position
56	Pneumatic Tube Blowers	"Off-On"	2 Position
57	Airlock Door Open	N/A	Push Button
58	Range Switch	N/A	18 position
59	Intercommunications Box	N/A	N/A

*The number on this table refers to the position of the control equipment on the reactor control console (Figure 7.2).

7.3.1 Annunciator

The annunciator is located in a space approximately 3 feet (0.9 m) wide by 2 feet (0.6 m) high and contains a six by ten array of plug-in type alarm modules. Each alarm module consists of a discretely labeled 3⅝-inch by 3⅜-inch translucent plastic face with an internal lamp bulb and relay assembly which actuates the module by illuminating the bulb and initiating an audible alarm. The modules are segregated by color illumination as follows: red for a reactor scram; blue for a rod run-in; and white for an abnormal event other than a scram or rod run-in. The annunciator control circuit is shown in Figure 7.4. Annunciation is initiated by inputs from neutron flux monitors, radiation monitors, or process logic circuits. When alarm functions are received, relays in the annunciator control de-energize, remove 115-VAC from the annunciator, and consequently cause visual and audible annunciation. In addition to the alarm function, other relay circuits and relays in the annunciator control provide logic networks and auxiliary outputs for control functions. The annunciator alarm sequence is described in Table 7-4. Alarm conditions displayed by the annunciator are shown in Table 7-5.

TABLE 7-3
INSTRUMENT PANEL INSTRUMENTATION

No.[a]	Function
1	Annunciator
2	Source Range Monitor Level Recorder
3	Intermediate Range Monitor Level Recorder - 2 Pen
4	Wide Range Monitor Level Recorder - 2 pen
5	Power Range Level Monitor Recorder - 3 Pen
6	Multiscaler
7	Neutron Flux Monitor - Signal Processor Drawer
8	Neutron Flux Monitor - Signal Processor Drawer
9	Neutron Flux Monitor - Signal Processor Drawer
10	Wide Range Neutron Flux Monitor - Signal Processor Drawer
11	Annunciator & Interlock Relay Drawer (K-1 Relays)
12	Servo Amplifier Drawer
13	Auxiliary Annunciator - Panalarm
14	Non-Coincidence Logic Unit - Reactor Safety System "Yellow Leg"
15	Trip Actuator Amplifier - Reactor Safety System "Yellow Leg"
16	Non-Coincidence Logic Unit - Rod Run-In System
17	Trip Actuator Amplifier - Rod Run-In System
18	Non-Coincidence Logic Unit - Reactor Safety System "Green Leg"
19	Trip Actuator Amplifier - Reactor Safety System "Green Leg"
20	+20 Vdc Regulated Power Supply Drawer (2PS1)
21	+20 Vdc Regulated Power Supply Drawer (2PS2)
22	Control Rod Drop Timer Circuit
23	Rod Position Indication Drawer
24	Reactor Safety System Relay Drawer (K-2 Relays)
25	Primary Coolant System Pressure Meter - PT 943
26	Primary Coolant HX503A Outlet Temperature Meter - TE 980A
27	Primary Coolant HX503B Outlet Temperature Meter - TE 980B
28	Dual Alarm Unit (EP 953A/B) - Primary Coolant High Temperature Scram
29	RTD Transmitter (EP 903B) - Primary Coolant T_h
30	Isolated Power Supply - Reactor Safety System "Yellow Leg"
31	Square Root Converter (EP 919A) - Primary Flow "A" Loop
32	Dual Alarm Unit (EP 920A/B) - Primary & Pool Low Flow Scrams
33	Square Root Converter (EP 919B) - Pool Flow "B" Loop
34	RTD Transmitter (EP 903A) - Primary Coolant T_c
35	Adder-Subtractor (EP 954) - Primary Coolant Differential Temperature
36	RTD Transmitter (EP 955) - In-Pool Heat Exchanger Differential Temperature
37	RTD Transmitter (EP 903C) - Pool Coolant T_c
38	Adder-Subtractor (EP 952) - Pool Coolant Differential Temperature
39	RTD Transmitter (EP 903D) - Pool Coolant T_h
40	Square Root Converter (EP 919C) - Primary Demineralizer Flow
41	Square Root Converter (EP 919D) - Pool Demineralizer Flow
42	Isolated Power Supply - Reactor Safety System "Green Leg"
43	Square Root Converter (EP 919E) - Primary Flow "B" Loop

TABLE 7-3 (cont.)
INSTRUMENT PANEL INSTRUMENTATION

No.[a]	Function
44	Dual Alarm Unit (EP 920C/D) - Primary & Pool Low Flow Scrams
45	Square Root Converter (EP 919F) - Pool Flow "A" Loop
46	Signal Resistor Unit - Pressurizer Water Level
47	Power Mode I, II, & III Indication Lights
48	Clock
49	"REACTOR ON" Light
50	Annunciator Alarm Power Switch - 3 Position
51	Area Radiation Monitor - Beamport Floor North Wall
52	Area Radiation Monitor - Beamport Floor West Wall
53	Area Radiation Monitor - Beamport Floor South Wall
54	Area Radiation Monitor - Containment Building Exhaust No. 1
55	Secondary Coolant System Radiation Monitor
56	Area Radiation Monitor - Reactor Pool Upper Bridge ALARA
57	Area Radiation Monitor - Reactor Pool Upper Bridge
58	Area Radiation Monitor - Beamport Floor East Wall
59	Area Radiation Monitor - Fuel Vault
60	Area Radiation Monitor - Mechanical Equipment Room (Room 114)
61	Area Radiation Monitor - Containment Building Exhaust No. 2
62	Fuel Element Failure Radiation Monitor
63	Circuit Fuses (7)
64	Reactor Pool Upper Bridge Radiation Monitor Upscale Switch & Light
65	Containment Ventilation Isolation Door 504 Stop Push Button
66	Containment Ventilation Isolation Door 504 Close Push Button & Light
67	Containment Ventilation Isolation Door 504 Open Push Button & Light
68	Containment Ventilation Isolation Door 505 Stop Push Button
69	Containment Ventilation Isolation Door 505 Close Push Button & Light
70	Containment Ventilation Isolation Door 505 Open Push Button & Light
71	Valve 552A Open Indication Light
72	Valve 552A Closed Indication Light
73	Valve 552B Open Indication Light
74	Valve 552B Closed Indication Light
75	Valve 552B Control Switch - 2 Position - "Open-Normal"
76	Valve 527D Open Indication Light
77	Valve 527D Closed Indication Light
78	Valve 527D Control Switch - 2 Position - "Open-Normal"
79	Valve Control Switches[b] - 2 Position - "Auto-Man"
80	Valve Control Switches[c] - 2 Position - "Open-Close"
81	Pump Control Switches[d] - 2 Position - "Off-On"
82	Cooling Tower Fan Control Switches - 3 Position - "Fast-Off-Slow"
83	Valve 547 Position Indication Light
84	Heavy Equipment Entry (Door 101) "Door Ajar" Indication Light
85	Primary Coolant T_h - T_c Recorder - 2 Pen
86	Primary Coolant System Temperature Controller (S-1)

TABLE 7-3 (cont.)
INSTRUMENT PANEL INSTRUMENTATION

No.[a]	Function
87	Pool Coolant T_h - T_c Recorder - 2 Pen
88	Pool Coolant System Temperature Controller (S-2)
89	Primary Coolant System Flow Recorder - 2 Pen
90	Pool Coolant System Flow Recorder - 2 Pen
91	Primary & Pool Coolant Demineralizer Flow Recorder - 2 Pen
92	Closed Circuit Television Monitor
93	Pool Coolant System Differential Temperature Meter
94	Primary Coolant System Differential Temperature Meter
95	Pressurizer Water Level Indication Meter
96	In-Pool Heat Exchanger Differential Temperature Meter
97	Reactor Pool Reflector Region Differential Pressure Meter - PT 917
98	Reactor Core Outlet Pressure Meter - PT 944A
99	Reactor Core Outlet Pressure Meter - PT 944B
100	Primary Coolant HX503A Differential Pressure Meter - DPS 928A
101	Primary Coolant HX503B Differential Pressure Meter - DPS 928B
102	Reactor Core Differential Pressure Meter - DPS 929
103	Valve 16A Closed Indication Light
104	Valve 16A Open Indication Light
105	Valve 16B Closed Indication Light
106	Valve 16B Open Indication Light
107	Fan Failure Alarm Panel
108	Secondary Coolant System Recorder - 3 Pen (Temperatures & Flow)
109	Secondary Coolant System High Temperature Alarm Light
110	Drain Collection System Control Panel
111	Digital Temperature Readout (TE 980/990)
112	Emergency Diesel Generator Alarm Panel
113	Hot Cell Isolation Valve Position Indication & Remote Operator
114	Reactor Power Calculator
115	Reactor Safety System Monitoring Circuit
116	Pool Coolant Flow Bypass Switch (2S40) - 4 Position
117	Primary Coolant Flow Bypass Switch (2S41) - 4 Position
118	Primary Coolant System Conductivity Meter - Demineralizer Inlet
119	Primary Coolant System Conductivity Meter - Demineralizer Outlet
120	Pool Coolant System Conductivity Meter - Demineralizer Inlet
121	Pool Coolant System Conductivity Meter - Demineralizer Outlet
122	Fire Main Low Pressure Alarm Light
123	Door Open Alarm (Room 114, Cooling Tower, Demineralizer Area)
124	Alarm Cutout Switches - Door, Firemain, Secondary pH
125	Domestic Cold Water (DCW) Low Pressure Alarm Light
126	Fire Main Low Pressure Alarm Panel
127	Pneumatic Tube System Irradiation Counter
128	Off-Gas Radiation Monitor Recorder - 3 Pen
129	Off-Gas Radiation Monitor Flow Alarms & Cutout Switch

TABLE 7-3 (cont.)
INSTRUMENT PANEL INSTRUMENTATION

No.[a]	Function
130	Off-Gas Radiation Monitor Recorder - 3 pen
131	Secondary Coolant System Flow Transmitter Selector Switch - 2 Position - "912P-912Q"
132	Emergency Diesel Generator Elapsed Time Run Meter
133	Radioactive Liquid Waste System Alarm Panel
134	Fire Protection System Alarm Panel

[a]The number on this table refers to the position of the control equipment and display instruments on the instrument panel (Fig. 7.3).
[b]Auto/Manual control switches for the following valves: V546A/B, V507A/B, V509, V545, V526, V527A, and V527B.
[c]Open/Close control switches and indication lights for the following valves: V546A/B, V507A/B, V509, V543A/B, V527E, V527F, V545, V526, V527A, V527B, and V527C.
[d]Off/On control switches and indication lights for the following pumps: P501A/B, P508A/B, P513A/B, and P533 (Off/Auto).
[d]Off/On control switches, speed raise/lower push buttons, and % load/speed meters for the following pumps: SP-1, SP-2, and SP-3.
[e]Off/On control switches and indication lights for the following fans: CTF-1, CTF-2, and CTF-3.

TABLE 7-4
ANNUNCIATOR ALARM SEQUENCE

Condition	Illumination	Audible Signal
Normal	Off	Off
Alarm Input	Flashing	On
Alarm Input Acknowledged	Steady-On	Off
Return to Normal	Dim Flashing	On
Reset	Off	Off
Operational Test	Flashing	On

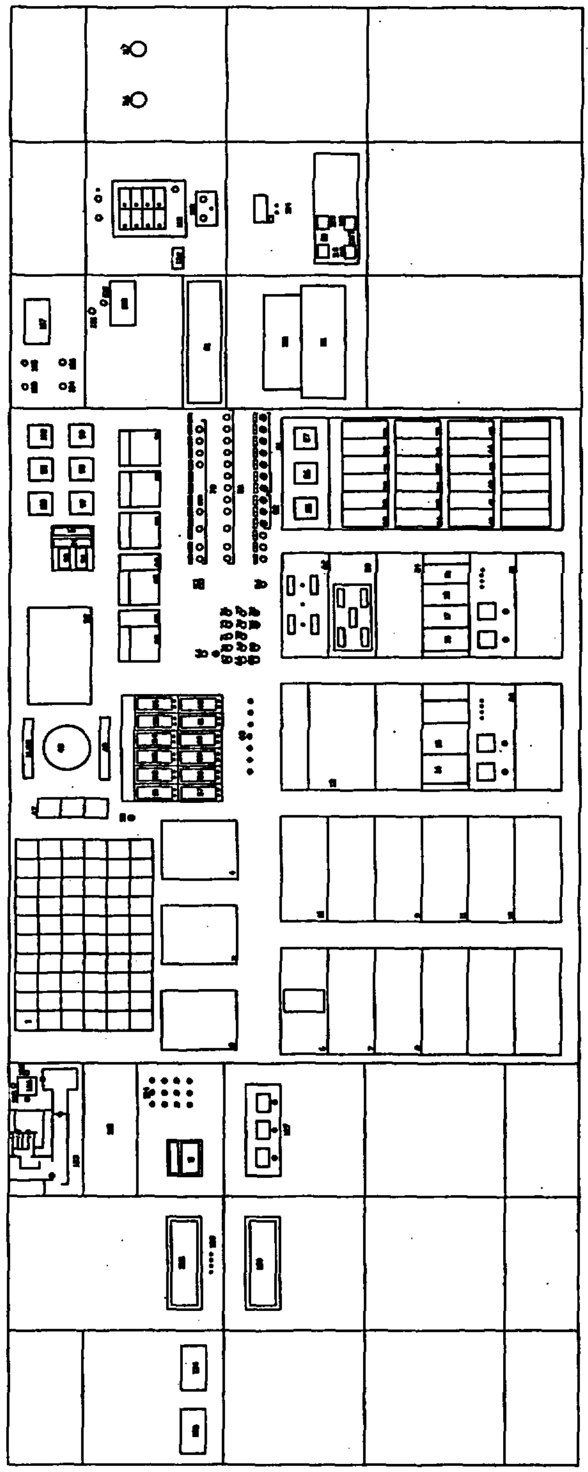


FIGURE 7.3
INSTRUMENT PANEL LAYOUT

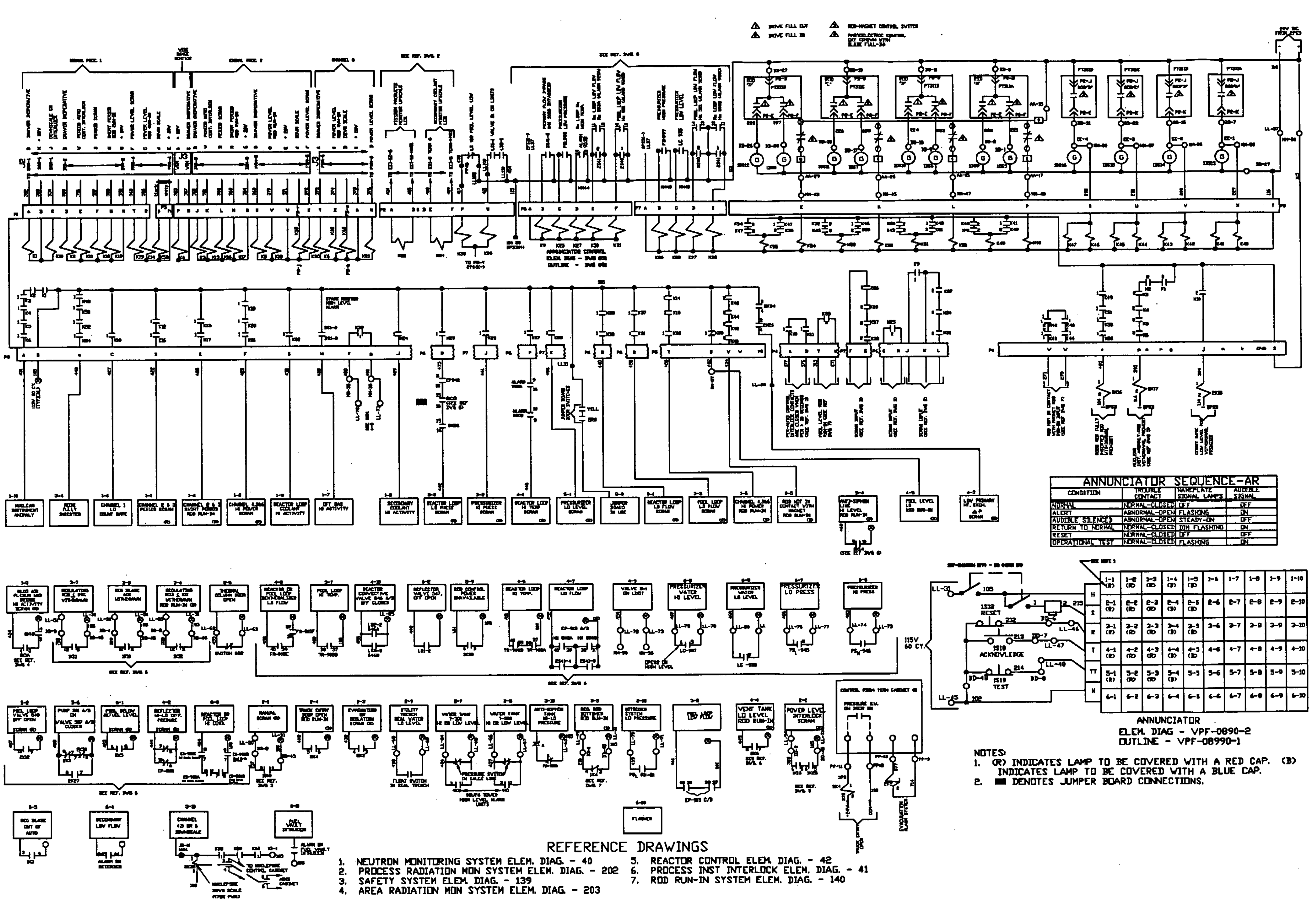


FIGURE 7.4
ANNUNCIATOR CONTROL

TABLE 7-5
ANNUNCIATOR ALARM CONDITIONS

1	Any Reactor Scram or Rod Run-In Condition (25)
2	Channel 1 Low Count Rate
3	Off-Gas High Activity
4	Secondary Coolant System High Activity
5	Primary Coolant System High Activity
6	Nuclear Instrument Anomaly
7	Thermal Column Door Open
8	Reactor Plant Make-Up Water Tank (T-301) High or Low Level
9	Reactor Plant Make-Up Water Tank (T-300) High or Low Level
10	Utility Entry Water Seal Low Level
11	Nitrogen System Low Pressure
12	Control Rod Drive Mechanisms Fully Inserted
13	Regulating Rod Withdrawn $\leq$ 20%
14	Regulating Rod Withdrawn 60%
15	Jumper Board In Use
16	Anti-Siphon System High or Low Pressure
17	Primary Coolant System High Temperature (T_h or T_c)
18	Primary Coolant System Low Flow
19	Primary or Pool Coolant Demineralizer Low Flow
20	Valve S-1 On Limit (Primary Coolant Temperature Control)
21	In-Pool Heat Exchanger Isolation Valve 546A or 546B Not Fully Closed
22	Regulating Rod Out of Automatic Control
23	Primary Coolant Circulation Pump 501A or 501B On With Primary Coolant Isolation Valve 507A or 507B Closed
24	Pool Coolant System High Temperature (T_h)
25	Pool Coolant System Low Flow
26	Rod Control Power Unavailable
27	Nuclear Instrumentation Channel 4, 5, or 6 Downscale
28	Reflector Plenum Natural Convection Valve 547 Not Fully Open
29	Fuel Vault Intrusion Alarm
30	Secondary Coolant System Low Flow
31	Primary or Pool Coolant System High Conductivity
32	Pressurizer High Pressure
33	Pressurizer Low Pressure
34	Pressurizer High Water Level
35	Pressurizer Low Water Level

7.4. Nuclear Instrumentation

7.4.1 Introduction

The Nuclear Instrumentation (NI) System consists of three (3) Neutron Flux Monitors, a Wide Range Neutron Flux Monitor, and a Multiscaler.

The Neutron Flux Monitors provide reliable neutron flux measurement from reactor shutdown through full power level. This is a fission chamber based system designed to measure neutron flux with the detector in a high gamma radiation and electrical noise environment. The system provides input signals to the reactor safety and rod run-in systems on sensing a high power level or short reactor period condition and in the case of the Source Range Channel, to the rod control system.

The Wide Range Neutron Flux Monitor provides an input signal to the rod control system for automatic control of reactor power level. This is a compensated ion chamber (CIC) based system providing neutron flux measurement from reactor shutdown through full power level.

The Multiscaler provides the reactor operator with a continuous indication of subcritical neutron source multiplication during a reactor startup. This information is used to validate the predetermined estimated critical position (ECP) during the approach to criticality.

7.4.2 Neutron Flux Monitors

The Neutron Flux Monitors provide the following six channels of neutron flux measurement from source level (shutdown) through 100% of full power operation: one source range, two intermediate ranges, and three power ranges. Each neutron flux monitor consists of the following main components: a detector assembly, an amplifier assembly, and a rack mount signal processor. The neutron flux monitor operates from the source (pulse counting mode) range through the Campbelling [mean-square-voltage (MSV) mode] range. The MSV is proportional to the average rate of neutron pulses and is not dependent on the pulses being individually identifiable. Individual pulses from the fission chamber are counted over the range of six decades from 10^{-1} to 10^{5} counts per second (cps). The sensitivity is established by the detector and a pulse amplitude discriminator in the amplifier assembly at 1 cps/nv (thermal). The MSV signal starts below the upper two decades of the source range to provide sufficient indication overlap for the intermediate range. Combining the pulse counting and MSV modes, a continuous output signal is produced over the range of 10^{-8}% to 200% of reactor power.

The detector assembly for these monitors consists of a gas-filled fission chamber with two aluminum electrodes electroplated with uranium. The output signal from the detector is composed of a series of charge pulses. The pulses result from the alpha decay of the uranium, from gamma photon interaction with material in the electrodes, and from the fissioning of the uranium atoms when a neutron is absorbed. The unwanted pulse signals generated by the alpha decay of the uranium and the gamma photon interactions are eliminated by amplitude discrimination in the amplifier assembly. The actual neutron flux impinging upon the detector is leakage flux and not reactor core flux; consequently the monitoring equipment is calibrated to present a readout of actual flux level from a proportional input signal. The detector assembly is located in a fixed position near the bottom of a water tight enclosure (drywell). The drywell is rigidly mounted within the reactor pool outside the reflector region, approximately at the elevation of the reactor core centerline.

The amplifier assembly contains modular plug-in subassemblies for a low-voltage power supply, a detector excitation high-voltage power supply, and the electronic circuitry which conditions the detector output signal prior to its transmission to the signal processor. The power supplies are incorporated into the amplifier assembly to minimize the number of interconnecting wires, and also to minimize the hazard, the voltage drop, or the noise pickup associated with long power supply cables. The signal conditioning electronics provide amplification, pulse shaping, discrimination against alpha, gamma, and electronic noise, and other signal conditioning. A self-diagnostic operation monitoring circuit provides a continuous analysis of the integrity of the detector, the cables, and the power supplies. A malfunction (inoperative) signal generated by this circuit will cause the following actions to occur:

1. Initiation of the "Nuclear Instrument Anomaly" annunciator alarm; and
2. Initiation of a reactor scram.

The inoperative signal indicates that one or more of the following conditions exists:

a. The ±15-VDC supply has failed;
b. Magnitude of the detector excitation high voltage supply is low;
c. Detector signal is low; or
d. One of the front panel switches (S1 thru S5) on the signal processor drawer has been actuated.

The amplifier is housed in a National Electrical Manufacturers Association (NEMA) steel enclosure mounted to the west side of the biological shield at the mezzanine level.

The signal processor provides the circuitry to further process the detector output signal into signals which are a measure of the logarithm of count rate, the rate-of-change of count rate, the logarithm of reactor power and the rate-of-change of reactor power. Each signal processor is capable of providing one source range, an intermediate range, and one power range channel. However, only six of the available nine channels are used. A dual bi-stable trip board provides input signals to the Reactor Safety and Rod Run-In Systems on detection of a high power level or short reactor period condition.

7.4.2.1 Source Range - Channel 1

The Source Range (start-up) Channel of the NI monitors count rate over a six-decade range of reactor power from 10^{-1} to 10^{5} cps. Source range levels and periods are displayed on each signal processor by linear bar graphs and on analog meters located on the reactor control console. A strip-chart, one-pen recorder mounted on the instrument panel records Channel 1 source range level.

If source range level indication decreases to less than 20 cps, a trip circuit will de-energize relay K10 and cause the following actions to occur:

a. Initiation of the "Channel 1 Lo Count Rate" annunciator alarm; and
b. Initiation of an input signal (relay 2K18 de-energizes) to the Rod Withdrawal Prohibit Circuit (Section 7.5.3.1).

The discriminated output signal of the source range channel is also transmitted to a multiscaler mounted on the instrument panel. The multiscaler is used to monitor the shutdown core, presence of the neutron source, and the proper response to changes in reactivity when the control rods are initially withdrawn. The multiscaler is described in greater detail in Section 7.4.4.

7.4.2.2 Intermediate Range - Channels 2 & 3

The Intermediate Range Channels of the NI monitor count rate over a ten-decade range of reactor power from 10^{-8} to 200%. Intermediate range levels and periods are displayed on each signal processor by linear bar graphs and on analog meters located on the reactor control console. A strip-chart, two-pen recorder mounted on the instrument panel records Channel 2 and 3 intermediate range levels.

The Intermediate Range Channels also provide the following reactor protective and interlock functions:

a. Reactor Scram: A reactor scram and a "Channel 2 & 3 Period Scram" annunciator alarm initiation if reactor period decreases below a predetermined value;

b. Rod Run-In: A rod run-in and a "Channel 2 & 3 Short Period Rod Run-In" annunciator alarm initiation if reactor period decreases below a predetermined value; and

c. Auto Control Prohibit: The reactor not being placed in automatic control unless reactor period is greater than 35 seconds (Section 7.5.4).

7.4.2.3 Power Range - Channels 4, 5, & 6

The Power Range Channels of the NI provide an indication of reactor power over a range from 0 to 125%. Power range levels are displayed on each signal processor by linear bar graphs and on analog meters located on the reactor control console. The power level signal is generated by squaring the mean-square-voltage signal. The signal is then displayed on a 0 to 125% linear scale. A strip-chart, three-pen recorder mounted on the instrument panel records Channel 4, 5, and 6 power range levels.

The Power Range Channels also provide the following reactor protective and alarm functions:

a. Reactor Scram: A reactor scram and a "Channel 4, 5 & 6 Hi Power Scram" annunciator alarm initiation if reactor power increases above a pre-determined value;

b. Rod Run-In: A rod run-in and a "Channel 4, 5 & 6 Hi Power Rod Run-In" annunciator alarm initiation if reactor power increases above a pre-determined value; and

c. Alarm: A "Channel 4, 5, or 6 Downscale" annunciator alarm initiation if reactor power decreases below a predetermined value once full power is obtained.

7.4.3 Wide Range Neutron Flux Monitor

The Wide Range Neutron Flux Monitor provides continuous neutron flux monitoring over a ten decade range, from source level (shutdown) to 125% of full rated power, in addition to providing an input signal to the Rod Control System for automatic control of reactor power level. Automatic control of the reactor is described in greater detail in Section 7.5.4. The Wide Range Neutron Flux Monitor is comprised of the following three main sections: flux level measuring, trip unit, and drawer power.

The flux level measuring section receives a current input from a compensated ion chamber (CIC) detector and converts it into a proportional DC voltage. This voltage is applied to analog meters located on the instrument drawer and the reactor control console, and to the trip units. The two main components of the flux level measuring section are the amplifier unit and the range switch (1S2). The range switch selects feedback elements of the amplifier unit, which have a value commensurate with the desired full-scale deflection input current. Thus, in essence, the range switch selects a different value feedback element each time it is set to a new position. The eighteen (18) -position range switch provides greater accuracy over the ten decades of input current. The first nine decades are divided into two ranges: a 0 to 125% range and a 0 to 40% range. This allows the scale to be expanded in the lower portion of a current decade, providing more accurate indication. The CIC detector is located in a fixed position near the bottom of a water tight enclosure (drywell) that is designed to allow both vertical and radial adjustment, if necessary. The drywell is mounted within the reactor pool outside the reflector region, approximately at the elevation of the reactor core centerline.

The trip unit section consists of two separate duel trip units with each trip unit containing two separate independent trip circuits. Presently, only one of the four trip circuits is used for alarm annunciation. If one or more of the following conditions are met, an instrument drawer inoperative trip will occur and a "Nuclear Instrument Anomaly" annunciator alarm will initiate:

a. A module is removed from the instrument drawer;
b. Drawer selector switch S1 is placed in any position other than OPERATE; or
c. The high voltage power supply output drops below a predetermined minimum voltage.

The remaining three trip circuits, one downscale and two upscale, can all be independently set to trip at different input voltages, if desired. There are no safety system trips associated with the Wide Range Neutron Flux Monitor, therefore, the MURR Reactor Safety System satisfies the intent of IEEE-279 and Criterion 24 of Appendix A, 10 CFR 50, with regard to control and protection system interaction. No instrument channel provides both safety and control functions.

The drawer power section provides regulated ±15-volt outputs for use throughout the drawer, and also contains two separate independent high voltage power supplies which supply the positive and negative high voltage potentials used as polarizing voltages for ion chambers.

7.4.4 Multiscaler

The Multiscaler is a versatile and highly accurate timer/multiple counter which provides a digital readout of the count rate generated by the Source Range Channel to assist the operator in reliably predicting criticality by the 1/M method during a reactor startup. It incorporates three 85-MHZ counting channels and a crystal controller timebase. The timer can be used as a fourth 85-MHZ counting channel. The data from all channels are shown on two seven-decade LED displays. The selection of the channels to be displayed is made via two pushbutton indicators.

The two seven-decade LED displays indicate either counts or time. The displays are switch-selectable between pairs of channels. The upper display shows Time or Channel A, and the lower display shows Channel B or C. Normally 9,999,999 (1×10^7-1) counts can be displayed, however, counting capacity can be expanded to 1×10^{28} -1 by use of its overflow capability.

7.4.5 Limiting Conditions for Operation

The limiting conditions for operation of the NI and their bases are discussed in the Technical Specifications. The objective of these conditions is to ensure that sufficient reliable information is presented to the operator to enable safe operation of the reactor.

7.4.6 Surveillance

The NI is periodically tested for operability as stated in the Technical Specifications. The objective of this specification is to reasonably assure proper operation of the system.

7.5 Rod Control System

7.5.1 Introduction

The reactivity of the reactor is controlled by five neutron absorbing control blades. Each control blade is attached to a control rod drive mechanism by means of a support and guide extension (offset mechanism). Four of the control blades, referred to as the shim blades, are used for coarse adjustments to the neutron density of the reactor core. The fifth control blade is a regulating blade. The low reactivity worth of this blade allows for very fine adjustments in the neutron density in order to maintain the reactor at the desired power level. The nominal speed of the shim blades is one inch per minute in the outward direction and two inches per minute in the inward direction. Nominal speed of the regulating blade is 40 inches per minute in both the inward and outward directions. The speed of the control blades cannot be adjusted without physically altering the system.

The reactor is operated from the control console in either of two control modes: manual or automatic. Manual control is used for reactor start-up, changes in power level, and steady-state operation for short periods of time. Automatic control is selected only after a minimum power level has been attained and is used for long term steady-state operation.

Control blade movements, interlocks and bypasses, and control modes are managed by the Rod Control System (Figure 7.5). The Rod Control System is a relay and switch logic system used to prohibit accidental or incorrect operation which could result in an unsafe condition. Rod position indication is determined and displayed in the reactor control room by a system of microcomputer controlled instruments. The system consists of a central control chassis and a remote display unit which provides the reactor operator with control rod height in inches.

Also part of the Rod Control System is the Rod Run-In System which initiates the automatic insertion of the control blades at a controlled rate should a monitored parameter exceed a predetermined value.

7.5.2 Control Rods

The shim blades are constructed of formed boral plate which is, by weight, nominally 50% boron carbide and 50% aluminum. The boron carbide-aluminum mixture is clad with aluminum alloy 1100. The active length of the neutron absorbing material is 34 inches (86.4 cm), and the overall blade length is approximately 40 inches (101.6 cm). The upper 6 inches (15.2 cm) of the shim blade is a mounting plate formed to the shape of the blade. The regulating blade is constructed of stainless steel with an overall nominal length of 30 inches (76.2 cm).

The four shim blades are actuated by electro mechanical control rod drive mechanisms that position, hold, and scram each shim blade. The control rod drive mechanisms are mounted on the upper (operating) bridge over the reactor pool surface. Each control rod drive mechanism consists of a 0.02-HP, 115-volt, one-amp, single-phase, 60-cycle motor connected to a ball-bearing lead screw assembly through a reduction gear box and overload clutch. The lead screw assembly converts the rotation of the worm gear reducer in the drive motor to the linear motion of the control blades. The motion is transmitted through ball bearings, which limits wear on the rotating member and produces a uniform rotation. The overload clutch limits the lifting force of the drive mechanism to approximately 55 to 60 pounds in the event of an abnormal alignment of the control blade in the core, thereby providing overload protection to the drive mechanism. A ball nut coupled directly to the drive tube is driven inward or outward by the lead screw. At a point above the reactor pool water level, an electromagnet, connected to the bottom end of the drive tube, engages a cadmium-plated carbon steel anvil that is attached to the end of the lift-rod assembly. The lift-rod assembly positions a shim blade through a support and guiding (offset) mechanism mounted on a bracket attached to the reflector tank. Cam-operated micro switches mounted on each control rod drive mechanism stop the drive motor at the upper and lower limits of travel. The following indications are displayed on the reactor control console for each shim blade:

1. "Power On" - Power is available to the electromagnets;
2. "Drive Full In"- Control rod drive mechanism is fully inserted;
3. "Drive Full Out" - Control rod drive mechanism is fully withdrawn;
4. "Magnet Engaged" - Electromagnet engaged to the anvil; and
5. "Rod Full In" - Shim blade is fully inserted.

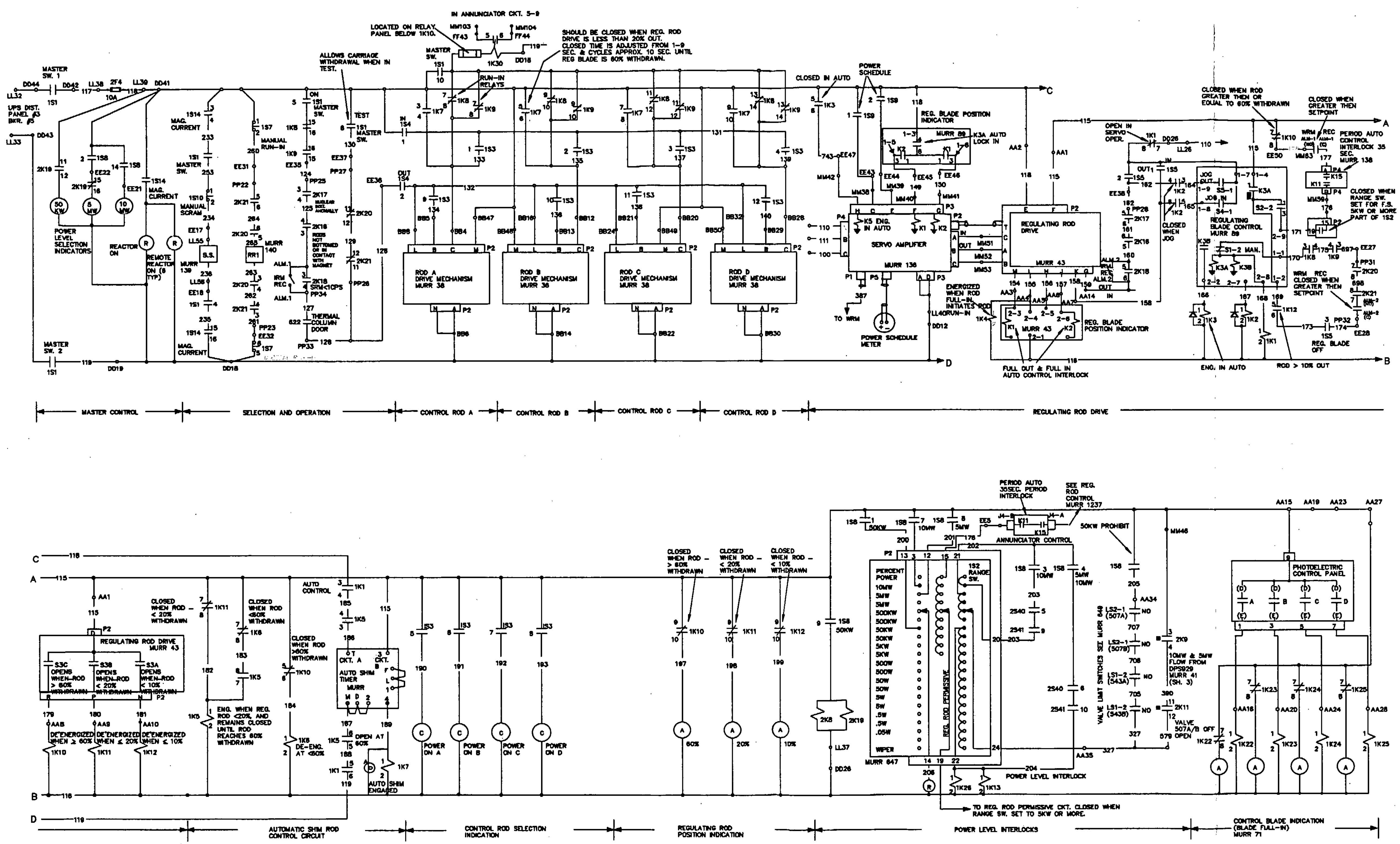
The shim blade can be withdrawn when the electromagnet is engaged with the anvil and is energized. When the reactor is scrammed, the electromagnet is de-energized, releasing the anvil, and allowing the shim blade and lift-rod assembly to drop. A dash pot assembly cushions the fall of the shim blade during the final 20% of travel.

The regulating blade is positioned by a control rod drive mechanism similar to the ones used for the shim blades, but with several differences. A scram signal does not insert the regulating blade, therefore, an electromagnet and anvil are not required. The lift-rod assembly is pinned directly to the drive tube. Also, the regulating blade is driven at a faster speed than the shim blades, requiring a different reduction gear box.

7.5.3 Manual Control

A three-position ("Off-Test-On") keylock Master Control Switch and a two-position ("Off-On") Magnet Current Switch located on the reactor control console controls power to the Rod Control System. The master control switch (1S1) and the magnet current switch (1S14) must both be in the "On" position to provide current to the shim blade electromagnets. The magnet current switch "On" position also illuminates the "REACTOR ON" lights throughout the reactor facility. Unauthorized operation of the Rod Control System is prevented by the keylock design.

The shim and regulating blades are withdrawn or inserted manually by three-position ("In-Normal-Out") switches located on the reactor control console. The switches are spring return to the



■ DENOTES JUMPER BOARD CONNECTIONS

FIGURE 7.5
ROD CONTROL SYSTEM

mid-position ("Normal") when released. A five-position ("A-B-C-D-Gang") selector switch enables the reactor operator to select the shim blades individually or as a group. The shim blade selector switch is designated 1S3 and the withdrawal-insertion switches for the shim and regulating blades are designated 1S4 and 1S5, respectively. Two push button switches located on the control console allow the regulating blade to be "jogged" inward and outward for fine adjustment of reactor power level in the manual control mode.

7.5.3.1 Rod Withdrawal Prohibit

The Rod Withdrawal Prohibit circuit prevents the control blades from being withdrawn unless the following control system logic conditions have been satisfied:

1. The Master Control Switch (1S1) in the "ON" position;

2. No Nuclear Instrument anomaly;

3. Shim rods bottomed and in contact with their electromagnets;

4. Source range level indication greater than 20 cps or intermediate range level recorder indication greater than 2 x 10^{-5}% power; and

5. Thermal Column door closed.

When all of the foregoing conditions have been satisfied, the Reactor Safety System and Rod Run-In System Trip Actuator Amplifiers (TAAs) may be reset and the control blades withdrawn. There is no interlock or prohibit circuit which prevents the control blades from being driven inward in either the manual or automatic control modes.

If a control rod loses contact with its electromagnet while being withdrawn, all shim blades must be fully inserted before they can be withdrawn again. The Rod Withdrawal Prohibit circuit may be bypassed (rod drive test permissive) for testing if the following conditions are met:

1. The Master Control Switch (1S1) is in the "Test" position; and

2. Power to the control rod electromagnets is secured (relays 2K20 and 2K21 are de-energized).

7.5.4 Automatic Control

The controlling parameter for automatic operation of the Rod Control System is the neutron flux monitored by a compensated ion chamber (CIC). The neutron flux is essentially proportional to the reactor power level. The pulse signal generated by the CIC is routed to the wide range neutron flux monitor located in the reactor control room. An output signal produced by the wide range signal processor drawer is then routed to a solid state servo amplifier. This output signal voltage is proportional to that of the input signal generated by the CIC. The wide range neutron flux monitor is described in greater detail in Section 7.4.3.

The servo amplifier is mounted on the control room instrument panel. The servo amplifier automatically controls the power level of the reactor by comparing the output signal from the wide range signal processor drawer with the power schedule setting. The power schedule set point is adjusted by a spring return switch (1S9) and displayed on a meter mounted on the reactor control console. The output from the signal processor drawer is a 0 to +10 VDC signal, corresponding to 0 to 125% reactor power level (+8.0 VDC = 100% power). The output from the power schedule potentiometer is a 0 to -10 VDC signal depending on the setting. This ± voltage window represents the ± control sensitivity of the servo amplifier. If a mismatch does exist, a positive or negative output signal is generated and sent to the relay-controlled drive motor of the regulating rod drive mechanism, which repositions the regulating blade, stepwise, in a direction which minimizes the discrepancy between the power schedule setting and the actual power level.

The reactor may be placed in automatic control after the following four conditions have been met:

1. Reactor period as indicated by Intermediate Range Channels 2 and 3 greater than 35 seconds;
2. Indicated reactor power level greater than the "auto control prohibit" set point on the wide range neutron flux monitor recorder;
3. The regulating blade position greater than 60% withdrawn; and
4. The Range Selector Switch (1S2) in the 5-kW red scale position or above.

After the above four conditions have been met, reactor control is placed in the automatic mode by depressing a push button switch (S2) located on the reactor control console. The regulating blade will then automatically maintain reactor power level as determined by the power schedule setting.

Once the reactor is placed in automatic control, the Rod Control System will remain in this mode unless any one of the following actions occur:

a. Control mode transferred to manual (push button S1-2 depressed);

b. Regulating rod control switch (1S5) actuated;

c. Regulating rod inserted to the 10% withdrawn position;

d. A reactor scram or rod run-in; or

e. Indicated reactor power reaching a level less than the "auto control prohibit" set point on the wide range neutron flux monitor recorder.

7.5.4.1 Automatic Shim Control

To compensate for cyclic poison effects, the regulating rod, in response to automatic control, may traverse its full stroke, thereby requiring the reactor operator to adjust the shim rods so that the regulating rod can be repositioned. To assist the operator in making the shim rod adjustments, the Automatic Shim Control circuit, which is activated when the regulating rod is inserted to its 20% withdrawn position, will automatically insert the shim rods, stepwise, a sufficient amount to drive the regulating rod to its 60% withdrawn position.

However, due to the difference in the reactivity worth and travel speed between the shim rods and the regulating rod, an adjustable interrupter circuit limits the travel time of the shim rods sufficiently to allow the regulating rod to maintain the power level of the reactor at a constant value during the shimming operation. When the regulating rod has only a limited amount of travel left (20% withdrawn), the Automatic Shim Control circuit will activate and insert the shim rods for a preset time and stop. After a "wait" period has elapsed, the cycle is repeated. The alternate "insert and wait" cycle will continue until the regulating rod, which remains in automatic control, has reached the 60% withdrawn position. At this point, the Automatic Shim Control circuit will deactivate until the regulating rod has again inserted to the 20% withdrawn position. The "insert" and "insert plus wait" periods are independently adjustable to facilitate setting an optimum automatic shimming cycle.

The Automatic Shim Control circuit is designed to only insert the shim blades; it cannot withdraw the shim blades to compensate for low neutron activity.

7.5.5 Rod Run-In System

The Rod Run-In System is designed to initiate the automatic insertion of the control blades at a controlled rate should a monitored parameter exceed a predetermined value. This system is not part of the Reactor Safety System, however, it does provide a protective function by introducing shim blade insertion to terminate a transient prior to actuating a Reactor Safety System set point trip. The Rod Run-In System also inserts the control rod drive mechanisms in the event of a scram condition. A rod run-in may be initiated manually by depressing a push button (1S7) on the reactor control console. The rod run-ins are listed in Table 7-6.

The Rod Run-In System is comprised of one Non-Coincidence Logic Unit (NCLU) and one Trip Actuator Amplifier (TAA). The NCLU receives input signals (12 or 24 volts) from the neutron flux monitors, process instruments, and safety interlocks (Table 7-7) and provides an output signal to control the TAA. Should an input signal from any one of the four devices in Board "A" (¼) or any one of the three devices in Board "B" (⅓) interrupt the logic unit, the TAA will actuate and interrupt current to the rod run-in relays (1K8 and 1K9), causing the control rod drive mechanisms to insert the control blades in the inward direction at a nominal speed of 2 inches per minute. The Rod Run-In System and all of the conditions which will initiate a rod run-in are shown in Figure 7.6.

TABLE 7-6
ROD RUN-INS

1. Manual	6. Regulating Rod ≤ 10% Withdrawn
2. Channel 2 & 3 Short Period	7. Regulating Rod Bottomed
3. Reactor Pool Level Low	8. Truck Entry Door (Door 101) Open
4. Vent Tank Low Level	9. Channel 4, 5, & 6 High Power
5. Anti-Siphon System High Level	10. Rod Not In Contact With Magnet

Once tripped, the TAA will remain in this condition until normal input voltage is restored, the rod run-in input signal is cleared, and the unit is reset. The TAA is reset by depressing the "Rod Run-In Reset" push button (1S13) on the reactor control console. If a reactor scram has also occurred, the reactor safety system TAAs must be reset prior to resetting the rod run-in TAA (relays 2K20 and 2K21 must be energized).

The rod run-in NCLU and TAA are similar to the ones used in the Reactor Safety System. Their operation is described in Sections 7.7.2.1 and 7.7.2.2.

TABLE 7-7
ROD RUN-IN NON-COINCIDENCE LOGIC UNIT SIGNAL INPUTS

Rod Run-In Non-Coincidence Logic Unit (1/N)			
Non-Coincidence Logic Board "A" (1/4)		Non-Coincidence Logic Board "B" (1/3)	
Input Signal	Voltage	Input Signal	Voltage
NI Channel 2	+12 VDC	NI Channel 3	+12 VDC
NI Channel 4	+12 VDC	NI Channel 5	+12 VDC
NI Channel 6	+12 VDC	Process Instrumentation	+24 VDC
Process Instrumentation	+24 VDC		

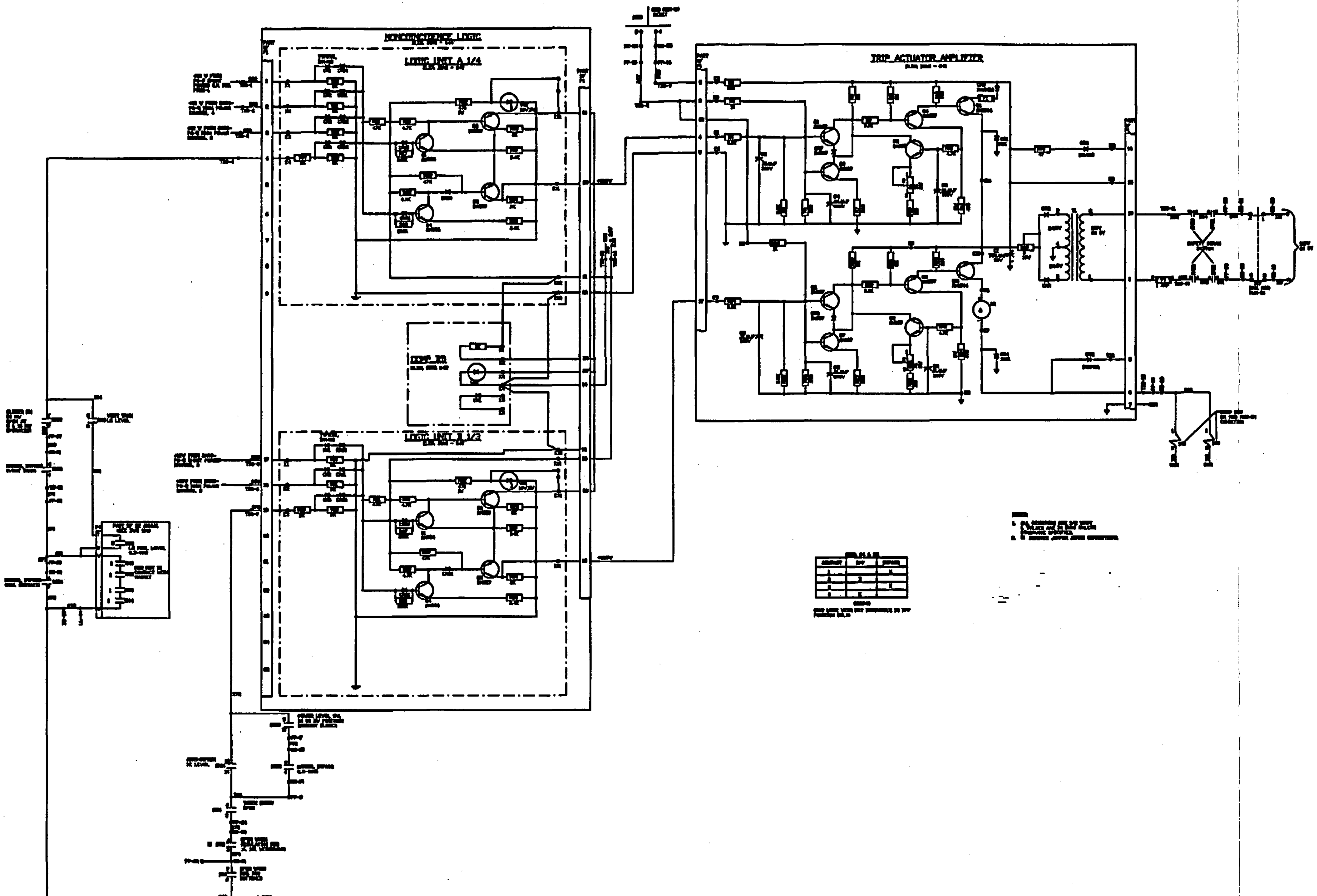


FIGURE 7.6
ROD RUN-IN SYSTEM

7.5.5.1 Limiting Conditions for Operation

The limiting conditions for operation of the Rod Run-In System and their bases are discussed in the Technical Specifications. The regulating blade rod run-ins will assure termination of a transient which, in automatic operation, is causing a rapid insertion of the regulating blade.

7.5.5.2 Surveillance

The Rod Run-In System is periodically tested for operability as stated in the Technical Specifications. The objective of this specification is to ensure proper operation of the system.

7.5.6 Control Rod Position Indication

Control rod position indication is provided by a system of microcomputer controlled instruments located in the reactor control room. The Rod Position Indication (RPI) system is designed to provide the following:

- Continuous precision rod position indication;
- Security against tampering or inadvertent alteration by use of keylock and password control; and
- Automatic self-test and internal diagnostics.

The RPI system consists of the following major components: five encoder transducers, one mounted on each control rod drive mechanism; a chassis with installed hardware; and a remote Operator Display Assembly (ODA).

Analog signals from the transducer encoders are converted into digital form at the RPI chassis. The chassis, mounted in the instrument panel, contains redundant AC to DC low voltage power supplies, a card file, and a motherboard holding the following circuit components: one analog module, five encoder modules, one display control module, and one computer module. A front panel assembly includes an electro-luminescent (EL) display, a keylock switch, and operating push buttons immediately below the display screen which perform the functions as specified by the screen display menus.

The ODA consists of an EL display and associated display driver and logic circuits. The ODA is mounted in the reactor control console and displays information received from the chassis.

The operating mode of the RPI system is controlled by a panel keylock switch located on the chassis. Two positions are available: OPER (operate) and INOP (inoperative). When the keylock switch is in the OPER position, the system is in the OPERATE mode and provides the following functions.

1. Control Rod Position Indication - The positions of all five control rods are displayed as inches (to two decimal places) from fully inserted. The range of indicated position is from -0.25 to 26.25 inches. This is the default display in the OPERATE mode.

2. Self-Test Status - This function displays the status of the self-test being performed on each of the RPI modules. The self-test continuously monitors the internal power supplies and compares them to the expected values. Any out-of-tolerance condition is considered a fault. This self-testing is automatically executed. The display indicates the status of each module after it is tested, and the current test cycle.

3. Input Status - The low voltage power supply output voltages and logic power bus voltages are displayed.

4. Display Off - This function provides the ability to blank the display during periods when the display is not in use, however all functional controller operations continue. The display is restored by the touch of any key.

When the keylock switch is placed in the INOP position, the system will be in the INOP-CAL mode. In this mode, the operator may request displays found in the OPER mode, initiate calibration of the system, test the display and keyboard, or run diagnostics.

7.6 Process Instrumentation and Control System

7.6.1 Introduction

The Process Instrumentation and Control System monitors, displays, and controls the following reactor plant parameters: temperature, pressure, flow, and pressurizer liquid level. Other parameters which are simply monitored and/or displayed are: primary and pool coolant conductivity, reactor plant make-up water storage tank levels, utility entry water seal level, secondary coolant flow, and nitrogen system pressure. The system also provides control interlocks for the primary and pool coolant circulation pumps and automatic isolation valves, as well as input signals to the Reactor Safety System. Generally, the Process Instrumentation and Control System permits manual and automatic operation within certain design limitations, e. g., in some instances, manual opening and closing of a particular valve cannot be accomplished because of valve and reactor control interlocks. The Process Instrumentation and Control System is shown in Figures 7.7, 7.8, and 7.9.

7.6.2 Temperature Measurement and Control

7.6.2.1 Primary Coolant System

Primary coolant system temperature is measured by Resistance Temperature Detectors (RTDs) at the following locations with the indicated Temperature Elements (TEs):

(a) Reactor Inlet (T_c) - TE 901A;

(b) Reactor Outlet (T_h) - TE 901B;

(c) Primary Coolant Heat Exchanger Outlet (HX503A) - TE 980A;

(d) Primary Coolant Heat Exchanger Outlet (HX503B) - TE 980B;

(e) In-Pool Heat Exchanger (HX505) Inlet - TE 950A; and

(f) In-Pool Heat Exchanger (HX505) Outlet - TE 950B.

Reactor inlet and outlet temperatures are recorded on a strip-chart, two-pen recorder mounted on the instrument panel. The low level DC millivolt output signal produced by the RTD is converted to a proportional DC milliampere signal (10 to 50 mA) by an RTD Transmitter. The output signal from the RTD Transmitter for TE 901A is directed to an Adder-Subtractor Module (ASM) and two-pen recorder. The output signal from the RTD Transmitter for TE 901B is directed to the ASM, the recorder, and a Dual Alarm Unit. The ASM provides a differential temperature input signal to the automatic power calculator mounted on the instrument panel. The power calculator uses an analog computer circuit to provide the reactor operator with a digital indication of reactor power level in megawatts (MW). In addition to providing temperature indication, the recorder initiates a "Reactor Loop Hi Temp" annunciator alarm if either reactor inlet or outlet temperature exceeds 120% of the normal operating value. If reactor outlet temperature exceeds 125% of the normal value, a reactor scram and a "Reactor Loop Hi Temp Scram" annunciator alarm is initiated. The Dual Alarm Unit opens a contact (K27-2) in the process input string to E4A of the Reactor Safety System NCLUs, thereby interrupting power to the control blade electromagnets.

The output signals from the primary coolant heat exchanger RTDs (TE 980A and TE 980B) are directed to Alarm Meter Units which initiate a reactor scram and a "Reactor Loop Hi Temp Scram" annunciator alarm if either primary coolant heat exchanger outlet temperature exceeds 125% of the normal operating value. The scram occurs when the Alarm Meter Units open contacts in the process input string to E3B of the Reactor Safety System NCLUs. The Alarm Meter Units also provide primary coolant heat exchanger outlet temperature indication to the instrument panel.

The reactor outlet high temperature scram provides a backup to the primary coolant heat exchanger outlet high temperature scram.

The temperature differential across the in-pool heat exchanger (decay heat removal system) is displayed on a meter mounted on the instrument panel. The output signals from TE 950A and TE 950B are directed to an RTD Transmitter which converts the DC millivolt signal produced by the RTDs to a proportional DC milliampere signal for use by the differential temperature meter.

Primary coolant system temperature is controlled by controlling reactor inlet (primary coolant heat exchanger outlet) temperature. The amount of secondary coolant passing through the heat exchangers is regulated by an automatic temperature control valve (S-1) located in a bypass line adjacent to the heat exchanger system. This valve is an 8-inch electro-hydraulic butterfly valve which responds to primary coolant outlet temperature from the heat exchangers, and maintains a constant cold leg temperature in the primary coolant system. The controller unit is installed adjacent to the reactor inlet and outlet temperature recorder and provides manual control of S-1 if desired. If valve S-1 reaches either its 80% open or 80% shut position, a "Valve S-1 On Limit" annunciator alarm is initiated informing the operator that minimal automatic temperature control remains available.

7.6.2.2 Pool Coolant System

Pool coolant system temperature is measured by Resistance Temperature Detectors (RTDs) at the following locations with the indicated Temperature Elements (TEs):

(a) Pool Coolant Heat Exchanger (HX521) Inlet (T_h) - TE 901C; and

(b) Pool Coolant Heat Exchanger (HX521) Outlet (T_c) - TE 901D.

Pool coolant heat exchanger inlet and outlet temperatures are recorded on a strip-chart, two-pen recorder mounted on the instrument panel. The low level DC millivolt signal produced by the RTD is converted to a proportional DC milliampere signal (10 to 50 *m*A) by an RTD Transmitter. The output signals from the RTD Transmitters for TE 901D and TE 901C are then directed to an Adder-Subtractor Module (ASM) and a two-pen recorder. The ASM provides a differential temperature input signal to the automatic power calculator. In addition to providing temperature indication, the recorder will initiate a "Pool Loop Hi Temp" annunciator alarm if pool coolant heat exchanger inlet temperature exceeds 110% of the normal operating value.

Pool coolant system temperature is controlled by controlling pool coolant heat exchanger outlet temperature. The amount of secondary coolant passing through the heat exchanger is regulated by an automatic temperature control valve (S-2) located in a bypass line adjacent to the heat exchanger system. This valve is a 6-inch electro-hydraulic butterfly valve which responds to pool coolant outlet temperature from the heat exchanger, and maintains a constant cold leg temperature in the pool coolant system. The controller unit is installed adjacent to the pool coolant heat exchanger inlet and outlet temperature recorder and provides manual control of S-2, if desired.

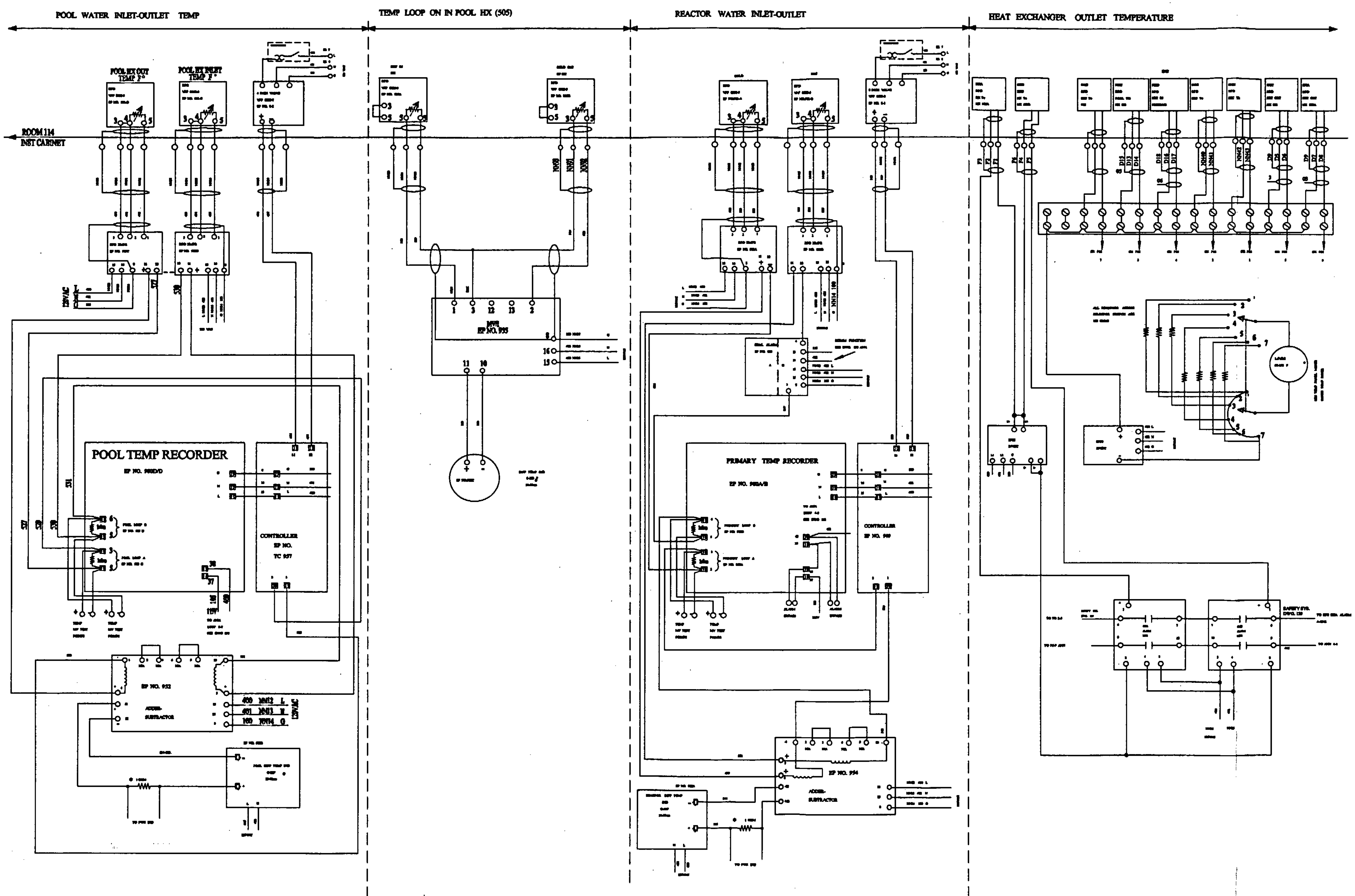


FIGURE 7.7
PROCESS INSTRUMENTATION CONTROL AND INTERLOCK

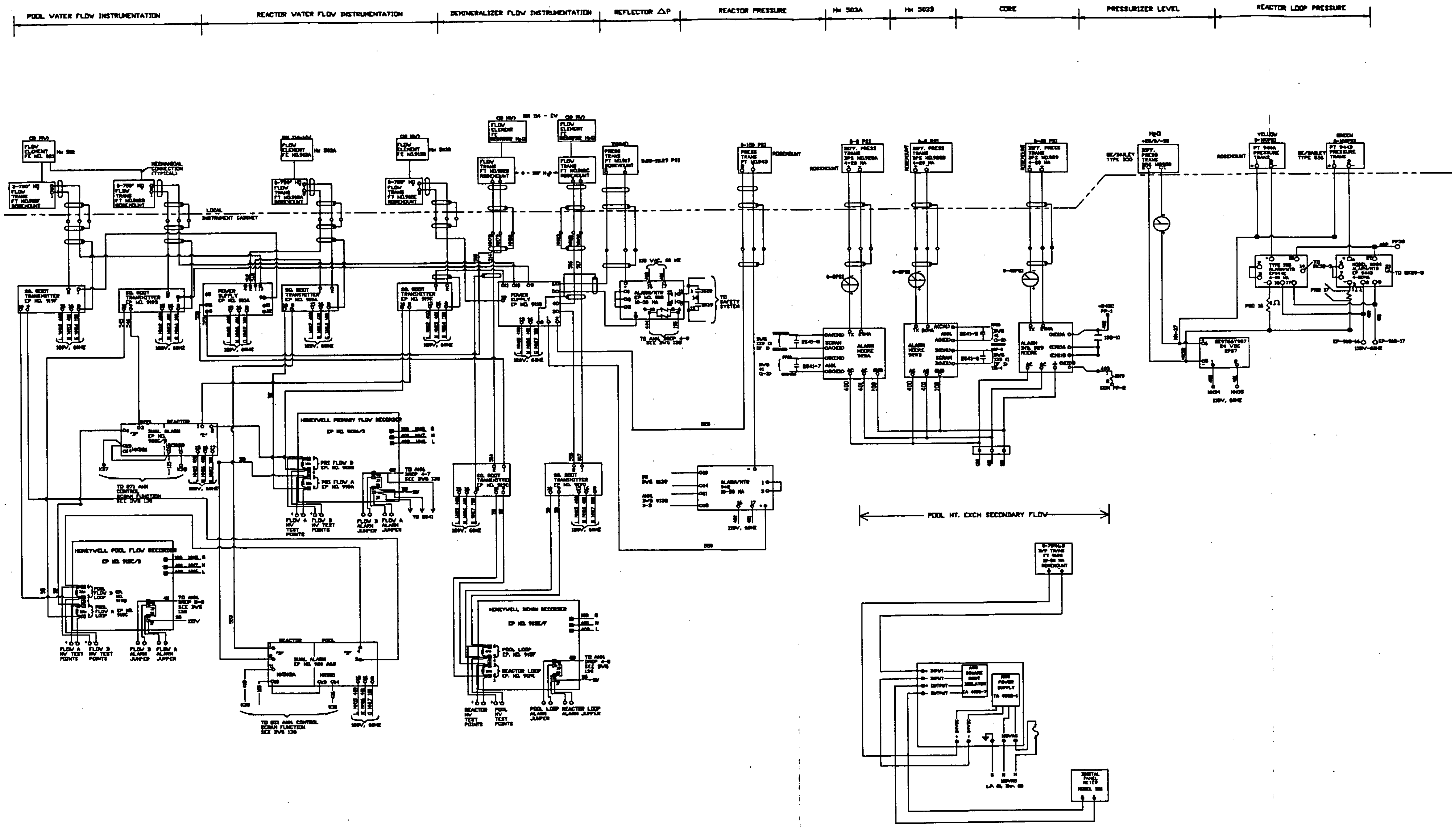


FIGURE 7.8
PROCESS INSTRUMENTATION CONTROL AND INTERLOCK

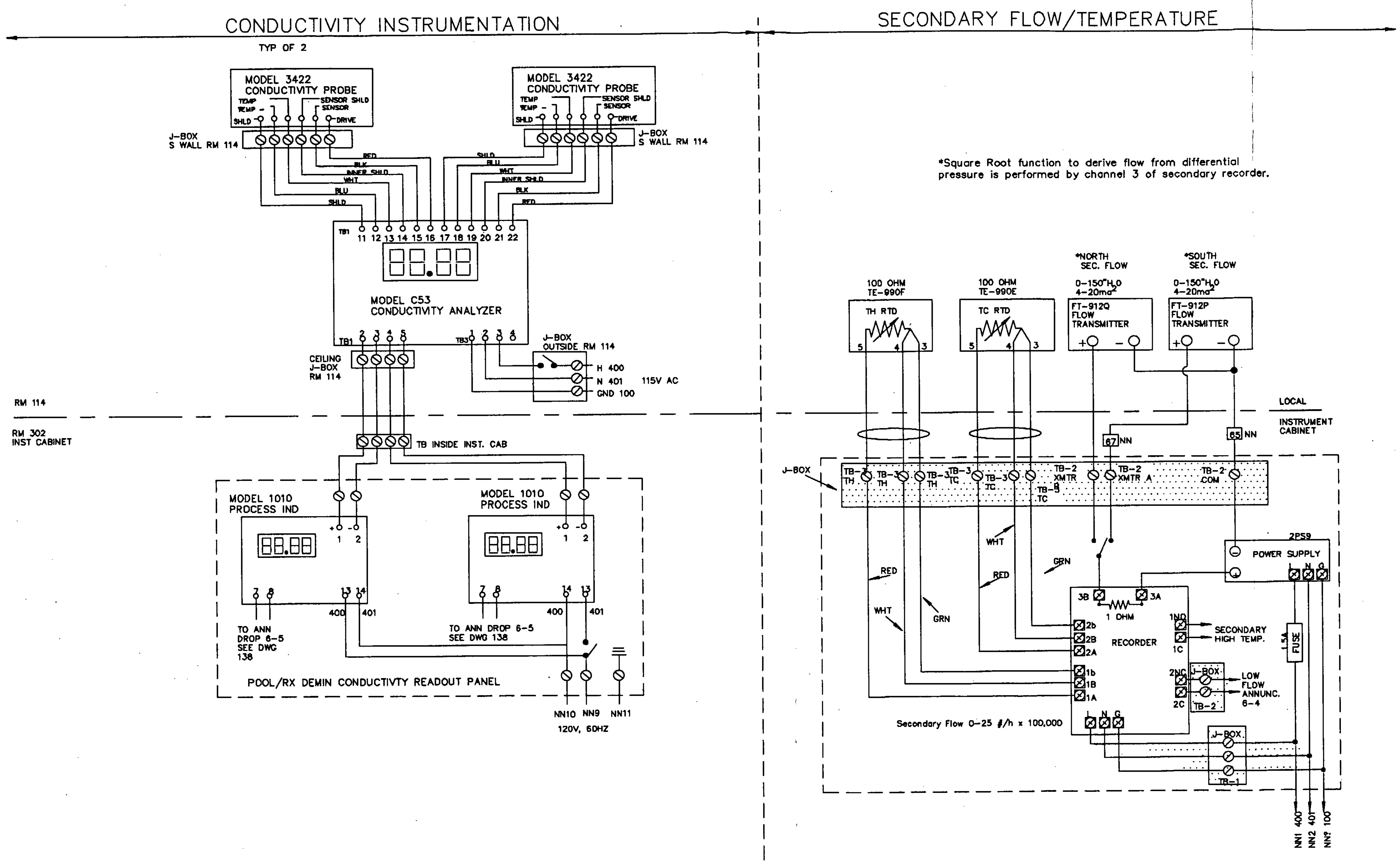


FIGURE 7.9
PROCESS INSTRUMENTATION CONTROL AND INTERLOCK

7.6.3 Pressure Measurement and Control

7.6.3.1 Primary Coolant System

Primary coolant system pressure is measured at the following locations with the indicated Pressure Transmitters (PTs) and Pressure Sensors (PSs):

(a) Reactor Core Outlet - PT 944A & PT 944B; and

(b) Primary Coolant Heat Exchanger Outlet - PT 943.

Primary coolant system pressure is measured at the reactor core outlet near primary coolant isolation valve V507A by PT 944A and PT 944B. The output signal produced by each pressure transmitter is directed to an Alarm Meter Unit. If primary coolant pressure decreases to 95% of the normal operating value, a reactor scram is initiated. The Alarm Meter Unit for PT 944A de-energizes relay 2K13 which opens a contact in the process input string to E4A of the Reactor Safety System NCLUs, thereby interrupting power to the control blade electromagnets. The Alarm Meter Unit for PT 944B de-energizes relay 2K28 which opens a contact in the process input string to E3B of the NCLUs. De-energizing either relay 2K13 or 2K28 will also cause the following actions to occur:

1. Primary Coolant Circulation Pumps 501A and 501B will stop;

2. Primary Coolant Isolation Valves V507A and V507B will close;

3. Anti-Siphon System Isolation Valves V543A and V543B will open; and

4. "Reactor Loop Lo Press Scram" annunciator alarm will be initiated.

The Alarm Meter Units also provide the reactor operator with primary coolant system pressure indication on the instrument panel.

Primary coolant system pressure is also measured by PT 943 at the point where the outlet piping from the primary coolant heat exchangers converge. The output signal produced by the pressure transmitter is directed to an Alarm Meter Unit. If primary coolant pressure decreases to 95% of the normal operating value, a reactor scram and a "Reactor Loop Lo Press Scram" annunciator alarm are initiated. The Alarm Meter Unit for PT 943 opens a contact in the process input string to E3B of the Reactor Safety System NCLUs, thereby interrupting power to the control blade electromagnets. The Alarm Meter Unit also provides the reactor operator with primary coolant system pressure indication on the instrument panel.

An additional primary coolant system low pressure scram is provided by a pressure sensor (PS 938) which measures pressurizer pressure. This instrument is discussed in Section 7.6.5.

7.6.4 Flow Measurement and Control

7.6.4.1 Primary Coolant System

Primary coolant system flow is measured at the following locations with the indicated Flow Transmitters (FTs) and Differential Pressure Sensors (DPSs):

(a) Downstream of Primary Coolant Heat Exchanger (HX503A) - FT 912A;

(b) Downstream of Primary Coolant Heat Exchanger (HX503B) - FT 912E;

(c) Differential Pressure across HX503A - DPS 928A;

(d) Differential Pressure across HX503B - DPS 928B; and

(e) Differential Pressure across the Reactor Core - DPS 929.

Primary coolant system flow is recorded on a strip-chart, two-pen recorder mounted on the instrument panel. Differential pressure across orifice plates (Flow Element 913A and 913B) located downstream of HX503A and HX503B is measured by FT 912A and FT 912E, respectively. The output signal (10 to 50 *m*A) generated by each flow transmitter is directed to a Square Root Converter which provides a linear output signal for the two-pen recorder and a Dual Alarm Unit. In addition to providing flow indication, the recorder will initiate a "Reactor Loop Lo Flow" annunciator alarm if primary coolant flow downstream of either heat exchanger decreases to 95% of the normal operating value. If primary coolant flow decreases to 90% of the normal value, a reactor scram and a "Reactor Loop Lo Flow Scram" annunciator alarm are initiated. The Dual Alarm Unit for FT 912A opens a contact (K30-2) in the process input string to E4A of the Reactor Safety System NCLUs, thereby interrupting power to the control blade electromagnets. The Dual Alarm Unit for FT 912E opens a contact (K38-2) in the process input string to E3B of the NCLUs.

A primary coolant system low flow scram may also be initiated by a low differential pressure across the primary coolant heat exchangers. Differential pressure across HX503A and HX503B, which provides an indication of flow, is measured by DPS 928A and DPS 928B, respectively. The output signal (4 to 20 *m*A) produced by each differential pressure sensor is directed to an Alarm Unit. If primary coolant flow decreases to 90% of the normal operating value, a reactor scram is initiated. The Alarm Unit for DPS 928A opens a contact in the process input string to E3B of the Reactor Safety System NCLUs, thereby interrupting power to the control blade electromagnets. The Alarm Unit for DPS 928B opens a contact in the process input string to E4A of the NCLUs. The Alarm Units will also de-energize relays 2K24 and 2K26, which will cause the following actions to occur:

1. "Low Primary HX ΔP Scram" annunciator alarm is initiated; and

2. In-Pool Heat Exchanger Isolation Valves V546A and V546B open.

The output signals produced by DPS 928A and DPS 928B also power meters on the instrument panel, thus providing the reactor operator with an indication of primary coolant heat exchanger differential pressure.

Differential pressure across the reactor core is monitored by DPS 929. This instrument also initiates a reactor scram, serving as a backup to the primary low flow scrams. The output signal (4 to 20 *m*A) produced by the differential pressure sensor is directed to an Alarm Meter Unit. If core differential pressure decreases to 90% of the normal operating value, a reactor scram is initiated. The Alarm Meter Unit will de-energize relay 2K9 which opens a contact in the Power Level Interlock circuit causing relays 1K13 and 1K26 to de-energize. Relay 1K13 opens a contact in the process input string to E3B of the Reactor Safety System NCLUs, thereby interrupting power to the control blade electromagnets. Relay 1K26 opens a contact in the process input string to E4A of the NCLUs. De-energizing either relay 1K13 or 1K26 also causes the following actions to occur:

1. "Power Level Interlock Scram" annunciator alarm will be initiated; and

2. In-Pool Heat Exchanger Isolation Valves V546A and V546B will open.

The output signal produced by DPS 929 also powers a meter on the instrument panel, thus providing the reactor operator with an indication of reactor core differential pressure.

In addition to the aforementioned primary coolant system scrams, in the event either primary coolant isolation valve V507A or V507B leaves its fully-open position, a limit switch mounted on each valve actuator causes relays 2K10, 2K11, and 2K27 to de-energize and the following actions occur:

1. Reactor will scram (relays 1K13 and 1K26 de-energize);

2. Primary Coolant Circulation Pumps 501A and 501B will stop;

3. In-Pool Heat Exchanger Isolation Valves V546A and V546B will open;

4. Anti-Siphon System Isolation Valves V543A and V543B will open;

5. Surge Line Isolation Valve V527C will close;

6. "Power Level Interlock Scram" annunciator alarm will be initiated; and

7. "Pump 501A/B On Valve 507A/B Closed" annunciator alarm will be initiated.

A reactor scram initiated by either primary coolant isolation valve V507A or V507B leaving its fully-open position provides a first line of protection for a loss of flow accident initiated by an inadvertent closure of the isolation valve(s).

Primary coolant demineralizer flow is recorded on a strip-chart, two-pen recorder mounted on the instrument panel. Differential pressure across an orifice plate (Flow Element 923A) located downstream of the primary coolant demineralizer pump is measured by FT 912B. The output signal generated by the flow transmitter is directed to a Square Root Converter which provides a linear output signal for the two-pen recorder. In addition to providing flow indication, the recorder initiates a "Reactor or Pool Loop Demineralizer Lo Flow" annunciator alarm if primary coolant demineralizer flow decreases to 85% of the normal operating value.

7.6.4.2 Pool Coolant System

Pool coolant system flow is measured at the following locations with the indicated Flow Transmitters (FTs) and Pressure Transmitters (PTs):

(a) Downstream of Pool Coolant Heat Exchanger (HX521) - FT 912D & FT 912F; and

(b) Differential Pressure across the Reactor Pool Reflector - PT 917.

Pool coolant system flow is recorded on a strip-chart, two-pen recorder mounted on the instrument panel. Differential pressure across an orifice plate (Flow Element 921A) located downstream of HX521 is measured by FT 912D and FT 912F. The output signal (10 to 50 *m*A) produced by each flow transmitter is directed to a Square Root Converter which provides a linear output signal for the flow recorder and a Dual Alarm Unit. In addition to providing flow indication, the recorder will initiate a "Pool Loop Lo Flow" annunciator alarm if pool coolant flow downstream of the heat exchanger decreases to 90% of the normal operating value. If pool coolant flow decreases to 85% of the normal value, a reactor scram and a "Pool Loop Lo Flow Scram" annunciator alarm are initiated. The Dual Alarm Unit for FT 912D opens a contact (K37-2) in the process input string to E3B of the Reactor Safety System NCLUs, thereby interrupting power to the control blade electromagnets. The Dual Alarm Unit for FT 912F opens a contact (K31-2) in the process input string to E3B of the NCLUs.

Differential pressure across the reactor pool reflector is monitored by PT 917. This instrument also initiates a reactor scram, serving as a back-up to the pool low flow scram. The transmitter senses pressure at the hot leg of the pool coolant system near pool coolant isolation valve V509 and provides an input signal to an Alarm Meter Unit. If pressure increases above or decreases below a predetermined set point, a reactor scram and a "Reflector Hi-Low Diff Pressure Scram" annunciator alarm are initiated. The Alarm Meter Unit opens contacts in the process input string to E3B of the Reactor Safety System NCLUs, thereby interrupting power to the control blade electromagnets. The Alarm Meter Unit also provides reactor pool reflector differential pressure indication on the instrument panel.

In addition to the aforementioned pool coolant system scrams, in the event the pool coolant isolation valve V509 leaves its fully-open position, a limit switch mounted on the valve actuator will cause relay 2K12 to de-energize and the following actions will occur:

1. Reactor will scram (a contact in the process input string to E4A opens);

2. Pool Coolant Circulation Pumps 508A and 508B will stop;

3. Pool Coolant Deminerlizer Pump 513B will stop; and

4. "Pool Loop Valve 509 Off Open Scram" annunciator alarm will be initiated.

A reactor scram initiated by the pool coolant isolation valve V509 leaving its fully-open position provides a first line of protection for a loss of flow accident initiated by an inadvertent closure of the isolation valve.

Pool coolant demineralizer flow is recorded on a strip-chart, two-pen recorder mounted on the instrument panel. Differential pressure across an orifice plate (Flow Element 923B) located downstream of the pool coolant demineralizer pump is measured by FT 912C. The output signal generated by the flow transmitter is directed to a Square Root Converter which provides a linear output signal for the two-pen recorder. In addition to providing flow indication, the recorder initiates a "Reactor or Pool Loop Demineralizer Lo Flow" annunciator alarm if pool coolant demineralizer flow decreases to 85% of the normal operating value.

7.6.5 Pressurizer Pressure and Liquid Level Control

Primary coolant system pressure is maintained by a nitrogen bubble on top of the water volume in the pressurizer tank. Six Pressure Sensors (PSs) are installed on the pressurizer to provide pressure control, as well as reactor safety and alarm functions.

If pressurizer pressure increases above the normal operating pressure, the following actions will occur:

1. At 105% of normal pressure, PS 940 provides the signal to open nitrogen vent valve V545 and vent the pressurizer to the exhaust system;

2. At 110% of normal pressure, PS 946 initiates a "Pressurizer Hi Press" annunciator alarm; and

3. At 115% of normal pressure, PS 939 initiates a reactor scram by opening a contact (K26-2) in the process input string to E3B of the Reactor Safety System NCLUs, thereby interrupting power to the control blade electromagnets. In addition, a "Pressurizer Hi Press Scram" annunciator alarm will be initiated.

If pressurizer pressure decreases below the normal operating pressure, the following actions occur:

1. At 95% of normal pressure, PS 941 provides the signal to open nitrogen addition valve V526 and emit nitrogen into the pressurizer;

2. At 90% of normal pressure, PS 945 initiates a "Pressurizer Lo Press" annunciator alarm; and

3. At 85% of normal pressure, PS 938 initiates a reactor scram by opening a contact (K25-2) in the process input string to E4A of the Reactor Safety System NCLUs, thereby interrupting power to the control blade electromagnets. In addition, a "Reactor Loop Lo Press Scram" annunciator alarm is initiated.

Water level is maintained in the pressurizer by adding water with a positive displacement pump if the level is low, and draining water to the drain collection system if the level is high. Three Level Controllers (LCs) and a Differential Pressure Sensor (DPS) are installed on the pressurizer to provide liquid level indication and control, as well as reactor safety and alarm functions.

If pressurizer liquid level increases above the normal operating level, the following actions occur:

1. At approximately 6 inches (15.24 cm) above center line, LC 936 signals water drain valve V527A to open and drain water from the pressurizer; and

2. At approximately 11 inches (27.94 cm)above center line, LC 937 initiates a "Pressurizer Water Hi Level" annunciator alarm and signals valve V527A to open (backup to LC 936).

If pressurizer liquid level decreases below the normal operating level, the following actions occur:

1. At approximately 7 inches (17.78 cm) below centerline, LC 936 signals water addition valve V527B to open and start coolant charging Pump 533, adding water to the pressurizer;

2. At approximately 11 inches below center line, LC 935 initiates a "Pressurizer Water Lo Level" annunciator alarm; and

3. At approximately 13 inches (33.0 cm) below center line, LC 935 initiates a reactor scram by opening a contact (K28-2) in the process input string to E3B of the Reactor Safety System NCLUs, thereby interrupting power to the control blade electromagnets. In addition, a "Pressurizer Lo Level Scram" annunciator alarm is initiated and the surge line isolation valve V527C closes to prevent an introduction of nitrogen gas into the primary coolant system.

A meter mounted on the instrument panel provides the reactor operator with a continuous indication of pressurizer water level. DPS 930 produces a DC milliampere signal proportional to the differential pressure created by a dry reference leg and a wet variable leg. The output signal is directed to the pressurizer liquid level meter.

The pressurizer is described in greater detail in Section 5.6, Primary Coolant Make-Up Water System.

7.6.6 Surveillance

The Process Instrumentation and Control System is periodically tested for operability as stated in the Technical Specifications. The objective of this specification is to ensure proper operation of the system.

7.7 Reactor Safety System

7.7.1 Introduction

The Reactor Safety System consists of the electronic circuitry which can initiate the instantaneous drop of the reactor control blades (reactor scram) by interrupting power to their electromagnets should a monitored parameter exceed a predetermined value. A reactor scram may also be initiated manually by depressing a push button (1S10) on the reactor control console. The reactor scrams are listed in Table 7-8.

TABLE 7-8
REACTOR SCRAMS

1. Manual	9. Pressurizer High Pressure
2. Channel 2 & 3 Short Period	10. Reflector Hi-Low Diff. Pressure
3. Reactor Loop Low Flow	11. Pool Valve 509 Off Open
4. Reactor Loop High Temp	12. Bldg Plenum & Bridge Hi Activity
5. Pressurizer Low Level	13. Evacuation or Isolation
6. Reactor Pool Below Refuel Level	14. Reactor Loop Low Pressure
7. Channel 4, 5, & 6 High Power	15. Low Primary HX Diff. Pressure
8. Power Level Interlock	16. Pool Loop Low Flow

The protection channels and protective responses are sufficient to ensure that no safety limit, limiting safety system settings, or reactor safety system limiting condition for operation will be exceeded. The system design ensures that the design bases can be achieved, and that the system can be readily tested and maintained in the designed operating condition. Sufficient safety system isolation and independence is provided to avoid malfunctions or failures caused by other systems. The Reactor Safety System reasonably achieves safe reactor shutdown in the event of a single random malfunction within the system. System operation prevents or mitigates hazards to the reactor or the escape of radiation, so that the full range of normal operations poses no undue radiological risk to the health and safety of the public, the facility staff, or the environment.

7.7.2 Description

The Reactor Safety System consists of two NCLUs and two TAAs located in the instrument panel. The NCLUs receive input signals from the neutron monitoring and process instruments and provide outputs to control the TAAs. The TAAs are arranged in parallel, each controlling current to two control blade electromagnets. Should an input signal to either logic unit be interrupted, the TAAs will actuate and interrupt current to the electromagnets. Manual initiation of a reactor scram by switch 1S10 opens an input (E4A and E3B) to each NCLU as well as interrupting power to the TAAs. In addition, the master control switch (1S1) and the magnet current switch (1S14) located on the reactor control console are available to secure 115-VAC supply power to the TAAs. A reactor scram may also be manually initiated by the actuation of the reactor isolation or the facility evacuation switches (Section 7.8). Figure 7.10 shows the Reactor Safety System, and the conditions which initiate a reactor scram.

The signal inputs from the process instruments are discussed in detail in Section 7.6, Process Instrumentation and Control System.

7.7.2.1 Non-Coincidence Logic Units

Each NCLU consists of three interconnected component boards: two non-coincidence logic (A and B) and one auxiliary. Each non-coincidence logic component board receives input signals (12 or 24 volts) from the neutron flux monitors, process instruments, and safety interlocks as shown in Table 7-9. The input signals to each NCLU are isolated from each other. When all of the required input signals are present, a continuous output voltage (approximately +16 volts) is supplied to the Trip Actuator Amplifiers (TAAs). If one or more of the input signals are interrupted, the output voltage will decrease and cause the TAAs to trip. When an input signal drops to 0 volts, the base-emitter circuits of the logic unit transistors are reverse-biased, causing the output voltage to approach 0 volts. This output is applied to one of the two inputs of both TAAs.

The non-coincidence logic circuit is designed for fail-safe operation. A component failure or malfunction will cause the output voltage to drop to less than 2 volts (safe condition). This is accomplished by the redundancy designed into the circuit.

[redacted]

FIGURE 7.10
REACTOR SAFETY SYSTEM

TABLE 7-9
REACTOR SCRAM NON-COINCIDENCE LOGIC UNIT SIGNAL INPUTS

Reactor Scram Non-Coincidence Logic Unit (1/N) - "Yellow Leg"			
Non-Coincidence Logic Board "A" (1/4)		Non-Coincidence Logic Board "B" (1/3)	
Input Signal	Voltage	Input Signal	Voltage
NI Channel 2 NI Channel 4 NI Channel 6 Process Instrumentation[a]	+12 VDC +12 VDC +12 VDC +24 VDC	Not Used	
Reactor Scram Non-Coincidence Logic Unit (1/N) - "Green Leg"			
Non-Coincidence Logic Board "A" (1/4)		Non-Coincidence Logic Board "B" (1/3)	
Input Signal	Voltage	Input Signal	Voltage
Not Used		NI Channel 3 NI Channel 5 Process Instrumentation[b]	+12 VDC +12 VDC +24 VDC

[a]This process instrumentation input string to the non-coincidence logic unit is also referred to as the "yellow leg" of the reactor safety system.
[b]This process instrumentation input string to the non-coincidence logic unit is also referred to as the "green leg" of the reactor safety system.

7.7.2.2 Trip Actuator Amplifiers

Each Trip Actuator Amplifier (TAA) consists of a power supply and two series-connected bi-stable driver amplifiers. The output signal from each NCLU is divided and fed to a bi-stable driver amplifier in each TAA. The bi-stable driver amplifiers control two series output transistors which turn the output current off in response to a low voltage input signal from the NCLUs. As a result, the current to the magnet coils for control rod drive mechanisms A, B, C, and D is interrupted and the electromagnets release the control blades thus scramming the reactor.

In addition to interrupting current to the electromagnets, a loss of TAA output current (approximately 0 amperes) will de-energize relays 2K20 and 2K21. De-energizing either relay causes the following actions to occur:

1. The Rod Run-In System is activated (supply power to the rod run-in TAA is interrupted); and

2. Automatic rod control mode is discontinued.

Once tripped, the driver amplifier remains in this condition until the normal input voltage is restored, the scram input signal is cleared, and the unit is reset. The TAA is reset by depressing the "Scram Reset" push button (1S11) on the reactor control console.

A malfunction in one TAA could result in, at most, the failure to turn off the current to two control blade electromagnets. The other two control blades would drop and successfully shut down the reactor.

7.7.3 Limiting Conditions for Operation

The limiting conditions for operation of the Reactor Safety System and their bases are discussed in the Technical Specifications. The objective of these conditions is to ensure that the system's instrument channels are operable.

7.7.4 Surveillance

The Reactor Safety System is periodically tested for operability as stated in the Technical Specifications. The objective of this specification is to ensure proper operation of the system.

7.8 Engineered Safety Features Actuation Systems

7.8.1 Introduction

The Engineered Safety Features Actuation Systems receive input signals from monitoring instruments and initiate the operation of the engineered safety features (ESFs). The ESFs are systems which are designed to mitigate the consequences of the Maximum Hypothetical Accident (MHA) and the Loss of Coolant Accident (LOCA) and to keep radiological exposures to the operating staff and the general public within the limits of 10 CFR 20. The Engineered Safety Features Actuation Systems' designs give reasonable assurance of reliable operation, if required. The two systems which are identified as ESFs for the MURR are the Containment System and the Anti-Siphon System. These systems are described in greater detail in Chapter 6, Engineered Safety Features.

The MHA and LOCA are discussed in detail in Chapter 13, Accident Analyses.

7.8.2 Containment Actuation (Reactor Isolation) System

The Containment Actuation System (CAS) is designed to completely isolate the reactor containment building, thereby preventing or mitigating any uncontrolled release of radioactive materials to the environment during an accident. Isolation of the reactor containment building can be automatically initiated by radiation detectors located at the reactor pool upper bridge and in the containment building exhaust plenum. Isolation can be manually actuated by switches in the reactor control room or the facility lobby (Room 202). The radiation monitors are described in detail in Section 7.9.

Manual or automatic actuation of the CAS causes the following actions to occur:

1. Reactor will scram;

2. All normally-open reactor containment building penetrations with automatic sealable closures will close;

3. An audible alarm will sound throughout the containment building; and

4. A flashing light at the entrance to the containment building personnel airlock will illuminate.

To ensure system reliability, the CAS is normally energized. This ensures that any system failure such as a relay coil "burnout" or a short in the system would place the CAS in a tripped condition. Redundancy is incorporated into the actuation system to ensure that no single component or circuit failure will render any portion of the system inoperative. Characteristics of the CAS which provide the redundancy and reliability are as follows:

(a) All relays which energize to perform their intended function are paralleled by a second relay such that if one fails to energize, the unaffected relay will operate as required;

(b) All relays which de-energize to perform their intended function are assumed to be fail-safe;

(c) Four (4) independent radiation detectors can initiate automatic isolation of the reactor containment building, two at each location (reactor pool upper bridge and containment building ventilation exhaust plenum);

(d) The radiation monitors are isolated from each other such that the operation of one does not affect the operation of another;

(e) The radiation monitors are powered by independent low voltage power supplies and are therefore not subject to a single failure event; and

(f) The emergency electrical power system provides a continuous supply of electrical power to the radiation monitors and the reactor isolation system in the event of a loss of normal electrical power.

At least two major failures must occur before the CAS would be unable to perform its intended function. This is not a credible event.

7.8.2.1 Automatic Initiation

Isolation of the reactor containment building can be automatically initiated by radiation detectors located at the reactor pool upper bridge and in the containment building ventilation exhaust plenum. A radiation level which is one decade above background at these locations will cause relays 2K1A and 2K1B to de-energize. Relay 2K1A is actuated by the containment building exhaust plenum No.1 and the reactor pool upper bridge ALARA radiation monitors. Relay 2K1B is actuated by the containment building exhaust plenum No. 2 and reactor pool upper bridge radiation monitors. De-energizing either of these relays will cause the following actions to occur:

1. A reactor scram is initiated by the "green leg" of the Reactor Safety System by a process input string to E3B of the NCLUs;
2. Parallel relays R2A and R2B and relay 2K2 in the CAS will de-energize;
3. Containment building ventilation exhaust plenum backup doors will close; and
4. "Bldg Air Plenum & Bridge Hi Activity Scram" annunciator alarm will be initiated.

De-energizing either relay R2A or R2B (see No. 2 above) of the CAS will cause the following actions to occur:

1. Containment building ventilation supply and exhaust duct electric-motor-driven isolation doors 504 and 505 will close;
2. Containment building hot exhaust line isolation valves 16A and 16B will close;
3. The reactor isolation horns will activate; and
4. The warning light at the entrance to the containment building personnel airlock will illuminate.

These radiation monitors feature a failure mode that occurs if they do not receive an input signal from the detector assembly for a specified time period (approximately one minute). This failure will de-energize relays 2K1A and 2K1B, thus preventing reactor operation with a failed radiation monitor.

7.8.2.2 Manual Initiation

Isolation of the reactor containment building can be manually initiated from either of two locations: the reactor control room or the facility lobby (Room 202). Manual initiation from the reactor control room is performed by the actuation of either of two switches located on the reactor control console: reactor isolation switch 1S15 and facility evacuation switch 1S16.

Actuation of the reactor isolation switch will cause the following actions to occur:

1. Relay 2K2 will de-energize and cause:

    a) A reactor scram from the "yellow leg" of the reactor safety system by a process input string to E4A of the NCLUs; and

    b) Initiation of the "Evacuation or Isolation Scram" annunciator alarm; and

2. All actions caused by relays R2A and R2B de-energizing.

Actuation of the facility evacuation switch will cause the following actions to occur:

1. Parallel relays R3A and R3B will de-energize and open contacts in the CAS thereby de-energizing relays 2K2, R2A, and R2B; and

2. Evacuation horns installed at strategic locations throughout the reactor facility laboratory building will activate.

A second facility evacuation switch is located at the reception desk in the facility lobby (Room 202).

Any postulated emergency necessitating the evacuation of all personnel from the reactor facility does not require instantaneous action, therefore, the actuation of the facility evacuation system is an administrative decision and no automatic trip function is required.

7.8.2.3 Surveillance

The CAS and its associated radiation monitors are tested for operability as stated in the Technical Specifications. The objective of this specification is to ensure proper operation of the system.

7.8.3 Anti-Siphon Actuation System

The Anti-Siphon Actuation System functions as a backup system to the various safety instrumentation and equipment (e. g., pressure sensors, pump and valve interlocks, etc.) which ensures that the reactor core does not become uncovered during a LOCA. The system is designed to admit a fixed volume of air to the high point of the reactor outlet piping, or invert loop, instantaneously establishing the pressure in this area at equal to or greater than atmosphere. This prevents a siphon action from being created due to a rupture of the primary coolant piping.

The Anti-Siphon Actuation System is automatically actuated upon detection of primary coolant system low pressure. System pressure is monitored by two electronic pressure transmitters

(PT 944A and PT 944B) located on the 12-inch primary coolant piping between the reactor pressure vessel and primary coolant isolation Valve 507A. Should primary coolant system pressure decrease below a predetermined value, PT 944A and PT 944B will de-energize relays 2K13 and 2K28, respectively, either of which will cause the following actions to occur:

1. Reactor will scram - 2K13 will open a contact in the process input string to E4A and 2K28 will open a contact in the process string to E3B of the Reactor Safety System NCLUs;

2. Primary Coolant Circulation Pumps 501A and 501B will stop;

3. Primary Coolant Isolation Valves 507A and 507B will close; and

4. Anti-Siphon System Isolation Valves 543A and 543B will open.

Redundancy is incorporated into the Anti-Siphon Actuation System to ensure that no single component or circuit failure will render any portion of the system inoperative. System reliability is achieved through instrumentation and equipment which fail-safe when de-energized. Characteristics of this system which provide the redundancy and reliability are:

(a) All relays which de-energize to perform their intended function are assumed to be fail-safe;

(b) Redundant primary coolant system pressure monitors PT 944A and PT 944B, either of which will initiate operation of this system are provided; and

(c) Redundant relays 2K13 and 2K28, either of which will initiate operation of this system are provided.

At least two major failures must occur before the Anti-Siphon Actuation System would be unable to perform its intended function. This is not a credible assumption.

7.8.3.1 Surveillance

The actuation of this system is periodically tested for operability as stated in the Technical Specifications. The objective of this specification is to ensure proper operation of this system.

7.9 Radiation Monitoring Systems

7.9.1 Introduction

The installed systems at the MURR which are used to detect and measure the radiation and activity levels and the releases of radioactive gases from the facility to the uncontrolled environment include the Area Radiation Monitoring System (ARMS), the Fuel Element Failure Monitoring System, the Secondary Coolant Monitoring System, and the Off-Gas Radiation Monitoring System.

Radiation and radioactivity measurements are made to demonstrate that exposures to radiation and gaseous effluent releases are within the established objectives of the MURR ALARA Program. The locations that have been selected to be monitored continuously include areas or systems where an increase in radiation level may indicate a change in plant conditions which could potentially have an adverse effect on the safe operation of the reactor facility and may constitute an undue risk to the health and safety of the facility staff and the general public.[1]

In addition, the ARMS can initiate an automatic isolation of the reactor containment building should a high radiation level occur at the reactor pool bridge or in the containment building ventilation exhaust plenum. This function is described in greater detail in Chapter 6, Engineered Safety Features and Section 7.8.2, Engineered Safety Features Actuation Systems.

This section discusses the operating principles, designs, and the functional performance of the instrumentation and control (I&C) aspects of the permanently installed radiation monitoring equipment at the reactor facility. The MURR Radiation Protection Program establishes the radiation monitoring criteria for the reactor facility which are necessary to provide an acceptable level of radiation protection for the staff and general public. This program is discussed in detail in Chapter 11, Radiation Protection Program and Waste Management. The MURR Radiation Monitoring System radiation detectors and monitors are applicable to measure the radiation encountered in a research reactor environment. These devices give reasonable assurance that all radiation sources will be identified and accurately evaluated. The Radiation Monitoring System also provides reasonable assurance that dose rates and effluents at the facility will be acceptably detected, and that the health and safety of the facility staff, the environment, and the public will be acceptably protected.

7.9.2 Area Radiation Monitoring System

7.9.2.1 Description

The Area Radiation Monitoring System (ARMS) is used to continuously monitor gamma radiation levels at various remote locations in the reactor facility. Radiation levels are displayed on four- or five-decade logarithmic scale meters positioned in a centrally-located control chassis. The control chassis is mounted on the reactor control room instrument panel. The locations of the remote detector assemblies and their detection ranges are shown in Table 7-10. The ARMS is shown in Figure 7.11.

Each wall-mounted remote detector assembly contains a high voltage power supply, a radiation detecting element, a pulse amplifier, a line driver, and a remotely-operated ^{90}Sr-^{90}Y check source. The pulse signal generated by the detecting element is amplified and processed, and then carried via cable to an electronics channel, where it is further processed and displayed on a logarithmic scale analog meter in millirem per hour. Each electronics channel is equipped with an

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[1]See Footnote on Page 7-2.

adjustable set point trip that initiates an audible and visual alarm on detection of a high radiation level. Adjustment of the potentiometer which sets the trip point is accomplished by inserting a small screwdriver through an access port on the front panel of each electronics channel. This arrangement reasonably ensures that an inadvertent or unmonitored adjustment of the trip point is highly unlikely. The power required to operate the detector assembly is supplied by the electronics channel through interconnecting wires.

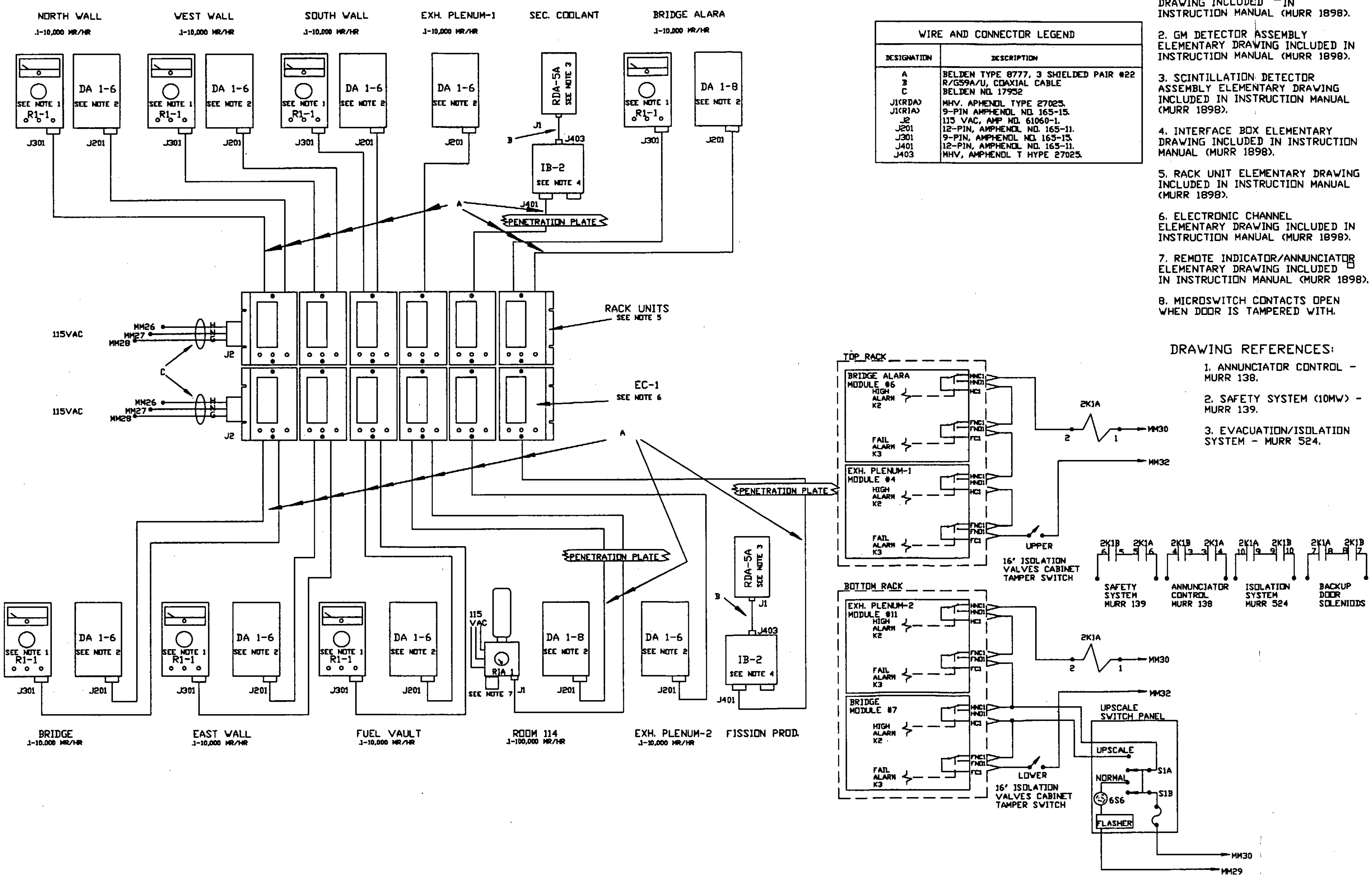
TABLE 7-10
AREA RADIATION MONITORING SYSTEM

Detector Assembly Location	Detection Range (millirem/hr)	Remote Indicator
Beamport Floor - South Wall	0.1 to 10K	Yes
Beamport Floor - West Wall	0.1 to 10K	Yes
Beamport Floor - North Wall	0.1 to 10K	Yes
Fuel Storage Vault	0.1 to 10K	Yes
Containment Building Exhaust Plenum - No. 1[a]	0.1 to 10K	No
Containment Building Exhaust Plenum - No. 2[a]	0.1 to 10K	No
Reactor Pool Upper Bridge (north side)[a]	0.1 to 10K	Yes
Reactor Pool Upper Bridge ALARA (south side)[a]	1.0 to 100K	Yes
Mechanical Equipment Room (Room 114)	1.0 to 100K	Yes
Beamport Floor - East wall	0.1 to 10K	Yes

[a]Radiation detectors at these locations can initiate an automatic isolation of the reactor containment building as described in Chapter 6, Engineered Safety Features. Redundancy is incorporated into the system (i.e., two detectors at each location) to ensure compliance with IEEE-279, Single Failure Criterion.

All remote detector assemblies, with the exception of the mechanical equipment room (Room 114) and the reactor pool bridge ALARA, utilize a Geiger-Mueller (GM) tube detecting element. The Room 114 and pool bridge ALARA detector assemblies utilize pressurized ion chambers as the detecting element due to the intermediate- to high-level radiation fields that they operate in.

A rack unit holds the electronics chassis for six channels of analog radiation meters and is the interconnecting tie point for those six channels in addition to supplying the AC line voltage. Two rack units are presently installed.



WIRE AND CONNECTOR LEGEND

DESIGNATION	DESCRIPTION
A	BELDEN TYPE 8777, 3 SHIELDED PAIR #22
B	R/G59A/U, COAXIAL CABLE
C	BELDEN NO. 17952
J1(RDA)	MHV, APHENOL TYPE 27025.
J1(RIA)	9-PIN AMPHENOL NO. 165-15.
J2	115 VAC, AMP NO. 61060-1.
J201	12-PIN, AMPHENOL NO. 165-11.
J301	9-PIN, AMPHENOL NO. 165-15.
J401	12-PIN, AMPHENOL NO. 165-11.
J403	MHV, AMPHENOL T HYPE 27025.

NOTE:

1. REMOTE INDICATOR ELEMENTARY DRAWING INCLUDED IN INSTRUCTION MANUAL (MURR 1898).
2. GM DETECTOR ASSEMBLY ELEMENTARY DRAWING INCLUDED IN INSTRUCTION MANUAL (MURR 1898).
3. SCINTILLATION DETECTOR ASSEMBLY ELEMENTARY DRAWING INCLUDED IN INSTRUCTION MANUAL (MURR 1898).
4. INTERFACE BOX ELEMENTARY DRAWING INCLUDED IN INSTRUCTION MANUAL (MURR 1898).
5. RACK UNIT ELEMENTARY DRAWING INCLUDED IN INSTRUCTION MANUAL (MURR 1898).
6. ELECTRONIC CHANNEL ELEMENTARY DRAWING INCLUDED IN INSTRUCTION MANUAL (MURR 1898).
7. REMOTE INDICATOR/ANNUNCIATOR ELEMENTARY DRAWING INCLUDED IN INSTRUCTION MANUAL (MURR 1898).
8. MICROSWITCH CONTACTS OPEN WHEN DOOR IS TAMPERED WITH.

DRAWING REFERENCES:

1. ANNUNCIATOR CONTROL - MURR 138.
2. SAFETY SYSTEM (10MW) - MURR 139.
3. EVACUATION/ISOLATION SYSTEM - MURR 524.

FIGURE 7.11
AREA RADIATION MONITORING SYSTEM

A remote indicator is mounted next to each detector assembly except for the two detector assemblies located in the reactor containment building ventilation exhaust plenum. The remote indicators display the analog meter reading and the alarm condition of the channel to which it is connected in addition to providing an audible alarm when the high radiation level set point trip is exceeded.

7.9.2.2 Limiting Conditions for Operation

The limiting conditions for operation of the radiation monitors which can initiate an automatic isolation of the reactor containment building, and their bases, are discussed in the Technical Specifications. The objective of these specifications is to ensure that sufficient reliable information is available to the reactor operator to ensure safe operation of the reactor.

7.9.2.3 Surveillance

The ARMS is periodically tested for operability using the check sources installed in the remote detector assemblies. The radiation monitors which can initiate an automatic isolation of the reactor containment building are tested as stated in the Technical Specifications. In addition, the trip points for the high radiation alarms are verified to be set in accordance with the Operations Procedures prior to reactor startup.

7.9.3 Fuel Element Failure Monitoring System

7.9.3.1 Description

The Fuel Element Failure Monitoring System continuously monitors the primary coolant system for fission product activity buildup that may indicate a fuel element failure.

This monitoring system functions by directing a controlled amount (approximately 100 cc/min) of primary coolant from the primary coolant demineralizer loop, upstream of the demineralizer, through a water sampling unit. The sampling unit is enclosed in a single console type steel cabinet and includes a shielded glass wool filter column, a shielded cation resin column, an anion resin column incorporated in the detector shield, detector shielding, and flow regulating equipment. A scintillation detector measures the activity collected on the anion resin column. The output from the detector is fed through an interface box that provides signal amplification to one of the electronics channels of the ARMS where the pulse signal is processed and displayed on a logarithmic scale analog meter in counts per minute. The electronics channel, mounted in one of the rack units of the ARMS, is equipped with an adjustable set point trip that initiates a "Reactor Loop Coolant Hi Activity" annunciator alarm on detection of a high radiation level.

7.9.3.2 Limiting Conditions for Operation

The Fuel Element Failure Monitoring System is required to be in service whenever the reactor is operated with forced circulation. If it is taken out of service, the primary coolant is required to be sampled as stated in the Technical Specifications.

7.9.3.3 Surveillance

The Fuel Element Failure Monitoring System's high radiation annunciator alarm is tested for operability prior to reactor startup.

7.9.4 Secondary Coolant Monitoring System

7.9.4.1 Description

The Secondary Coolant Monitoring System continuously monitors the secondary coolant system for the presence of radioactive isotopes which could indicate a leak from the primary or pool coolant systems through their respective heat exchangers.

This system consists of a scintillation detector that measures the gross activity of the secondary water. The output from the detector is fed through an interface box which provides signal amplification to one of the electronics channels of the ARMS where the signal is processed and displayed on a logarithmic scale analog meter in counts per minute. The electronics channel, mounted in one of the rack units of the ARMS, is equipped with an adjustable set point trip that initiates a "Secondary Coolant Hi Activity" annunciator alarm on detection of a high radiation level.

The secondary coolant monitor is located in the return leg of the secondary piping, downstream of the pool and primary heat exchangers. This location insures a faster response in the event of a leak from either heat exchanger.

7.9.4.2 Surveillance

The Secondary Coolant Monitoring System's high radiation alarm is tested for operability prior to reactor startup.

7.9.5 Off-Gas Radiation Monitoring System

7.9.5.1 Description

The air exiting the facility through the ventilation system exhaust stack is continuously-monitored for airborne radioactivity by the Off-Gas Radiation Monitoring System. The maximum rate of discharge through the facility ventilation exhaust stack shall not exceed limits specified in the

Technical Specifications. These limits ensure that exposure to the general public resulting from radioactivity released to the environment will not exceed the limits of 10 CFR 20.

The monitoring equipment of this system consists of a three-channel radiation detection system designed to measure the airborne concentrations of radioactive particulate, iodine, and noble gas in the facility exhaust air which is sampled by an isokinetic probe located in the ventilation exhaust plenum. The radiation detection equipment is a self-contained unit consisting of a fixed filter monitored by a beta scintillation detector, a charcoal cartridge monitored by a gamma scintillation detector, and a gas chamber monitored by a beta scintillation detector shielded by three inches of lead (4π configuration). One-inch (2.54 cm) lead shields separate the individual detectors. The output from each radiation detector is displayed on a local meter in counts per minute (cpm) and on a strip-chart, three-pen recorder mounted on the instrument panel in the reactor control room. An audible and visual alarm alerts the operator to high activity or abnormal air flow through the radiation detection equipment.

7.9.5.2 Limiting Conditions for Operation

The limiting condition for operation of the Off-Gas Radiation Monitoring System is discussed in the Technical Specifications. The objective of this specification is to ensure that sufficient reliable information is available to the reactor operator to ensure safe operation of the reactor.

7.9.5.3 Surveillance

The Off-Gas Radiation Monitoring System is periodically tested for operability.

CHAPTER 8

ELECTRICAL POWER SYSTEMS

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REFERENCES

8.1 Preliminary Hazards Report, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, March 1961.

8.2 Hazards Summary Report, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, July 1965.

8.3 Hazards Summary Report, Addendum 1, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, February 1966.

8.4 Hazards Summary Report, Addendum 2, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, May 1966.

8.5 Hazards Summary Report, Addendum 3, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, August 1972.

8.6 Hazards Summary Report, Addendum 4, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, October 1973.

8.7 Hazards Summary Report, Addendum 5, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, January 1974.

8.8 "Specifications - Covering Mechanical and Electrical Work for the Construction of the Research Reactor Facilities," Volume 3 of 4, Cornelius L. T. Gabler and Associates, Detroit, Michigan, February 1963.

8.0 ELECTRICAL POWER SYSTEMS

This chapter discusses and describes the electrical power systems designed to support reactor operation at the facility.[1] The information in this chapter is provided under two categories: normal and emergency electrical power systems.

8.1 Normal Electrical Power System

8.1.1 Introduction

Normal electrical power is supplied to the Missouri University Research Reactor (MURR) by the University of Missouri at Columbia Power Plant and/or the City of Columbia through an electrical distribution system which serves the entire campus. This electrical distribution system from the University's power plant has five (5) major substations, serving different sections of the campus, as well as a tie-in (Hinkson Creek Substation) to the City of Columbia electrical transmission system. Load sharing between the University and the City of Columbia is dependent upon demand and the cost of production.

The electrical distribution system for the MURR was originally designed and constructed according to the following codes and standards (Ref. 8.8):

- State of Missouri and City of Columbia Electrical Codes;
- National Electrical Code;
- Insulated Power Cable Engineers Association Standards;
- Association of Electrical Illuminating Companies Standards; and
- National Electrical Manufacturers Association (NEMA) Standards.

The design of the MURR does not require electrical power to safely shut down the reactor or to maintain an acceptable shutdown condition. A loss of normal electrical power and a subsequent failure-to-start of the emergency power generator (i.e., a complete loss of electrical power to the facility) is analyzed in Chapter 13, Accident Analyses.

8.1.2 Description of System

The MURR receives its electrical power from the Research Park Substation located south of the facility. Two independent primary transmission lines, routed through underground duct banks, supply 13.8 kV, 3-phase, 60-cycle electrical power to the cooling tower and laboratory building through a series of disconnect switches. One transmission line is directed to a pad mount 1,500-kVA transformer located just outside the cooling tower while the other transmission line is directed to a

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[1]The original safety evaluation of the MURR is documented in the Preliminary Hazards Report (Ref. 8.1), the Hazards Summary Report (Ref. 8.2), and Hazards Summary Report, Addenda 1-5 (Ref. 8.3-8.7).

pad mount 2,000-kVA transformer located just outside the laboratory building, on the southwest side.

The 1,500-kVA transformer is a fused-switch, 3-phase, oil-filled, step-down transformer which reduces the incoming line voltage from 13.8 kV to 480/277 V. This transformer supplies electrical power to Substation A, which is located in the electrical equipment room (Room 101) of the cooling tower. Substation A houses two main feeder breakers. One feeder breaker provides electrical power to a Motor Control Center (MCC) also located in the electrical equipment room, while the other breaker is an installed spare.

The 2,000-kVA transformer is a fused-switch, 3-phase, oil-filled, step-down transformer which reduces the incoming line voltage from 13.8 kV to 480/277 V. This transformer supplies electrical power, through a 1,600-amp main breaker, to Substation B, which is located in the mechanical equipment room (Room 278) of the laboratory building. Substation B houses five main feeder breakers. The feeder breakers provide electrical power to three Motor Control Centers (MCC), a 120/208 V Distribution Center, and the Emergency Distribution Center (CTR-1) through an Automatic Transfer Switch (ATS). In the event of a loss of normal electrical power, the ATS transfers the emergency electrical load (CTR-1) between normal and emergency power. Operation of the emergency electrical power system, including the ATS, is described in Section 8.2.

To reduce spurious (false) reactor scrams caused by voltage fluctuations or a momentary interruption in electrical power, a 15-kVA Uninterruptible Power Supply (UPS) is installed to provide precise regulated and transient free 120-VAC electrical power to the reactor Instrumentation and Control (I&C) System. The UPS also ensures that the reactor control console and instrument panel indications are maintained during the transition period from the time of a loss of normal electrical power to the time the emergency load (CTR-1) is supplied by the diesel generator. This allows the shutdown condition of the reactor to be monitored continuously by the reactor operator.

The UPS system (See Figure 8.1) consists of a static inverter, a rectifier charger, and a lead acid battery bank. The battery bank is sized in accordance with the Institute of Electrical and Electronic Engineers Standard 485-1978 (IEEE 485-1978). The primary AC source (CTR-1) supplies power to the rectifier charger. The rectifier charger provides regulated DC power to support the inverter and simultaneously maintain the battery bank in a fully charged condition. The inverter converts the DC power into regulated AC power for the reactor I&C System. Upon failure of the primary AC power supply, input power for the inverter is automatically supplied from the connected battery bank. If input AC power does not return (i.e., complete loss of electrical power to the facility), the UPS will automatically secure when the discharge limit of the battery bank is reached (approximately two hours at a typical load current of 60 amps and a rated battery bank life of 120 amp-hours). An alarm circuit provides local indication of an abnormal condition in the operation of the UPS system. The circuit also provides a common output signal which activates an audible and visual alarm in the reactor control room.

The MURR UPS is not required for safe reactor shutdown or maintenance of safe shutdown conditions. However, the UPS system does supply power to the necessary reactor instrumentation (e. g., rod position indication, power level, etc.) to affirm a complete reactor shutdown. Should a failure of the UPS occur, a reactor shutdown may also be affirmed upon visual observation that the control blades are fully-inserted and that reactor power has been reduced by noting a reduction or decrease in the intensity of the Cerenkov radiation in the region of the reactor core. The UPS also supplies power to the Area Radiation Monitoring System (ARMS) to ensure radiation levels are known.

The MURR electrical distribution system is shown in Figure 8.2. This system will not interfere with safe shutdown or lead to reactor damage if the system malfunctions during normal reactor operation.

8.1.3 Surveillance

The remote alarm system for the UPS system is periodically tested for operability.

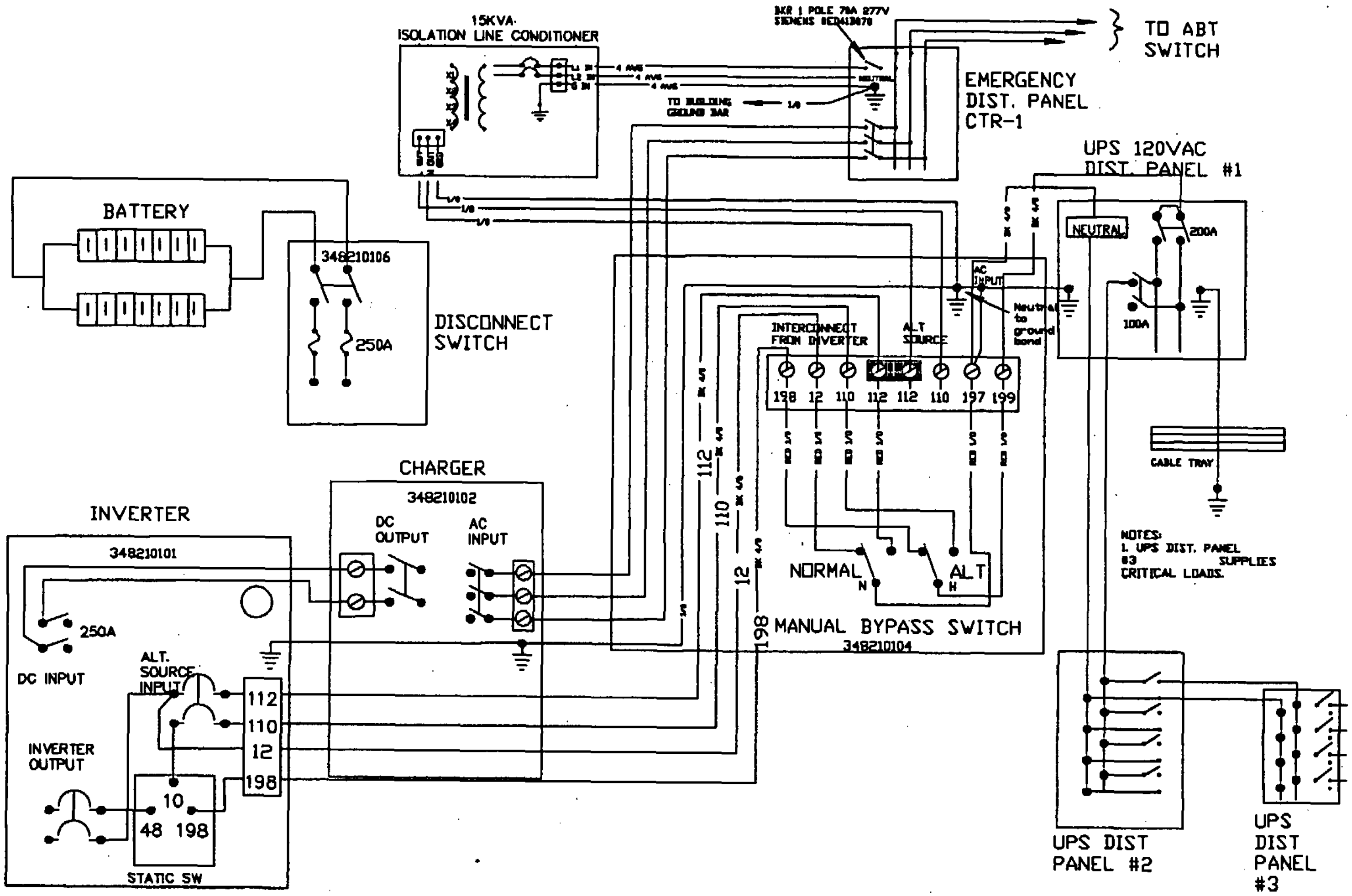

FIGURE 8.1
UNINTERRUPTIBLE POWER SUPPLY

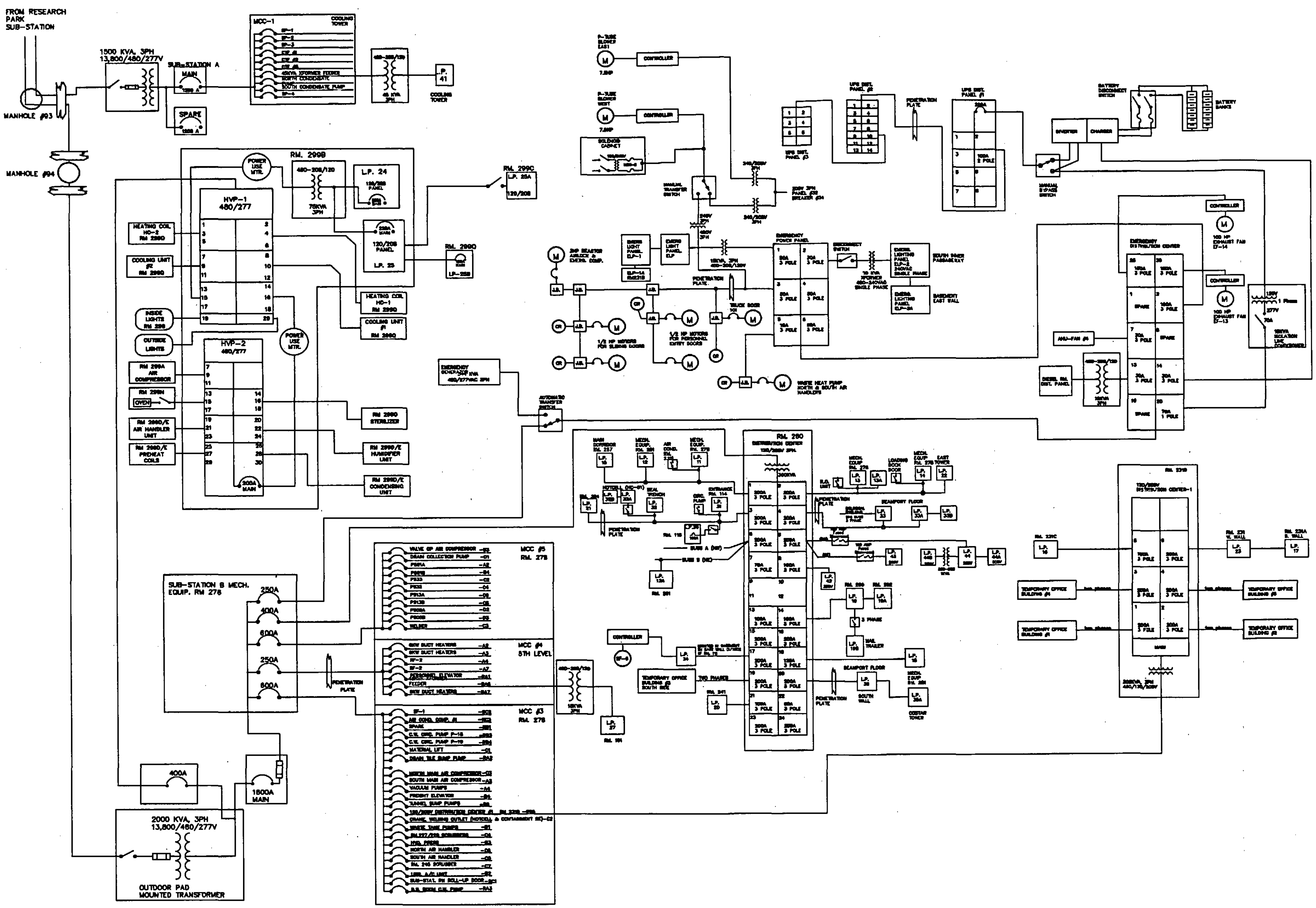

FIGURE 8.2
MURR ELECTRICAL DISTRIBUTION SYSTEM

8.2 Emergency Electrical Power System

8.2.1 Introduction

The emergency electrical power system is designed to provide electrical power to essential reactor components in order to monitor systems and assure personnel safety should the facility suffer a loss of normal electrical power. The system design is basic and reliable, ensuring availability, if needed. The emergency electrical power system will not interfere with safe shutdown or lead to reactor damage if the system malfunctions during normal reactor operation.

Emergency power is supplied by a 275-kW diesel generator located in the southwest area of the laboratory building (Room 231E). Electrical power is supplied to the selected loads through an Automatic Transfer Switch (ATS) which transfers source power from the normal electrical power system to the emergency electrical power system.

The design of the MURR does not require electrical power to safely shut down the reactor or to maintain an acceptable shutdown condition. In addition, the emergency electrical power system is not required for protection of the integrity of the fuel elements. A loss of normal electrical power and a subsequent failure of the emergency power generator to start (i.e., complete loss of electrical power to the facility) is analyzed in Chapter 13, Accident Analyses.

8.2.2 Emergency Power Generator

The emergency power generator is driven by a water-cooled Cummins turbo- charged diesel engine. It is a 855-cu in, 395-HP, six-cylinder diesel unit with a mechanically-driven fuel injection system. Fuel is stored in a skid-mounted tank with a capacity of 270 gallons (1,022 l). This capacity allows for a continuous run time of approximately 10 hours under a full load condition. Two parallel sets of two 12-volt lead-acid batteries connected in series generate 24 volts for the starting system. A static-type, dual rate, float/equalizer charger with automatic and manual charge control maintains the startup batteries fully-charged.

This four pole generator is rated for an output of 344 kVA (275 kW at 0.8 PF), 277/480-volt, three-phase, 60-cycle electrical power. It is equipped with a brushless permanent magnet exciter. The design of the exciter and regulator provides for voltage regulation of less than ± 2%. Stable generator output voltage and frequency are established within two seconds after the transition between no load and full load conditions. The unit is designed to assume full load within seven seconds from a cold start and is sized to meet current and anticipated loads with an excess capacity approaching 50%.

8.2.3 Automatic Transfer Switch

The automatic transfer switch (ATS) transfers the emergency electrical load between normal and emergency power. The transfer switch is equipped with the following time delays as described.

(1) Start Time Delay - Adjustable from 0 to 15 seconds. This time delay prevents the diesel generator from starting during power interruptions of a short duration. The timing starts the moment the normal power source is interrupted.

(2) Stop Time Delay - Adjustable from 0 to 10 minutes. This time delay allows the diesel generator to cool down with no load before stopping. The timing starts when the emergency electrical load is retransferred to the normal power source.

(3) Transfer Time Delay - Adjustable from 0 to 120 seconds. This time delay allows the diesel generator to reach rated voltage and frequency prior to transfer to the emergency bus. This allows the diesel generator to stabilize before the emergency electrical load is applied.

(4) Retransfer Time Delay - Adjustable from 0 to 30 minutes. This time delay controls retransfer once the normal power source returns. This allows the normal power source to stabilize before the emergency electrical load is applied.

To prevent simultaneous closing to both power sources, the transfer switch is mechanically interlocked. In addition to the adjustable time delays for starting, stopping, and transfer, there is an exerciser clock which automatically starts and runs the emergency generator once a week for 30 minutes under no-load conditions to verify its operability.

8.2.4 Emergency Power Loads

The emergency power bus is routed through the ATS to an emergency distribution panel (CTR-1) as shown in Figure 8.3. This distribution panel feeds the following emergency electrical loads:

1. Facility Exhaust Fan EF-13;
2. Facility Exhaust Fan EF-14;
3. Diesel Generator Room Distribution Panel;
4. Emergency Power Panel:
    a. Emergency Air Compressor;
    b. Electric-motor-driven Containment Isolation Doors 504 and 505;
    c. West Pneumatic Tube Blower;
    d. Emergency Lighting Panel No. 2 (through a 10-kVA transformer);
    e. Waste Heat Pump No. 2; and
    f. Emergency Lighting Panel (through a 15-kVA transformer):
        1) Exit Lights;
        2) Stairway Lights;
        3) Fan Failure Alarm;
        4) Intercommunications System;
        5) Off-Gas Stack Monitor;
        6) Evacuation/Isolation Alarm System; and
        7) Emergency Lighting Panel No. 1;

5. 120 VAC Loads through 3 Distribution Panels:
   a. Area Radiation Monitoring System (ARMS);
   b. Annunciator Panel;
   c. Neutron and Process Monitoring Instruments; and
   d. Reactor Control Power (Control Rods, Rod Run-In System, Reactor Safety System, Servo Amplifier).

8.2.5 Description of Operation

Operation of the emergency power generator, and the transfer of electrical power from the normal source to the emergency power bus to supply the emergency electrical loads, is automatic. The emergency power generator starts approximately one second after a loss of the normal electrical power source. After reaching rated voltage and frequency, it will assume the emergency electrical load after the ATS shifts to the emergency power bus. Upon restoration of the normal electrical power source, the emergency load will be transferred back to the normal source after an adjustable time delay. After an additional adjustable time delay, which allows the unit to cool down with no load, the emergency power generator will shut down and assume its normal standby mode.

8.2.6 Limiting Conditions for Operation

The limiting condition for operation of the emergency power generator and its bases are discussed in the Technical Specifications.

8.2.7 Surveillance

The ability of the emergency power generator to assume the emergency load is tested, and normal operability checks are performed, as required by the Technical Specifications. The operability checks indicate that the emergency power generator is available for automatic operation prior to startup of the reactor. These checks also provide reasonable assurance that the emergency power generator will remain available throughout an extended reactor run. The periodic load test provides reasonable assurance that the emergency power generator electrical control and distribution system will remain operable. Periodic maintenance is performed according to the manufacturer's recommendations.

8.2.8 Design Features

The specifications and bases for the design features of the emergency electrical power system are discussed in the Technical Specifications. The objective of these specifications is to ensure that adequate emergency electrical power is supplied to the facility in the event of normal electrical power failure.

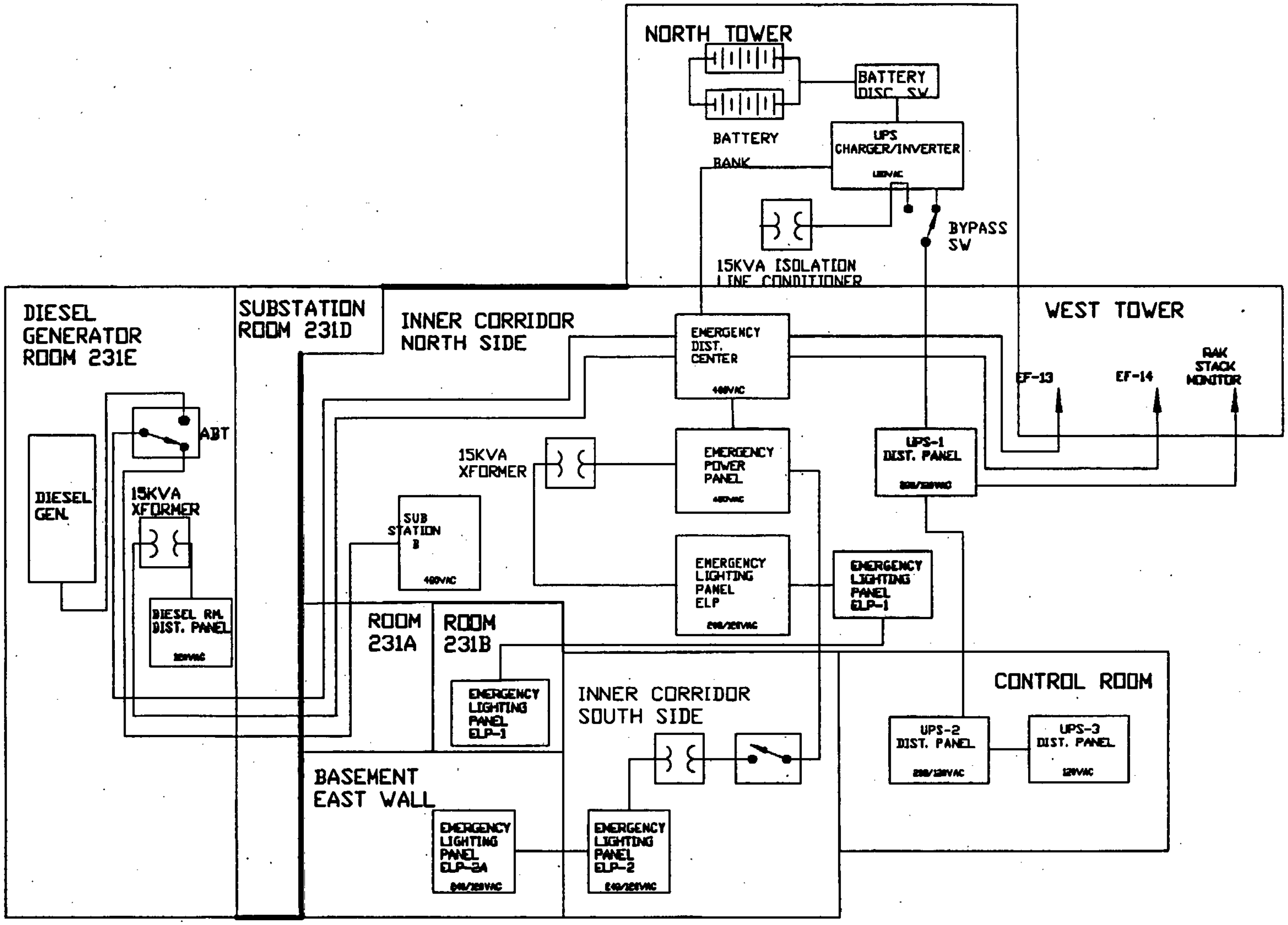

FIGURE 8.3
MURR EMERGENCY ELECTRICAL DISTRIBUTION SYSTEM

CHAPTER 9

AUXILIARY SYSTEMS

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9.3 Hazards Summary Report, Addendum 1, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, February 1966.

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9.7 Hazards Summary Report, Addendum 5, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, January 1974.

9.8 Bono, R., "Reactor Fuel Dose Measurements Using Lithium Fluoride Thermoluminescent Dosimeters," University of Missouri Research Reactor Facility, 1979.

9.0 AUXILIARY SYSTEMS

This chapter discusses the auxiliary systems which support operation of the reactor facility. Auxiliary systems are those systems not fully described in other chapters of the SAR that are important to the safe operation and shutdown of the reactor and to the protection of the health and safety of the public, the facility staff, and the environment.

9.1 Heating, Ventilation, and Air-Conditioning Systems

9.1.1 Introduction

The Missouri University Research Reactor (MURR) heating, ventilation, and air-conditioning systems provide heated and conditioned air for the laboratory and reactor containment buildings, creating an acceptable working atmosphere for personnel and equipment. In addition, the facility ventilation system (Figure 9.1) is designed[1] to perform the following radiological control functions:

- Maintain the reactor containment and laboratory buildings at a slightly negative pressure with respect to the surrounding environment to prevent the spread of radioactive contamination;

- Provide the necessary air exchanges to ensure that concentrations of radioactive gases in the laboratory and reactor containment buildings are maintained at levels which are below the limits of 10 CFR 20, Appendix B, Table 1 for restricted areas;

- Ensure that maximum dilution of potentially contaminated air is attained, resulting in minimum concentrations of radioactive gases being released to the environment; and

- Continuously monitor all radioactive gases discharged through the facility ventilation exhaust stack.

The ventilation system also prevents the uncontrolled release of radioactive materials to the environment in the event of an accident. This function is discussed in greater detail in Chapter 6, Engineered Safety Features.

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[1]The original safety evaluation of the MURR is documented in the Preliminary Hazards Report (Ref. 9.1), the Hazards Summary Report (Ref. 9.2), and Hazards Summary Report, Addenda 1-5 (Ref. 9.3-9.7).

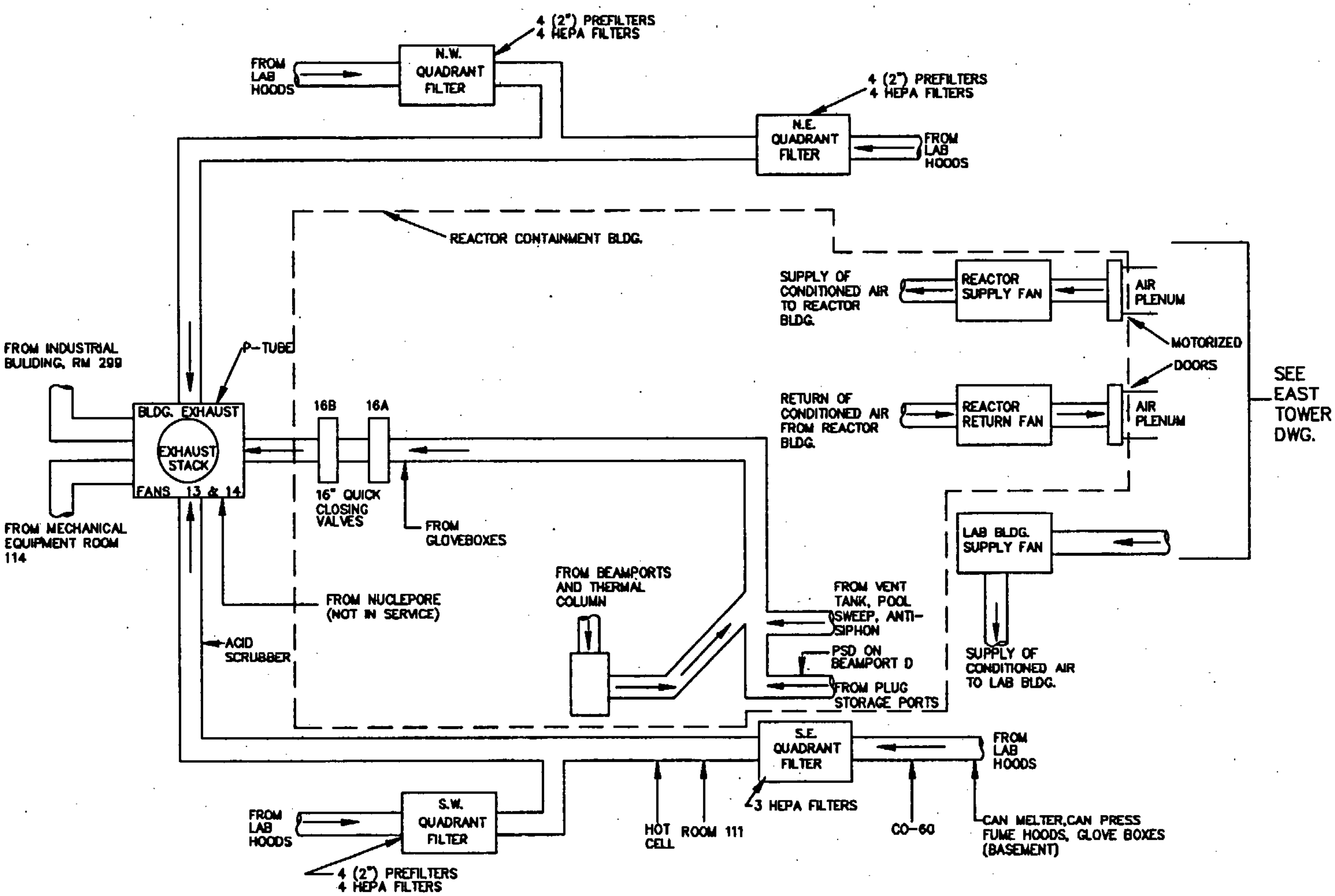

FIGURE 9.1
MURR FACILITY VENTILATION SYSTEM

9.1.2 Description of Operation

9.1.2.1 Supply System

Fresh air is supplied to both the laboratory and reactor containment buildings through louver dampers on the north and south facades of the reactor building east tower. Most of the supply air entering the east tower is diverted to the laboratory building; a smaller portion is for make-up air to the containment building. Figure 9.2 shows the air flow paths through the reactor containment building east tower.

Outside air is preheated, as necessary, by a steam coil and then filtered before entering the laboratory building via the laboratory building supply fan (SF-1). The supply air is then directed to two receiving plenums (hot and cold decks), each containing an independent coil system. Steam is supplied to the hot deck to heat the supply air; chill water from the air-conditioning system is supplied to the cold deck to cool the air. The heated and conditioned air from these plenums is then circulated throughout the laboratory building via a double duct air distribution system. A ceiling register and mixing box in each laboratory and office allows the hot and cold air from the distribution system to be combined and circulated within that space to create a suitable environment for personnel comfort and equipment cooling.

Additional fresh air is supplied to the laboratory building through two roof top air handlers (RTAHs) located on the laboratory roof above the north and south outer corridors. Each RTAH has a heating and a cooling coil to heat and/or condition the supply air before its discharge into the laboratory building through registers in the ceilings of these corridors.

A third RTAH services an addition attached to the southwest corner of the laboratory building which houses the 275-kW emergency diesel generator. Fresh heated and/or conditioned air is supplied to this space through the RTAH and subsequently removed by a roof ventilator.

Ventilation for the reactor containment building is primarily recirculation with only a small amount of air being exhausted through a 16-inch line. Air from the containment building is returned to a plenum in the east tower at a rate of approximately 17,500 cubic feet per minute (cfm) by the reactor containment building return fan (RF-2). Approximately 2,500 cfm of fresh make-up air is mixed with the return air before passing through a dust filter and a set of heating and cooling coils. The mixed air then collects in a supply plenum before entering the containment building via the reactor containment building supply fan (SF-2) at a rate of approximately 20,000 cfm. The supply and return fans are interlocked such that securing SF-2 will automatically secure RF-2, thus preventing an extreme negative pressure condition from occurring within the containment building. However, operation of SF-2 with RF-2 secured is allowed.

The supply and return plenums for the reactor containment building each contain two isolation doors. One door is air-operated-to-open, gravity-to-close, and is designated as a "back-up door." The other door is driven open and closed by a 3-phase motor connected to a chain drive

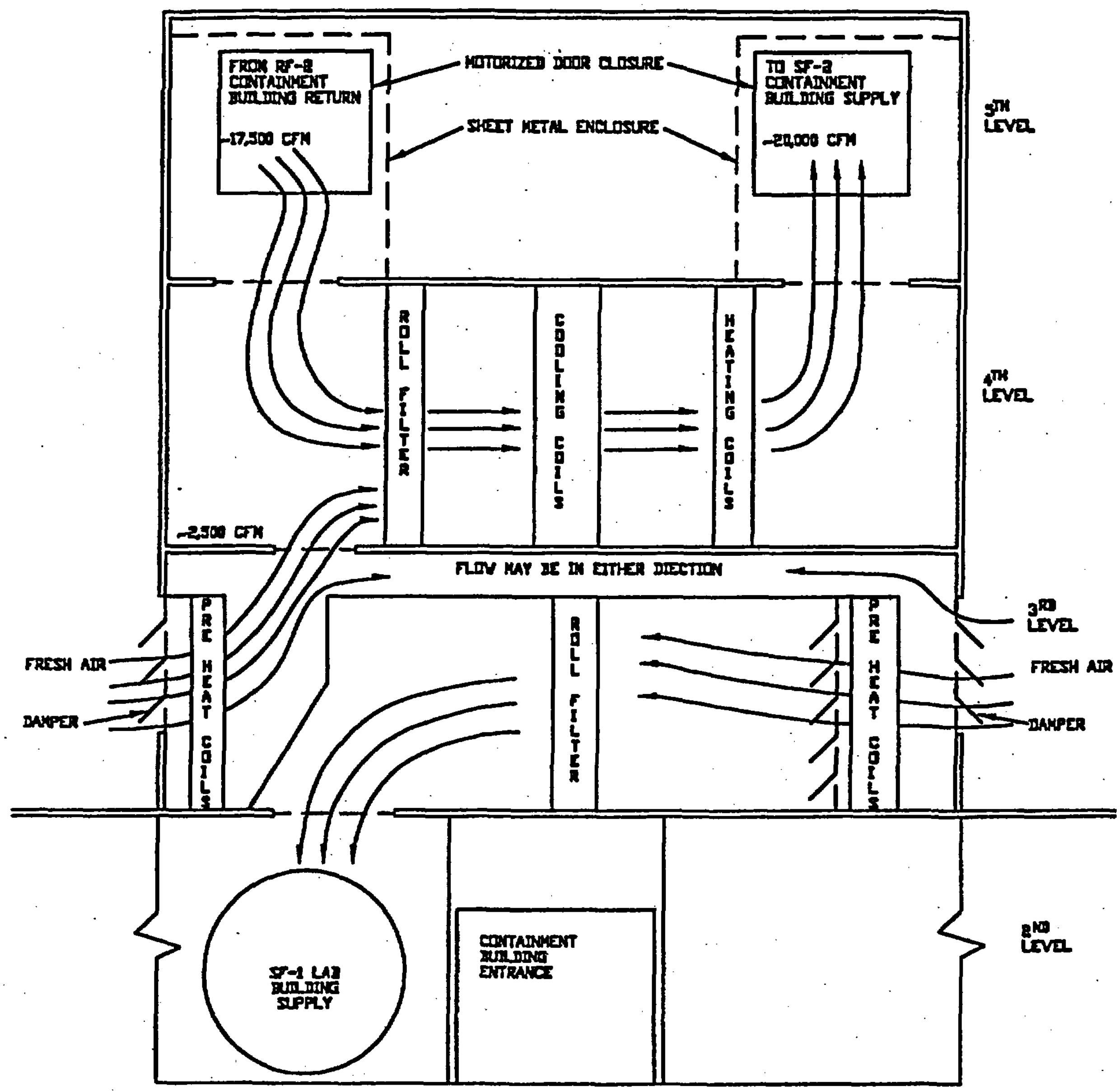

FIGURE 9.2
MURR CONTAINMENT AIR FLOW PATHS

assembly. The motorized door for the supply plenum is designated as Door 504 and the motorized door for the return plenum is designated as Door 505. Closure of either door will secure both RF-2 and SF-2, preventing operation of the fans with no flow path. The isolation doors are discussed in greater detail in Chapter 6, Engineered Safety Features.

9.1.2.2 Exhaust System

Exhaust air from both the laboratory and reactor containment buildings is combined in an exhaust plenum prior to being discharged to the atmosphere. Since air from both buildings is never mixed until this point, potentially contaminated air is diluted by mixing with uncontaminated air, resulting in minimum concentrations of radioactive gases being released to the environment.

The exhaust system for the laboratory building is divided into four quadrants, each servicing approximately one quarter of the building. Each quadrant consists of a stainless steel filter housing containing a bank of pre-filters and a bank of high efficiency particulate air (HEPA) filters. Air is ducted from the quadrants to an exhaust plenum located in the reactor building west tower. Laboratory building exhaust fans (EF-13 and EF-14), located within the exhaust plenum, discharge the air through the facility exhaust stack to the atmosphere. The top of the exhaust stack is approximately 70 feet (21 m) above grade level of the containment building. One exhaust fan is in operation while the other is in standby. This condition is indicated by a green light on the fan failure alarm panels located in the reactor control room and the facility lobby (Room 202). Any condition other than one fan in "fast speed" and the other in "stand-by" will de-energize the green light and indicate an abnormal condition. Malfunction of the operating fan will automatically start the standby fan and de-energize the green light indicating a loss of the operating fan. Failure of both fans, or significantly degraded flow, actuates a pressure switch which initiates an audible alarm in the reactor control room.

Exhaust air from the mechanical equipment room (Room 114), MURR Industrial Building (Room 299), hot cell, and the pneumatic tube system passes through separate filtration systems before discharging into the exhaust plenum.

Reactor containment building exhaust air discharges to the exhaust plenum at approximately 2,500 cfm through a 16-inch line which penetrates the west containment building wall. This line contains two quick-closing isolation valves designated 16A and 16B. Exhaust air from areas which produce radioactive gases or airborne contamination is ducted to the 16-inch line. These areas include the reactor pool sweep system, the beamport storage ports (hot storage ports), the thermal column, and the beamport vestibules. The hot exhaust line isolation valves (16A and 16B) are discussed in greater detail in Chapter 6, Engineered Safety Features.

9.1.3 Surveillance

The reactor control room Fan Failure Alarm Panel is periodically tested for operability.

9.2 Handling and Storage of Reactor Fuel

9.2.1 Introduction

The fuel handling system provides a safe, effective, and reliable means of transporting and handling reactor fuel from the time it enters the facility until it leaves. [redacted] All fuel handling is performed in accordance with Special Nuclear Material (SNM) Control and Accounting Procedures and as outlined in the Operations Procedures. These procedures provide reasonable assurance that all special nuclear material will be accounted for and that fuel will meet procurement specifications. Prior to being irradiated, fuel may be handled without special equipment or radiological precautions. Irradiated fuel is handled with specially designed remote tools.

Fuel, new or irradiated, may be stored in any one of the 88 in-pool fuel storage locations (not including the core). These storage locations are designed to the following specifications:

(a) A geometry such that the calculated K_{eff} is less than 0.9 under all conditions of moderation and irrespective of the number of fuel elements stored or the amount of burnup per element;

(b) Sufficient natural convection cooling to prevent a fuel element from exceeding its design temperature;

(c) Location within the reactor pool at a sufficient depth to provide adequate radiation shielding;

(d) Arrangement in the reactor pool to permit efficient handling during the insertion, removal, or interchange of the fuel elements; and

(e) Fabrication from materials compatible with the fuel elements.

The fuel handling cycle consists of (1) receiving new, unirradiated fuel elements, (2) transferring the new fuel elements to the fuel storage vault, (3) transferring the new fuel elements from the storage vault to the reactor pool, (4) unloading irradiated fuel elements from the reactor into the in-pool storage locations, (5) loading new or previously irradiated (partially used) fuel elements from the in-pool storage locations into the reactor, and (6) transferring the spent fuel elements from the in-pool storage locations to the fuel transfer cask for shipment.

There are no specific accidents in this type of research reactor associated with the storage of spent fuel when stored in accordance with the Technical Specifications.

9.2.2 Methods of Storage and Transfer

New, unirradiated fuel arrives at the facility in Department of Transportation (DOT) approved shipping containers. The new fuel [redacted], is removed from the shipping containers by hand and placed in a storage vault [redacted]. The storage vault is equipped with special racks that allow storage of the fuel in a criticality-safe geometry such that the calculated K_{eff} is less than 0.9. All new fuel elements are inspected before being placed in the reactor. During inspection, fuel elements are handled singly and are replaced to their proper storage location prior to inspecting a different element. The Assistant Reactor Manager-Physics ensures that a complete and accurate fuel inventory is maintained.

As new fuel is required for loading into the reactor, each element is transported individually from the storage vault to the reactor pool. Each storage location has adequate clearance for inserting or withdrawing a fuel element without interference. Only one element is allowed out of any storage location, or its position in the reactor, at any given time. The Assistant Reactor Manager-Physics, or an authorized delegate, provides a step-by-step fuel movement procedure anytime fuel is handled.

The reactor must be shut down and the reactor pressure vessel cover removed before transferring fuel from the reactor. The normal fuel handling tool is air-operated with a buoyancy assist tank. The tool is designed to provide a positive indication of latching prior to movement of an element. This feature and indication of the unlatched condition of the tool are tested prior to any fuel handling sequence. The tools for inserting and removing fuel from the core are specifically designed to avoid damaging a fuel element. Reactor operators, working from the upper bridge, move fuel out of the core to designated storage locations in the reactor pool. During all fuel handling, water in the reactor pool is maintained at a level to provide maximum shielding for the operators. The highest dose rate during fuel handling is approximately 25 millirem/h during the short interval of time when an element is passed into the weir area for storage, or when an element is raised to clear the top of the pressure vessel. These dose rates are consistent with the facility ALARA program. Following each refueling sequence, a verification is performed to ensure that the fuel elements are fully seated, and a visual inspection of the core is performed to verify that no debris is present. These steps are completed prior to bolting the reactor pressure vessel cover in place. A post-refueling map check is also performed to verify that the appropriate positions in the visible in-pool storage baskets contain fuel elements as indicated on the fuel location map.

Irradiated fuel does not leave the facility until it is loaded into a U.S. Nuclear Regulatory Commission (NRC) approved cask for shipment. Transfer of spent fuel from in-pool storage locations to the cask is done manually with the cask underwater and resting on the shelf behind the weir. The 15-ton capacity overhead rectilinear crane is used to lower and remove the cask from the reactor pool. The cask is decontaminated prior to release for shipment.

A spent fuel element is not loaded into a shipping cask for shipment until a predetermined cooling period has elapsed since the element was last removed from the reactor core. Cooling times are based on a thermal analysis[1] of the decay heat generated by a spent fuel element and by the storage requirements at the Department of Energy (DOE) site. The thermal analysis is included in a separate Safety Analysis Report for Packaging (SARP) for each cask. The Certificate of Compliance for the shipping cask will state the required minimum cooling time.

All cask lifting equipment, including the 15-ton capacity crane, is rigorously maintained, including preventive maintenance and magnetic particle testing (MT) as appropriate. A dye penetrant inspection is also performed on the shipping cask. Therefore, the dropping of a spent fuel shipping cask during fuel transfers is considered a highly unlikely event.

If the air-operated fuel handling tool becomes inoperative, the Reactor Manager or the Assistant Reactor Manager-Operations may authorize the use of a manually-operated fuel handling tool. Fuel movements using this tool are performed with the pool level lowered to the height of the lower bridge with an operator stationed at this level. A special maintenance procedure is used in conjunction with the Operations and the SNM Control and Accounting Procedures when using the manual tool.

9.2.3 Limiting Conditions for Operation

The limiting conditions for operation for the storage of reactor fuel outside the reactor core and their bases are discussed in the Technical Specifications. The objective of these specifications is to ensure that the reactor fuel is maintained within acceptable design considerations thus maintaining fuel integrity.

9.2.4 Surveillance

Fuel elements are inspected for anomalies as stated in the Technical Specifications. The objective of this specification is to reasonably ensure proper performance of the reactor fuel.

9.3 Fire Protection Systems and Programs

The MURR fire protection system and program are designed to protect the facility and staff, and to mitigate any property loss in the event of a fire. The system provides two primary functions: (1) detection, which affords an early warning of an actual or potential fire condition by a combination of heat, smoke, and remote manual devices, and (2) suppression, which incorporates a normal sprinkler system with a pre-action system that is used in areas with sensitive electronic equipment, and a deluge, non-freezing system used in the cooling tower. It should be noted that fire protection is not required to accomplish a safe shut down of the reactor or to maintain a safe shutdown condition.

The fire detection system is a combination of thermal, photoelectronic, and ionization-type

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[1]See Footnote on Page 9-1.

sensors, flow switches, and manual pull stations. A central control station, located in Room 204, and a repeater station, located in the reactor control room, monitors each system component. These stations will annunciate an alarm if any component is not in its normal condition.

The fire suppression system is a combination of many types of systems: a deluge system used in the cooling tower; a pre-action system used in areas that contain highly sensitive electronic equipment; a dry fire main system used in the reactor containment building; and a traditional sprinkler system used throughout the rest of the laboratory building. The containment building fire suppression system consists of three fire hose cabinets connected to a dry fire main. Cross-connecting it to the rest of the facility's wet system by a manual isolation valve located in the laboratory basement can flood this system.

The fire protection system receives a virtually unlimited supply of water from the combination University fire and domestic cold water main. Four (4) siamese or storz hose fittings are also connected to the MURR fire main. These fittings are located outside the facility and they facilitate connecting a pumper truck between the fittings and fire hydrants, which are located in the vicinity of the hose fittings, thus providing an additional water supply path to the fire main. In addition, fire extinguishers are strategically located throughout the facility.

The fire protection system is powered from the emergency electrical distribution system (ELP-2). The system also has a self-contained 24-hour battery backup.

The laboratory building is constructed of noncombustible materials such as concrete blocks with brick veneer exterior walls. Interior walls are mostly concrete block. The containment building walls are poured reinforced concrete. Exits from the facility meet the requirements of the National Fire Protection Association (NFPA) 101, "Life Safety Code." All areas in the facility have access to at least two exits, with the exception of the reactor containment building. The reactor containment building is considered "low hazard Industrial Occupancy" from the view point of fire safety. Only one exit is required for these spaces provided that the travel distance to the exit is adequate. NFPA 101, Table A-5-6.1, shows that a distance of travel to an exit for these spaces should be a maximum of 300 feet (91.4 m). All areas of the containment building are well within this distance to an exit.

Additionally, the MURR is located in a sparsely developed area. Furthermore, the facility site is not a wildland interface area, so no exposure threats from burning wildlands exist.

9.4 Communication Systems

The MURR utilizes two principal communication systems: a computerized telephone system and an intercommunication system which allows two-way communication between a master station and a staff station. A paging feature which allows several different telephones to originate a page is incorporated into the telephone system. The paging system allows a facility-wide announcement of an emergency. The communications systems have provisions for summoning emergency assistance from designated personnel, as detailed in the physical security and emergency plans.

Master and staff station locations for the intercommunication system are shown in Table 9-1. Any staff station may be called from the master control station and any staff station may call the master control station. Voice paging may also be accomplished from the master control station.

A third method of communication at the facility is by portable hand-held radio transmitter-receivers ("walkie-talkies"). This method of communication is mainly used during operational tests of facility equipment or during emergency conditions.

9.5 Possession and Use of Byproduct, Source, and Special Nuclear Material

In addition to the facility operating license (R-103), there is currently one other NRC-issued license granted to The Curators of the University of Missouri governing work at the MURR: the Broad Scope Material License (No. 24-00513-39). The Broad Scope Material License authorizes the receipt, possession, and use of radioactive materials throughout the reactor facility.

The physical protection system and the security organization which will detect the attempted theft or diversion of Special Nuclear Material (SNM) at the MURR are described in the MURR Physical Security Plan. It outlines the objectives that are met by the Security Procedures, a separate document which describes the security requirements and security measures for the facility.

9.5.1 Types and Quantities of Radionuclides

Typical of a research reactor of its size, the MURR handles or produces a spectrum of byproduct materials. The MURR has been licensed to handle isotopes ranging from tritium through transuranics. The MURR utilizes radioisotopes it produces by the activation of materials, radioactive sources, and that are used to calibrate and verify the operation of radiation detection instruments. Isotopes that the MURR is currently licensed to possess and use under the Broad Scope Material License are summarized in Table 9-2. Because byproduct material continuously changes as part of the normal operation of the reactor and the experimental program, the information presented in Table 9-2 should be considered representative rather than an exact listing.

TABLE 9-1
INTERCOMMUNICATION SYSTEM MASTER AND STAFF STATION LOCATIONS

Master Station	Location Description
Room 302	Reactor Control Console
Staff Stations	**Location Description**
Cooling Tower Basement	Mechanical Equipment Room Below-Grade Level
Cooling Tower Main Entrance	Door Entrance
Laboratory Building Roof	Door Entrance
Room 101	Beamport Area
Room 105	Laboratory Building Below-Grade Level
Room 114 (2)	Mechanical Equipment Room Below-Grade Level
Room 112	Demineralizer Area Below-Grade Level
Room 210	Laboratory Building Outer Corridor
Room 216	Laboratory
Room 218	Laboratory
Room 227	Laboratory
Room 228	Laboratory
Room 242	Laboratory
Room 271	Machinery Shop
Room 282	Electronic Shop
Room 286	Personnel Airlock
Room 278	Mechanical Equipment Room
Room 285	Containment Building Lobby
Room 288	Health Physics Office
Room 307	West Tower Third Level
Room 503	Containment Building Fifth Level
Room 507	West Tower Fifth Level

TABLE 9-2
TYPICAL BYPRODUCT MATERIAL THAT MURR MAY POSSESS AND USE UNDER THE BROAD SCOPE MATERIAL LICENSE

Byproduct Material	Form	Quantity
Isotopes with atomic number 1-83	Any	3 Ci each, 15 Ci total
Hydrogen-3 (^{3}H)	Any	6 Ci
Nickel-63 (^{63}Ni)	Encapsulated metallic powder	300 Ci
Samarium-153 (^{153}Sm)	Any	40 Ci
Uranium-238 (^{238}U) (depleted)	Any	500 grams
Neptunium-237 (^{237}Np)	Any	5 mCi
Plutonium-239 (^{239}Pu)	Any	10 grams
Plutonium-240 (^{240}Pu)	Any	10 grams
Americium-241 (^{241}Am)	Any	10 Ci
Cesium-137 (^{137}Cs)	Sealed source	4 Ci
Californium-252 (^{252}Cf)	Sealed source	23 μgrams
Polonium-210 (^{210}Po)	Activation product	1 mCi

9.5.2 Rooms, Spaces, and Equipment

Radioactive materials may be used in any room or area designated by MURR management and approved by Health Physics. Generally, these rooms or areas are laboratories specifically-designed and constructed for such use. Laboratories typically contain a fume hood and sink which are connected to the facility ventilation exhaust and liquid waste retention systems. Air exiting the facility through the ventilation exhaust system is continuously monitored for airborne radioactivity. Areas which are not specifically-designated as laboratories are evaluated on a case-by-case basis by the Health Physics staff to ensure that they are a safe environment for the work planned or conducted in that area.

While a complete list of equipment in each room would be unnecessary and subject to constant change as determined by the experimental program, each room or area could have some, all or none of the following equipment:

- Radiation detection equipment such as friskers, dose rate meters, and High Purity Germanium (HPGe) gamma-ray spectroscopy detectors;

- Gas-flow proportional swipe counters;
- Liquid scintillation counters;
- Cabinets for the storage of check sources and calibration sources; and
- Lead caves for the storage of activated materials produced during analysis activities or for the storage of materials during the analytical process.

9.5.3 Procedures

Procedures developed for the possession and use of byproduct material under the facility operating license can be divided into the following three (3) categories: Operations, Health Physics, and Project Specific. Operations' procedures are used to control the introduction of new or similar uses of radioactive materials that are introduced into the reactor experimental facilities. The Reactor Manager reviews the proposed use of materials and the accompanying procedure to ensure the physical safety of the reactor and its components. Health Physics' procedures are developed by the Reactor Health Physics Manager and serve to direct the overall use and control of radioactive materials as they relate to facility and personnel safety. Project Specific procedures are developed to ensure operational and personnel safety based upon specific project needs. Individual projects are reviewed either by the Radiation Safety Committee (see Section 11.1.2.3), if the project falls under the Broad Scope Material License, or the Isotope Use Subcommittee (see Section 12.2.5.4) if the project is related to the Facility Operating License.

Health Physics procedures are issued by the Reactor Health Physics Manager to ensure that all facility-wide operations are conducted in a manner that promotes the ALARA philosophy. These procedures spell out how the Health Physics staff provide support to the Operations and Research staff and how duties and functions necessary for the maintenance and support of the Technical Specifications are carried out and assigned. Additionally, Health Physics procedures provide the basis for general radiation protection throughout the facility.

9.5.4 Uses of Radioactive Materials

Radioactive materials usage can be divided into two categories: research and development, and processing. Research and development radioactive materials are generally those materials that are used to investigate the properties of radionuclides which may be beneficial to some discipline which studies the effects and potential uses of that material. Examples would be the production of isotopes for the determination of their potential cancer fighting properties or the detection of impurities in irradiated silicon ingots to identify better methods of silicon production in order to enhance silicon's ability to be used in the semiconductor industry. Another use of the reactor is the production of radioactive materials utilizing the pneumatic tube system. Trace elemental analysis is also done, both on a research and development or purely a research basis.

Production of radioactive material includes those materials that are beyond the initial research and development phase and show more promise in, for example, the cancer-fighting properties of a certain radiolabeled compound. Many of the current routinely-produced radioactive compounds at the MURR have found regular use in cancer treatment. In addition, compounds and isotopes are produced which are destined for further chemical processing which eventually find their way into the basic life sciences research and development community, both domestically and internationally.

The MURR has a fully-staffed shipping department capable of shipping radioactive materials within the United States and abroad. Shipments can include isotopes produced during routine production runs as well as small quantity research and development isotopes where an end user or additional researcher(s) may wish to continue testing the subject material. Shipments are made according to U.S. Department of Transportation (DOT) or International Air Transport Association (IATA) regulations depending on the modality of the shipment. The MURR also has the capability to ship Type A, Type B, and Limited Quantity material as well as Surface Contaminated Objects (SCO) and Low Specific Activity (LSA) waste.

9.6 Nitrogen Supply System

9.6.1 Introduction

The nitrogen supply system supplies nitrogen gas to the pressurizer to ensure that pressure in the primary coolant system is maintained within the Limiting Safety System Settings (LSSSs) for both 5- and 10-MW operation. Pressurizer pressure is maintained at 85 psia (586 kPa) for both Mode I and Mode II operation.

9.6.2 Description of System

The nitrogen supply system consists of two banks of nitrogen cylinders, with each bank comprised of three cylinders connected in parallel. One bank is in service while the other bank is in standby. Nitrogen cylinder pressure is normally 2,200 psig (15 MPa above atmosphere) when first placed in service. This pressure is reduced to around 140 psig (965 kPa above atmosphere) by a pressure regulator at the nitrogen station. The nitrogen gas is then piped to the mechanical equipment room (Room 114) where it is further reduced to approximately 95 psig (655 kPa above atmosphere) by a pressure regulator upstream of the pressurizer nitrogen addition valve V526. Nitrogen gas is added to the pressurizer as required by a control signal from pressure switch 941 signaling valve V526 to open.

The control system for the nitrogen station switches the nitrogen supply to the standby bank if pressure falls below 120 psig (827 kPa above atmosphere) in the on-service bank. If nitrogen pressure continues to lower, a "Nitrogen System Lo Press" annunciator alarm occurs at 115 psig (793 kPa above atmosphere), indicating low nitrogen system pressure.

9.7 Emergency Pool Fill System

9.7.1 Introduction

The emergency pool fill system provides an emergency raw water supply deliverable at a rate in excess of 1,000 gpm (3,785 lpm) to the reactor pool in the event of a leak in the reactor pool or pool coolant system. There are no functions or malfunctions of the emergency pool fill system that could initiate a reactor accident, prevent safe reactor shutdown, or initiate the uncontrolled release of radioactive material.

9.7.2 Description of System

The emergency pool fill system consists of an 8-inch wet fire line that extends from the fire protection water loop outside the perimeter of the laboratory building, reducing to a 6-inch line in the laboratory basement, before entering the reactor containment building by way of the utility entry water seal (seal trench). This line is further reduced to a 4-inch line in the containment building just before entering the biological shield and terminates at a goose-neck extending over the ledge of the reactor pool. The opening of the goose-neck is approximately 10 inches (25.4 cm) above the ledge, precluding the possibility of any siphoning action of reactor pool water into the emergency pool fill system.

The emergency pool fill system is actuated by the opening of a 4-inch ball valve, located in a recessed box adjacent to the east end of the reactor pool. This valve requires operation of only a quarter turn from closed to full open.

Water pressure for the emergency pool fill line is monitored by a spirahelic pressure indicating transmitter located in the laboratory building basement. This instrument provides local indication on an analog meter and also produces an electrical output signal displayed on a digital meter located in the reactor control room. This meter is equipped with an adjustable set point trip that provides an audible and visual alarm upon detection of low water pressure.

9.7.3 Water Sources

The University of Missouri at Columbia water supply system provides a virtually unlimited source of raw water for the emergency pool fill system. Five deep wells, each with varying flowrates, supply water to a 10-inch fire main that services the campus. The Southwest well, which is located approximately two hundred feet (61 m) south of the reactor facility, provides water at a flow rate of 1,000 gpm (3,785 lpm) to maintain a 1.5-million gallon (5.7-million l) reservoir near capacity. Three pumps, each with a 1,000-gpm (3,785-lpm) capacity, take suction from this reservoir and discharge into the 10-inch main to provide a portion of the campus water supply. The 8-inch wet fire line that provides the flow path for the emergency pool fill system is connected to this 10-inch main. To ensure that a continuous supply of water to the emergency pool fill system is maintained, a 1,000-kW diesel generator provides emergency electrical power to the supply pumps should a loss of normal electrical power occur.

In addition to the five deep wells, a 10-inch main from the City of Columbia water supply system can be directed either to the 1.5-million gallon (5.7-million l) reservoir or into the campus system. This provides an additional source of water for the emergency pool fill system.

9.7.4 Basis

The emergency pool fill system is capable of supplying raw water to the reactor pool at a rate in excess of 1,000 gpm (3,785 lpm). This rate of water addition is adequate to maintain greater than 3 feet (0.9 m) of water above a completely-severed 6-inch beamport with no flow impediments in the port.

9.7.5 Limiting Conditions for Operation

The limiting condition for operation of the emergency pool fill system is discussed in the Technical Specifications. The objective of this specification is to protect the fuel integrity and personnel.

9.7.6 Surveillance

Normal operational checks of the emergency pool fill system are performed as stated in the Technical Specifications. The objective of this specification is to reasonably assure proper operation of this auxiliary system.

9.8 Pool Skimmer System

9.8.1 Introduction

The pool skimmer system is designed[1] to remove particulates that may accumulate on the reactor pool surface at the normal operating level. The system also provides a means for lowering the pool water level to the elevation of the lower bridge in addition to providing a path for make-up water to the reactor pool during normal reactor operation. The pool skimmer system consists of a skimmer surface box, centrifugal pump, filter, and associated piping and valves. There are no functions or malfunctions of the pool skimmer system that could initiate a reactor accident, prevent safe reactor shutdown, or initiate an uncontrolled release of radioactive material.

[1]See Footnote on Page 9-1.

9.8.2 Skimmer Surface Box

The pool skimmer surface box is an aluminum box, 29¾ inches (75.6 cm) wide with a flexible floating bar which allows pool surface water to be skimmed. The skimmer box is adjustable in the vertical direction over a 10-inch (25.4-cm) span to allow for the most efficient positioning of the skimmer box relative to the reactor pool surface. A mesh screen strainer is installed in the box to prevent large objects from entering the pool skimmer system.

9.8.3 Pump

The pool skimmer pump is a horizontal, centrifugal, single-stage pump that is direct-connected to the driving unit through a flexible coupling. The pump is designed to provide sufficient discharge head to transfer water from the reactor pool to the demineralized water storage tank (T301) at a capacity of 50 gpm (189 lpm). The skimmer system pump is designated as P532.

9.8.4 Filter

The pool skimmer filter is a replaceable cartridge type, located in a stainless steel housing. The filter cartridges that are in use are sized to remove 1 micron or larger particles from the pool coolant. Isolation valves are installed to allow for filter cartridge replacement. The skimmer system filter is designated as F531.

9.8.5 Description of Operation

The pool skimmer system may be arranged to perform any of three operations: remove debris from the pool surface; lower pool water level to the elevation of the lower bridge; or provide a path for make-up water to the reactor pool.

When removing debris from the pool surface, the skimmer pump takes a suction on the skimmer box and circulates this water through a filter before it is returned to the reactor pool via a 2-inch aluminum header that encircles the reactor pool within the biological shield. A ½-inch line allows this return water to flow from the 2-inch header to a gap between the fixed port liner and the movable port liner for each beamport and then back into the reactor pool volume. This flow path prevents stagnation of pool water in these areas, helping to minimize corrosion to the fixed and movable beamport liners. In addition to return lines to each beamport, two ½-inch lines are routed to the spent fuel storage area ("Z" basket) in the reactor pool. These lines have been capped to prevent the possibility of siphoning water from the "Z" basket when the pool level is lowered.

Lowering reactor pool water level utilizing the skimmer system requires routing the suction of the skimmer pump from the skimmer box to a location immediately below the lower bridge. The valves required to perform this operation are located in the reactor pool approximately 4 feet (1.2 m) below the surface. Operation of these valves is accomplished from the upper bridge by the use of reach rods. During this operation, the discharge from the skimmer pump is directed to a demineralized water storage tank (T301) by two air-operated valves. One valve opens to allow a flow path to T301 and the other valve shuts, securing the normal discharge path back to the pool. Both valves are actuated by a single switch; ensuring only one valve is open at a time. The skimmer pump is then operated as required to lower the reactor pool water level to any height between the upper and lower bridge.

Make-up water is added to the reactor pool during normal operation through the pool skimmer system. Remote operation of an air-operated valve creates a path for water to gravity-flow from the demineralized water storage tank (T301) to the skimmer pump suction and then into the reactor pool. Operation of this valve is from the reactor control room. If water is required to be added at a faster rate, the skimmer pump may be started.

9.9 Closed Circuit Television System

An industrial type closed circuit television (CCTV) system is installed in the reactor facility to allow the reactor operators to monitor selected areas of the reactor containment and laboratory buildings. A selector box located in the reactor control room allows the reactor operators to select any one of up to six locations to be viewed. Four are presently used for reactor operations. Of these four, one camera is installed in the personnel airlock for surveillance and identification of personnel entering or leaving the containment building. A second camera is located on the beamport floor and can be positioned to view different areas of the floor. A third camera views the entrance to the laboratory building basement from the cooling tower tunnel. The fourth camera is located within the fuel storage vault.

A rack mounted 17-inch video monitor, designed to provide a clear bright picture of any camera location under continuous operation, is installed in the reactor instrument panel. The monitor is provided with manual control of horizontal and vertical hold, contrast, and brightness.

Additional security cameras are strategically located throughout the facility, both internally and externally, for the purpose of surveillance.

9.10 Battery Operated Emergency Lights

9.10.1 Description

Battery operated emergency lights are strategically positioned throughout the reactor containment and laboratory buildings. These battery operated emergency lights, in conjunction with selected lights powered by the emergency electrical power system, provide the lighting during a loss of normal electrical power. Each light has a self-charging battery pack and a switching circuit that actuates the emergency light upon electrical power failure.

9.10.2 Surveillance

All battery operated emergency lights are checked for operability on a periodic basis.

9.11 Radioactive Waste Disposal Systems

9.11.1 Introduction

Gaseous, solid, and liquid radioactive wastes are disposed of in accordance with 10 CFR 20. In addition to 10 CFR 20, procedures are established within the reactor facility to ensure proper handling, collection, and control of radioactive waste.

The Health Physics Branch maintains the responsibility for the monitoring and record keeping of the radioactive gases released through the facility ventilation and air treatment system, the solid radioactive waste program, and the radioactive liquid waste retention and disposal system. The radioactive waste programs are discussed in greater detail in Chapter 11, Radiation Protection Program and Waste Management. There are no functions or malfunctions of the radioactive waste disposal system that could initiate a reactor accident, prevent safe reactor shutdown, or initiate the uncontrolled release of radioactive material.

9.11.2 Gaseous Waste Disposal

Gaseous wastes (effluents) are disposed of through the facility ventilation and air treatment system described in Section 9.1. Air exiting the facility through the ventilation exhaust stack is continuously monitored for airborne radioactivity. Radioactive gases are diluted to minimum concentrations by the mixing the potentially contaminated air with uncontaminated air.

The maximum release of radioactive gases through the facility ventilation exhaust stack shall not exceed the limits as specified in the Technical Specifications. These limits ensure that exposure to the general public resulting from the radioactivity released to the environment will not exceed the limits of 10 CFR 20.

9.11.3 Solid Waste Disposal

Receptacles are located in laboratories and other work areas that create solid radioactive waste. The waste is collected on a routine basis and stored on the below-grade level of the laboratory building until a sufficient volume has accumulated to be packaged in sealed containers. The containers are then processed by a member of the Health Physics staff and prepared for shipment to a waste processing site prior to final disposal. In order to minimize the amount of radioactive waste generated, contaminated and noncontaminated material is segregated, and the efficient use of experiment material is encouraged.

9.11.4 Liquid Waste Disposal

All potentially contaminated liquids either drain or are pumped to a liquid waste retention system located on the below-grade level of the laboratory building. The liquid waste retention system consists of four tanks, three with a capacity of approximately 5,000 gallons (18,927 l) and the fourth with a capacity of 550 gallons (2,082 l), three transfer pumps, three filter banks, and associated piping and valves. Liquid waste is retained or chemically treated until an assay indicates activity levels are less than the limits for disposal specified in 10 CFR 20 and then is released into sanitary sewerage. In addition to having the activity levels measured, all liquid waste is circulated through a filter bank to ensure no suspended solids of visible size are present prior to release. It is standard practice to hold all liquid waste as long as practical to minimize the total activity released to sanitary sewerage.

9.12 Demineralized Water Supply System

9.12.1 Introduction

The demineralized water supply system provides primary grade water for daily reactor plant and facility operations. This system consists of three principle components: a water treatment system; demineralized water storage for reactor plant make-up; and demineralized water storage for general use throughout the facility. There are no functions or malfunctions of the demineralized water supply system that could initiate a reactor accident, prevent safe reactor shutdown, or initiate the uncontrolled release of radioactive material.

9.12.2 Water Treatment System

The water treatment system consists of a water conditioning unit, an activated charcoal filter, cartridge type pre-filters, a reverse osmosis (RO) unit, a mixed bed resin column, and associated piping and valves. The effluent from this system will contain no greater than 0.1 ppm total hardness at a conductivity of less than 2.0 μmho.

Raw water from the domestic hot and cold water systems is supplied to the water conditioning unit to generate soft water. The soft water is then directed to a charcoal filter and then four pre-filters for removal of chlorine and any suspended particles. This pretreatment of the supply water is necessary to ensure the RO unit will operate at its maximum efficiency. After being processed by the RO unit, the supply water is circulated through a mixed bed resin column to produce demineralized water at a flow rate of about 1.5 to 2.5 gpm (5.7 to 9.5 lpm). The water treatment system is operated as required to maintain a sufficient inventory of demineralized water for the facility and the reactor plant.

In addition to the water treatment system discussed above, an ion exchange demineralizer system (DI-300) serves as a backup. This system consists of a 300-gallon (1,136-l) rubber-lined carbon steel tank sized to hold 24 cubic feet (0.68 m^3) of mixed bed resin, filters before and after the demineralizer tank, and associated piping and valves.

Demineralized water from either system may then be directed to the water storage systems for the facility or the reactor plant.

9.12.3 Reactor Plant Make-Up Water Storage Tanks

Two lined carbon steel tanks, each with a capacity of 7,000 gallons (26,498 l), provide storage of demineralized make-up water for both the primary and pool coolant systems. These tanks, designated as T300 and T301, are located in the south tower of the reactor containment building, external to the containment structure.

A 2-inch line connects T300 to the suction of the primary coolant charging pump. Demineralized make-up water from T300 is added automatically to the primary coolant system by the primary coolant charging pump as required by system demand. Make-up water is added to the reactor pool during normal operation through the pool skimmer system. Demineralized water from T301 is gravity-fed to the skimmer system via a 1¼-inch line and into the reactor pool. A 4-inch line from T301 to the pool coolant system is also provided to allow rapid lowering or raising of pool water level when the reactor is shutdown. T300 and T301 may be cross-connected to provide a total capacity of 14,000 gallons (52996 l) for reactor pool water level lowering operations.

To provide local water level indication, pressure gauges calibrated in gallons are connected to the bottoms of both T300 and T301. Connected to these gauges are pressure switches with adjustable set point trips that initiate an annunciator alarm upon detection of low water level. Water level monitoring sensors are also installed in each tank, and will initiate an annunciator alarm upon detection of high water level.

9.12.4 Facility Demineralized Water Storage Tanks

Three water storage tanks, each with a capacity of 420 gallons (1,590 l), supply demineralized water for general use throughout the reactor facility. The tanks are constructed of 304 stainless steel and are connected in parallel, resulting in a total capacity of 1,260 gallons (4,770 l). A pressurization tank and pump ensure that sufficient demineralized water pressure is maintained at all service locations. A sightglass is installed on one of the storage tanks to provide local level indication. Level monitoring sensors provide an audible and visual alarm in the reactor control room upon detection of high and low water level. An automatic high level shutdown feature is incorporated into the RO unit control system to allow unattended filling of the facility demineralized water storage tanks.

The pressurization tank and pump and the facility demineralized water storage tanks are located in the north tower of the reactor building, external to the containment structure.

9.12.5 Limiting Conditions for Operation

Demineralized water is kept at an inventory as specified in the Technical Specifications. This ensures that an adequate supply of primary grade make-up water is available for reactor plant make-up during all modes of operation.

9.13 Reactor Loop Vent System

9.13.1 Introduction

The reactor loop vent system is designed to provide a path for the venting of any gases that may accumulate within the in-pool portion of the primary coolant system. This system consists of a vent tank, two solenoid-operated vent valves, two liquid level controllers, a pressure regulator, and associated piping and valves. There are no functions or malfunctions of the reactor loop vent system that could initiate a reactor accident, prevent safe reactor shutdown, or initiate the uncontrolled release of radioactive material.

9.13.2 Vent Tank

The vent tank is constructed of two lengths of 6-inch diameter aluminum pipe connected near the top and bottom by 3-inch diameter aluminum pipe. The tank has an overall length of 60 inches (1.5 m) and is located at the lower bridge level in the reactor pool. Located within the vent tank are two liquid level controllers, designated LC925A and LC925B, which control the operation of two solenoid-operated vent valves.

9.13.3 Description of Operation

The vent tank is connected to the highest points on the invert loop and in-pool heat exchanger. Gases in the primary coolant system will accumulate at these points and flow via a ½-inch line into the vent tank. Any gases collected in the vent tank are released through an exhaust line which diverges into two parallel ¼-inch lines, each containing a strainer, check valve, and solenoid-operated vent valve. These two lines then converge into a single line which contains a pressure regulator. The pressure regulator reduces the flow rate and pressure of the venting gas prior to exhausting through an absolute and charcoal filter. The solenoid-operated vent valves are designated V552A and V552B.

As gases are collected, the water level in the vent tank will recede. When the water level recedes to a predetermined value, level controller 925A will signal valve V552A to open and vent the collected gas. If the level continues to lower, level controller 925B will signal valve V552B to open. Operation of level controller 925B will also initiate an annunciator alarm and rod run-in. The rod run-in prevents operation of the reactor with a vent tank level which could result in the introduction of air into the primary coolant system.

9.14 Compressed Air Systems

9.14.1 Introduction

The MURR utilizes the following four compressed air systems: main, emergency, valve operation, and instrument. The main air system supplies compressed air for general use throughout the laboratory and the reactor containment buildings. The emergency air system provides a backup supply of compressed air to the isolation valves and the sealable closures of the containment isolation system. The valve operation air system supplies compressed air to the air-operated isolation valves of the reactor plant. The instrument air system supplies compressed air to the pneumatically-operated temperature and humidity controls of the facility heating, ventilation, and air-conditioning systems. All four compressed air systems are interconnected, thereby ensuring that an alternate source of compressed air is available should an air compressor fail. There are no functions or malfunctions of the compressed air systems that could initiate a reactor accident, prevent safe reactor shutdown, or initiate the uncontrolled release of radioactive material.

9.14.2 Main Air System

The main air system utilizes two compressors, designated the North and South Main Air Compressors, which supply the majority of the compressed air needs of the reactor facility. Both compressors have a discharge capacity of around 100 cfm. The north compressor is set to run as the dedicated air compressor, assuming the majority of the main air system demand. The compressor will automatically start when main air system pressure reduces to approximately 95 psig (655 kPa above atmosphere) and will automatically secure when air system pressure increases to approximately 120 psig (827 kPa above atmosphere). If the main air system pressure drops to approximately 90 psig (621 kPa above atmosphere), the south compressor will automatically start and run to increase pressure to approximately 105 psig (724 kPa above atmosphere) before securing.

9.14.3 Emergency Air System

The emergency air system supplies compressed air to the isolation valves and sealable closures of the containment system should the main air system become inoperable. Operation of the emergency air system is described in Chapter 6, Engineered Safety Features, Sections 6.2.3.8 and 6.2.5.

9.14.4 Valve Operation Air System

The valve operation air system is comprised of an air compressor which supplies compressed air to the reactor plant. The valve operation air compressor will automatically start when system pressure reduces to approximately 100 psig (689 kPa above atmosphere). The compressor will then automatically secure when pressure in the valve operation air system increases to approximately 115 psig (793 kPa above atmosphere). Since the main air system is set to operate at a higher pressure, the valve operation air compressor does not typically run and is essentially a backup air supply to the valve operation air system. However, should the main air system become inoperable or system pressure decrease below the set points of the valve operation system, the valve operation air compressor will assume the valve operation compressed air load and force shut a check valve in the air supply line from the main air system, isolating the main air system from the valve operation air system. The check valve ensures that the valve operation air compressor will supply compressed air to only the valve operation air system.

9.14.5 Instrument Air System

The instrument air system is comprised of an air compressor which supplies compressed air to the pneumatically-operated temperature and humidity controls of the facility heating, ventilation, and air-conditioning systems. These controls include thermostats, humidostats, flow control valves (steam and chill water), and the damper operators which position the louver dampers on the north and south facades of the reactor building east tower. Should the instrument air compressor fail, a normally-shut isolation valve which connects the main air system to the instrument air system may be opened. This will ensure that a continuous supply of compressed air is available to the instrument air system.

9.14.6 Surveillance

The emergency air compressor is periodically tested for operability.