MURR Safety Analysis Report Index

Chapter 1 – The Facility

Chapter 2 – Site Characteristics

Chapter 3 – Design of Structures, Systems, and Components

Chapter 4 – Reactor Description

> **Chapter 4, Attachment A** – Current Safety Limit Analysis for MURR – Hazards Summary Report, Appendix F – Safety Limit Analysis for MURR
>
> **Chapter 4, Attachment B** – Current Bases for Limiting Safety System Settings, Hazards Summary Report, Appendix H – Bases for Limiting Safety System Settings for Modes I and II Operation
>
> **Chapter 4, Attachment C** – MURR Safety Limit Analysis Prior to Amendment #36

Chapter 5 – Reactor Coolant Systems

Chapter 6 – Engineered Safety Features

Chapter 7 – Instrumentation and Control Systems

Chapter 8 – Electrical Power Systems

Chapter 9 – Auxiliary Systems

Chapter 10 – Experimental Facilities and Utilization

Chapter 11 – Radiation Protection Program and Waste Management

Chapter 12 – Conduct of Operations

Chapter 13 – Accident Analyses

Chapter 14 – Technical Specifications

Chapter 15 – Financial Qualifications

Chapter 16 – Other License Considerations

Chapter 17 – Decommissioning

Chapter 18 – High-Enriched to Low-Enriched Uranium Conversions

Appendix A – Removed – Previously Contained the MURR Technical Specifications

Appendix B – Radiological Impact of Ar-41 During Normal Operations

Appendix C – Thermal-Hydraulic Transient Analysis

CHAPTER 1

THE FACILITY

TABLE OF CONTENTS

1.0 THE FACILITY 3
1.1 Introduction 3
1.1.1 Location of the Facility 4
1.1.2 Purpose of the Facility 4
1.2 General Description of the Facility 5
1.2.1 Introduction 5
1.2.2 Reactor 6
1.2.3 Reactor Containment Building 10
1.2.4 Laboratory Building 11
1.3 Comparison With Similar Facilities 12
1.4 Summary and Conclusions on Principal Safety Considerations 12
1.4.1 Introduction 12
1.4.2 Reactor 13
1.4.3 Reactor Containment and Laboratory Buildings 15
1.5 Summary of Operations 16
1.6 Compliance With the Nuclear Waste Policy Act of 1982 16
1.7 Facility Modifications and History 16
1.7.1 Introduction 16
1.7.2 Original Design and Construction 17
1.7.3 License Amendments 18
1.7.4 Reactor Systems Modifications 24

LIST OF TABLES

Table 1-1 SUMMARY OF REACTOR PARAMETERS 8

LIST OF FIGURES

Figure 1.1 – BELOW GRADE LEVEL PLAN (BASEMENT) 28
Figure 1.2 – GRADE LEVEL PLAN 29
Figure 1.3 – THIRD LEVEL PLAN 30
Figure 1.4 – FOURTH LEVEL PLAN 31
Figure 1.5 – FIFTH LEVEL PLAN 32
Figure 1.6 – NORTH AND EAST ELEVATIONS 33
Figure 1.7 – SOUTH AND WEST ELEVATIONS 34
Figure 1.8 – BUILDING SECTIONS A-A AND B-B 35
Figure 1.9 – BUILDING SECTIONS C-C, D-D, E-E, F-F, G-G, AND H-H 36

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1.5 Hazards Summary Report, Addendum 3, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, August 1972.

1.6 Hazards Summary Report, Addendum 4, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, October 1973.

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1.0 THE FACILITY

1.1 Introduction

The original submittal of this Safety Analysis Report (SAR) by The Curators of the University of Missouri supported an application to the U.S. Nuclear Regulatory Commission (NRC) for authorization to continue operation of the University of Missouri Research Reactor (MURR), dated August 31, 2006. MURR is a pressurized, reflected, open pool-type, light water moderated and cooled, heterogeneous system designed for operation at a maximum steady-state power level of 10 Megawatts thermal (MW). The application requested a twenty-year renewal of the Class 104c Amended Facility License No. R-103 (NRC Docket No. 50-186). On January 4, 2017, the NRC issued Renewed Facility Operating License No. R-103 for the University of Missouri-Columbia Research Reactor. NRC rulemaking, effective January 29, 2025, eliminated license terms for research and development research reactors, such as MURR, licensed under 10 CFR 50.21(c).

This SAR update is a compilation of the August 31, 2006, SAR submission, supplemented with responses to requests for additional information provided during the relicensing process, and with SAR related information that was included in the annual reports required by TS 6.6.e. since the August 31, 2006, submission.

References 1.1 through 1.7 provide the bases for the original reactor license granted October 11, 1966, on which the SAR submittal dated August 31, 2006, and this SAR submittal updated through December 31, 2024, were based.

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 provide an educational opportunity for students in these fields. MURR also provides radiation and isotope production services for public and private consumers.

This document addresses the safety issues associated with the operation of MURR reflecting the as-built condition of the reactor facility and includes the experience observed in the operation and performance of the reactor systems. MURR has reanalyzed postulated accidents and transients using the latest analysis programs. The updated analyses confirmed the conclusions of the analyses performed when the reactor was first licensed and demonstrate that the continued operation of MURR will not present any undue hazard to any member of the general public or staff. Accident scenarios are analyzed in Chapter 13, “Accident Analyses.”

Within this document resides some information which has been identified as “historical” in that it provides valuable historical context for the siting and design of MURR. This information, however, would provide little value if updated with contemporary revisions of the document. As such, it may not be revised with subsequent revisions of this document and is denoted as historical information by appearing in italics; *example historical text*.

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 (now referred to as Research Commons), an 84-acre (34.0-hectare) tract of land approximately one mile (1.6 km) southwest of the University of Missouri (MU) main campus. The campus is situated in the southern portion of Columbia; a city with an estimated December 2024 population of 131,200 people. Columbia is the county seat and largest city in Boone County, Missouri.

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

1.1.2 Purpose of the Facility

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.

MURR has six (6) 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 (6) 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 isotope production for the development of radiopharmaceuticals and other life-science research, and neutron activation analysis. The six (6) 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 thermal column is designed for the purpose of performing neutron radiographs and large sample irradiations.

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. 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.

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

MURR is a heterogeneous, 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 23.5 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 7.5 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) long by 62 feet (18.9 m) wide by 64.5 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 reactor containment buildings at levels below 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 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. MURR contains 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 (5) levels of the reactor facility.

1.2.2 Reactor

The reactor core consists of eight (8) identical fuel assemblies, each occupying a 45° segment of a cylindrical annulus. The reactor fuel is described in detail in Section 4.2.1. The reactor core is in an annular aluminum pressure vessel, approximately 12 inches (30.5 cm) in diameter and 13 feet (4.0 m) high.

The reflector region consists of two (2) 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 (6) beamports.

The reactivity of the reactor is controlled by five (5) control rods (blades) 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 (4) 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 around the pressure vessel.

The reactor utilizes three coolant systems: primary, pool, and secondary. The reactor core is cooled by the circulation of pressurized primary coolant through two (2) 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 (2) 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 (2) 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 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 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
Peak 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)
Active Fuel Height	24.0 inches (60.96 cm)
Active Volume of the Core	33.0 liters
Fuel Loading for 5 MW Operation	5.2 kg U-235
Fuel Loading for 10 MW Operation	6.2 kg U-235
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 above atmosphere)
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
U-235 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 (13,608-kg) overhead rectilinear crane, and a 2-ton (1814-kg) capacity passenger elevator which services four (4) 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, the water level within the trap will elevate and create a path to equalize pressure. The water trap will re-seal the reactor containment building once the pressure transient has terminated.

In order to minimize potential leakage points from the reactor 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 entryways, 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 grade level is a 33,184 sq.ft. (3,083 m2) 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 sq.ft. (836 m2) of floor space. This area contains a 15-ton (13,607-kg) capacity hydraulic freight elevator and a primary 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.

Three (3) temporary office buildings (TOBs), a north office addition (NOA), and an industrial building provide additional office and laboratory space for the facility. In December 2024, a MURR West building was in the final stages of construction to the north-west of the laboratory building. It will provide necessary additional space for offices and laboratories.

1.3 Comparison With Similar Facilities

The uranium-aluminide (UAlx) dispersion fuel system currently used at MURR was developed at the Idaho National Engineering Laboratory (INEL) in southwest Idaho. The UAlx 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 MURR in August 1971 and continues to be used in the Advanced Test Reactor (1967-present).

The excellent performance of aluminide UAlx fuel has been consistently demonstrated at MURR. MURR has used over 1000 fuel elements since 1971 with only one detectable leaking fuel element.

The leaking fuel element was detected by a slight elevation in the primary coolant sample I-131 analysis for the week of October 20 and November 3, 1997. The highest concentration of I-131 activity in the primary coolant was determined to be 2.3 x 10-5 µCi/ml, which was a factor of 200 less than the Technical Specification (TS) limit for I-131 of 5 x 10-3 µCi/ml at that time (TS 3.9.c), which is the same limit in current TS 3.3.c. The fuel element was removed from service. The cause of the problem was determined to be a slight defect in the fabrication of fuel element MO-460.

1.4 Summary and Conclusions on Principal Safety Considerations

1.4.1 Introduction

The analyses presented in this SAR demonstrate that 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 decades of established operating experience MURR has demonstrated a 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 MURR, this document shows 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. The four principal elements of reactor safeguards include:

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;

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;

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

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 TSs; 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 of the features and the characteristics of the site selected for 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 in the following sections.

1.4.2 Reactor

MURR is a heterogeneous, pressurized, reflected, open pool-type, which is light water moderated and cooled. As such, it has many of the inherent safety features characteristic of a heterogeneous, 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 through years of continuous operation at MURR and 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/sq.ft.-hr. 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 (4) shim blades are measured at intervals as specified in the TSs. 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.

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

The reactor is located near the bottom [approximately 11.5 feet (3.8 m) below ground level] 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 exposure 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, the bulk pool and the 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 U.S. They have well-established operating experience with no significant new design alterations required.

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

- Fueled Experiment Failure (Maximum Hypothetical Accident (MHA));
- Fuel Failure during Reactor Operations (FFA)
- Fuel Handling Accident (FHA)
- Insertion of Excess Reactivity;
- Loss of Primary Coolant (LOCA);
- Loss of Primary Coolant Flow (LOFA);
- Experiment Malfunction;
- Loss of Electrical Power;
- External Events; and
- Mishandling or Malfunction of Equipment.

The MHA is an accident condition which postulates the complete failure of a fueled experiment and the subsequent release of fission products to the pool coolant system. This accident is selected to postulate off-site 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.

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.

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 released through the facility exhaust stack during normal operation of the reactor, have been evaluated and are discussed in Chapter 11, "Radiation Protection Program and Waste Management."

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 (30.48-cm) 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 reactor containment building, as verified by periodic surveillance testing, ensures that this objective is met.

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 reactor 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 (13,608-kg) capacity overhead rectilinear crane, is rigorously maintained, including preventive maintenance as appropriate. 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 MURR facility to monitor radiation levels. Each radiation meter is equipped with an adjustable set point trip that initiates an audible and visual alarm upon detection of a high radiation level.

1.5 Summary of Operations

Since September 1, 1977, MURR has had an operating schedule of 24 hours per day, 6.5 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 preventive maintenance. This operating schedule equates to operating at full power (10 MW) for approximately 90% of all available hours in the year.

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 DOE for the disposal of high-level radioactive wastes and spent nuclear fuel. DOE provides fuel assistance for 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.

There is a standard research subcontract (No. 00072718), “Reactor Fuel Assistance and Fuel Elements” between Battelle Energy Alliance, LLC (BEA) and the University of Missouri that provides the applicable requirements of the Nuclear Waste Policy Act of 1982. This subcontract is issued under Prime Contract No. DE-AC07-05ID14517 between Battelle Energy Alliance, LLC (BEA) and the United States Department of Energy (DOE) for the management and operation of the Idaho National Laboratory (INL).

1.7 Facility Modifications and History

1.7.1 Introduction

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 complied 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 50 years 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 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 (UAl_x) dispersion 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.
January 4, 2017	The NRC issues Renewed Facility Operating License No. R-103 for the University of Missouri-Columbia Research Reactor.

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 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 40 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
Authorized increasing the amount of contained U-235 that MURR can receive, possess, and use from 15 to 45 kilograms.

Amendment 2 - Date of Issuance: July 9, 1974
Authorized operation at a steady-state power level up to 10 MW, after installation of additional heat exchanger equipment;

Authorized receipt, possession, and use of a 100-Curie source of antimony-beryllium.

Amendment 3 - Date of Issuance: February 14, 1975
Authorized TSs Change No. 11 which permitted operation with increased fuel burnup. The amendment also increased the allowable surveillance interval for uranium-aluminide (UAl_x) dispersion fuel elements.

Amendment 4 - Date of Issuance: September 7, 1976
Authorized possession of up to 20 grams of Pu-239 as neutron filters used in conjunction with the operation of neutron spectrometers;

Authorized the use of up to 60 kilograms of source material under Facility Operating License R-103 rather than under the Source Material License No. SUD-287.

Amendment 5 - Date of Issuance: May 18, 1977
Increased the possession limit of U-235 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 MURR prior to the shutdown of the Oak Ridge, Tennessee facility.

Amendment 6 - Date of Issuance: July 8, 1977
Increased the possession limit of unirradiated U-235 from 19 to 20.5 kilograms. This amendment accounted for the storage of three (3) 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
Changed the fuel inspection intervals specified in the TSs. This amendment reduced the number of fuel inspections required to be performed to coincide with the revised operating schedule.

Amendment 8 - Date of Issuance: February 24, 1978
Changed the limitations on fueled experiments specified in the TSs. This amendment allowed expansion of the experimental program by revising the restrictions placed on fueled experiments.

Amendment 9 - Date of Issuance: March 23, 1978
Incorporated control blade surveillance requirements into the TSs. This amendment formalized a facility practice of inspecting a control blade (shim blade) prior to use.

Amendment 10 - Date of Issuance: July 13, 1978
Authorized performance of 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
Revised the MURR Physical Security Plan.

Amendment 12 - Date of Issuance: July 5, 1979
Revised the limitations on gaseous radioactivity release specified in the TSs. This amendment increased the TS 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 6, 1981
Revised the facility organizational structure specified in the TSs to reflect the administrative organization of the facility.

Amendment 14 - Date of Issuance: April 15, 1981
Changed the definition of "Reactor Secured" specified in the TSs. 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
Revised the MURR Physical Security Plan.

Amendment 16 - Date of Issuance: October 20, 1988
Increased 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;

Allowed for the receipt, possession, and use of 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;

Removed the requirement that the pool coolant circulation pumps operate in parallel; testing has shown that either pump will provide the minimum TSs for pool flow and differential pressure across the reactor reflector region;

Changed the facility organizational structure specified in the TSs to reflect the administrative organization of the facility; and

Corrected a typographical error and replace references to the AEC with NRC in the TSs.

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

Amendment 18 - Date of Issuance: May 8, 1989
Shifted administrative authority for MURR from the MU system to the MU at Columbia. This amendment allowed 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 MURR research programs.

Amendment 19 - Date of Issuance: June 6, 1990
Temporarily increased the possession limit for U-235 from 45 to 60 kilograms for an interim period from the effective date until May 31, 1991. The two (2) spent fuel shipping casks with the capability to transport MURR spent fuel were removed from service for various reasons. This amendment allowed MURR to retain spent fuel on site until a suitable shipping cask was found, maintaining uninterrupted reactor operation.

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

Amendment 21 - Date of Issuance: May 8, 1991
Temporarily increased the possession limit for U-235 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 (2) spent fuel shipping casks with the capability to transport MURR spent fuel were removed from service for various reasons. The task of finding and becoming an authorized user of a new shipping cask was not completed by the May 31, 1991, expiration date. This amendment allowed MURR to retain spent fuel onsite until a suitable shipping cask was found, maintaining uninterrupted reactor operation.

Amendment 22 - Date of Issuance: August 17, 1992
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 Amended Facility License No. R-103. A spent fuel shipping cask was acquired by DOE and made available to MURR. This amendment re-established the possession limit for U-235 at 45 kilograms.

Amendment 23 - Date of Issuance: May 17, 1993
Added to the TSs the requirement to have written procedures for the shipping of byproduct material produced under Amended Facility License No. R-103.

Amendment 24 - Date of Issuance: July 21, 1993

Temporarily increased the possession limit for U-235 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 MURR on a schedule that would allow MURR to continue operation without exceeding the current possession limit of 45 kilograms.

Amendment 25 - Date of Issuance: March 2, 1994

Changed 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

Temporarily increased the possession limit for U-235 from 45 to 60 kilograms for an interim period from the effective date until May 31, 1995. This amendment was required because the SRP was not able to accept spent fuel from MURR on a schedule that would allow MURR to continue operation without exceeding the current possession limit of 45 kilograms.

Amendment 27 - Date of Issuance: November 4, 1994

Authorized the installation of a single pool coolant system heat exchanger in place of the two (2) existing heat exchangers. This amendment was required because the two (2) existing heat exchangers had reached the end of their design operational lifetime. The two (2) existing heat exchangers were replaced by a single heat exchanger of plate-type design.

Amendment 28 - Date of Issuance: March 15, 1995

Authorized an increase the permanent U-235 possession limit from 45 to 60 kg.

Authorized an extension the temporary increase in the U-235 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 SRP could not receive spent fuel from MURR on a continuous basis to maintain the fuel inventory at MURR below 45 kilograms.

Amendment 29 - Date of Issuance: March 6, 1997

Revised aspects of the administrative structure having oversight authority for MURR and corrected an inconsistency in the TSs concerning meeting requirements of the Reactor Advisory Committee.

Amendment 30 - Date of Issuance: February 26, 1998

Clarified the TS requirements for written procedures for the preparation for shipment of byproduct material produced under the Amended Facility License.

Amendment 31 - Date of Issuance: September 20, 1999

Authorized the conduct of movable and unsecured experiments in the center test hole of the reactor flux trap.

Amendment 32 - Date of Issuance: October 19, 2001
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 Amended Facility License No. R-103.

Amendment 33 - Date of Issuance: January 29, 2004
Revised the description and structure of the management organization within the TSs.

Amendment 34 - Date of Issuance: October 10, 2008
Revised the requirements for the conduct of fueled experiments. This amendment allowed TSs to be changed such that fueled experiments, with inventories of iodine-131 through -135 greater than 1.5 Curies or strontium-90 greater than 5 millicuries, could be placed in irradiation containers that are designed and tested for pressure exceeding the maximum expected pressure by at least a factor of two.

Amendment 35 - Date of Issuance: February 9, 2012
Authorized installation of a "Flux-trap Irradiations Reactivity Safety Trip" (FIRST) device in the reactor protective system. This amendment allowed an increase in irradiation capacity in the center test hole by removing the reactivity contribution of the sample holder while also allowing the use of the 6-barrel sample holder for movable samples.

Amendment 36 - Date of Issuance: July 8, 2013
Revised Section 2.1, 'Reactor Core Safety Limit,' of the TSs. The Safety Limits were revised to correct an error caused by misinterpreting the definition of "diameter of heated surface" (D_i) in the Bernath correlation as "heated diameter" (D_h). Additionally, updated peaking factors were used based on modern analyses.

Amendment 37 - Date of Issuance: March 11, 2016
Imposed the controls necessary to allow irradiation and processing of non-fueled experiments to produce iodine-131. This amendment allowed MURR to safely produce this isotope for domestic and international distribution.

Renewed Facility Operating License No. R-103 was issued to MURR on January 4, 2017.

Amendment 38 - Date of Issuance: December 19, 2018
Consisted of changes to the TSs regarding "Radiation Monitoring Systems and Airborne Effluents" (TS 3.7) and "Review and Audit" (TS 6.2).

Amendment 39 - Date of Issuance: May 26, 2020
Consisted of changes to TSs regarding the definition of "Reactor Secured" (TS 1.26) and the testing frequency for the Reactor Containment Building (TS 4.4)

Amendment 40 - Date of Issuance: June 8, 2021
Consisted of changes to TSs regarding "Review and Audit" (TS 6.2) and the frequency of procedure review in "Procedures" (TS 6.4).

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 enhance the capabilities of the experimental facilities. Safety Evaluations (SEs) are performed on all system modifications, providing the basis for the determination as to whether the modification does or does not involve a question as defined in 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 redundancy into the reactor containment system traversal pathways.

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, establishing 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 (4) fuel plates (MHA) will be retained in a closed volume, 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, establishing 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-V, 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 (1034-m^3/min) fans, each driven by a 2-speed, 100-HP (74.57-kw) motor. The existing fans (16,500-SCFM (467-m^3/min) 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 (38.1 m/min)) across each laboratory fume hood necessary to maintain the operational standards for a safe working environment.

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

The two (2) 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, removing a major heating load from inside the laboratory building.

June 2002 - Construction of the MURR Industrial Building

The MURR Industrial Building (MIB) was constructed and attached to the west side of the laboratory building to provide future expansion of laboratory space. The 6257 sq.ft. (581 m^2) MIB was constructed with the poured concrete flooring resting on pilings driven to bedrock in anticipation of future hot cell locations.

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 (2) 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.

February 2005 - Replacement and Upgrade of Substation "A" 500-kVA Transformer

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, removing a major heating load from inside the cooling tower.

December 2006 - Replacement of the Two Primary Coolant System Heat Exchangers

The two (2) existing primary coolant system shell-and-tube-type heat exchangers (HX503A and HX503B) were replaced by heat exchangers of a plate-type design. This replacement was necessitated by the degradation of the heat transfer capability of the original heat exchangers since they were approaching the end of their design operational lifetime.

August 2007 - Construction of the North Office Addition

A North Office Addition (NOA) Building was constructed on-site to provide additional staff office space, meeting rooms, classrooms, and a cyclotron facility with associated laboratory space.

October 2008 - Replacement of Demineralizer Water Storage Tanks T300 and T301

The old demineralizer water storage tanks T300 and T301 were lined carbon steel tanks each with a capacity of 7,000 gallons (26,498 l). Due to continued degradation of the phenolic liner, the tanks were replaced with 316L stainless steel tanks. Each tank was substantially identical to the existing tanks with only minor exceptions.

June 2009 - Construction of the Shipping and Receiving Building

A Shipping and Receiving Building was constructed on-site to enhance the security at MURR by creating a location external to the main facility where packages are received, opened, and examined prior to being transported into the facility. It also serves as a staging area for outgoing packages.

January 2012 - Replacement of the Cooling Tower

The old cooling tower, which was a wood-framed, induced-draft, crossflow type, with three (3) cells and two-speed fan assemblies per cell, was replaced with a stainless steel, induced-draft, cross-flow type, with three (3) cells and two-speed fan assemblies per cell. The new tower is designed to cool 5,400 gallons (20,440 l) of water per minute to a temperature of 85 °F (29.4 °C) from an initial temperature of 115 °F (46.1 °C) at a maximum wet bulb temperature of 80 °F (26.7 °C).

February 2012- Approval of the Flux-Trap Irradiations Reactivity Safety Trip (FIRST) device

Amendment 35, dated February 9, 2012, approved the addition of the FIRST device. The FIRST device provides protection against an excessive reactivity step insertion in the Flux Trap region.

June 2014 - Construction of the MURR Industrial Building Eastward Expansion

An eastward expansion to the MURR Industrial Building was constructed in order to provide additional

laboratory space at MURR. The expansion provides 1,440 sq.ft. (134 m^2) and 1,432 sq.ft. (133 m^2) of net usable space on the grade and the below grade levels, respectively. The grade level floor is poured concrete designed to support four (4) 100,000-lb (45,359-kg) hot cells positioned adjacent to each other along the north-south direction. The steel and concrete structure rests on pilings driven to bedrock.

2018 Fabrication of a New Control Rod Drive Mechanism

This Modification Record documents the fabrication of a new control rod drive mechanism (CRDM). The original CRDMs have been in operation for nearly the entirety of the facility's operating lifetime, requiring only minimal preventive maintenance over that time period. Although they have been reliable, the objective of fabricating a new CRDM is to systematically remove from service each installed CRDM with the intent to completely refurbish them and return them back to service. This also creates a readily available spare CRDM should one fail.

2019 Replacement of Primary and Pool Coolant System Demineralizer Tanks

This modification record documents changes to the primary and pool coolant demineralizer (DI) system. The scope of this work included full replacement of the DI tanks (DI-200, DI-201, and DI-202), the resin storage tank, and necessary piping to connect the new tanks to the existing system. The original resin tanks were carbon steel, cylindrical tanks that had an internal rubber lining for corrosion resistance. The primary motive for implementing this modification was significant degradation of the existing resin tank rubber liners. The tanks were replaced with a stainless-steel design that does not require a liner. These tanks were designed to utilize the same or similar type of mixed bed resin that was previously in use at MURR.

2020 Fabrication of a New Control Rod Drive Mechanism for the Regulating Rod

This modification record documents the new design and installation of a control rod drive mechanism for the regulating rod – referred to as the regulating rod (blade) drive mechanism (RRDM). Prior to installation of the new RRDM, the previous RRDM had been in operation for nearly 54 years and, since 1974, there have been 25 licensee event reports submitted to the NRC that have been attributed to various failures of the RRDM during reactor operation. The new design is expected to be more robust and dependable. It was designed with newer technology replacement parts currently available that should significantly increase the reliability of the RRDM. A second RRDM has been fabricated as an available spare.

Figure Withheld per 10 CFR 2.390

FIGURE 1.1 – BELOW GRADE LEVEL PLAN (BASEMENT)

Figure Withheld per 10 CFR 2.390

FIGURE 1.2 – GRADE LEVEL PLAN

Figure Withheld per 10 CFR 2.390

FIGURE 1.3 – THIRD LEVEL PLAN

Figure Withheld per 10 CFR 2.390

FIGURE 1.4 – FOURTH LEVEL PLAN

Figure Withheld per 10 CFR 2.390

FIGURE 1.5 – FIFTH LEVEL PLAN

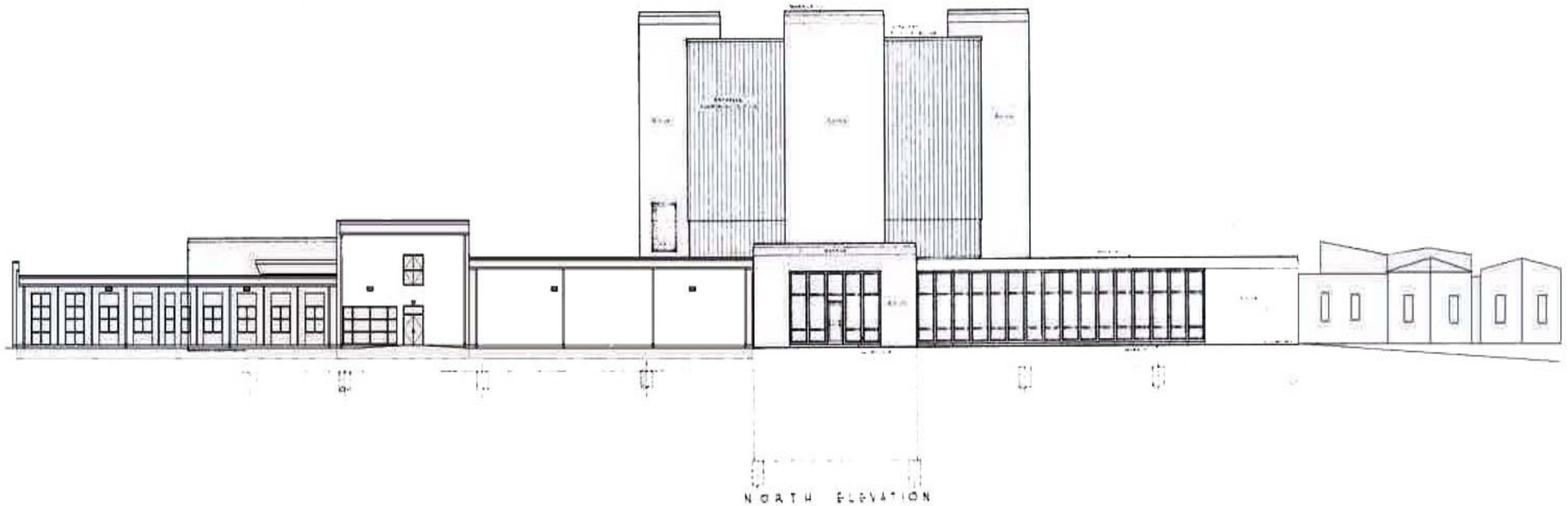
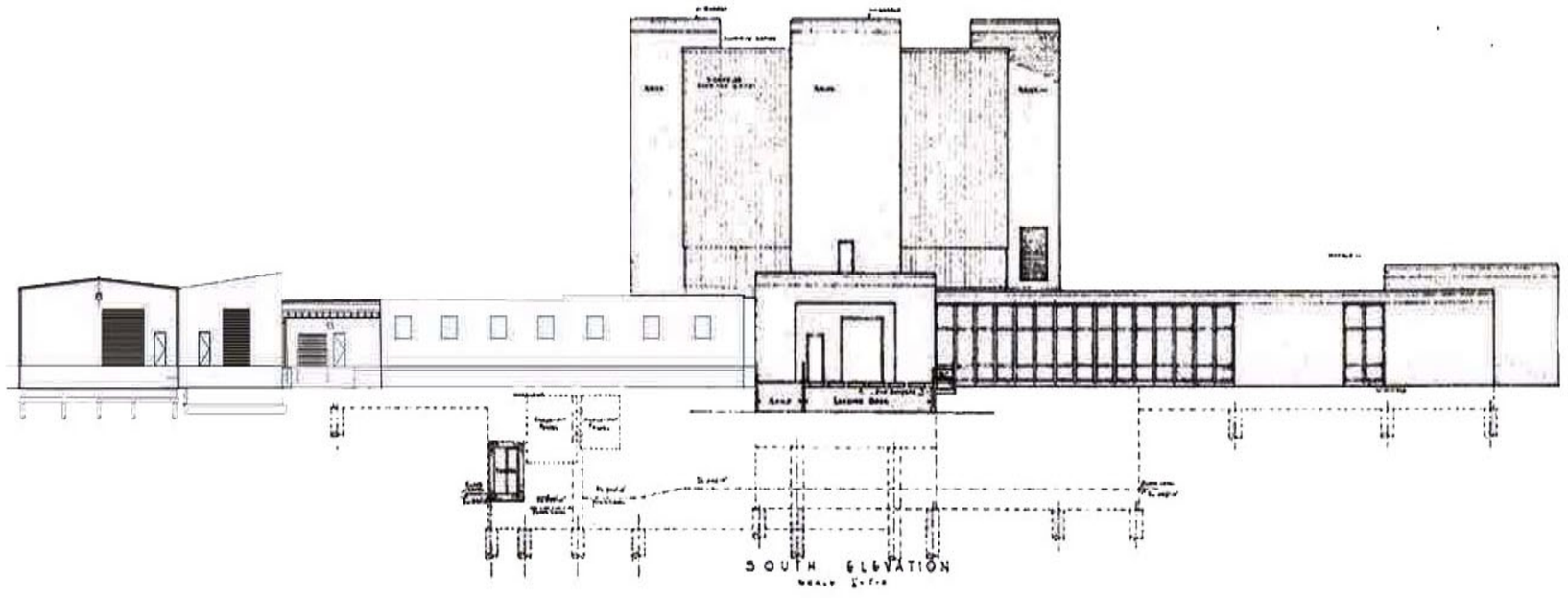


FIGURE 1.6 – NORTH AND EAST ELEVATIONS



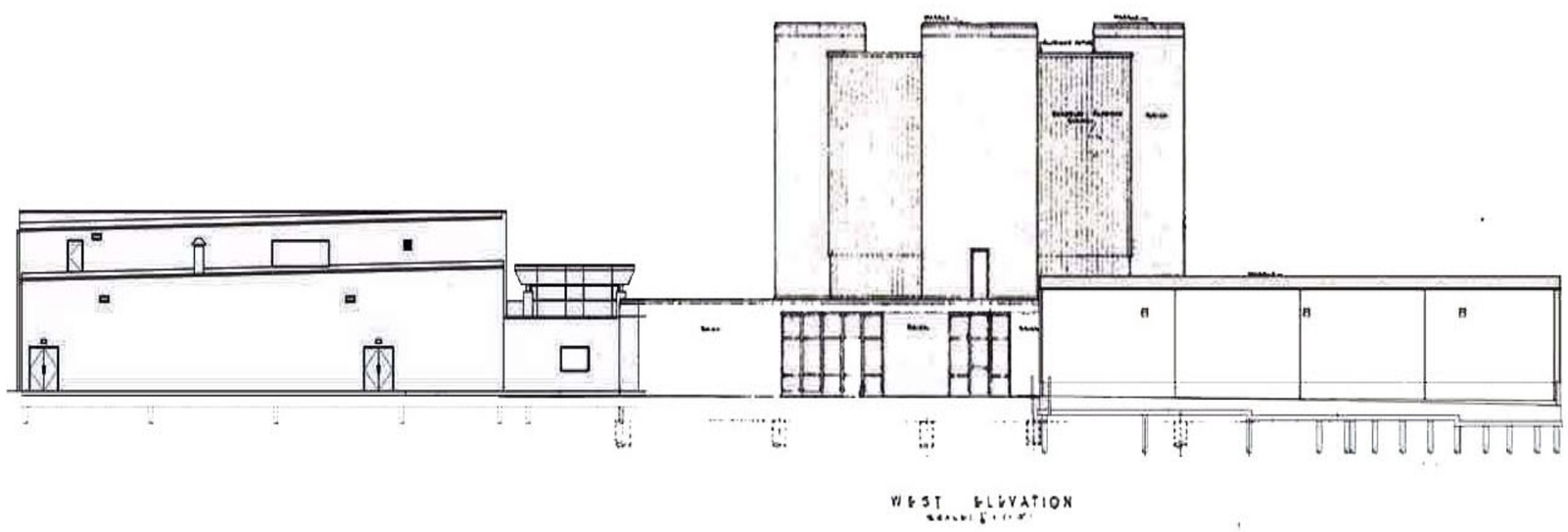


FIGURE 1.7 – SOUTH AND WEST ELEVATIONS

Figure Withheld per 10 CFR 2.390

FIGURE 1.8 – BUILDING SECTIONS A-A AND B-B

Figure Withheld per 10 CFR 2.390

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

CHAPTER 2

SITE CHARACTERISTICS

CONTENTS

2.0 SITE CHARACTERISTICS 5
2.1 Geography and Demography 5
2.1.1 Site Location, Description and Historical Construction 5
2.1.2 Operational Boundaries 6
2.1.3 Population Distribution 7
2.1.4 Physiography 9
2.2 Nearby Industrial, Transportation, and Military Facilities 10
2.2.1 Industry 10
2.2.2 Transportation 10
2.2.3 Military Facilities 12
2.2.4 Analysis of Potential External Accidents at Facilities near MURR 13
2.3 Meteorology 13
2.3.1 General and Local Climate 13
2.3.2 Site Meteorology 15
2.4 Hydrology 21
2.4.1 Surface Water 21
2.4.2 Stormwater Drainage System 23
2.4.3 Sanitary Sewer System 24
2.4.4 Ground Water 25
2.4.5 Flooding 26
2.4.6 Accidental Release of Liquid Effluents 28
2.5 Geology, Seismology, and Geotechnical Engineering 29
2.5.1 Regional and Site Geology 29
2.5.2 Seismology 31

LIST OF TABLES

Table 2-1 POPULATION DISTRIBUTION WITHIN 8 KILOMETERS OF THE REACTOR FACILITY (December 2024 Estimated)[d] 7
Table 2-2 HISTORICAL AND PROJECTED POPULATION GROWTH FOR THE CITY OF COLUMBIA 9
Table 2-3 MAJOR LAND TRANSPORTATION ROUTES WITHIN 5 MILES OF THE REACTOR FACILITY 12
Table 2-4 *PRECIPITATION DATA FOR THE STATE OF MISSOURI* 14
Table 2-5 AVERAGE THUNDERSTORM DAYS FOR COLUMBIA, MISSOURI (Period of Record 1991 - 2020) 16
Table 2-6 NORMAL TEMPERATURES AND RECENT EXTREME TEMPERATURES FOR COLUMBIA, MISSOURI 17
Table 2-7 NORMAL AVERAGE AND EXTREME PRECIPITATION TOTALS FOR COLUMBIA, MISSOURI 18
Table 2-8 NORMAL AND EXTREME SNOWFALL TOTALS FOR COLUMBIA, MISSOURI 19
Table 2-9 AVERAGE RELATIVE HUMIDITIES FOR COLUMBIA, MISSOURI (1991-2020) 20
Table 2-10 NORMAL AND EXTREME WINDS FOR COLUMBIA, MISSOURI 21

Table 2-11 SUMMARY OF MONTHLY AVERAGE HINKSON CREEK FLOW-RATE AT COLUMBIA, MISSOURI (January 2010-October 2024) 23
Table 2-12 UNIVERSITY OF MISSOURI DEEP WELLS 25
Table 2-13 RAINFALL INTENSITY (inches/hour) 27
Table 2-14 RAINFALL DEPTH (inches) 27
Table 2-15 HINKSON CREEK WATERSHED FLOWS (ft^3/sec) 28
Table 2-16 HYDROLOGICAL SOILS GROUPS 29
Table 2-17 EARTHQUAKES OF MISSOURI ORIGIN AS CLASSIFIED BY SEISMIC AREAS 1920 to 1939 33
Table 2-18 INTENSITIES OF EARTHQUAKES FELT IN MISSOURI 35

LIST OF FIGURES

Figure 2.1 – MAP OF MISSOURI SHOWING BOONE COUNTY 39
Figure 2.2 – LOCATION OF MAJOR CITIES, RAILROADS, AND HIGHWAYS IN BOONE COUNTY, MISSOURI 40
Figure 2.3 – EMERGENCY PLANNING ZONE (EPZ) 41
Figure 2.4 – CLIMATOLOGICAL STATIONS IN THE STATE OF MISSOURI 42
Figure 2.5 – SURFACE WIND ROSE AT COLUMBIA, MISSOURI 43
Figure 2.6 – PERCHE CREEK DRAINAGE IN BOONE COUNTY, MISSOURI 44
Figure 2.7 – WATERSHEDS FOR THE COLUMBIA AREA 45
Figure 2.8 – STORMWATER DRAINAGE SYSTEM FOR THE REACTOR FACILITY SITE 46
Figure 2.9 – CITY OF COLUMBIA SANITARY SEWER SYSTEM 47
Figure 2.10 – UNIVERSITY SANITARY SEWER SYSTEM 48
Figure 2.11 – REACTOR FACILITY SITE - 100 YEAR FLOOD PLAIN 49
Figure 2.12 – SOILS OF BOONE COUNTY, MISSOURI 50
Figure 2.13 – GEOLOGIC COLUMN AT COLUMBIA, MISSOURI 51
Figure 2.14 – GEOLOGIC MAP OF THE COLUMBIA AREA 52
Figure 2.15 – NEW MADRID SEISMIC ZONE 53
Figure 2.16 – EARTHQUAKE DISTRICTS IN THE STATE OF MISSOURI 54
Figure 2.17 – TOPOGRAPHICAL MAP 55
Figure 2.18 – TOPOGRAPHIC FILL AND CUT AREAS 56

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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, Description and Historical Construction

The University of Missouri Research Reactor (MURR) is located in the Columbia quadrangle, Missouri-Boone County, Township 48N, Range 13W and Section 24, 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 (Reference 2.1). The site location latitude and longitude coordinates are 38° 55' 53" North Latitude, 92° 20' 31" West Longitude and the Universal Transverse Mercator (UTM) Coordinates are Zone 15, 4,309,362 meters north and 557,037 meters east. 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. Attached is a topographical map of where MURR is located (Figure 2.17).

The building area of the MURR site at the time construction commenced had a gradual rise from south to north. In addition to the removal of existing trees on the site, approximately 15,000 cubic yards of fill was deposited on the south and central portions of the site. Fill material was removed from the east side of Research Park Drive (formerly known as "Reactor Park Road"). The attached topographic map (Figure 2.18) indicates relevant fill and cut areas.

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 (Reference 2.2).

In addition to 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 Dalton Cardiovascular 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. The Molecular Imaging and Theranostics Center, Life Science Business Incubator, and the American Red Cross Blood Center, are located to the east of the facility. 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 MURR.

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 is a restricted access area and consists of the outer walls of the Research Reactor Facility (a 30,000 ft^2 north office addition, and laboratory and reactor containment buildings).

The North Office Addition Consists of offices, some additional laboratory space and a 16 MeV cyclotron and its support equipment. In addition, the adjacent cooling tower building and shipping and receiving building are contained within the restricted access area. The 3,127 sq. ft. shipping and receiving building was constructed on the 7.5-acre facility lot to enhance security at the facility. Incoming and outgoing packages are received and staged for shipment in this building so that transport vehicles are not required to enter the gated parking lot which surrounds the facility.

The area within the operations boundary is a “restricted access” area where the Reactor Facility Director has direct authority and control over all activities, normal and emergency. The operations boundary is within the site boundary.

The site boundary consists of the following: Stadium Boulevard; Providence Road (Route K); 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. (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.)

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).

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 in December 2024 had a population of approximately 131,220 people. Nearly the entire city, as well as parts of the outlying metropolitan area, live within a 5-mile (8-km) radius of the Reactor Facility (Table 2-1). Approximately 75% 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 (DECEMBER 2024 ESTIMATED)[d]

Distance from Facility[a]	**Area (sq. km)**	**Population[b]**	**Population[c]**
0 to 1 kilometer (0.62 mile)	3.14	328	328
1 to 2 kilometers (1,24 mile)	9.42	21,021	21,349
2 to 4 kilometers (2.48 mile)	37.70	52,173	73,522
4 to 6 kilometers (3.73 mile)	62.83	24,892	98,414
6 to 8 kilometers (5.00 mile)	87.96	32,806	131,220
Total	201.06		131,220

a. Represents the area surrounding the reactor facility as divided into a series of concentric rings.
b. Population within each concentric ring (estimated).
c. Cumulative population to 8 kilometers (estimated).
d. Population estimate based on December 2024 data provided by Missouri-Demographics (https://www.missouri-demographics/columbia-demographics)

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 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 (15,060 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 MU'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, four 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 commercial development.

6 to 8 kilometers: The only substantial residential development is to the northeast. A shopping center, three middle/junior high schools, and six elementary schools are located in this area. Recreational areas include one park. Substantial amounts of land presently exist for residential or commercial 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 (Reference 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. Student enrollment is difficult to project but has remained relatively constant.

Due to the seasonal occupation of certain buildings, significant population variations do occur. Approximately 14,000 of the nearly 38,000 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 Arenas are, at most, occupied several times per week but are rarely occupied at the same time as Memorial Stadium.

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

Year	1990	2000	2010	2020	2024*
Population	69,101	84,531	108,500	126,254	131,220*

*U.S. Census data for 1990, 2000, 2010, and 2020 for Columbia, Missouri, with December 2024 estimate from Missouri Demographics by CUBIT (https://missouri-demographics.com/columbia-demographics)

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 (Reference 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 (Reference 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 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 one of the area's largest employers, employing approximately 9,500 people in 2024. Columbia, in 2024, had an estimated university and college population of about 38.000 students. Columbia is also the home of Stephens College (employing approximately 200 people) and Columbia College (employing approximately 1,800 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. Typical carloads of freight transported on this track is about 1,500 cars per year. The primary freight includes 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 1,538 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: 7402 feet by 150 feet (2,256 m by 46 m) concrete strip which supports most of the commercial air traffic, and a smaller 5500 feet by 100 feet (1,676 m by 30 m) cross-wind runway primarily for private aircraft.

The Columbia Regional Airport averaged around 39,000 commercial, private, and military flight operations in 2024.

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 6,500 military and civilian personnel.

2.2.4 Analysis of Potential External Accidents at Facilities near MURR

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 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 below grade level and 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 2024 mean maximum temperatures in Columbia ranged from 34.3 °F (1.3 °C) in January to 87.6 °F (30.9 °C) in June. The city's average relative humidity is 72%.

Precipitation in the Columbia area was approximately 41.8 inches (106.2 cm) in 2024. 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.

TABLE 2-4
PRECIPITATION DATA FOR THE STATE OF MISSOURI

Precipitation (inches)			
Average		***Maximum***	
Station	***Yearly Total***	***Monthly Total***	***24-Hour Total***
Columbia	*42.6*	*12.31 – May 1995*	*5.95 – July 1989*
Kansas City	*39.1*	*12.38 – Oct 2009*	*4.33 – July 2016*
St. Louis	*41.7*	*12.8 – May 2019*	*9.00 – July-2022*
Springfield	*45.6*	*13.4 – June 2008*	*6.00 – Dec-2015*

Data published by the National Weather Service, National Oceanic and Atmospheric Administration

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.

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. Tornadoes are

rare in Boone County, accounting for fewer than 1% of all tornadoes recorded in Missouri between 1950 and 2025. Historically, Boone County is considered a low-risk area. 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 August are shown to have the greatest thunderstorm occurrence, averaging 32.7 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.

The general meteorological conditions for the City of Columbia are summarized below. The included tables delineate the seasonal distribution of temperature, precipitation, and other recent climatological data.

TABLE 2-5
AVERAGE THUNDERSTORM DAYS FOR COLUMBIA, MISSOURI (PERIOD OF RECORD 1991 - 2020)

Month	***Thunderstorm Days***	***Month***	***Thunderstorm Days***
January	*0.6*	*July*	*7.0*
February	*1.0*	*August*	*6.3*
March	*2.8*	*September*	*4.3*
April	*4.9*	*October*	*2.9*
May	*7.3*	*November*	*1.5*
June	*7.2*	*December*	*0.7*
Annual			*46.5*

Data published by the National Centers for Environmental Information (NCEI), Asheville, North Carolina (30-year normal 1991-2020)

2.3.2.1 Temperature

The normal and most recent extreme temperatures for the City of Columbia are shown in Table 2-6. The coldest daily low temperature occurred in January 2024 [-11 °F (-23.9 °C)] while the warmest daily high temperature occurred in June 2012 [107 °F (41.7 °C)].

2.3.2.2 Precipitation

The normal and extreme precipitation totals for the City of Columbia are shown in Table 2-7. The annual precipitation for 2024 was 41.75 inches (106.01 cm). The maximum twenty- four-hour total for the period 1991-2024 was 5.73 inches (14.55 cm) which occurred in 1995. 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 72%. The lowest value occurs in April (66%) while the highest occurs in December (75%). The monthly and average annual relative humidity's during the 30-year reporting period for midnight, 6:00 a.m., noon, and 6:00 p.m. (Local Standard Time) 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 City of Columbia shows a southerly prevailing wind direction. 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 3-second winds are based on a 30-year period of record (1991-2022).

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 TEMPERATURES AND RECENT EXTREME TEMPERATURES FOR COLUMBIA, MISSOURI

Temperature (°F)							
Normal 2024				**Recent Daily Extremes**			
Month	**Daily Max.**	**Daily Min.**	**Average Monthly**	**High**	**Year**	**Low**	**Year**
Jan	34.3	20.4	27.3	77	2013	-11	2024
Feb	57.7	33.5	45.6	80	2024	-8	2021
Mar	62.2	40.1	51.1	85	2012	-2	2014
Apr	69.7	49.1	59.4	89	2012	19	2007
May	78.4	59.3	68.8	94	2018	35	2002
June	87.6	67.7	77.7	107	2012	50	2017
July	86.1	68.1	77.1	106	2012	52	2013
Aug	86.0	67.4	76.7	105	2011	50	2004
Sept	81.8	60.1	70.9	98	2022	37	2001
Oct	75.4	49.2	62.3	94	2006	22	2013
Nov	56.9	39.8	48.4	82	2022	7	2019
Dec	45.6	30.1	37.9	76	2021	2	2006
Year				107	June 2012	-11	Jan 2024

Normal Daily Data for 2024 published by the National Centers for Environmental Information, Columbia, Missouri (01/16/2025).
Extreme Daily Temperatures published by National Weather Service Climatological Data for Columbia, Missouri (02/04/2025).

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

Precipitation (inches)							
Normal		Most Recent Extremes					
Month	Monthly	Record Maximum	Year	Record Minimum	Year	24-hour Maximum	Year
Jan	2.41	5.94	2005	0.26	2009	2.26	2008
Feb	2.32	3.86	2013	0.21	2017	2.28	2001
Mar	3.24	7.25	2004	0.92	2005	3.94	2004
Apr	4.83	8.28	2014	0.45	2018	4.89	2017
May	5.20	9.79	2013	1.31	2012	5.73	1995
June	5.12	10.85	2021	1.46	2016	5.26	2021
July	4.62	10.91	2016	0.62	2005	5.07	2016
Aug	4.11	9.56	2023	1.25	2021	4.06	2002
Sept	4.26	7.33	2016	0.53	2023	4.72	2008
Oct	3.53	10.18	2014	0.36	2010	5.25	2014
Nov	3.00	8.43	2015	0.36	2017	4.37	2015
Dec	2.18	7.04	2016	0.23	2018	2.67	2015
Year	44.82	10.91	July 2016	0.21	Feb 2017	5.73	May 1995

Most Recent Extremes published by the National Weather Service for Columbia, Missouri (11/1/2023).
24-Hour Maximum Data published by the National Weather Service Climatological Data for Columbia, Missouri (6/28/2021)
Summary of Monthly Normal Precipitation for 1991-2020 published by the National Center for Environmental Information, Asheville, North Carolina (11/14/2023)

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

Precipitation (inches)							
Normal		**Most Recent Extremes**					
Month	**Monthly**	**Record Maximum**	**Year**	**24-hour Minimum**	**Year**	**Depth Maximum**	**Year**
Jan	6.0	23.5	1979	19.7	1995	21.6	1995
Feb	4.6	23.3	2011	17.5	2011	23.3	2011
Mar	1.5	18.0	1978	10.0	2013	10.3	2013
Apr	0.2	7.1	1980	2.5	1996	2.5	1996
May	0.0	Trace	2013	Trace	2013	0	N/A
June	0.0	none	N/A	none	N/A	0	N/A
July	0.0	none	N/A	none	N/A	0	N/A
Aug	0.0	none	N/A	none	N/A	0	N/A
Sept	0.0	none	N/A	none	N/A	0	N/A
Oct	0.0	0.4	2020	0.4	2020	0.4	2020
Nov	0.8	8.3	1971	2.3	2006	5.2	2003
Dec	3.4	17.8	1973	13.0	2006	13.0	2006
Year	16.5	23.5	Jan 1979	19.7	Jan 1995	23.3	Feb 2011

Monthly Normal Snowfall for 1991-2020 published by National Centers for Environmental Information, Asheville, North Carolina (11/14/2023).
Extremes published by the National Weather Service Climatological Data for Columbia, Missouri (11/1/2023).
24-hour Maximum Snowfall for 1991-2020 published by the National Weather Service Climatological Data for Columbia, Missouri (11/14/2023).
Maximum Depth Data published by the National Centers for Environmental Information, Asheville, North Carolina (11/14/2023).

TABLE 2-9
AVERAGE RELATIVE HUMIDITIES FOR COLUMBIA, MISSOURI (1991-2020)

Relative Humidity (%)													
	Jan	**Feb**	**Mar**	**Apr**	**May**	**Jun**	**Jul**	**Aug**	**Sep**	**Oct**	**Nov**	**Dec**	**Yr**
Mean	74	72	67	66	72	73	72	73	72	70	72	75	72
00 LST	77	76	72	71	80	83	83	83	81	77	77	78	78
06 LST	80	81	79	79	86	87	88	89	88	85	83	82	84
12 LST	67	64	58	56	61	60	58	57	58	57	63	69	61
18 LST	70	66	58	55	61	62	60	61	64	62	68	71	63

Data published by the National Centers for Environmental Information (NCEI), Asheville, North Carolina (30-year normal 1991-2020)

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

Wind								
Normal			Maximum 2-Minute			Maximum 3-Second		
Month	Mean Speed (mph)	Prevail Dir (deg)	Speed (mph)	Dir (deg)	Year	Speed (mph)	Dir (deg)	Year
Jan	10.5	310	44	340	2014	53	50	2021
Feb	10.5	310	49	260	2014	63	220	2019
Mar	11.2	310	54	230	2011	64	230	2011
Apr	11.2	190	54	230	2002	68	330	2001
May	9.2	170	52	300	2009	64	270	2011
June	8.3	190	53	340	2011	74	320	2006
July	7.4	190	46	310	2014	58	320	2014
Aug	7.3	180	59	10	2003	81	360	2003
Sept	8.0	160	39	270	2015	55	260	2014
Oct	9.2	170	47	250	1996	59	250	1996
Nov	10.3	170	45	190	1998	63	220	2022
Dec	10.2	310	39	230	2021	60	220	2021
Year	9.4	190	59	10	Aug 2003	81	360	Aug 2003

Data published by the National Centers for Environmental Information (NCEI), Asheville, North Carolina (30-year normals, 1991-2020, including 2021 and 2022 data)

2.4 Hydrology

2.4.1 Surface Water

The principal stream of the Boone County drainage basin is Perche Creek1 (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. (In older records of the State of Missouri, Perche Creek was identified as Roche 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 km2) of Boone County. Few of the tributaries of Hinkson Creek are named and all are short with steep gradients (Reference 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 1/2 mile (0.8 km) south of Stadium Boulevard. Flat Branch enters the Hinkson approximately 3/4 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 (Reference 2.6).

Hinkson Creek is a relatively shallow stream with frequent riffles, many pools, and rocky and/or dirt banks (Reference 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 ft3/sec (283,168 lps) which occurred on April 11, 1979. The average flow over the period (January 2010 to October 2024) was 67.6 ft3/sec (1914.2 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 (Reference 2.11)

TABLE 2-11
SUMMARY OF MONTHLY AVERAGE HINKSON CREEK FLOW-RATE AT COLUMBIA, MISSOURI (JANUARY 2010-OCTOBER 2024)

Month	Average Flow (ft^3/s)
January	39
February	55
March	100
April	145
May	117
June	93
July	96
August	31
September	28
October	42
November	20
December	45

Information based on United States Geological Service Surface-Water Monthly Statistics for Columbia, Missouri, January 2010-October 2024 [data taken by gauging station No. 06910230, (Hydrologic Unit 10300102, on Old South Providence Rd. bridge, 0.10 miles downstream of Highway 163 bridge, 1 mi upstream from Flat Branch.)].

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 (Reference 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;*
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 (Reference 2.8).

Within the city limits, the watershed contains residential, commercial, and undeveloped land. Outside the city limits, the watershed is considered undeveloped (Reference 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 (Reference 2.8). The stormwater drainage system in the vicinity of the reactor facility site is shown in Figure 2.8.

2.4.3 Sanitary Sewer System

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 32 million gallons (121 million liters) per day. 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 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 (Reference 2.9).

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 15 alluvial wells approximately 100 feet (30.5 m) deep with a total design capacity of 28 million gallons (106 million liters) per day. 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.

TABLE 2-12
UNIVERSITY OF MISSOURI DEEP WELLS

Well Designation	Year	Depth (feet)	Supply Rate (gpm)	Reservoir Capacity (gallons)
West Well (No. 1)	1938	1,175	600	425,170
East Well (No. 2)	1938	1,188	600	287,246
South Well (No. 3)	1959	1,595	600	582,990
Southwest Well (No. 4)	1968	1,442	1,000	994,399
North Well (No. 5)	1992	1,415	900	256,831

Updated 2/10/2025, email from Director, Campus Facilities Energy Management

The City of Columbia's (current usage) wellfield is located on the Missouri River alluvial flood plain near the town of McBaine (Reference 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 (Reference 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. 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) (Reference 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) (Reference 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 (Reference 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 (Reference 2.8).

Rainfall data for the City of Columbia was 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 (Reference 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;
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 (Reference 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 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, 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, “Radiation Protection Program and Waste Management.” 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 (Reference 2.10). Coal, formerly an important mineral of Columbia, was at one time mined a few miles north of the city from this Pennsylvanian system (Reference 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 (Reference 2.9). Table 2-16 provides a description of the soils shown in Figure 2.12.

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 (Reference 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 2.13. Figure 2.14 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 (Reference 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 sulfur.

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 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 (Reference 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 2.15). 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 (Reference 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 (Reference 2.14). Boone County, however, shows considerable stability and has little evidence of past seismic activity.

2.5.2.2 Earthquake Districts

Sections 2.5.2.2 and 2.5.2.4 provide historical information related to the seismic stability within and near Boone County and the MURR site. The terminology used by the Federal Emergency Management Agency (FEMA) has since changed and the use of the term "earthquake districts' has been replaced by "seismic zones."

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 2.16. 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 (Reference 2.16).

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 Poplar 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) (Reference 2.29). No subsequent earthquake in the United Sates has been felt

over a wider area (Reference 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.

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.

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

1939, November 23, Western Illinois
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).

1963, March 3, South-East Missouri
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).

1965, October 21, Eastern Missouri
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).

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.

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 (Reference 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 psia (20.68 MPa); and
   b. Steel rebar yield strength: 40,000 psia (275.79 MPa);

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

3. Load combinations considered in this assessment are:
   a. 1.5 x DL + 1.8 x LL,
   b. 1.25 x DL + 1.25 x OBE + 1.25 x LL,
   c. 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 17,200 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 (Reference 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 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 (Reference 2.4)."

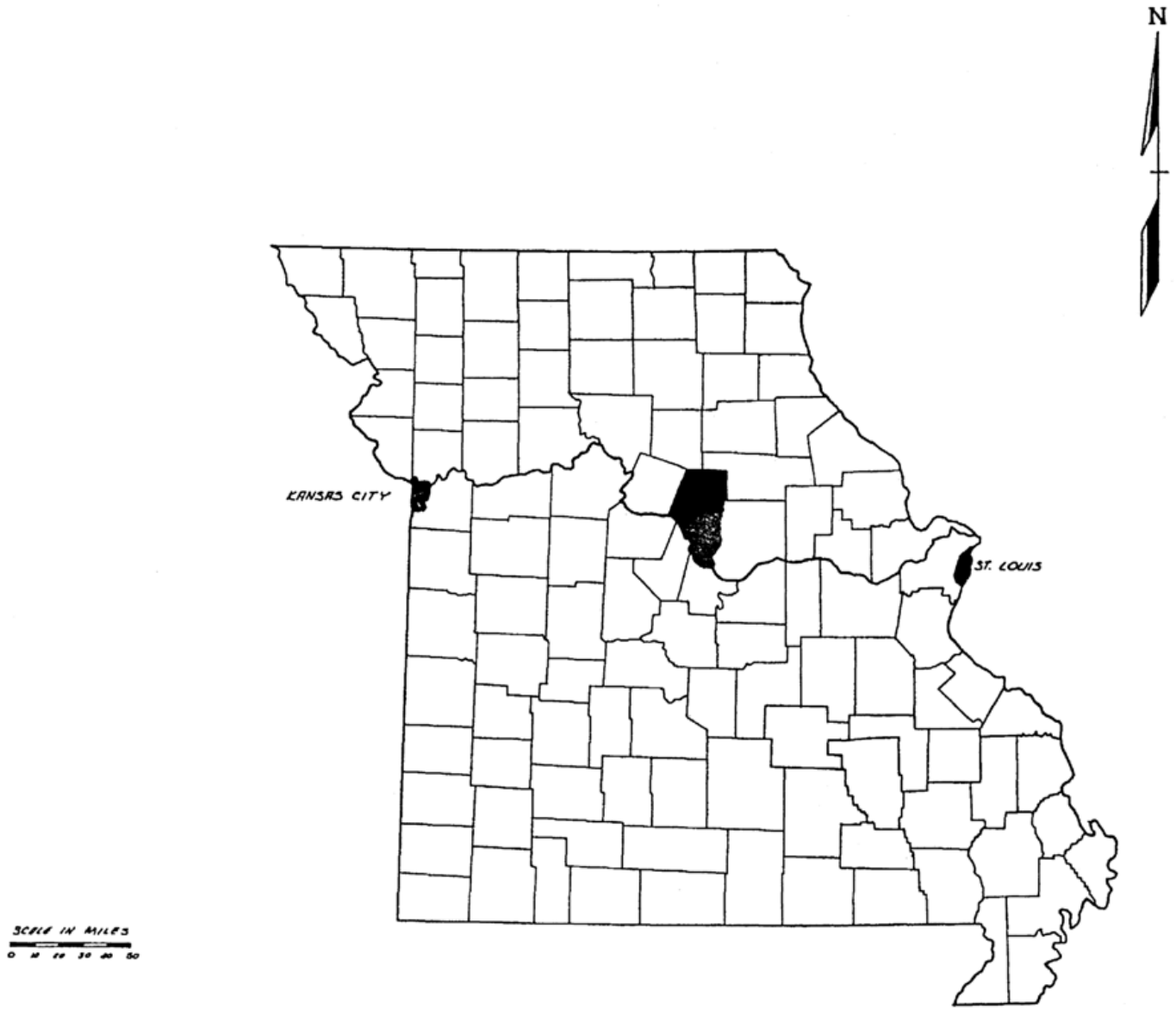


FIGURE 2.1 – MAP OF MISSOURI SHOWING BOONE COUNTY

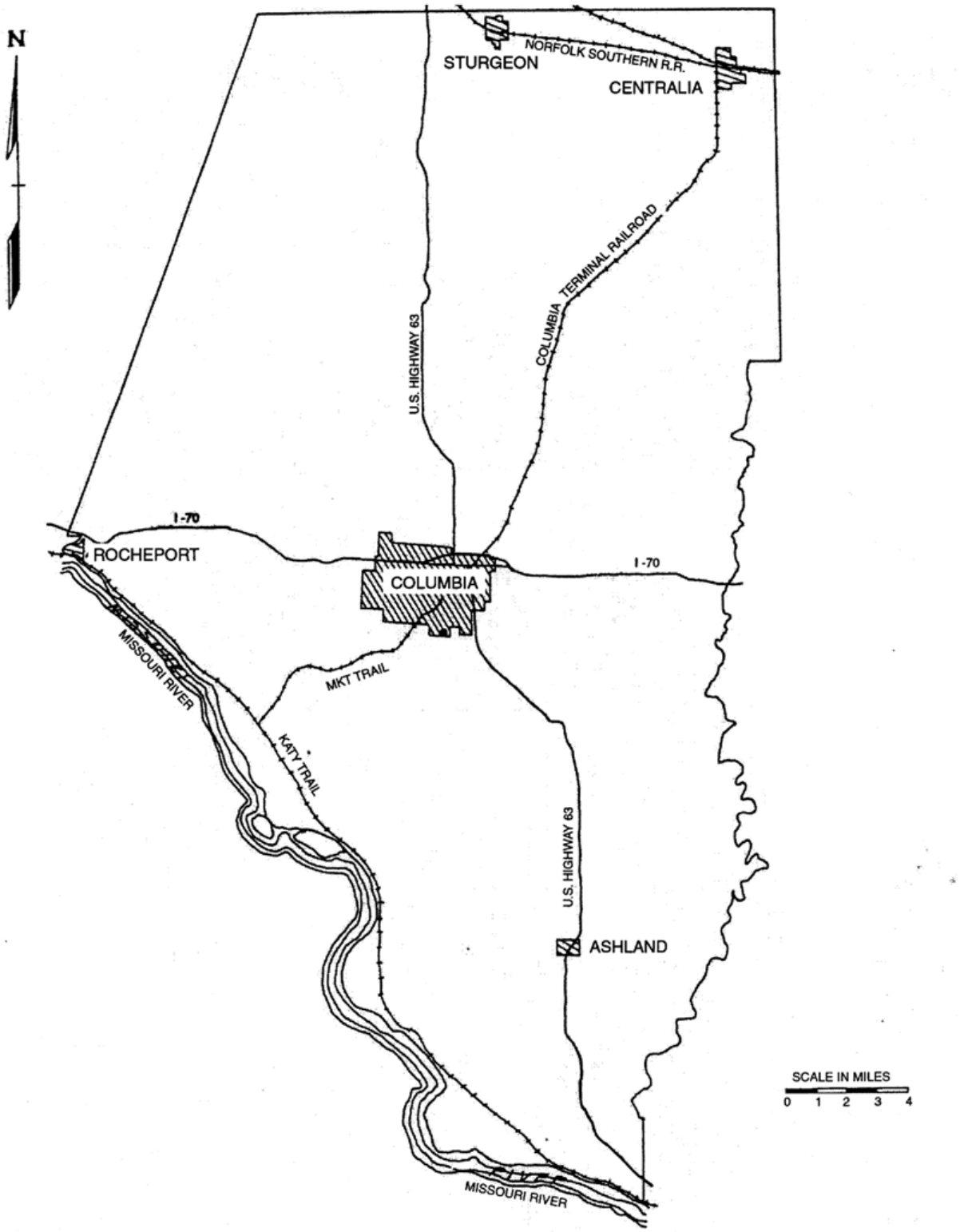
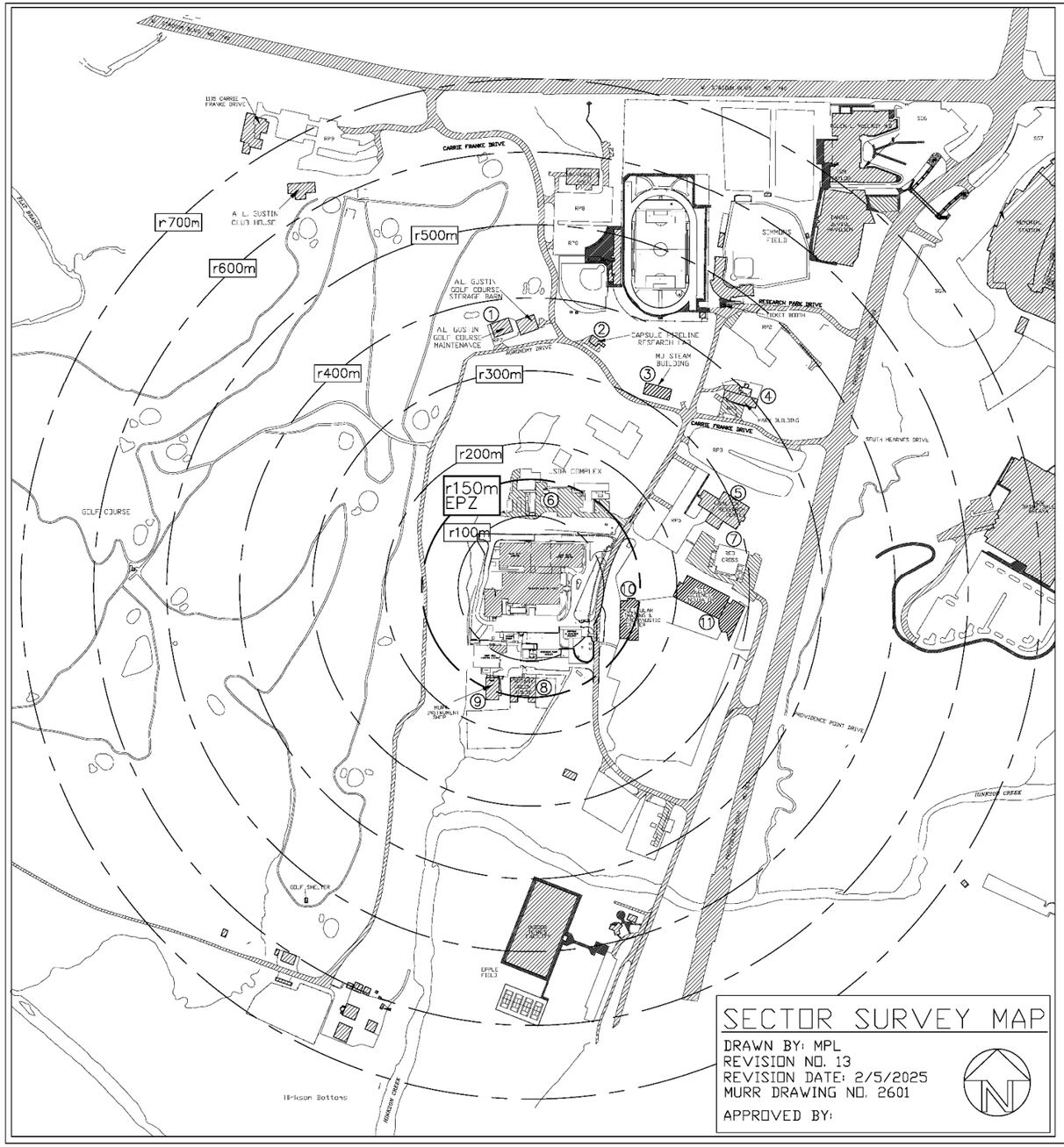


FIGURE 2.2 – LOCATION OF MAJOR CITIES, RAILROADS, AND HIGHWAYS IN BOONE COUNTY, MISSOURI



EPZ AND SITE AREA EVACUATION BUILDINGS

1). Golf Course Greens Keeper Building.....2-6226

2). Civil Engineering Freight Pipline Research Lab.....2-3081

3). Research Commons Thermal Plant.....
4). Marx Building (Art Department).....4-6075

5). Dalton Cardiovascular Research Center.....882-7588

6). USDA Biological Control of Insects Research Lab.....(573)-875-5361

7). American Red Cross Blood Center.....449-2656

8). Greenhouse.....2-6276

9). MURR Instrument Shop.....2-3711 or 2-3712

10). Molecular Imaging and Theranostic Center.....882-7016

11). Life Science Business Incubator at Monsanto Place..4-0492

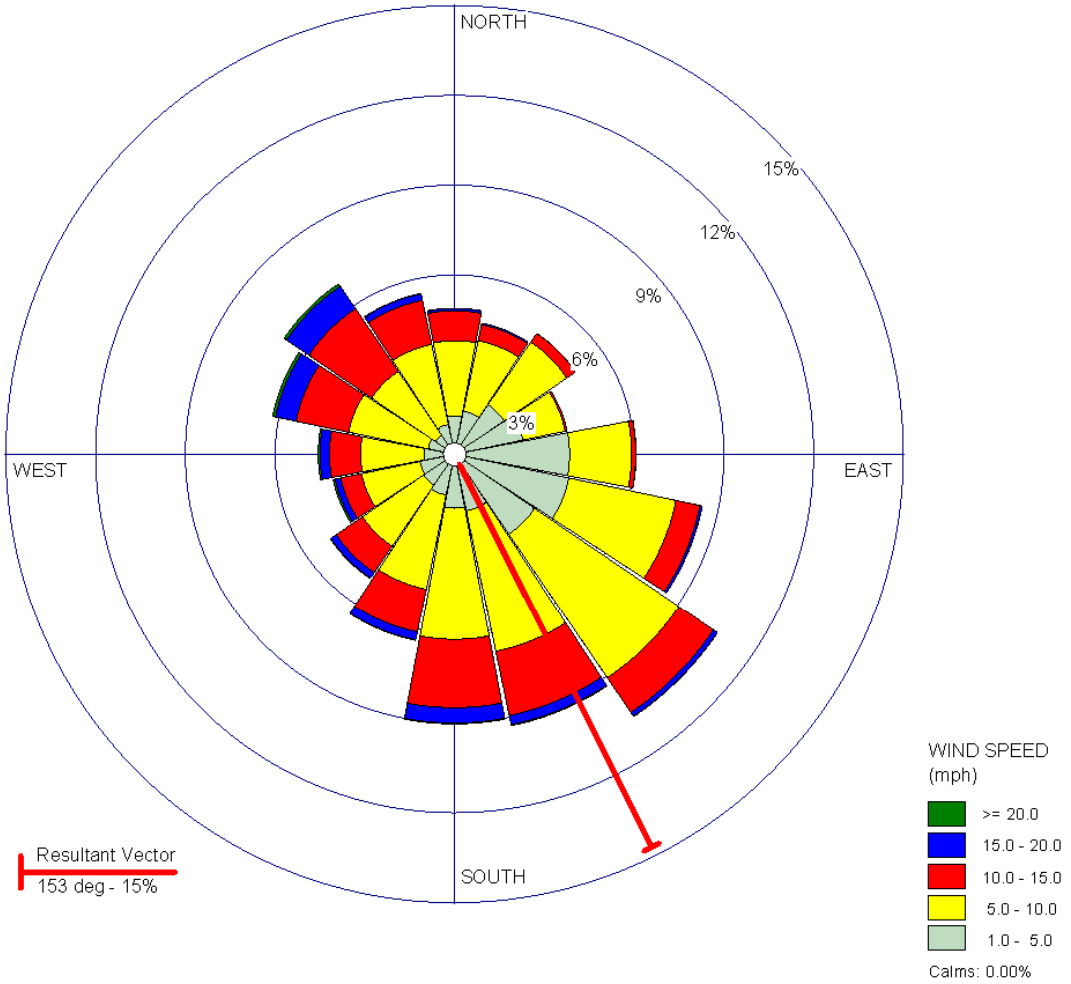
FIGURE 2.3 – EMERGENCY PLANNING ZONE (EPZ)



FIGURE 2.4 – CLIMATOLOGICAL STATIONS IN THE STATE OF MISSOURI

WIND ROSE PLOT:
Columbia, MO (South Farms)
MU Extension Commercial Agriculture Weather Station

DISPLAY:
Wind Speed
Direction (blowing from)



COMMENTS:	DATA PERIOD:		
Based on hourly wind speed and direction from the MU Ag Weather Station at Columbia (South Farms) (measured at 3 meters)	**2000-2010** **Jan 1 - Dec 31** **00:00 - 23:00**		
	CALM WINDS: (less than 1 mph) **0.00%**	TOTAL COUNT: **96431 hrs.**	UNIVERSITY OF MISSOURI Extension COMMERCIAL AGRICULTURE
	AVG. WIND SPEED: **7.00 mph**	DATE: **2/22/2011**	

WRPLOT View - Lakes Environmental Software

FIGURE 2.5 – SURFACE WIND ROSE AT COLUMBIA, MISSOURI

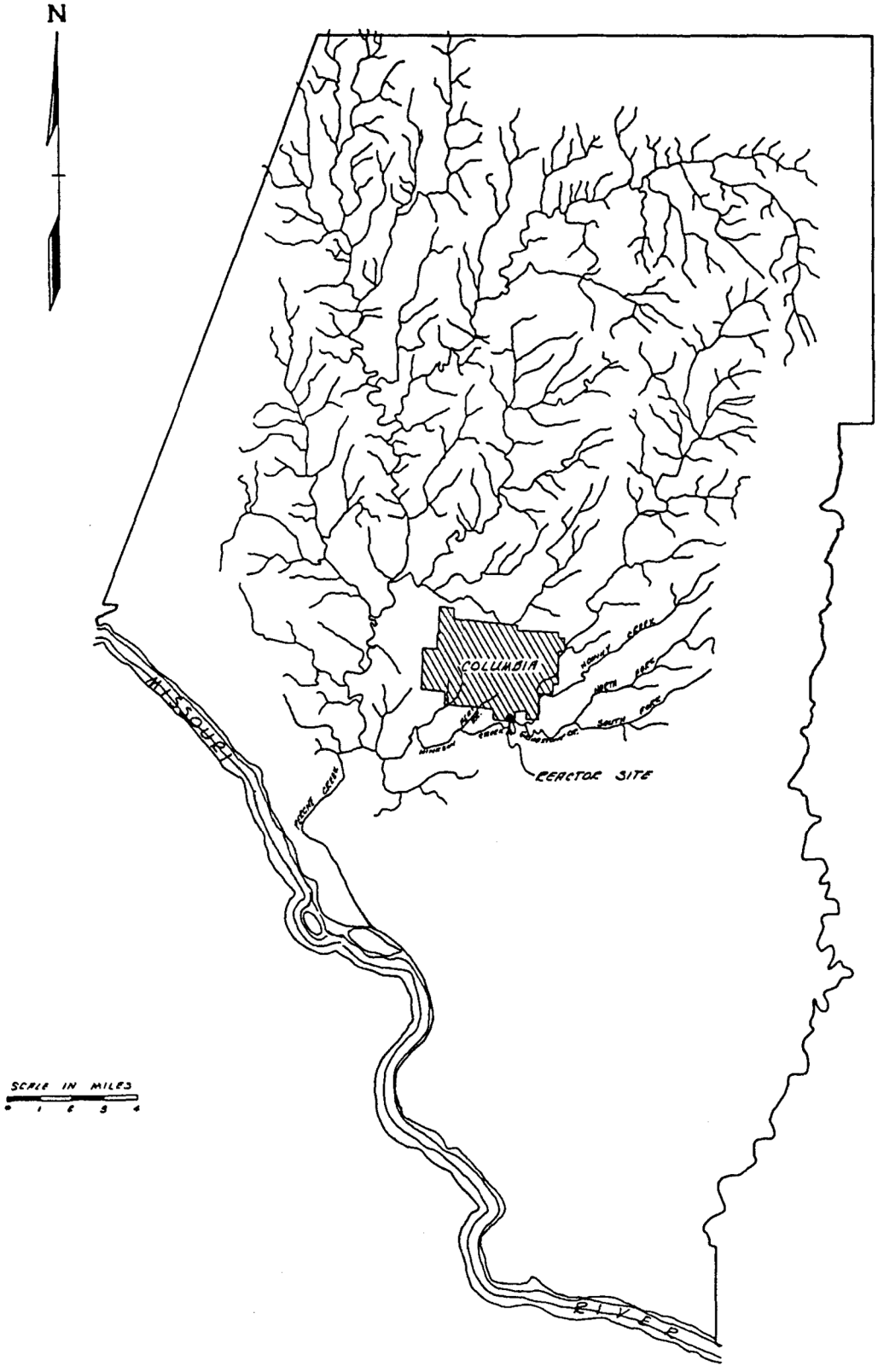


FIGURE 2.6 – PERCHE CREEK DRAINAGE IN BOONE COUNTY, MISSOURI

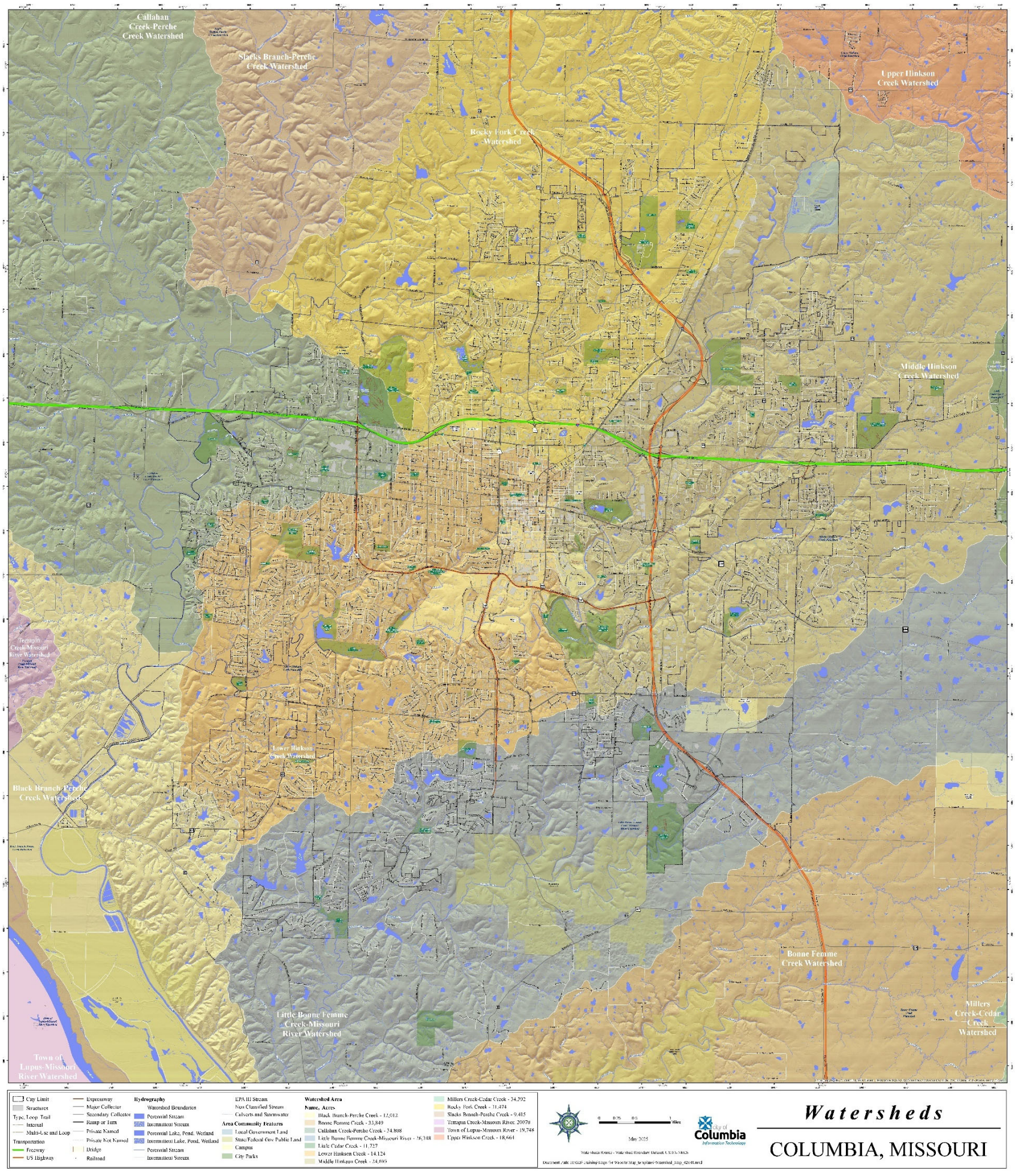


FIGURE 2.7 – WATERSHEDS FOR THE COLUMBIA AREA

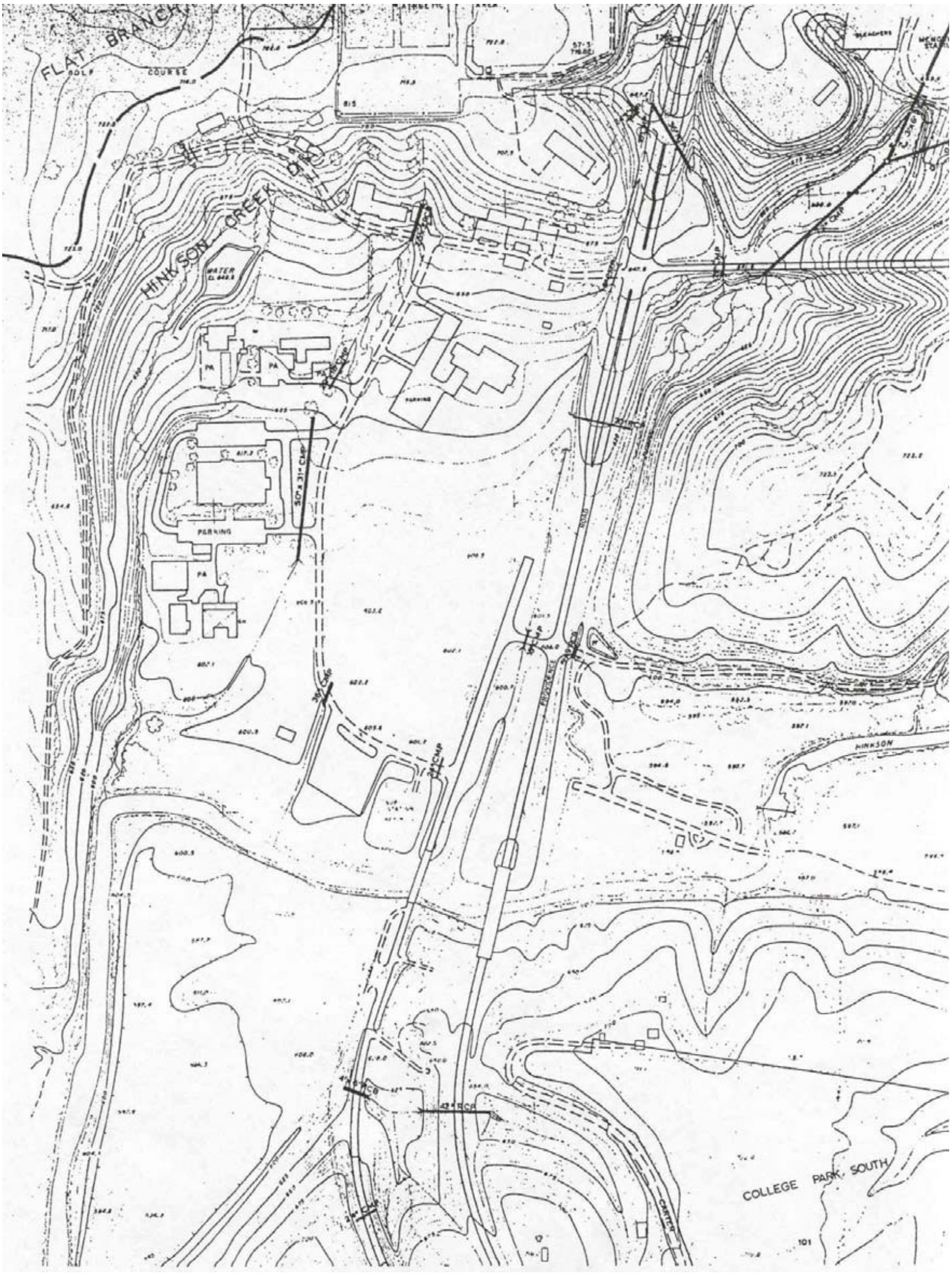
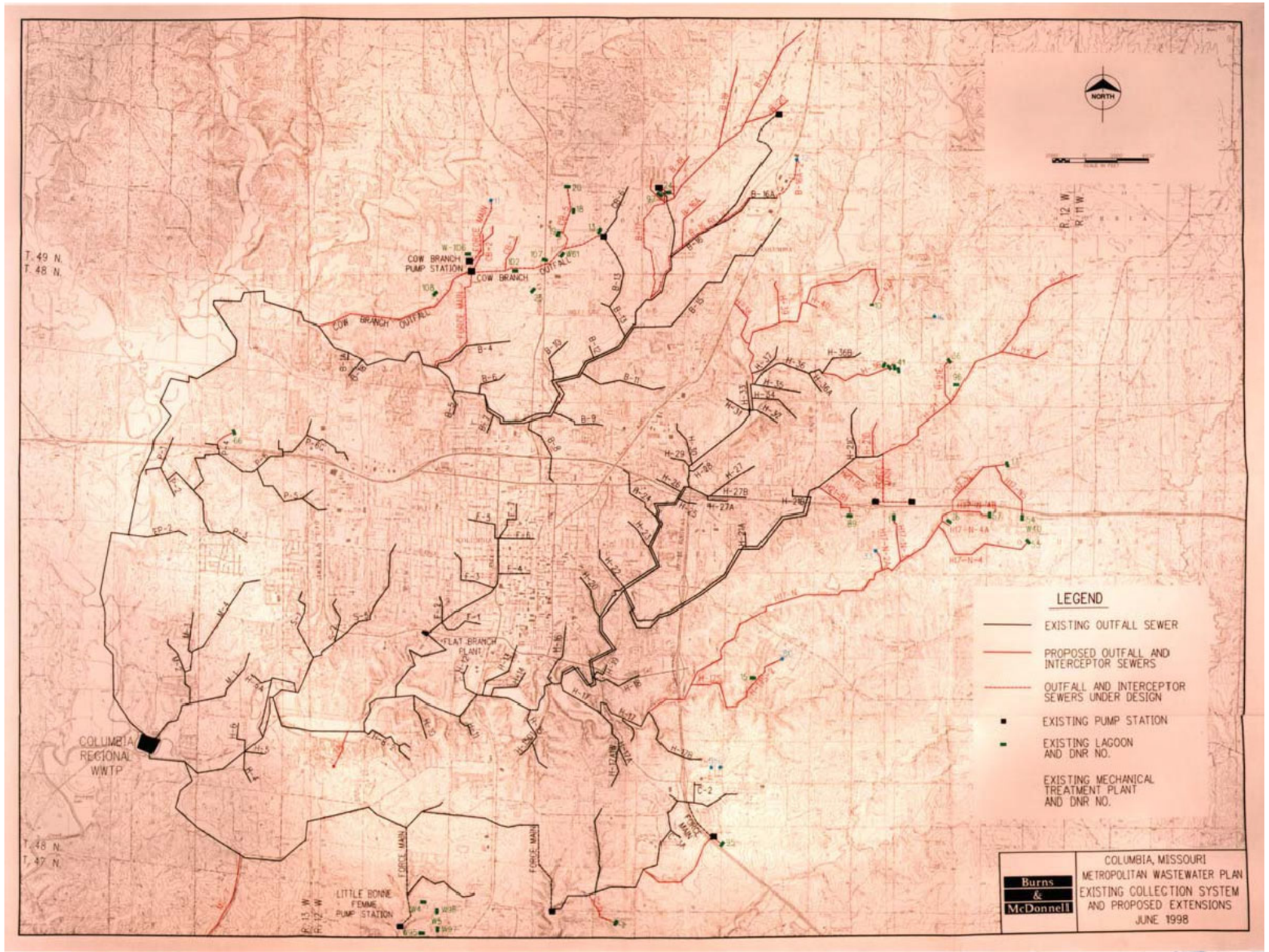


FIGURE 2.8 – STORMWATER DRAINAGE SYSTEM FOR THE REACTOR FACILITY SITE


NORTH
T. 49 N.
T. 48 N.
R. 12 W.
R. 11 W.
COW BRANCH PUMP STATION
COW BRANCH
COW BRANCH OUTFALL
FORCE MAIN
B-KA
H-36B
H-36A
H-27B
H-27A
H-29
H-30
H-31
H-32
H-33
H-34
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H-71
H-72
H-73
H-74
H-75
H-76

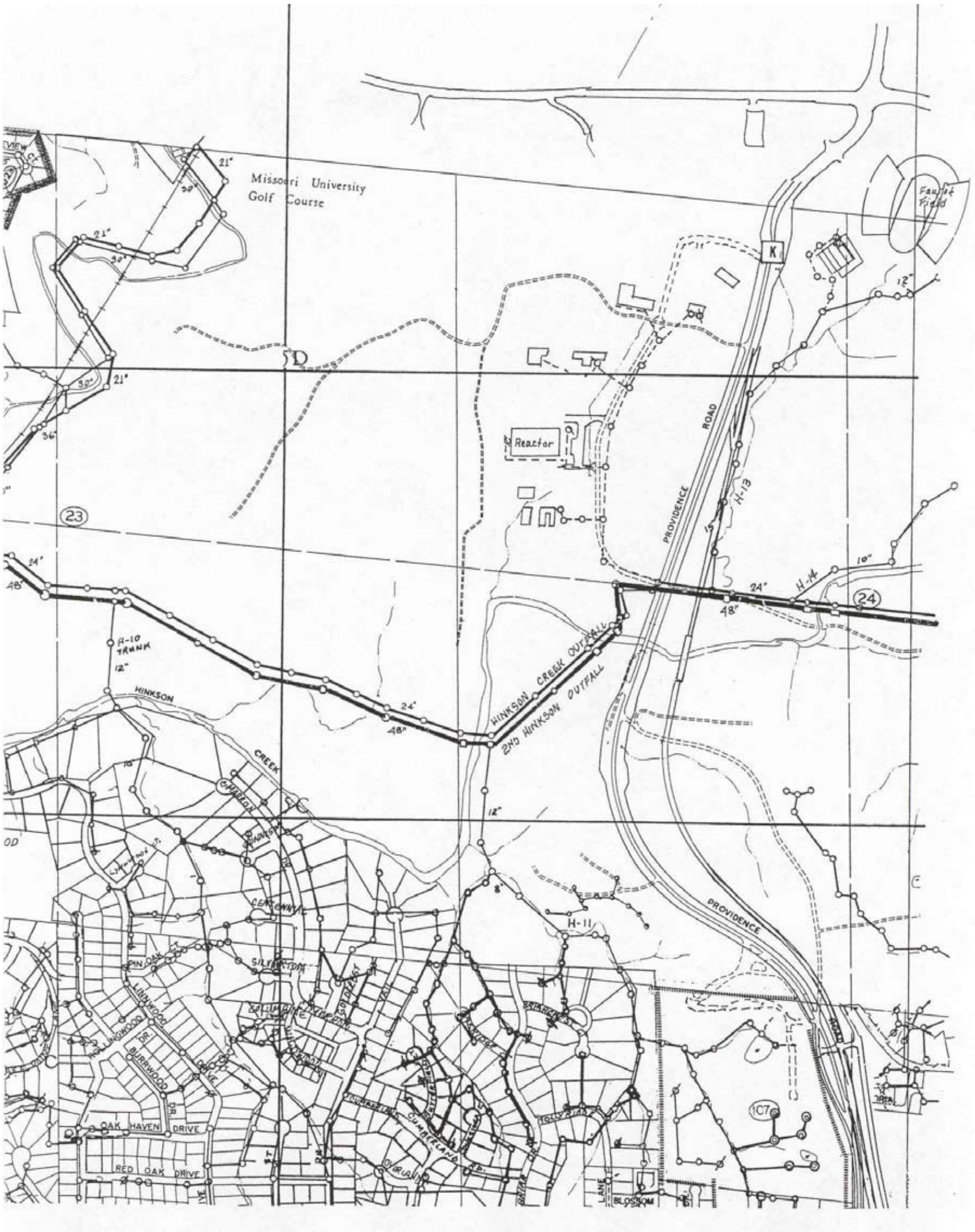


FIGURE 2.10 – UNIVERSITY SANITARY SEWER SYSTEM

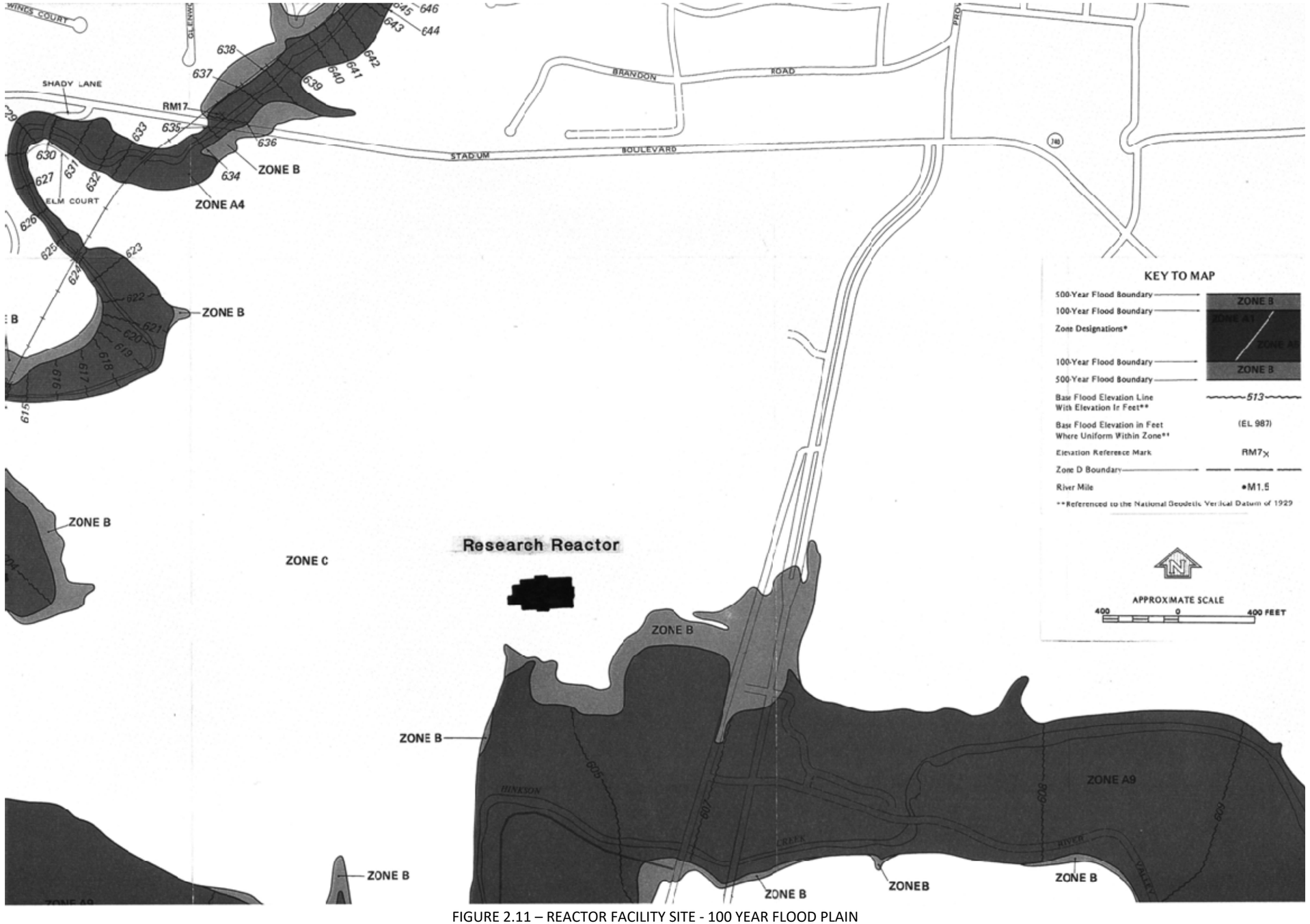
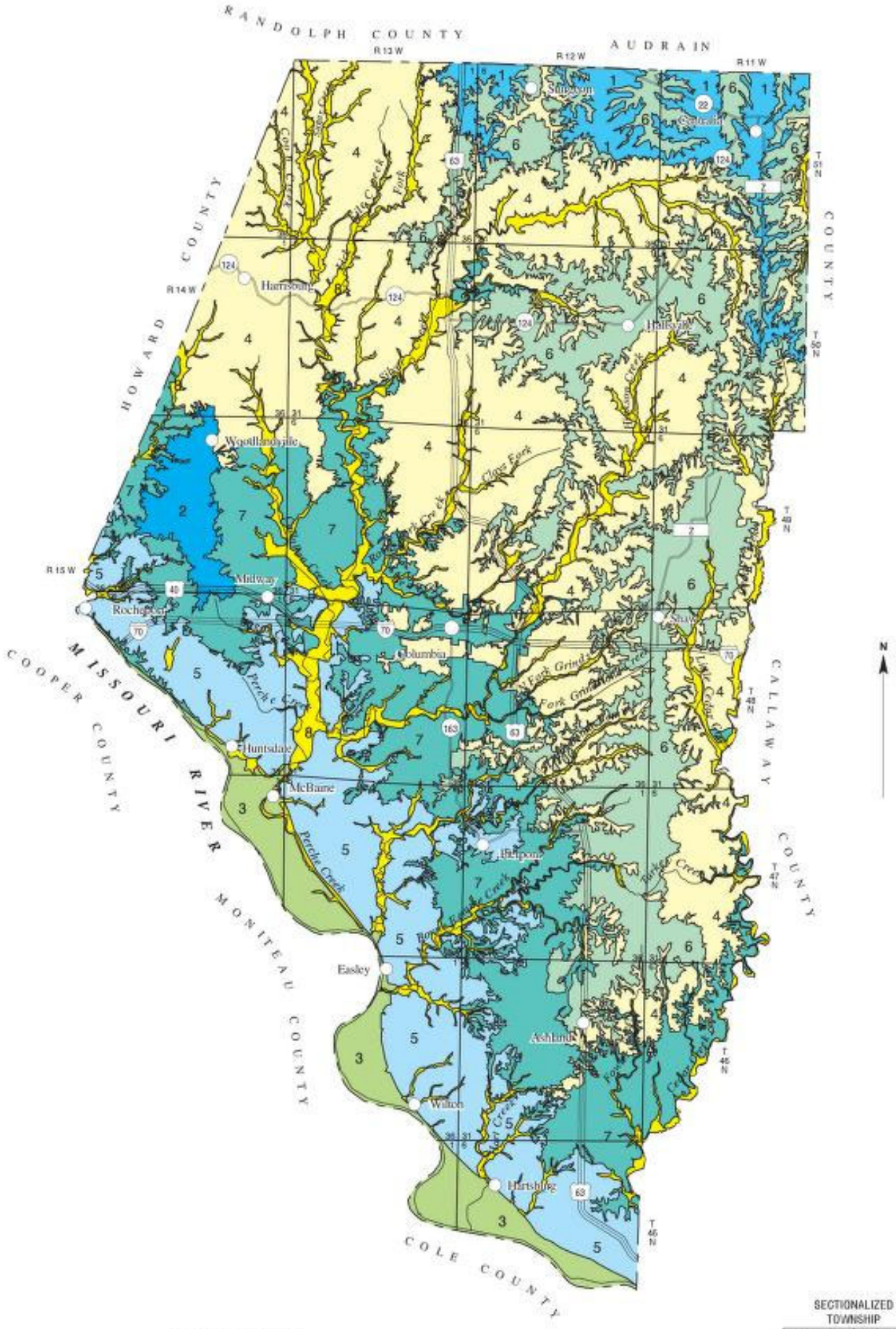


FIGURE 2.11 – REACTOR FACILITY SITE - 100 YEAR FLOOD PLAIN



SOIL LEGEND*

1	Adco-Mexico-Putnam
2	Arisburg-Armstrong
3	Haynie-Blake-Darwin-Leta
4	Keswick-Hatton-Winnegan
5	Menfro-Winfield-Rocheport
6	Mexico-Leonard
7	Weller-Bardley-Clinkenbeard
8	Wilbur-Moniteau-Perche-Haymond

*The units on this legend are described in the text under the heading "General Soil Map Units."

Compiled 1998

Each area outlined on this map consists of more than one kind of soil. The map is meant for general planning rather than a basis for decisions on the use of specific tracts.

SECTIONALIZED TOWNSHIP

6	5	4	3	2	1
7	8	9	10	11	12
18	17	16	15	14	13
19	20	21	22	23	24
30	29	28	27	26	25
31	32	33	34	35	36

UNITED STATES DEPARTMENT OF AGRICULTURE
NATURAL RESOURCES CONSERVATION SERVICE
in cooperation with
MISSOURI DEPARTMENT OF NATURAL RESOURCES
and MISSOURI AGRICULTURAL EXPERIMENT STATION

GENERAL SOIL MAP
BOONE COUNTY, MISSOURI

Scale 1:253440

1 0 1 2 3
MILES
1 0 1 2 3 4 5 6
KILOMETERS

FIGURE 2.12 – SOILS OF BOONE COUNTY, MISSOURI

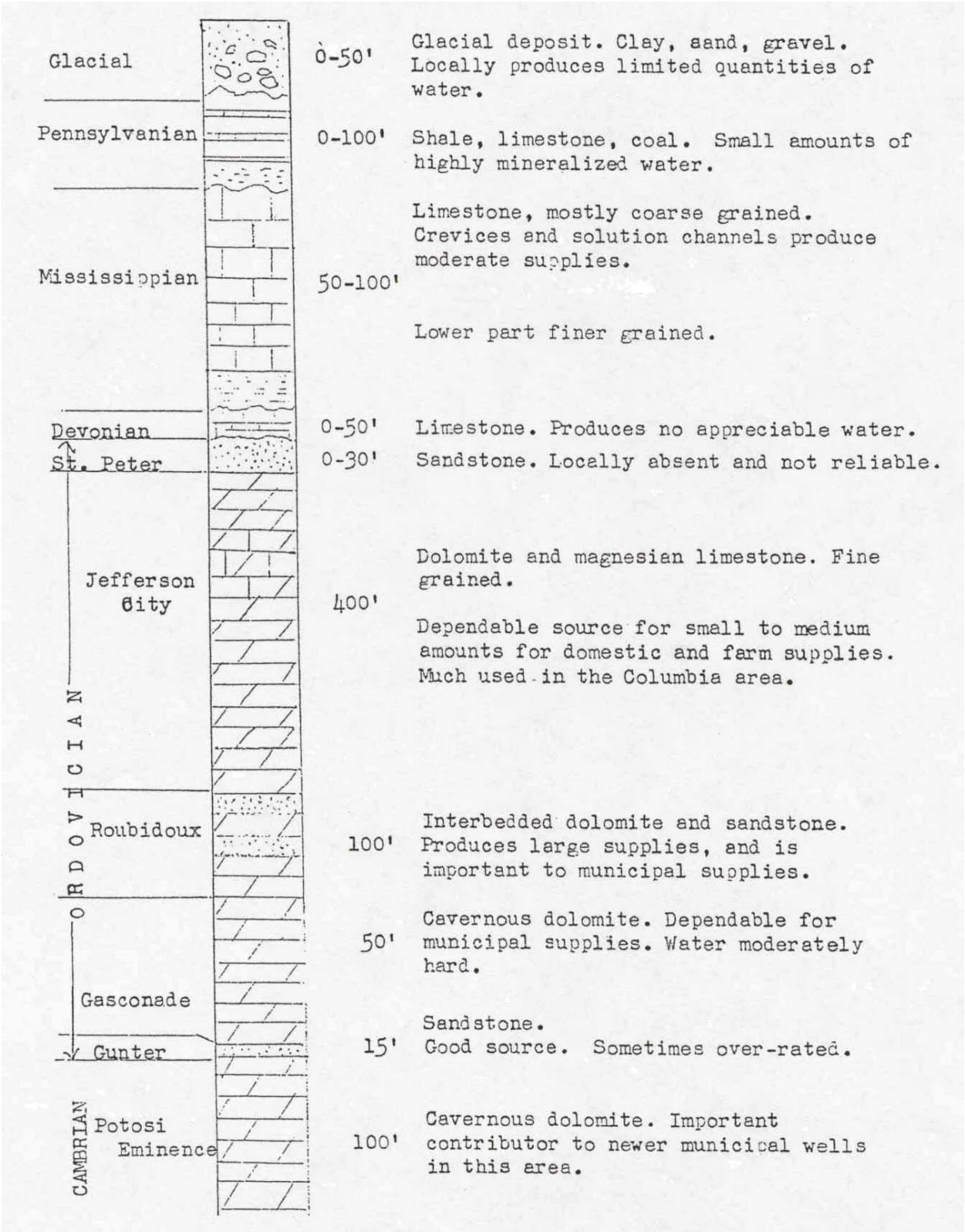
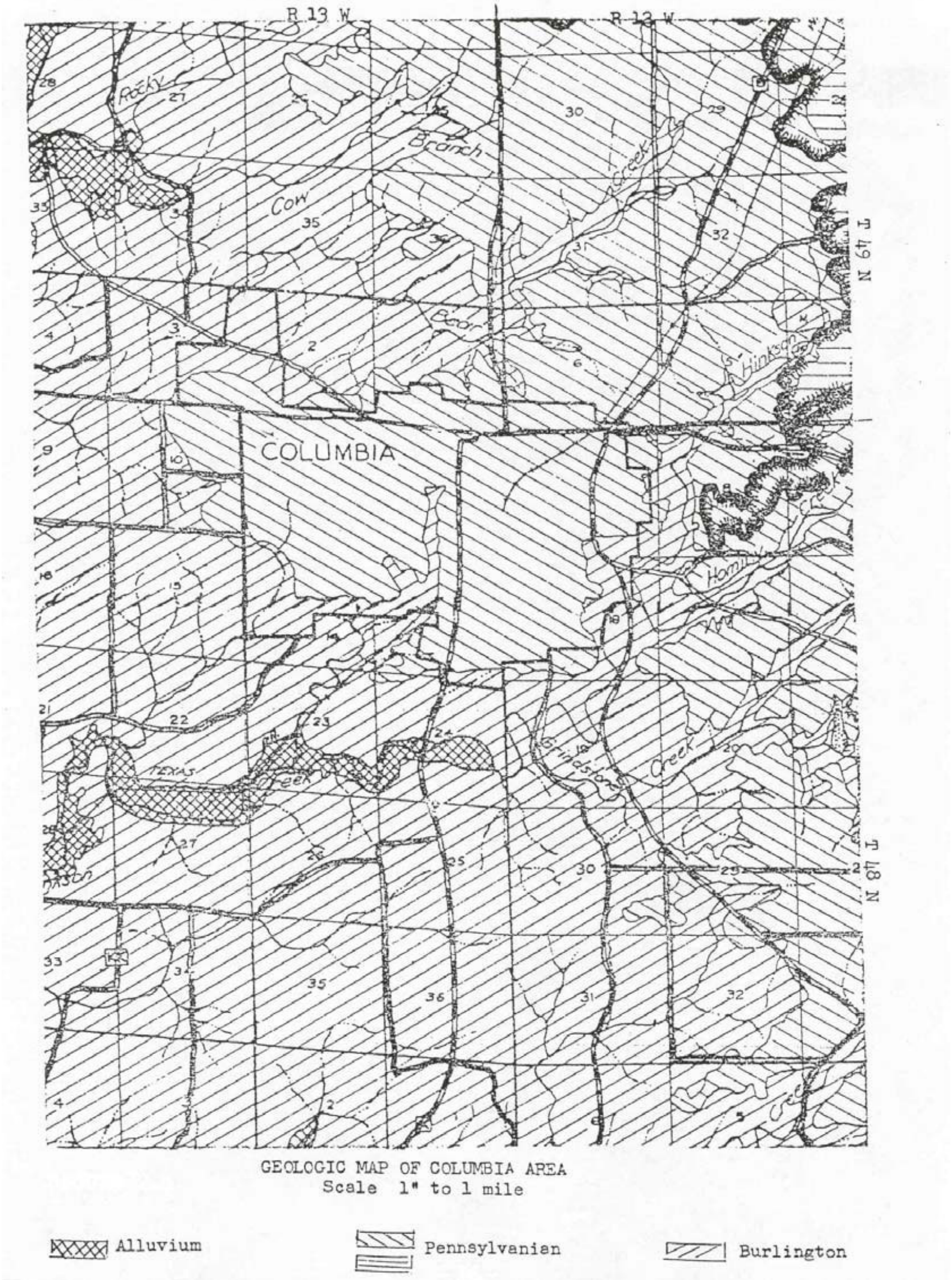


FIGURE 2.13 – GEOLOGIC COLUMN AT COLUMBIA, MISSOURI



GEOLOGIC MAP OF COLUMBIA AREA
Scale 1" to 1 mile

FIGURE 2.14 – GEOLOGIC MAP OF THE COLUMBIA AREA

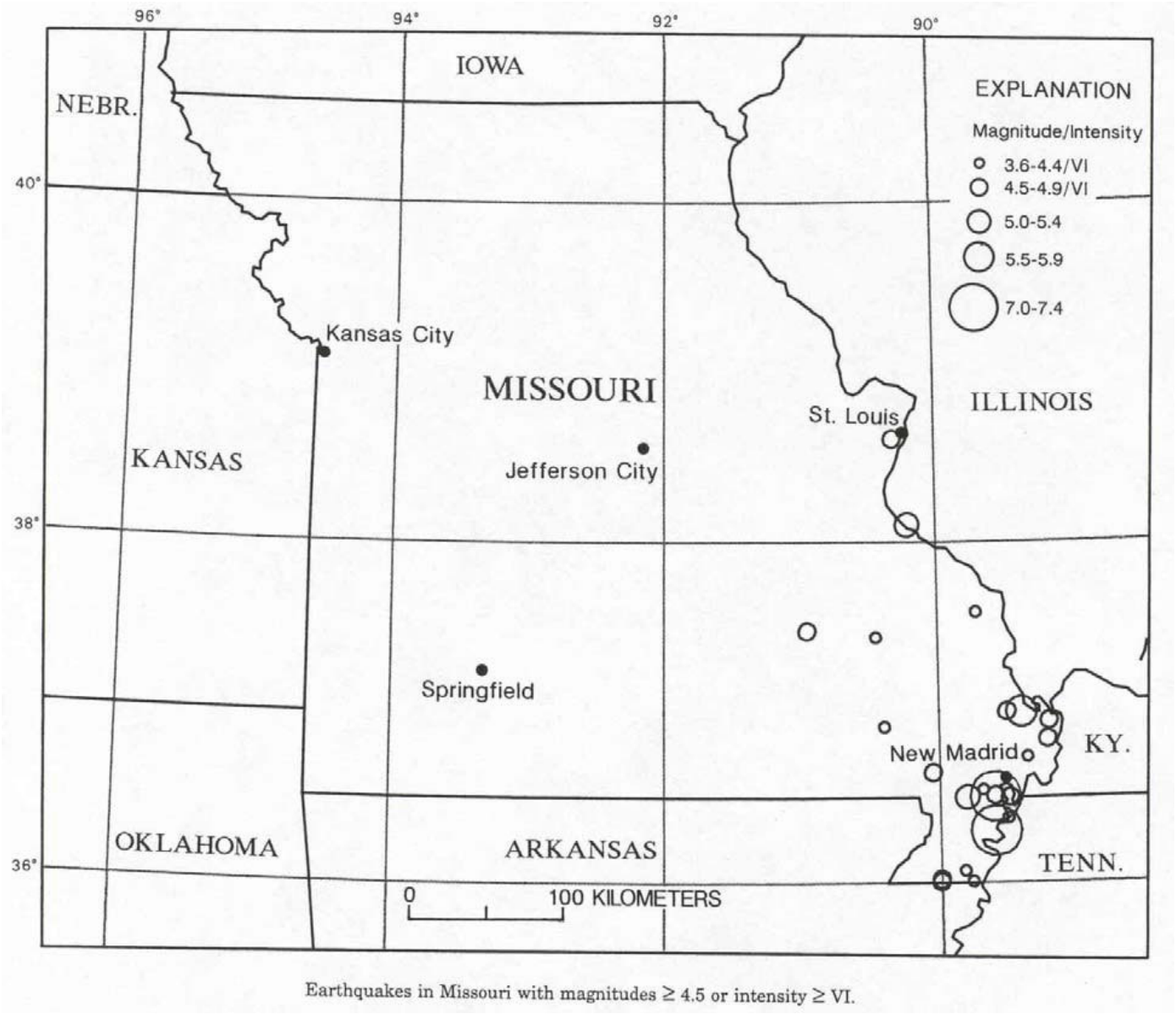


FIGURE 2.15 – NEW MADRID SEISMIC ZONE

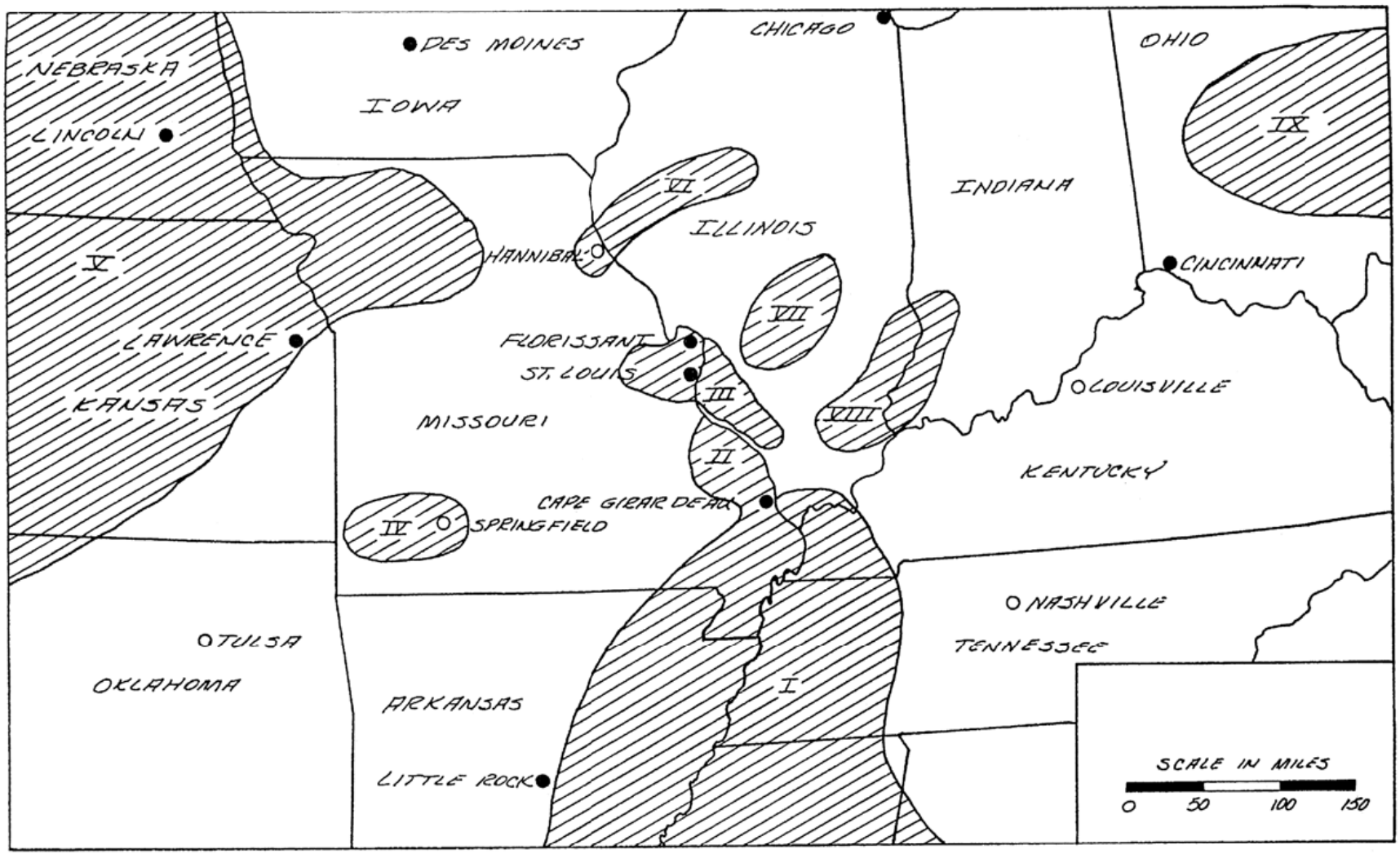
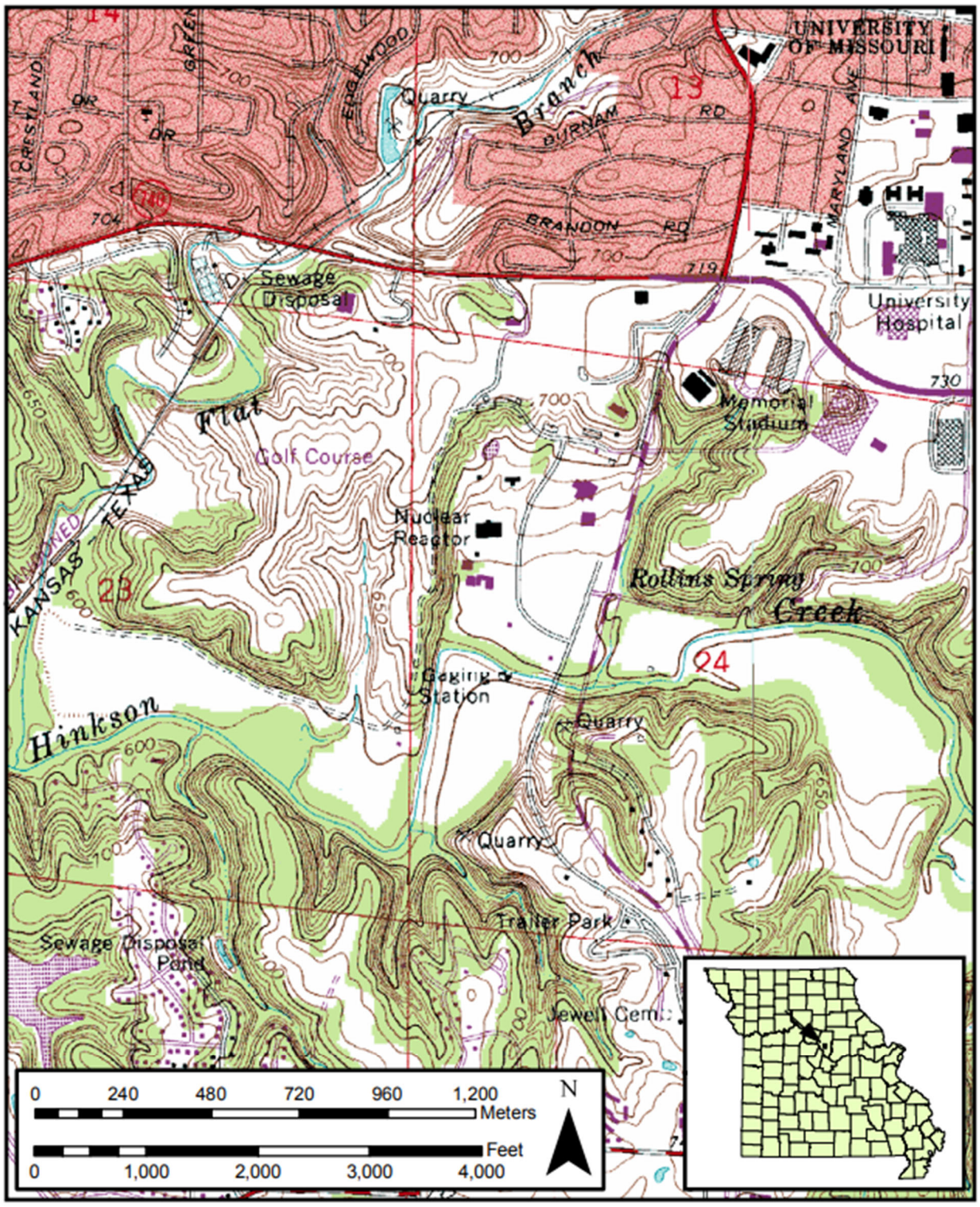


FIGURE 2.16 – EARTHQUAKE DISTRICTS IN THE STATE OF MISSOURI



The MURR facility (designated "Nuclear Reactor") is located in the Columbia quadrangle, Township 48N, Range 13W, and Section 24. Basemap: USGS 7.5-minute Columbia, MO topographic quadrangle (1967, photorevised in 1981). Prepared by Matthew T. Boulanger, May, 2010.

FIGURE 2.17 – TOPOGRAPHICAL MAP

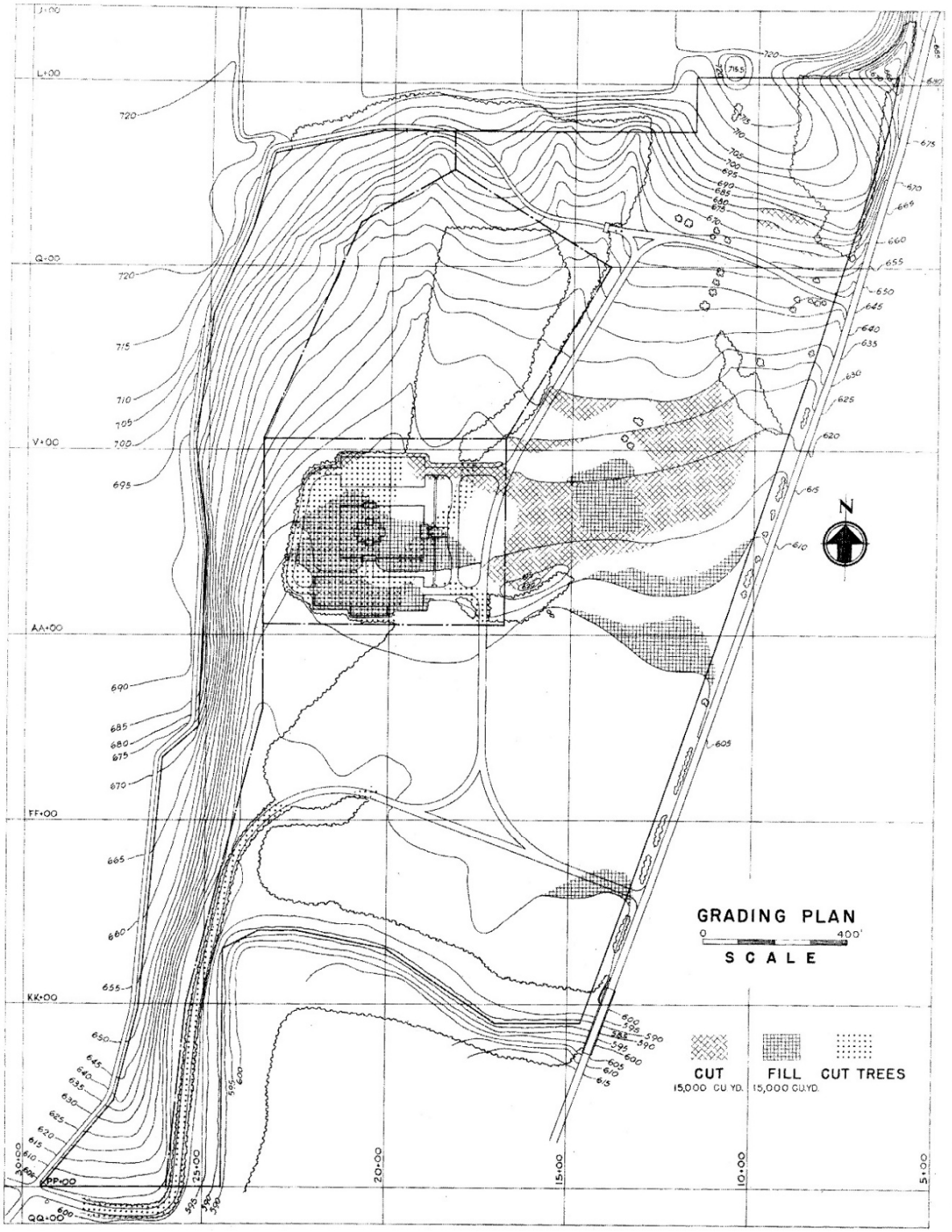


FIGURE 2.18 – TOPOGRAPHIC FILL AND CUT AREAS

CHAPTER 3

DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

TABLE OF CONTENTS

3.0 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS 3
3.1 Conformance with NRC General Design Criteria 3
3.1.1 Introduction 3
3.1.2 Overall Requirements (Criteria 1-5) 4
3.1.3 Protection by Multiple Fission Product Barriers (Criteria 10-19) 6
3.1.4 Protection and Reactivity Control Systems (Criteria 20-29) 11
3.1.5 Fluid Systems (Criteria 30-46) 15
3.1.6 Reactor Containment (Criteria 50-57) 19
3.1.7 Fuel and Radioactivity Control (Criteria 60-64) 22
3.2 Meteorological Damage 28
3.3 Water Damage 28
3.4 Seismic Damage 28
3.5 Mechanical System and Components 29
3.5.1 Fuel System 29
3.5.2 Control Rod Drive Mechanisms 30
3.5.3 Reactor Core Assembly Support Structure 31
3.5.4 Reactor Safety System 31
3.5.5 Containment System 31
3.5.6 Anti-Siphon System 32

LIST OF TABLES

Table 3-1 APPLICABILITY OF COMPLIANCE WITH GENERAL DESIGN CRITERIA 24

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.

3.6 Safety Evaluation Report Related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-Power Reactors, NUREG-1313, July 1988.

3.7 “University of Missouri at Columbia – Staff Assessment of Applicability of Fukushima Lessons Learned to University of Missouri - Columbia Research Reactor” NRC letter dated December 8, 2016 (ADAMS Accession No. ML15161A387).

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 (Reference 3.1), as they apply to the University of Missouri 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 (GDC) were formulated for the purpose of establishing minimum requirements for the principal design criteria for water-cooled nuclear power plants. The GDC 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. MURR is a research facility, its systems are not designed for and are not categorized according to power-plant application. The descriptions that follow are oriented toward the individual criterion rather than toward identification of areas of noncompliance with power plant criteria.

MURR is designed and licensed for operation at a maximum steady-state power level of 10 Megawatts (MW) thermal. MURR's fission-product inventory is substantially less than that of conventional nuclear power plants for which the GDC were primarily developed. In addition, MURR operates at relatively low temperatures and pressures. No active engineered safety features (ESFs) 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 MURR is in a much lower risk category when reviewing for compliance with the GDC.

The GDC are divided into the following six (6) 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 GDC to 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 (6) contractors: two (2) for design and construction, and four (4) for construction and testing. All design and construction work were 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 the existing standards and requirements. The contractors and their responsibilities are discussed in Section 1.7.2 and Reference 3.5.

Criterion 2: Design Bases for Protection Against Natural Phenomena

Tsunamis and large seiches do not occur in central Missouri. The influence of hurricanes on the climatology of the site is normally insignificant. Flooding in the area could be caused by 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 but infrequent tornadoes. Since 1950, only 1.3 percent of all tornadoes recorded in the state of Missouri were in Boone County. 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. 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:

- Periodic inspections are made for the purpose of reducing nonessential combustible material inventory.
- A fire detection/suppression system is installed in the facility.
- Fire extinguishers are strategically located throughout the facility.
- Established emergency procedures will be used in the event of a fire.
- The large volume of water in the reactor pool would protect the core from any conceivable fire.
- The reactor is failsafe and will shut down should fire damage the instrumentation and control (I&C) systems.
- Fire protection is not required to accomplish a safe shut down of the reactor or to maintain a safe shutdown condition.

Safe shutdown of the reactor is not dependent on the availability of forced cooling, decay heat removal (see Section 13.2.9.3), coolant charging systems, or emergency electrical power (see Section 13.2.7). Power availability and drive train thermal delay protection and/or fire suppression for these systems is not required from the standpoint of fire protection.

All physical components involved in shutting down the reactor are below the pool water line. These components are not vulnerable to the typical weaknesses, embrittlement, and deformation failures associated with a fire. As a result, thermal delay protection and/or fire suppression for these components is not required from the standpoint of fire protection.

All remaining components involved in safely shutting down the reactor are essentially electrical in nature and are utilized in a non-coincidence logic system. Under fire conditions, these control components, circuits, and instrument channels may fail, but will fail in either a grounded or an open circuit state. Either and any combination of these failure states in any of the circuits significant to safety will still result in the desired loss of power, and a resultant safe shutdown of the reactor. The failure state where ungrounded continuity could be maintained due to fire conditions would have to occur in multiple locations simultaneously with no grounding and no open circuit failures. As a result, thermal delay protection and/or fire suppression for these instrument channels, circuits, and control components is not required from the standpoint of fire protection.

Criterion 4: Environmental and Dynamic Effects Design Bases

The design of 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 reactor 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 near the bottom [approximately 11.5 feet (3.5 m) below ground level] of the reactor pool liner, 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 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 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 a single parameter criterion for maintaining the integrity of the fuel cladding. Maintaining the integrity of the fuel cladding requires that the cladding remains below the blistering temperature of 986 °F (530 °C) (Reference 3.6). Limiting Safety System Settings (LSSSs) are settings for automatic protection devices which prevent the safety limit from being exceeded. The LSSSs for MURR have been chosen such that the true value of any of the four (4) safety-related variables will not exceed the operating limits under the most severe anticipated transient.

Accident analyses presented in Chapter 13, "Accident Analyses," show that, under credible accident conditions, MURR safety limits will not be exceeded, and a fission product release exceeding the allowable radiation limits as stated in 10 CFR 20, would not occur.

Criterion 11: Reactor Inherent Protection

MURR has many of the inherent safety features characteristic of a heterogeneous, 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 the 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 system. 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 MURR operates, the reactor is tightly coupled and inherently stable with respect to space-time and xenon oscillations.

Criterion 13: Instrumentation and Control

The 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 I&C system; reactor safety system; ESF 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 eight (8) channels (only seven (7) 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 movement and interlocks 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 independent from the reactor safety system; however, it does provide a protective function by introducing shim blade insertion to terminate a transient prior to reaching any reactor safety system set point trip.

The process I&C 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 (4) 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 ESF actuation systems receive input signals from monitoring instruments and initiate the operation of the ESFs. The two (2) 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 210). Manual or automatic actuation of the reactor isolation system causes the following actions to occur:

- Reactor scram;
- Closing of all normally-open reactor containment building penetrations with sealable closures;
- Sounding of an audible alarm throughout the reactor containment building; and
- 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.) and ensures 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 inverted loop, instantaneously

establishing the pressure in this area at equal to or greater than atmospheric. This prevents 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:

- Reactor scram;
- Stopping primary coolant circulation pumps P501A and P501B;
- Closing primary coolant isolation valves V507A and V507B; and
- Opening 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 provide input signals to the ESF actuation system, which actuates the containment system, an ESF 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. MURR utilizes three (3) 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), providing 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, 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 emergency power 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 emergency power 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 two (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.

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) 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 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, the MURR control room was not designed with radiation protection 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. Should a radiological accident require complete evacuation of the reactor containment building, radiation levels within the reactor containment building can be displayed on the ARMS remote indicator located in the Instrumentation Support Shop (Room 282). Equipment at appropriate locations outside the reactor containment building is 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 (reactor safety system) has been designed to (1) initiate automatic actions to assure that the reactor safety limit is not exceeded as a result of anticipated operational occurrences, and (2) sense accident conditions and to initiate the operation of systems and components important to safety. The reactor protective 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 (4) control rod drive mechanism electromagnets, or to activate the ESFs.

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" (Reference 3.2). 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 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 were considered in the analysis of protective system reliability.

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 (1) of the reactor safety system trip actuator amplifiers (TAAs) could result in, at most, the failure to interrupt the current to two (2) shim blade electromagnets, in which case the other two (2) shim 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 (1) 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 (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.

The instrument channels to the reactor safety system from the process I&C 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 (4) 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 limit is not exceeded.

The failure of one (1) instrument safety channel does not affect the operation of another. The signal inputs are arranged in series such that action by any one (1) interrupts current to the control blade electromagnets resulting in a shutting down of the reactor. Additional redundancy is provided by two

(2) 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 an automatic 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 ensure reliability. The reactor protective system 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."

Criterion 22: Protection System Independence

The channels used in the reactor protective system are redundant. The circuitry for 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 are under continuous surveillance. Signal lines from the process sensors, transmitters, and controllers which provide redundant protective functions are separated and clearly identified. The reactor protective system satisfies the intent of IEEE-279 with regard to channel independence.

Criterion 23: Protection System Failure Modes

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 (1) 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 TAAs. A malfunction in one (1) of the TAAs could result in, at most, the failure to interrupt the current to two (2) shim blade electromagnets, in which case the other two (2) shim 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 (4) 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

MURR has five (5) independent reactivity control rods (blades): four (4) shim rods and one (1) regulating rod. Each of the control rods has its own drive mechanism and may be operated individually. The four (4) 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 (4) 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) have been demonstrated through decades 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.

Criterion 27: Combined Reactivity Control Systems Capability

This criterion is not applicable.

Criterion 28: Reactivity Limits

No conceivable malfunction of the reactivity control system will result in core damage. The continuous withdrawal of all four (4) shim blades, as discussed in Section 13.2.2.1.2, 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 (6) 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 MURR experiments ensure that (1) 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) the magnitude of the potential reactivity worth of each unsecured experiment is less than the value of reactivity which would cause a violation of the safety limit. As shown in Chapter 13, “Accident Analyses,” MURR can withstand a positive step reactivity insertion of 0.007 Δk with no core damage. Reactivity limits placed on the center test hole and each secured experiment of 0.006 Δk 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.

The discussion under Criterion 17 covers the facility response to situations involving a loss of normal electrical power.

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 or connected to the pool coolant system. 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. 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.

Criterion 31: Fracture Prevention of Reactor Coolant Pressure Boundary

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), providing sufficient margin to assure that the primary coolant system design pressure of 125 psig (0.86 MPa above atmosphere) will not be exceeded.

Criterion 32: Inspection of Reactor Coolant Pressure Boundary

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, 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.

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

Criterion 33: Reactor Coolant Make-Up

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 (2) stainless 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, "Accident Analyses," 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 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 reactor containment building would occur as a result of normal changes in atmospheric pressure and the pressure equilibrium between the reactor containment building and atmosphere. An extreme assumption would be a pressure change on the order of 0.7 inches of Hg (25.4 mm of Hg at 60 °C) from an initial atmospheric pressure of 15.0333 psia (103.651 kpa). 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. 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, 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 (1) 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 is 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, "Accident Analyses," no credible accident has been identified which can cause an increase in pressure within the reactor 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 building. In the event of an overpressure condition within the reactor 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 reactor 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 (3) lines which penetrate the containment structure. One line each is used for: pressurizing the reactor containment building, attaching the instruments necessary for measuring internal pressure and leakage rate, depressurizing the reactor containment building upon completion of the leakage rate test.

Criterion 53: Provisions for Containment Testing and Inspection

The reactor containment building leakage rate test is performed biennially to ensure that the reactor

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 reactor containment building, pass through two (2) steel penetration plates located in the containment structure walls. Each penetration plate contains sealed connectors which allow electrical lines to enter or exit the reactor 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 the reactor containment building, and therefore, compliance with this criterion is conditional.

Criterion 56: Primary Containment Isolation

Penetrations through the reactor 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 (2) 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 (1) door is always closed and sealed, ensuring containment integrity is maintained.

The truck entry door (Door 101) 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 reactor containment building supply and exhaust air ducts contain two (2) 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 reactor containment building supply and exhaust plenums, immediately adjacent to Doors 504 and 505. These doors also shut on an automatic closure signal from the containment actuation system. A separate ventilation duct discharges potentially contaminated gases from the reactor containment building. Exhaust air from areas which produce radioactive gases or airborne contamination is ducted to a 16-inch (0.41-m) line which penetrates the west wall of the containment structure and discharges to the facility exhaust plenum. A 12-inch (0.31-m) long steel collar surrounding the 16-inch (0.41-m) pipe is cast into the reactor containment building wall. The 16-inch (0.41-m) pipe is welded to this collar. Two (2) quick-closing isolation valves are located on this line within the reactor 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.

The 6-inch (0.15-m) fire protection water line and the 6-inch (0.15-m) emergency pool fill line have manual isolation valves in close proximity to the laboratory building.

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 the reactor containment building, compliance with this criterion is conditional.

Criterion 57: Closed System Isolation Valves

Piping systems which enter the reactor containment building that are neither 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 the reactor containment building.

The reactor 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.). Identification and measurement of individual component leakage is not performed.

3.1.7 Fuel and Radioactivity Control (Criteria 60-64)

Criterion 60: Control of Releases 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 (2) buildings is never mixed until this point. Potentially contaminated air is 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. All irradiated fuel elements are stored in special storage baskets (Criterion 62) located in the reactor pool. When fuel is stored in the deep pool, the water provides a minimum shield thickness of at least 26 feet (7.9 m). When fuel is stored in the spent fuel storage location, the water provides a minimum shield thickness of 20 feet (6.1 m). 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.

New, unirradiated fuel elements are stored in a criticality-safe configuration in a fuel storage vault. These elements require no special handling arrangements or radiation shielding.

Criterion 62: Prevention of Criticality in Fuel Storage and Handling

Fuel is stored in baskets located in the reactor pool. 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.

Procedurally, only one (1) 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 (0.41-m) line. The recirculated containment air is monitored by the ARMS. A radiation level greater than the set point of either one (1) of the two (2) radiation monitors in the exhaust plenum will provide an input signal to the ESFs actuation system, which initiates the containment isolation system, an ESF which provides a complete isolation of the reactor containment building.

Exhaust air from both the laboratory and reactor containment buildings’ 16-inch (0.41-m) 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 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 sewerage.

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. Suppression 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. Electric Power Systems	X			
18. Inspection and Testing of Electric Power Systems	X			
19. Control Room			X	

TABLE 3-1
APPLICABILITY OF COMPLIANCE WITH GENERAL DESIGN CRITERIA (CONT.)

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			

TABLE 3-1
APPLICABILITY OF COMPLIANCE WITH GENERAL DESIGN CRITERIA (CONT.)

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		
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			

TABLE 3-1
APPLICABILITY OF COMPLIANCE WITH GENERAL DESIGN CRITERIA (CONT.)

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 but infrequent tornadoes. Between 1950 and 2025, fewer than one percent of all tornadoes recorded in the state of Missouri were in Boone County. 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. 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 and built for a peak internal pressure of 2.0 psig (13.8 kPa above atmosphere), which is the dominant architectural feature regarding loading. Among other elements, this feature results in eight (8) right angle shear walls external to the containment structure. The shear walls are enclosed to form the four (4) towers surrounding the containment structure. In addition to significant shear strength provided against wind loading, the resulting geometry presents an approximation to a more circular structure, rather than the base rectangular structure. This geometry results in significant reductions to both direct wind pressure and vortex induced negative pressure down-wind. A typical 3-second wind gust design speed for structures in the Midwest is 90 mph (40.2 m/s) and was the most likely design speed for this structure. Compare this with Table 2.10, "Normal and Extreme Winds for Columbia, Missouri", which shows that the maximum recorded 3-second wind gust speed during the period 1991-2022 was 81 mph (36.2 m/s).

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

3.3 Water Damage

The MURR facility is situated about 200 feet (61 m) from the nearest 100-year floodplain. As discussed in Section 2.4.5, there are no credible hydrological events which can lead to flooding or other water-induced damage to the MURR 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 reactor containment building has been determined to be structurally adequate to resist the OBE and SSE seismic events.

The NRC staff identified the need for additional information for three (3) high-power research and test reactors, including MURR, in a preliminary assessment dated March 2, 2015, "Draft White Paper Applicability of Fukushima Lessons Learned to Facilities Other than Operating Power Reactors." The

assessment was further updated, finalized, and provided to the Commission in SECY 15-0081, "Staff Evaluation of Applicability of Lessons Learned from the Fukushima Daiichi Accident to Facilities Other than Operating Power Reactors." The NRC staff assessed the seismic hazard of MURR using the following present-day methodologies:

- Electric Power Research Institute, 2012, "Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic;"
- "Design Response Spectra for Seismic Design of Nuclear Power Plants." Regulatory Guide 1.60, Revision 2; and
- "Design-Basis Tornadoes and Tornado Missiles in Nuclear Power Plants." Regulatory Guide 1.76, Revision 1.

Based on the re-evaluated seismic hazard, the Ground Motion Response Spectra (GMRS) is enveloped by the SSE in the 1 to 16 Hz range. While the GMRS exceeds the SSE at frequencies larger than 16 Hz, the "fail safe" failure mode of the reactor safety features will drop the control rods in such an event and the passive design of the reactor does not require electric power for safe shutdown. Therefore, the NRC staff concluded that no additional assessments would be needed for seismic-related hazards. Additionally, no other assessment would be needed for seismic-induced sloshing and high-wind-related hazards (Reference 3.7).

3.5 Mechanical System and Components

MURR does not have structures, systems, or components that are important to safety in the same context as water-cooled nuclear power plants. For MURR, a Loss of Coolant Accident (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" (Reference 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.

MURR's design does not require electrical power to safely shut down the reactor, maintain an acceptable shutdown condition, or for protection of fuel element integrity. However, certain systems and components are required to function as designed to ensure safe operation and shutdown 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 through decades of continuous operation at MURR and 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 (UAl_x) dispersion matrix with a maximum loading of 775 grams of uranium-235 (U-235). 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 MURR. Several features of the UAl_x dispersion fuel system contribute to its extended performance capability in the high flux reactors.

- Powder dispersion allows voids to be fabricated into the fuel matrix for the accommodation of the increased volume of fission products.
- Burnable poisons can be readily dispersed in the fuel matrix.
- 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 U-235 (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 analysis is maintained, the reactor core consists of eight (8) fuel assemblies. However, operation to 100 watts above shutdown power on less than eight (8) 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 (2) relatively new fuel assemblies, four (4) fuel assemblies that are approximately at the "half-way" point, and two (2) fuel assemblies that are approaching their burnup limit. 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 (4) 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 gearbox and adapter. 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 (2) inches (5.1 cm) per minute in the inward direction and one (1) 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 dashpot 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 (0.1 m) per minute in both the inward and outward directions, thereby requiring a different reduction gearbox.

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.

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. (Reference 3.4)

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 reactor containment building exhaust plenum. A reactor isolation can be manually actuated by switches in the reactor control room (Room 302) or the facility lobby (Room 210).

The reactor containment building is designed to hold a peak internal overpressure of 2.0 psig (13.8 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 an overpressure condition within the reactor 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 various safety instrumentation and equipment (e.g., pressure sensors, pump and valve interlocks, etc.) and ensures that the reactor core does not become uncovered during a LOCA. The 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, 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

TABLE OF CONTENTS

4.0 REACTOR DESCRIPTION 8
4.1 Introduction 8
4.2 Reactor Core 9
4.2.1 Reactor Fuel 9
4.2.1.1 Reactor Fuel System 9
4.2.1.2 Fuel Element Description 12
4.2.1.3 Fuel Element Materials of Construction 14
4.2.1.4 Fuel Element Acceptance Criteria 16
4.2.1.5 Core Loadings 16
4.2.1.6 Technical Specifications 16
4.2.2 Control Blades 17
4.2.2.1 Description 17
4.2.2.2 Evaluation of the Control Blades 20
4.2.2.3 Evaluation of Control Blade Thermal Distortion 20
4.2.2.4 Technical Specifications 21
4.2.3 Neutron Moderator and Reflector 21
4.2.4 Neutron Startup Source 23
4.2.5 Reactor Core Assembly Support Structure 24
4.2.5.1 Description 24
4.2.5.2 Evaluation of Radiation Damage, Coolant Chemistry and Reactivity Effects 25
4.3 Reactor Pool 26
4.4 Biological Shield 27
4.5 Nuclear Design 30
4.5.1 Early Nuclear Design Methods (Historical Methods) 31
4.5.2 Recent Nuclear Design Methods (Since August 2006) 36
4.5.3 Normal Operating Conditions 39
4.5.4 Reactor Core Physics Parameters 40
4.5.5 Operating Limits 40
4.6 Thermal-Hydraulic Design 44
4.6.1 Natural Convective Cooling Analysis 45
4.6.2 Steady-State Forced Cooling Analysis 46
4.6.3 Safety Limit Analyses 50
4.6.4 Bases for Limiting Safety System Settings 50
4.6.5 Decay Heat Removal Analysis 50

LIST OF TABLES

Table 4-1 URANIUM LOADING FOR 775-GRAM FUEL ELEMENTS 11
Table 4-2 SUMMARY SHEET FOR MURR FUEL ELEMENTS 12

Table 4-3 SUMMARY OF FUEL ELEMENT SPECIFICATIONS 14
Table 4-4 ISOTOPIC COMPOSITION OF URANIUM METAL FOR UALx DISPERSION FUEL SYSTEM 15
Table 4-5 MAXIMUM IMPURITIES IN URANIUM METAL FOR UAL_X DISPERSION FUEL SYSTEM 15
Table 4-6 FUEL PLATE TOLERANCES 16
Table 4-7 MURR R-Z ZONE SPECIFICATIONS 33
Table 4-8 BOLD VENTURE R-Z RESULTS AT COLD CLEAN CORE CONDITION 34
Table 4-9 BOLD VENTURE R-Z DEPLETION RESULTS 35
Table 4-10 MURR 3D MODEL RESULTS 36
Table 4-11 TOTAL CONTROL BLADE WORTH 36
Table 4-12 SUMMARY OF TYPICAL REACTOR AND MAJOR NEUTRON PHYSICS DATA 42
Table 4-13 SUMMARY OF HEAT TRANSFER DATA 48

LIST OF FIGURES

Figure 4.1 – REACTOR ASSEMBLY SECTION IN ELEVATION VIEW 52
Figure 4.2 – REACTOR ASSEMBLY SECTIONS IN PLAN VIEW 53
Figure 4.3 – FUEL ELEMENT - PICTORIAL VIEW 54
Figure 4.4 – MURR CORE LOADING 55
Figure 4.5 – OFFSET MECHANISM 56
Figure 4.6 – REACTOR ASSEMBLY AND ARRANGEMENT 57
Figure 4.7 – PLOT OF THE MCNP5 PREDICTED VOLUMETRIC HEATING FUNCTION FOR THE FIRST SIX INCHES (15.24 CM) OF THE BIOLOGICAL SHIELD CONCRETE 58
Figure 4.8 – SCHEMATIC OF THE ONE-DIMENSIONAL HEAT TRANSFER MODEL USED TO STUDY THE TEMPERATURE PROFILE IN THE SHIELD DUE TO GAMMA ENERGY DEPOSITION 59
Figure 4.9 – TEMPERATURE PROFILE DUE TO GAMMA HEATING ACROSS THE BOLOGICAL SHIELD 60
Figure 4.10 – TEMPERATURE PROFILE FOR ADIABATIC HEAT TRANSFER TO THE AIR OUTSIDE THE BIOLOGICAL SHIELD 61
Figure 4.11 – REACTOR AND BIOLOGICAL SHIELD - PLAN AND SECTION VIEW 62
Figure 4.12 – CALCULATIONAL SEQUENCE OF AMPX-II 63
Figure 4.13 – COMPUTATIONAL MODULES OF BOLD VENTURE 64
Figure 4.14 – R-Z MODEL OF THE MURR REACTOR 65
Figure 4.15 – Θ-R MODEL OF THE MURR CORE AND BERYLLIUM REFLECTOR 66
Figure 4.16 – COMPARISON OF PEAK BURNUP IN FUEL PLATES OF ELEMENT 775F3 67
Figure 4.17 – LINKAGE OF CODES USED IN THE ANALYSIS SUITE 68
Figure 4.18 – CODE SUITE FLOW DIAGRAM IMPLEMENTED AT MURR FOR ROUTINE CORE PHYSICS ANALYSIS 69

ATTACHMENTS

ATTACHMENT A	Current MURR Safety Limit Analyses
ATTACHMENT B	Current Bases for Limiting Safety System Settings
ATTACHMENT C	Past MURR Safety Limit and Limiting Safety System Settings Analyses

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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 University of Missouri Research Reactor (MURR) is a pressurized, reflected, heterogeneous, open pool-type reactor, 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 (2) cylindrical reactor pressure vessels. Each fuel assembly is comprised of 24 circumferential plates containing uranium enriched to approximately 93% in the isotope uranium-235 (U-235) 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 (2) 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 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 (6) 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 650 grams of U-235. This type of fuel system 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 U-235 needed per MWD of energy produced at MURR, a conversion was performed in 1971 to switch to a uranium-aluminide (UAl_x) dispersion fuel material with a maximum loading of 775 grams of U-235 per assembly.

The MURR fuel elements are 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 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 voids 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. (Reference 4.52) 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 high-enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel (Reference 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. (Reference 4.57) A more detailed discussion of MURR's conversion status is included in Chapter 18, "High-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 U-235 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 in test and research reactors such as the ATR and MURR. MURR has used more than 1000 UAl_x fuel elements since 1971 with no failures. ATR has used more than 5,190 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 twenty years. There also have been no aluminide fuel element failures at MURR due to pitting corrosion (Reference 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 (Reference 4.47). A pitting corrosion 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). Based on the many years of experience that 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 (Reference 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 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), the Fuel Failure During Reactor Operations Accident or the Fuel Handling Accident as described in Chapter 13, "Accident Analyses."

TABLE 4-1
URANIUM LOADING FOR 775-GRAM FUEL ELEMENTS

Plate Number	**U-235 Content (grams) +/- 1% per plate**
1	19.26
2	20.39
3	21.53
4	22.66
5	23.79
6	24.93
7	26.06
8	27.19
9	28.33
10	29.46
11	30.59
12	31.73
13	32.86
14	33.99
15	35.12
16	36.26
17	37.39
18	38.52
19	39.66
20	40.79
21	41.92
22	43.06
23	44.19
24	45.32
Total	775.0

TABLE 4-2
SUMMARY SHEET FOR MURR FUEL ELEMENTS

Description	**Nominal Value**
Fuel Content (grams U-235 per element)	775
Type of Fuel	Aluminide-UAl_x (mostly UAl_3 phase)
Fuel Density (grams of U-235 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

4.2.1.2 Fuel Element Description

A drawing of the MURR fuel element is shown in Figure 4.3. Eight (8) such fuel elements comprise the fixed MURR core. This design has proven to be reliable through decades of continuous operation at MURR and in other research reactors with no significant failures. Fuel element specifications are summarized in Table 4-3.

The fuel elements have a radial dimension of 3.21 inches (8.15 cm) as determined from radial length from the inside of the inner roller to the outside of the outer roller and an overall length of 32.5 inches (82.55 cm). 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. (Reference 4.12) 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 (68.04 kg) 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). Fuel assemblies of similar construction have withstood severe hydraulic tests (Reference 4.51) at flow velocities up to 50 feet/sec (15.24 m/sec) without distortion, indicating an adequate design margin in this regard. The side plates are 31.75 inches (80.65 cm) long by 3.16 inches (8.03 cm) wide and 0.15 inches (3.81 mm) thick.

A handling and guide fixture (end fitting) at each end hold 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 6 kg.

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).

TABLE 4-3
SUMMARY OF FUEL ELEMENT SPECIFICATIONS

Description	Nominal Value
Fuel Material	
Enrichment U-235	93%
Thickness	0.020 inches (0.508 mm)
Cladding	
Material	Aluminum
Thickness	0.015 inches (0.381 mm)
Fuel Assembly	
Number of Fuel Plates	24
*Innermost Fuel Plate Center Radius	2.795 inches (7.099 cm)
*Outermost Fuel Plate Center Radius	5.785 inches (14.694 cm)
Overall Fuel Assembly Length	32.5 inches (82.550 cm)
Overall Fuel Plate Length	25.5 inches (64.770 cm)
Overall Active Fuel Length	24.0 inches (60.960 cm)
Fuel Plate Thickness	0.050 inches (1.270 mm)
Distance Between Fuel Plates	0.080 inches (2.032 mm)
Maximum U-235 Loading UAl_x Dispersion Fuel	775 grams
Weight	≈ 6 Kg

*The fuel plate center radius is calculated by adding half the thickness on the fuel plate (0.025 inches) to the nominal radius of the innermost (2.77 inches) and outermost (5.76 inches) fuel plates.

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 U-235 (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. The isotopic composition of the uranium metal is listed in Table 4-4.

The maximum individual impurities in the fissionable material are shown in Table 4-5. The aluminum for the fuel material is 99.8% pure alloy melt stock (Ref 4.49). 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.

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

Isotope	**Limit (WT% of Total)**
U-234	1.20 Maximum
U-235	93.0 ±1.0
U-236	0.70 Maximum
U-238	6.0 ±1.0

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

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

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 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 Technical Specifications

Technical Specification (TS) 4.1.d. states that one out of every eight (8) fuel elements that have reached their end-of-life shall 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 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 externally to the pressure vessels. Each control blade is coupled to a drive mechanism by means of a support and guide extension (offset mechanism). This offset mechanism is shown in Figure 4.5. Four (4) 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 control blades are constructed of formed BORAL® plate which is, by weight, 52% ± 2% boron carbide and 48% ± 2% 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 ± 0.007 inches (4.445 ± 0.178 mm). The minimum weight of boron-10 per unit cross sectional area is 0.0418 gm/cm^2. All four sides of the blades have a 0.25-inch (6.35-mm) thick aluminum edge clad channel welded to the BORAL® plate so that the blade edge 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. The shim blade has a nominal inner radius of 6.430 inches (16.33 cm). Each shim blade occupies approximately 72° of a circular arc around the pressure vessel.

The radius of the shim control blade provides a certain amount of inherent stiffness that aids the hydraulic stability of the blade in its operating gap between the outer reactor pressure vessel and the beryllium reflector. During normal operation, only a portion of the shim blade operates in this gap, resulting in a minimized downward force due to flow. The flow of pool coolant past the portion of the blade that is in the gap is of a low velocity, resulting in very little downward force and a minimal likelihood of flow instabilities. Based on the current measured flow induced differential pressure across the reflector elements and the control blade gaps, the pool coolant flow velocity in the control blade gaps is approximately 3.5 feet-per-second (1.07 m/s).

The BORAL® and aluminum 1100 that comprise the shim blade have an excellent service history, both at MURR and other facilities, for the radiation conditions of the reactor. From 1966 to 1983, one (1) shim blade and its associated offset mechanism were removed and inspected each calendar quarter. Since 1983, one shim control blade and its associated offset mechanism is removed semi-annually and inspected with the control blade and offset mechanism submerged in the reactor pool to a depth that

provides the required radiation shielding. Inspection of the control blade is required by surveillance TS 4.2.c. A rebuilt offset mechanism (bearings replaced and offset mechanism aligned) is installed in its place using either a new or used control blade that has been inspected and approved. The inspection and approval of the control blade is performed visually and using geometric templates to ensure no swelling, distortion, blistering, rubbing or other abnormality is present prior to placing a control blade on service. This inspection is performed in accordance with preventative maintenance procedure RX-S-1, "Inspection of Control Blade and Offset Mechanism, and Preparing an Offset for Service." (Reference 4.50)

When a control blade and associated offset mechanism has been removed, a visual inspection is immediately performed with regard to the above criteria. The detailed results of these inspections as well as the signed approval by the Reactor Manager or his designee are captured as part of the Preventive Maintenance (PM) and Surveillance programs. The resulting experience has shown that shim control blades suffer no operational limitations or flow erosion, and minimal dimensional changes due to the hydraulic and radiation environment. (Reference 4.50)

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 (4) 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 (0.75 kw), 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.

The physical separation of the control blades and the drive mechanisms prevents mechanical failure of one blade from potentially affecting the insertion or withdrawal of any other control blade. Each blade is located in its own curved annular gap between the beryllium reflector and the outer reactor pressure vessel. The control blade gaps are separated by five (5) long aluminum spacer bars that run the full vertical height of the beryllium reflector. The horizontal cross-section of the spacer is similar to that of a thick "T" with the thickness of the top of the "T" establishing the proper blade gap spacing (approx. 0.55 inches) between the beryllium reflector and the outer pressure vessel. The vertical parts of the spacers are set into vertical slots in the inner radius of the beryllium reflector. These vertical slots effectively lock the spacer bars into their proper position and clearly establish properly sized and shaped independent

control blade gaps. The physical separation of the control blades and the drive mechanisms prevents mechanical failure of one blade from potentially affecting the insertion or withdrawal of any other control blade.

Control blade drive controls, including their independence with regard to operation, are discussed in detail in Section 7.5. These include descriptions of general control independence in the manual and automatic control modes, as well as circuit features such as Rod Withdrawal Prohibit, Rod Run-In, and Automatic Shim Control. These descriptions can be summarized by stating that there are no automatic circuits which could withdraw one or more control blade drives, and there are two automatic control circuits, Rod-Run-In and Automatic Shim Control, which could insert the control blade drives. All other blade drive operations are performed manually by the operator.

Independence with regard to malfunction is primarily protected against by the eight (8) independent drive relays that provide insertion or withdrawal for each of the four (4) drives. Each drive has its own insertion relay and its own withdrawal relay, requiring power to be supplied from the rod control circuits mentioned above to operate. The failure of any one of these relays can only impact operation of that one drive in one direction. A malfunction in one blade drive would require one or more additional malfunctions in one of the control circuits to affect the insertion or withdrawal of any other control blade. The special case where both insertion and withdrawal relays are energized simultaneously would still be limited to a single drive but would have additional protection via Rod Control Fuse 2F4. (Reference 4.50)

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 installed in the reactor control console. The RPI system is described in greater detail in Section 7.5.6.

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. The 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). The minimum width of the control blade gap is:

(13.680 - 12.566) / 2 = 0.557 inches (1.415 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 thermal distortion to cause control blade binding, 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) (Reference 4.19). 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:

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

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 (Ref 4.56). 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. Control blade binding due to thermal distortion of the control blades is a highly unlikely event, and has not occurred at MURR in the decades of operating at 10 MW. Further analysis of the possibility of control blade binding due to thermal distortion is provided in the following reference as the response to a Request for Additional Information. (Reference 4.56)

4.2.2.4 Technical Specifications

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. Control blade insertion times are determined by a Control Blade Drop Timer Circuit. When the reactor scram signal is initiated, the control current to the electromagnets, which engages the control rod drive mechanism to the anvil of the control blade-lift rod assembly, is removed by an electro-mechanical relay contact which allows the control blade to drop and start a blade drop timer/chronometer count. At the 20% withdrawn position (80% inserted), a digital fiber optic sensor, which provide a NPN (Not-Pointing-In) output to the control unit when triggered, causes the electro-mechanical relay to change state, stopping the blade drop timer/chronometer. The control blade drop time is then displayed on a meter on the reactor control room instrument panel. (This test provides a means for detecting degradation of the control blades which could affect their mechanical operability.

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, 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 aluminum used to fabricate these reflector elements has a long service history with regard to radiation fields and is well suited for a neutron rich environment. 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. Thermal expansion in the graphite reflector is not an issue due to the generous design tolerances within and between the reflector elements. The potential hazards associated with stored energy in graphite (Wigner energy) and the probability of a graphite fire is very minimal since the energy generation in MURR graphite elements is small; the MURR approach to canning graphite in a thermally insulated helium gap promotes thermal annealing to reduce the storage of Wigner energy. Additionally, the MURR graphite elements are enclosed in several non-contiguous wedges under several feet of water. Thus, there is not adequate energy available to ignite combustion, nor is there adequate air or other oxidizer available to sustain combustion. (Reference 4.50)

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.

The timing for replacement of the beryllium reflector is an operational consideration without a corresponding safety issue. This is performed approximately every 26,000 MWD, which is roughly eight years of operation at the current operating schedule of 150 hours per week (nominal). The replacement interval is based on past operational experience from six beryllium reflector replacements, after the

original beryllium reflector had cracked while 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. (Reference 4.50)

There are no potential experiment facility malfunctions in the reflector that could affect reactor core components. The mechanism used to verify that each experiment, or single experiment sample, complies with all of the Technical Specifications and other limitations based upon good operating, engineering, and health physics practices is called the Reactor Utilization Request. This formalized process specifically requires that a safety analysis be prepared, reviewed, and approved by both the Reactor and Reactor Health Physics Managers before an experiment can be conducted. Additionally, any proposed experiment significantly different from any previously reviewed or which involves a question pursuant to 10 CFR 50.59 shall be reviewed by the Reactor Advisory Committee.

Each safety analysis includes, but is not limited to, the following major considerations: criticality and/or reactivity impact, heat generation, shielding, and off gassing and/or chemical reactions. It also includes all credible accident and transient scenarios to ensure that the experiment does not jeopardize the safe operation of the reactor or constitute a hazard to the safety of the facility staff and general public.

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 and 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 5.5 inches (13.97 cm) long and 1.25 inches (3.18 cm) in diameter. The source was originally licensed for a maximum antimony-124 activity of two (2) curies, however, the neutron emission rate was found to be too low for the accurate counting statistics needed for subcritical measurements. During the upgrade to 10-MW operation the license was amended to increase the neutron startup source strength to a maximum of 100 curies.

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. The gamma ray responsible for inducing the (γ, n) reaction in beryllium originates from Sb-124 with a half-life of 60 days, so the source has to be irradiated periodically to reactivate the antimony target. Conservative calculations have determined that, to achieve a “steady state” activity of 100 Curies, the antimony target has to be irradiated for more than 5.0 hours every thirty days. In order to maintain the source strength less than the license limit, the Sb-

Be source is irradiated only for 2.0 hours every thirty days. The calculations indicate that the activity achieved with this monthly 2.0-hour irradiation at the peak reflector irradiation position is approximately 36 Curies. This periodic irradiation is tracked through the PM program. (Reference 4.50)

A handling line allows the neutron source to be lowered into an irradiation position located in the graphite reflector region. The normal storage location for the neutron source is in the deep section of the reactor pool. An identification tag is attached to the end of the handling line.

4.2.5 Reactor Core Assembly Support Structure

4.2.5.1 Description

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 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 (30.48 cm) 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 55 inches (1.4 m) in height and the lower tank cylinder, including the base flange, is approximately 39.25 inches (1 m) in height.

The vessel support ring which supports the outer reactor pressure vessel and the lower reflector cylinder is 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 (5) horizontal brackets are installed around the upper periphery of the upper tank cylinder outer wall. Four (4) 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.

4.2.5.2 Evaluation of Radiation Damage, Coolant Chemistry and Reactivity Effects

The greatest change due to radiation effects on core support structure materials occurs in the inner reactor pressure vessel just below core centerline where fast and thermal fluxes are at their peaks. The effect is a slight strengthening of the aluminum 6061-T6 inner pressure vessel in this area with a slight reduction in ductility. This is primarily due to the transmutation of some of the aluminum into silicon. This change is detailed in Section 16.1.2.

Conductivity and pH are monitored and maintained in an appropriate range to minimize corrosion and degradation of core support structure materials. Both the primary and pool coolant systems have a

clean-up system for demineralization and corrosion control. These ion-exchange systems maintain low conductivity and maintain the pH in a range between 5 to 6. The slightly acidic pH maintained in the primary and pool coolant systems maximizes aluminum aqueous corrosion resistance. In addition to routine monitoring, water samples are taken weekly from the primary and pool coolant systems and are analyzed for pH, conductivity, and contained radioisotopes.

The small negative reactivity of the aluminum 6061-T6 core support structure materials is slightly reduced with increased fluence. This is due to the transmutation of the aluminum and the partial burnout of the trace alloy elements and impurities in the aluminum. (Reference 4.50)

4.3 Reactor Pool

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 mrem/hr. at the maximum power level of 10 MW. The liner is constructed of 0.25-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.

Chemical damage to the pool liner is minimized by the pool coolant demineralizer loop, described in Section 5.5. Additionally, MURR re-examines the reactor pool liner during the periodic replacement of the beryllium reflector. During replacement, the reactor pool level is lowered to a depth of 10 feet (9.1 m), which allows for a more effective evaluation. (Reference 4.50)

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 were 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 (2) 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 (2) 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.

The design of MURR is such that the entire reactor pool and pool liner are above the elevation of the beamport floor and not in contact with any soil. With the exception of the base, all areas of the reactor pool are visible and accessible. The base of the pool is situated on a 16-foot-square (4.87-m-square) monolith of reinforced concrete, which is then supported by a 3.5-foot-thick (1.07-m-thick) concrete pad poured directly onto a 12-foot (3.66-m) caisson. The caisson extends horizontally one (1) foot (0.30 m) out beyond the reactor pool in all directions and extends downward to a minimum depth of 6 inches (0.15 m) below “sound” bedrock at the lowest point around the edge of the caisson. The entire biological shield structure that surrounds the pool is effectively separated from its surroundings and surrounded by two drain trenches in the containment basement. Either of these trenches accommodate approximately 30 gallons (113.6 l) prior to initiating an audible and visual level alarm in the Control Room. In the event of a pool leak, whether the leak is in the pool proper or the pool coolant system (see Section 13.2.9.1), all probable leakage paths provide both retention and detection. Therefore, the possibility for pool water to enter the environment is essentially eliminated and the potential for radiological impact on members of the public is minimal. (Reference 4.50)

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:

- At one (1) 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 mrem/hr;
- At one (1) 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 mrem/hr;

- At three (3) 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 mrem/hr; and
- At any location at the top surface of the reactor pool, the radiation level shall not exceed 20 mrem/hr.

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

The biological shield is a massive bulk shield structure created by the placement of three (3) 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 0.25-inch (6.35-mm) thick steel plating fastened to the structure by tie-rods. The biological shield is shown in Figure 4.11.

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.

The potential for degradation of concrete shielding from thermal stress due to exposure to long term heating for MURR is minimal. The reactor preliminary design study, final design, and design specifications were provided by the Internuclear Company of St. Louis, Missouri. The design was based on assuming the biological shield would be constructed of barytes (also known as barites) concrete. The minimum pool radius was calculated to achieve the following two design criteria:

- The temperature differential between the concrete surface and the maximum temperature in the barytes concrete shall not exceed 50 °F.
- The maximum temperature in the barytes concrete shall not exceed 150 °F.

With the design based on a pool coolant temperature of 90 °F (32.2 °C), the limiting criterion in sizing the minimum pool radius was a temperature differential of less than 50 °F (27.7 °C). Assuming an adiabatic outside surface of the biological shield, it was calculated that a minimum thickness of 27 inches (0.69 m) of water was required between the outer surface of the reactor reflector and the concrete inner surface.

However, since the calculations did not factor the actual reactor final design, it was necessary to validate this earlier result with more detailed modeling. Therefore, the three-dimensional transport code MCNP5 was utilized to calculate the heating rate of the biological shield. The MCNP5 model (Ref 4.53)

includes final design details that were decided on after the 1962 calculation was made which includes such factors as a pool radius of 5 feet (1.52 m), the biological shield made of magnetite concrete and not barytes, and the core centerline offset 1.5 feet (0.46 m) from the pool centerline towards a lead shielded thermal column filled with graphite (Ref 4.51).

The heat generated in the magnetite concrete of the biological shield is predominantly due to the gamma-rays emitted from the core as fission (prompt and delayed) and activation gammas of reactor materials including the reactor core, pressure vessels, reflector components and water. This gamma heating was calculated using detailed MCNP5 models of MURR, with the gamma and neutron fluxes tallied in regions of the biological shield where the peak fluxes would occur (i.e., regions whose radial distances are closest to the core). Three feet is the closest the core is to the magnetite concrete portion of the biological shield, with water in this space between the reactor reflector and the biological shield. This occurs on each side of the thermal column, on the south side between Beamport C and the thermal column and on the north side between Beamport D and the thermal column.

For these locations, the MCNP5 F6 photon energy deposition tally was used to calculate the volumetric heating as a function of distance into the biological shield based on the code determined gamma interactions within the concrete. Figure 4.7 provides a plot of the MCNP5 volumetric gamma heating function within the first six inches of the concrete shield, which drops off exponentially with distance. The peak gamma flux in the shield was calculated to be 1.0×10^{11} photons/cm^2-s by MCNP5 while the peak total neutron flux (at the same position) is 8.5×10^6 neutron/cm^2-s. The gamma energy deposition rate in the concrete at the leading surface corresponds to 2.48 mW/cm^3. The contributions from neutrons to the heat generations were insignificant compared to the gammas (less than 1.1×10^{-9} W/cm^3)

To verify that the neutron flux does not significantly contribute to the heating due to the production of capture gammas within the concrete, ORIGEN 2.2 was used to calculate activation gamma thermal power produced by it in the magnetite concrete at the inner surface. ORIGEN 2.2 calculated radiation heating in the inner surface due to activation from the neutrons was less than 1.1×10^{-9} W/cm^3, which is insignificant compared to the direct reactor produced gamma heating of 2.48 mW/cm^3 at the surface.

Therefore, most of the gamma energy is deposited within the first few inches of the shield, with insignificant heating occurring in the rest of the approximately 6.7-foot-thick (2.04-m-thick) biological shield. Since the pool water is essentially in contact with the inner surface of the concrete (from a heating perspective), it is the major heat-sink for the excess heating within the biological shield. A heat-transfer model was developed with the pool water/concrete surface having one boundary condition and the concrete surface/air on the other side having another. Figure 4.8 is a depiction of the model, where it shows a bi-directional heat flow across the concrete in the thickness dimension only. This is conservative since it does not factor in any heat transfer in the other two directions. In the actual hot spot location, there is more than just radial heat conduction.

The differential equation for this one-dimensional model allows heat transfer to both surfaces of the biological shield. It also assumes a slab shield instead of a cylindrical sleeve shield. However, its

simplicity allows for an exact analytic solution to be obtained using the appropriate boundary conditions.

Second-order differential equation for the model was solved to obtain the results shown in Figure 4.9 that provides the temperature profile through the biological shield with the pool water temperature at 100 °F (37.78 °C) and the air temperature at 77 °F (25 °C). The peak temperature in the biological shield is 115.5 °F (46.4 °C). This results in a peak ΔT = 12.4 °F (6.9 °C) across 7.87 inches of the inner surface. The ΔT = 12.4 °F (6.9 °C) across 7.87 inches from the pool surface, corresponding to a 1.85 °F ΔT per inch (2.6 °C/cm) of concrete. Across the rest of the 6.5 feet (1.98 m) of biological shield, the ΔT = 35.6 °F (19.8 °C), which corresponds to a 0.51 °F ΔT per inch (0.72 °C/cm) of concrete. This means the thermal stress in the inner face would be more than three times the stress outside the peak temperature in the concrete.

To better compare the results to the 1962 study, the air temperature was adjusted to result in no heat transfer in or out of the outer surface of the biological shield. This matched their assumed adiabatic outer surface. Figure 4 provides the temperature profile through the biological shield with the pool water at 100 °F (37.78 °C) and the air at 121.15 °F (49.53 °C). The outside wall temperature is also the peak temperature in the biological shield. This results in a peak ΔT = 17.7 °F (9.84 °C) across the biological shield. However, most of the peak thermal stress is across the inner 9.84 inches (25 cm) of the biological shield, which has a ΔT = 17.01 °F (9.45 °C) or 1.73° F ΔT per inch (2.44 °C/cm) of concrete. Across the rest of the 6.5 feet (1.98 m) of biological shield, there is a 0.01 °F ΔT per inch (0.014 °C/cm) of concrete. As calculated, the thermal stress would be almost totally in the inner few inches of the biological shield. In both figures 4.9 and 4.10 the values are based on barytes concrete heat conductance. If the magnetite heat conductance was used the temperature differentials and corresponding thermal stress would be reduced by half.

The potential for degradation of concrete shielding from thermal stress due to exposure to long term heating has been addressed in more recent studies. In an extensive study done by Kassir and Bandyopadhyay (Ref 4.59) on the thermal degradation of concrete, it shows that the long-term heating effect on concrete at ambient temperatures and up to 250 °F (121 °C) serves to increase its mechanical strength. However, for long term exposure to temperatures at or greater than 300 °F (149 °C), the mechanical properties begin to fail. In the 1962 study for MURR, the limiting temperature is conservatively set at 150 °F (65.5 °C) or a ΔT = 50 °F (27 °C), half of the limiting value reported by Kassir and Bandyopadhyay. Since the worst-case analysis using the baryte heat conductance coefficient, which is half of the magnetite coefficient, shows a maximum ΔT over the 6.5 feet (1.98 m) of biological shield, when assuming no heat transfer to the air, is only 17.7 °F (9.8 °C) for the MURR biological shield, the potential for degradation from long term gamma heating is minimal.

4.5 Nuclear Design

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 MURR and elsewhere, the core life limit specified in condition (1) above was subsequently extended to 1,200 MWD.

4.5.1 Early Nuclear Design Methods (Historical Methods)

Early 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, and using the two-dimensional diffusion theory program PDQ. These historical calculational methodologies are described in detail in MURR Design Data, Volume II (Reference 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. (Reference 4.16) Addendum 4 also provides the changes and bases for the Technical Specifications associated with the upgrades. The safety limits and limiting safety system settings (prior to Amendment # 36) for 6.2-Kg cores (775-gram 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 fuel elements. The peaking factor and other

nuclear parameters for the cores were determined by employing two-dimensional neutron diffusion code models using the EXTERMINATOR-II code (Reference 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 (Reference 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 U-235 per element (Extended Life Aluminide Fuel or ELAF). (Refs. 4.13 and 4.14) However, the ELAF elements were never fabricated on a production scale or used at MURR. During the ELAF analyses, newer computer codes were used to verify and confirm many of the nucleonics analyses done earlier for the in-use MURR core and fuel 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.

To analyze the MURR core with various fuel loadings, the BOLD VENTURE-IV (Refs. 4.22, 4.23) and AMPX-II (Reference 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 MURR fuel element with an 82.5-MWD power history. The results of the benchmark indicated 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" (Reference 4.21). The report describes the use of AMPX-II to generate the cross-section sets for the various nuclides in 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 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 major components of the computational system are shown in Figure 4.12. 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 shown in Figure 4.13. 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 4.14 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 reactor. 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 4.15 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 Θ-R 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 Figure 4.16. The "Δ" 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.

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

The benchmark study used the only experimental data available, the results from the destructive analysis of a single MURR fuel element, 775-F3. The fuel element had reached a burn-up of 82.5 MWDs during normal operations with a loaded center test hole canister inserted. In order to compare the fuel

plate burn-up values from the BOLD VENTURE neutronics model against those from the destructive analysis of the 775-F3 element, the BOLD VENTURE neutronics model had as much detail as possible of the core and the surrounding regions. Since the experimental samples and the center test hole canister in the center test hole (flux trap region) can impact the flux (and power) as seen by the inner fuel plates, the three-tube test hole canister and the samples generally present in the central flux trap region were also included in the BOLD VENTURE neutronics model. The three-tube test hole canister shown in Figure 10.2 was modeled using cylindrical rings to closely approximate the material densities and locations within the center test hole. This detailed model is referred to as the "Flux Trap" model. 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 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. MURR provided detailed analyses to justify the conclusion that the calculated safety limit for the reactor based on an all-water flux trap region of the core is limiting in a Request for Additional Information (RAI) response dated July 30, 2012. (Reference 4.61) The NRC agreed to this conclusion in the SER to Amendment No. 36. (Reference 4.60)

Table 4-8 summarizes the BOLD VENTURE R-Z results for the cold clean 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

	Control Blades Full-Out	**Control Blades Full-In**
K-eff.	1.1124	0.9315

The total worth of the four control blades for the 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 three columns of values listed in Table 4-9 were taken from Table 7 of a MURR internal report (Reference 4.21) generated for the Extended Life Aluminide Fuel (ELAF) amendment submittal. This table compared the reactivity with and without boron included in the

proposed ELAF core design to the in-use MURR core. The BOLD VENTURE calculations were performed with the control blades at various heights and fuel compositions. The calculations were readjusted at discrete burn-up steps based on the isotope burner module included in the BOLD VENTURE suite of codes for this comparison. For the last two burnup steps, the control blades are kept at a constant height for the 330 MWD burnup. Assuming the reactivity change is linear with fuel burnup, the change in reactivity can be calculated. If that is done, the decrease in reactivity due to the burnup of fuel is approximately 3.3%.

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 in-use MURR 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%.

When comparing the K-eff. values for the "Control Blades Full-In" case for both the R-Z and the Θ-R-Z models, the error in the 2D result is approximately 2.3%. This is higher than the "Control Blades Full-Out" case but still relatively small considering the inherent approximations that are present in the 2D model. For example, the control blade modeling in the R-Z model assumes full 360-degree coverage around the core as opposed to the actual 288-degree coverage offered by the four shim control blades. As expected, the difference in the K-eff. values between the 2D and 3D cases will be lower in the "Control Blades Full-Out" case since the effect of the control blades on the core is virtually eliminated.

TABLE 4-9
BOLD VENTURE R-Z DEPLETION RESULTS

Cumulative Burnup (MWD)	**Control Blade Position (inches withdrawn from full-in) (cm)**	**MURR Core K-eff.**
0	16.38 (41.61)	1.06697
325	17.61 (44.73)	1.02537
650	19.00 (48.26)	1.02588
970	20.50 (52.07)	1.02605
1,300	20.50 (52.07)	1.0173

In the 3D model of the core, two beamport holes and the impurities in the beryllium reflector were not considered. According to a previous study (Reference 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 1.1139, which is higher than the measured value of 1.0795 by approximately 3.2%. 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.

The total control blade worths calculated from the R-Z and the Θ-R-Z runs are summarized in Table 4-11.

TABLE 4-10
MURR 3D MODEL RESULTS

	Control Blades Full-Out	Control Blades Full In
K-eff.	1.11725	0.95340

TABLE 4-11
TOTAL CONTROL BLADE WORTH

	R-Z Model	Θ-R-Z Model	Experimentally Measured
Reactivity (Δk/k)	0.1746	0.1538	0.1655

Core loadings of fresh and depleted 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 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 fuel elements can be used together at burnups up to 2.3 x 10^{21} fissions/cm^3 and operated with greater margins from the Departure from Nucleate Boiling (DNB) and flow instability than those on which the safety limits are based.

4.5.2 Recent Nuclear Design Methods (Since August 2006)

Historically, neutron physics modeling and analyses at MURR have been performed using several multi-group and multi-dimensional neutron diffusion theory codes such as PDQ, EXTERMINATOR-II and BOLD VENTURE. Since BOLD VENTURE core model was benchmarked against the destructive analysis of a high-enriched uranium (HEU) MURR fuel element for the license renewal application submitted to the NRC in August 2006, MURR used results provided by the neutronics codes listed above.

MURR has since switched to using newer, state-of-the-art programs, such as MCNP for neutronic analysis. For a compact core such as MURR, it is preferable to use a transport theory code to capture the rapidly changing spectra across the various regions. Therefore, MCNP (in combination with other activation and depletion programs such as ORIGEN) is now routinely used for all calculations of core K-eff., critical control blade height, detailed power distribution, and experimental fluxes/reaction rates.

As part of the on-going collaboration between MURR staff and Argonne National Laboratory (ANL) analysts for the purpose of determining the feasibility of converting MURR from HEU to low-enriched uranium (LEU) fuel, ANL has assembled a neutronics analysis code suite utilizing WIMS-ALNL, REBUS-DIF3D and REBUS-MCNP. Figure 4.17 illustrates the linkage of the codes in the analysis suite.

The suite of programs, or codes, were used to provide detailed (radial, axial and azimuthal) fuel composition for partially burned fuel elements. Since MURR routinely operates with a fuel cycle utilizing a mixed burnup core, realistic experimental flux, reaction rates and power peaking values have to be evaluated for the typical core weekly cycles rather than for an all-fresh core. The detailed fuel composition data obtained is then subsequently used in a MCNP calculation to obtain the worst-case power peaking factors and heat flux values used in the thermal-hydraulic analysis. These newer analysis methodologies have mainly been used to confirm that the results obtained by earlier computer programs are still valid and conservative.

WIMS-ANL

WIMS-ANL is a one-dimensional lattice physics code used to generate burnup dependent, multi-group cross sections. The code utilizes either 69- or 172-group libraries of cross-section data for 123 isotopes generated from ENDF-6. A customized 10-group structure was developed by ANE based on the neutron spectrum that exists in the MURR core. This multi-group data can be used in MCNP and REBUS-MCNP analyses of depleted cores.

REBUS-DIF3D

DIF3D is a multi-dimensional, multi-group neutron diffusion code that can model systems in a number of geometries. REBUS is a depletion code that utilizes neutron fluxes from a neutronics solver and cross-section data to solve isotopic transmutation calculations. A 16 of 86 detailed O-R-Z diffusion MURR model was developed for DIF3D. The depleted core characteristics (plate-by-plate and axially segmented atom densities) can be saved and passed on to MCNP for more detailed neutronics analyses.

MCNP

MCNP is a continuous energy Monte Carlo neutron transport code. MCNP is capable of modeling the heterogeneous details of the MURR fuel elements, core structures, and experimental facilities while capturing the rapidly changing spectra across these various regions. Using the 69- group lumped fission product library generated by WIMS-ANL, the code can be used to model cores of depleted and fresh elements.

ANL had performed extensive work to validate the above set of neutron physics codes and models for application to MURR. The MCNP and DIF3D models were benchmarked against available experimental data for the 1971 and 2008 core configurations, as described in the Argonne Technical Memorandum titled “Technical Basis in Support of the Conversion of the University of Missouri Research Reactor (MURR) Core from Highly-Enriched to Low-Enriched Uranium – Core Neutron Physics” [ANL/RERTR/TM-12-30].

To speed up routine neutronics calculations, where such detailed axial, radial and azimuthal fuel composition is not necessary, MURR utilizes the MONTEBURNS program to determine a uniform fuel composition for different levels of burnup. MONTEBURNS is a coupled MCNP-ORIGEN code system developed by Los Alamos National Laboratory (LANL). It utilizes the capabilities of ORIGEN 2.2 for isotope generation and depletion calculations, and MCNP5 for continuous energy neutron flux, reaction rates, and criticality calculations. Neither MCNP nor MONTEBURNS are designed to handle transient

calculations such as during the period from reactor startup through criticality and then on to steady-state reactor operation. This process involves control blade motion due to poison buildup and fuel depletion. However, with the help of in-house developed routines, a code system including MCNP and MONTEBURNS was developed to perform routine reactor physics calculations during a typical mid-week cycle at equilibrium Xenon.

These "routine" calculations utilize a very detailed MCNP MURR core model that has the following key capabilities:

- It can model MURR's "mixed-core" weekly fuel configuration, i.e., atom densities of various isotopes in the fuel matrix, for a range of fuel element burnups from fresh (0 MWd or no fuel depletion) to spent status. For the Estimated Critical Position (ECP) calculations, fuel element definitions can be individually selected from a fuel burnup database to simulate any combination of eight (8), xenon-free fuel elements.
- Similarly, it can include a mixture of four (4) independently depleted BORAL® shim control blades - each with a different axial and radial boron depletion profile based on its operational history (or core residence time).
- It has the ability to account for poison and gas buildup, and the reduction of beryllium atom density within the beryllium reflector based on its run time (from 0 to 8 years).
- The multiple samples that are irradiated in the high worth central flux trap region of MURR, as well as in the various positions within the graphite reflector region, are modeled accurately in order to reduce the error in the ECP calculation.
- With the help of a critical control blade height search routine, starting from an initial estimate of the critical control blade height, a series of MCNP5 criticality (KCODE) calculations can be performed in order to calculate the critical control blade height.
- In order to predict control blade travel during startup and subsequent steady-state operation, as well as recovery following an unplanned reactor shutdown, it can track the buildup of xenon-i135 and other poisons in the core during reactor operation as well as the buildup and decay of the poisons during shutdown and restart using the isotope buildup and decay/loss capability of MONTEBURNS.

The system of codes and calculation methodology described previously has been benchmarked extensively using actual weekly core refueling and reactor startup data.

The code system that is currently used for reactor physics analysis at MURR has been benchmarked extensively. One method used for benchmarking was comparing the Estimated Critical Position (ECP) calculations from the detailed MCNP MURR model against the actual startup critical control blade height data from several weekly reactor startups. The detailed MCNP MURR core model includes depleted control blade data, beryllium aging effect (i.e., more and more gas molecules taking up the place of beryllium atoms with increasing run time), as well as detailed sample information present in the central flux trap region of the reactor core.

Detailed description of eight separate cores selected for comparison to verify the consistency in the model's ability to predict the ECP accurately under various core stages (mixed burnup) and flux trap sample conditions are provided in pages 19 through 25 (of 86) in reference 4.58, Responses to RAIs dated October 1, 2015.

4.5.3 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 MURR is introduced in Section 4.6.3, which refers to Attachment A, the current MURR Safety Limit analyses. 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 Departure from Nucleate Boiling Ratio (DNBR) 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 that a fuel plate reaches during the transient is 280.3 °F (137.9 °C), that is still more than 500 °F (277.8 °C) below the 900 °F (482 °C) temperature at which MURR fuel plates are tested to during fabrication to verify that no blistering will occur.

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 introduces analyses to show that the LSSSs for operating in Modes I, II and III meet this criterion. The bases for the LSSSs are provided in Attachment B.

4.5.4 Reactor Core Physics Parameters

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

4.5.5 Operating Limits

Reactivity limitations ensure 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 maximum rate of reactivity addition shall not exceed 1.5×10^{-4} Δk/k/sec which 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 assurance that the required shutdown margin can always be satisfied as shown below:

   Technical Specification (TS) 3.1 a. limits the excess reactivity above cold, clean, critical not to exceed 0.098 Δk/k. The excess reactivity is verified after any changes to the core are made - which is currently during every startup following a weekly shutdown for a core refuel. The verification is done during the reactor startup, when the cold, clean critical rod height is measured.

After this verification, the only change to this overall excess reactivity would be due to the addition or removal of movable experiments, which are defined in TS. Unsecured experiments are defined in TS 3.8 and the maximum worth of each one is limited to 0.0025 Δk/k. The maximum worth of all unsecured experiments is limited by TS to 0.0060 Δk/k. All movable experiments are unsecured experiments, but not all unsecured experiments are movable experiments. All unsecured experiments that are not movable experiments must be inserted before the reactor is started up and not removed until it is shutdown. Therefore, following a reactor startup, the maximum possible change in excess reactivity due to experiments is 0.0060 Δk/k.

MURR has only approached the excess reactivity limit when loading eight fresh fuel elements; this was done in 1971 as physics testing for the current uranium-aluminide fuel element design and subsequently for criticality checks on eight new fuel elements before their use in the fuel cycle. Since the restriction prohibiting possession of more than 5.0 kg of unirradiated uranium-235 was enacted around 1980, this type of testing has not been and cannot be performed.

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 blades shall not exceed one inch (2.54 cm) when the reactor power is above 100

kilowatts. A detailed analysis is provided in the MURR response to RAIs dated September 8, 2011 (Reference4.54) and shows that a four-inch (10.16-cm) difference in control rod heights does not exceed the SAR peaking factors. Thus, the one-inch (2.54-cm) limit between the highest and lowest control blades when the reactor is at a power level greater than 100 kW ensures that an excess in flux tilting is not created.

TABLE 4-12
SUMMARY OF TYPICAL REACTOR AND MAJOR NEUTRON PHYSICS DATA

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.	1.0795
Power Density; Max/Avg:	
a.Uniform Loading	3.306
b. Non-uniform Loading	3.676
Control Blade Characteristics: Total Worth of Four Shim Blades Total Worth of the Regulating Blade Total Worth of One Stuck Blade Max Worth Per Blade (One Blade In) Max K-eff. With One Stuck Blade	 0.1655 Δk/k 0.0023 Δk/k 0.10923 Δk/k 0.0425 Δk/k 0.938 Δk/k
Xenon Worth:	
a. Equilibrium	- 0.0273 Δk/k
b. After 16-Hour Shutdown	- 0.0607 Δk/k
c. Peak to Equilibrium Xenon Ratio	2.22

TABLE 4-12
SUMMARY OF TYPICAL REACTOR AND MAJOR NEUTRON PHYSICS DATA (CONT.)

Parameter	Description
Samarium Worth: Equilibrium Weekday Maximum 5-MW Operation 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 U-235
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	0.0736 (Δk/k)/(Δm/m)
Temperature Coefficients: a. Primary: (*) b. Pool: 85 to 115 °F (29.4 to 46.1 °C)	 (*) $+ 1.34 \times 10^{-4}$ Δk/k/°F
Void Coefficients: Primary Flux Trap	 $- 2.51 \times 10^{-3}$ Δk/k/% void $+ 0.865 \times 10^{-5}$ Δk/k/cc void
Reactivity Increase - Flooding One Beamtube	~ 0.001 (Δk/k)/beamtube

* MURR recalculated the primary temperature coefficient using the newer sets of computer programs that are described below. (Reference 4.58)

Coefficient	MURR Technical Specification Limit	ANL-MCNP Calculation (294 to 400 K)
Primary Coolant Temperature (Isothermal)	Average Core Temperature Coefficient Shall be More Negative Than: -6.0×10^{-5} Δk/k/°F	All Fresh Core, BOC: $-13.2 \times 10^{-5} \pm 2.3 \times 10^{-7}$ Δk/k/°F Mixed Core, BOC: $-12.8 \times 10^{-5} \pm 2.2 \times 10^{-7}$ Δk/k/°F Mixed Core, Eq. Xe: $-11.8 \times 10^{-5} \pm 2.2 \times 10^{-7}$ Δk/k/°F Mixed Core, BOC (ENDF7): $-12.5 \times 10^{-5} \pm 2.2 \times 10^{-7}$ Δk/k/°F

Note: BOC = Beginning of Cycle.

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 in-use MURR cores are presented in Table 4-12.

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 x 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 fewer 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 reactor vessel 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 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:

- 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;
- Simultaneous occurrence of hot channel and hot spot conditions; and
- 10% overload relative to nominal power level.

As required by the fuel fabrication specifications, every fuel plate that is assembled into a MURR fuel element is heated to 900 °F (482.2 °C) to verify that no blisters form. This temperature is then used as the basis for the thermal design and operating limits. Steady-state thermal analysis predicts the actual fuel temperature is 311.3 °F (155.2 °C) during operation (Reference 4.55). This temperature occurs in fresh fuel rather than in high-burnup fuel and is far from the cladding blister temperature.

MURR provided an analysis of the effect on the thermal performance of MURR fuel elements due to the buildup of an oxide layer on the cladding surface of fuel plates. This analysis was provided in response to an NRC RAI dated January 6, 2012. (Reference 4.55)

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 (Reference 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 provided from the bulk pool at a temperature of 100 °F (37.8 °C). This leads to the basis for Technical Specification 2.2.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.

The revised Safety Limit Analysis submitted to the NRC as Attachment 11 to an application for license amendment dated August 24, 2011, stated the analysis in Section 5.5.3, "Analysis of Natural Convective Cooling of the Core," still conservatively envelops Mode III operation of MURR. (Reference 4.63)

4.6.2 Steady-State Forced Cooling Analysis

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.

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 (Reference 4.32). Tests were performed at Argonne in 1963 on three channel thicknesses (0.054, 0.072, and 0.094 inches (0.137, 0.183, and 0.239 cm)), and it was found that, for the two thinnest channels (0.054 and 0.072 inches (0.137 and 0.183 cm)), 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 (0.239-cm) channel before the coolant reached saturation conditions at the channel exit. (References 4.44 and 4.45)

Subsequent full-scale ATR testing (Reference 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 the 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. A summary of heat transfer data is provided in Table 4-13.

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 U-235
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	0.86×10^5	1.72×10^5	Btu/ft^2-h
Fuel Assembly Average Outlet Temp.	159.3 (70.7)	159.3 (70.7)	°F (°C)
$\frac{\text{Ave. Power Density in Hot Channel}}{\text{Ave. Core Power Density}}$	2.469[a]	2.469[a]	
$\frac{\text{Max. Power Density in Hot Channel}}{\text{Ave. Power Density in Hot Channel}}$	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	
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	2.0 (0.6)	4.0 (1.2)	ft/sec (m/sec)
b. Unobstructed Test Hole	4.7 (1.4)	9.4 (2.9)	ft/sec (m/sec)
c. Control Blade Gaps	2.2 to 2.9 (0.7 to 0.9)	4.5 to 5.9 (1.4 to 1.8)	ft/sec (m/sec)
d. Reflectors	variable	variable	
Design Temperature Rise in Island	5.8 (3.2)	5 (2.8)	°F (°C)

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

TABLE 4-14
SUMMARY OF MURR HOT CHANNEL FACTORS (Refs. 4.60 and 4.62)

	Safety Limits Basis
On Enthalpy Rise	
Power-related Factors	
Nuclear Peaking Factors:	
Radial	2.215
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	1.150
Overall Product	3.03
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	2.215
Azimuthal Within Plate	1.070
Axial	1.3805
Add. Allowable Factor	1.062
Engineering Hot Channel Factors on Flux:	
Fuel Content Variation	1.030
Fuel Thickness/Width Variation	1.150
Overall Product	4.116
Energy Fraction Generated in Fuel Plate	0.930

4.6.3 Safety Limit Analyses

Early calculations and analyses of the MURR Safety Limits (SLs) are retained in Chapter 4, Attachment C. (Refs. 4.26, 4.27, 4.28, 4.29, 4.30, 4.31, 4.32, 4.33, 4.34, 4.35. 4.36, 4.37, 4.38, 4.39, 4.40, 4.41, 4.42, and 4.43)

Attachment A provides the most current SL analyses which were approved as part of Amendment #36 by the NRC on July 8, 2013. These SL analyses were part of a MURR license amendment request dated August 24, 2011, as a revision to Original Addendum 4, Section F, to the MURR Hazards Summary Report (HSR). It replaced the information in Section 3.3 of Addendum 3; Appendix F of Addendum 4; and Section 6.0 of Addendum 5 to the MURR HSR.

4.6.4 Bases for Limiting Safety System Settings

Early calculations and analyses of the MURR LSSSs are retained in Attachment C.

Attachment B provides the most current LSSS analyses which were approved as part of Amendment #36 by the NRC on July 8, 2013. These LSSS analyses were part of a MURR license amendment request dated August 24, 2011, as a revision to Original Addendum 4, Section H. to the MURR Hazards Summary Report (HSR).

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, obviating the need for an additional heat removal system to protect the integrity of the fuel. 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-fined tubes approximately 5 feet (1.5m) long with an overall finned length of about 4.5 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.

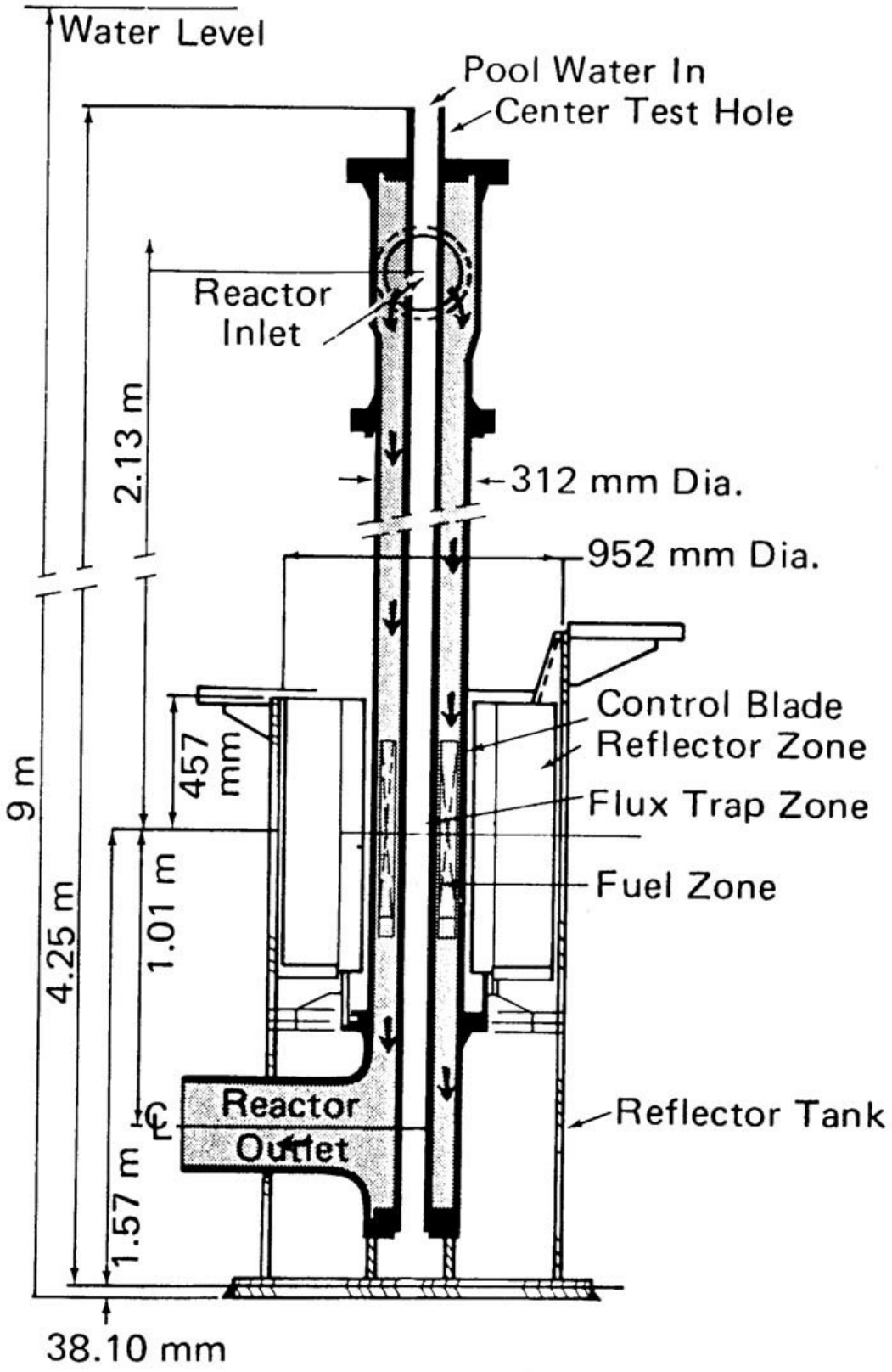


FIGURE 4.1 – REACTOR ASSEMBLY SECTION IN ELEVATION VIEW

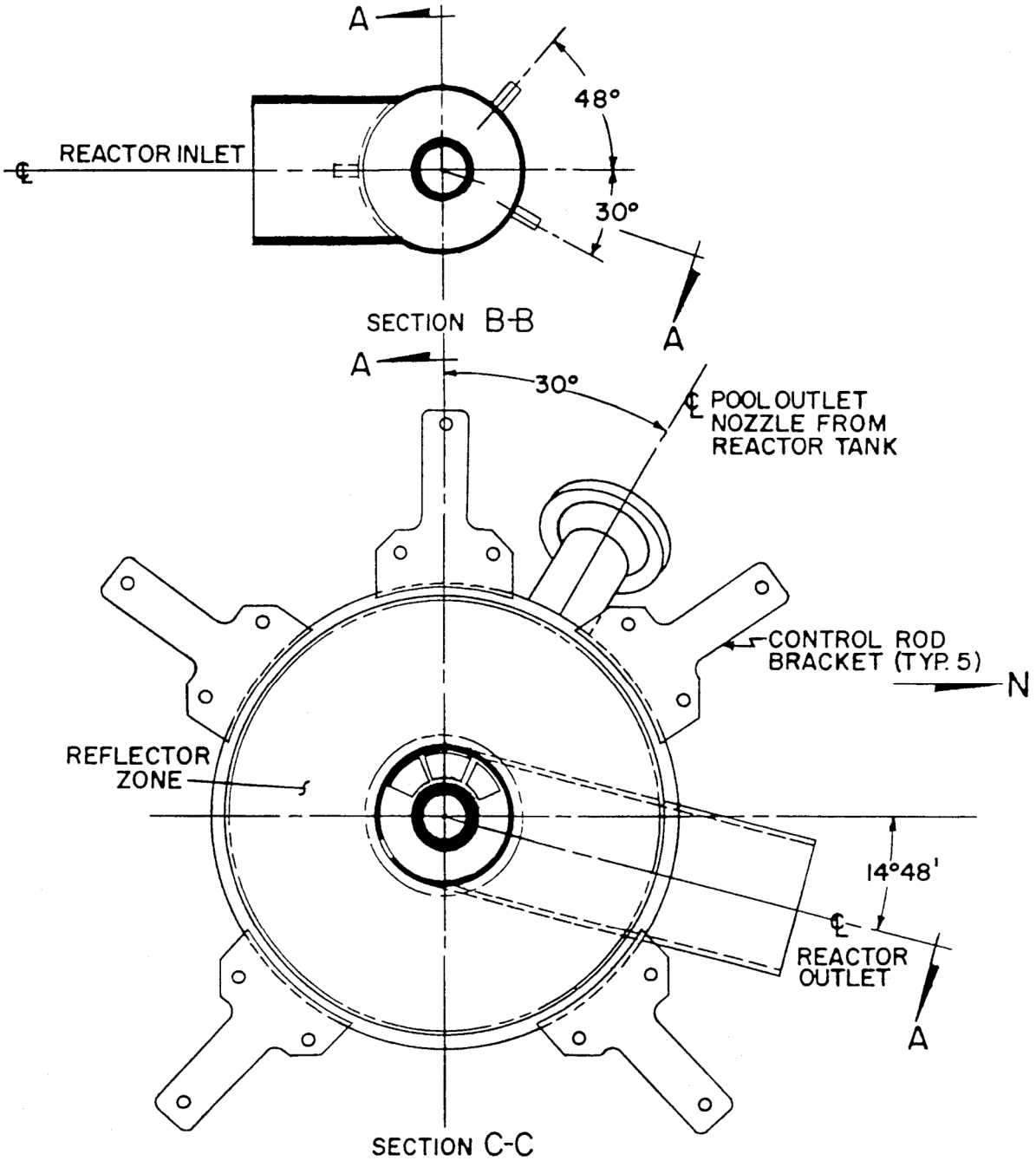


FIGURE 4.2 – REACTOR ASSEMBLY SECTIONS IN PLAN VIEW

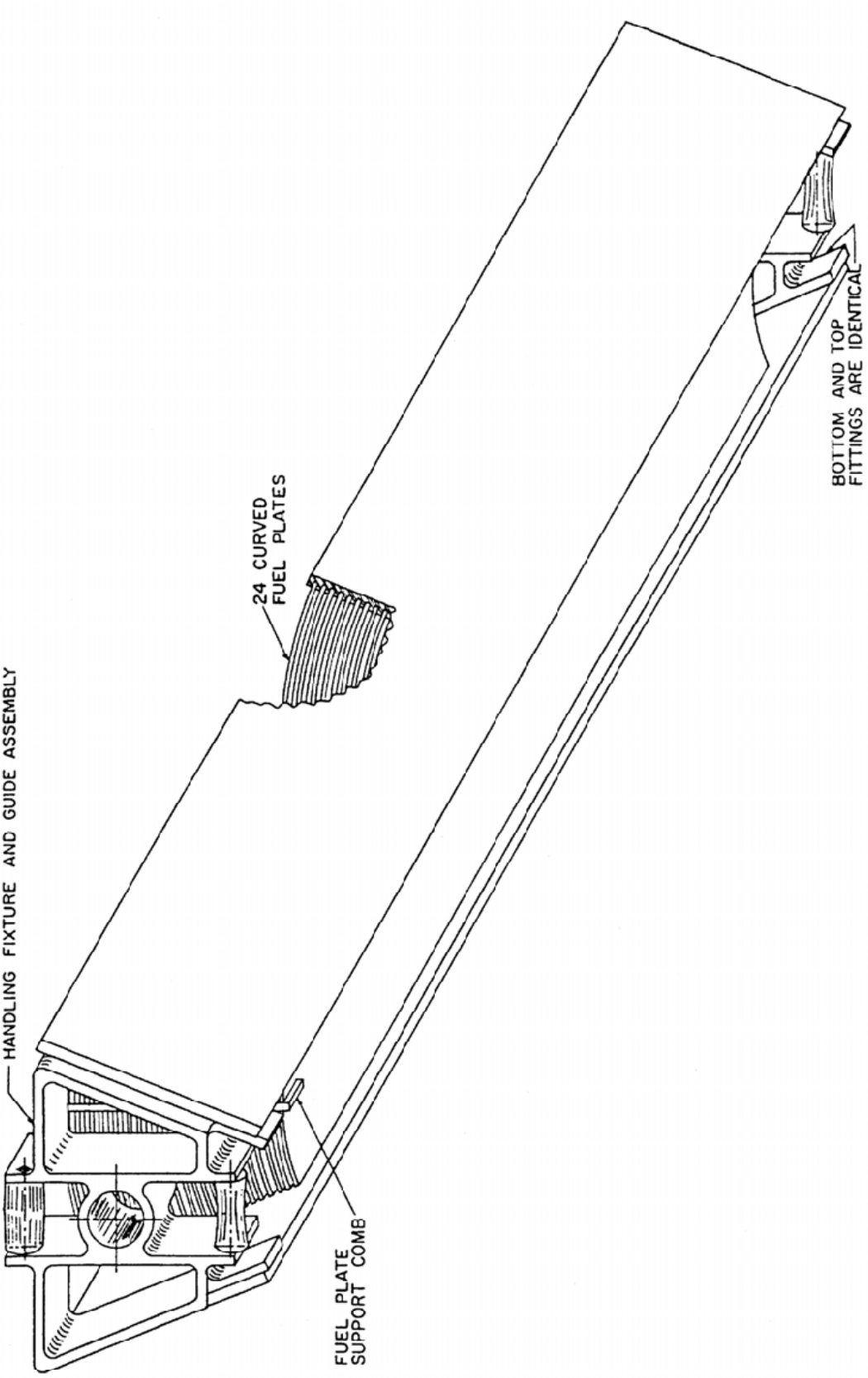


FIGURE 4.3 – FUEL ELEMENT - PICTORIAL VIEW

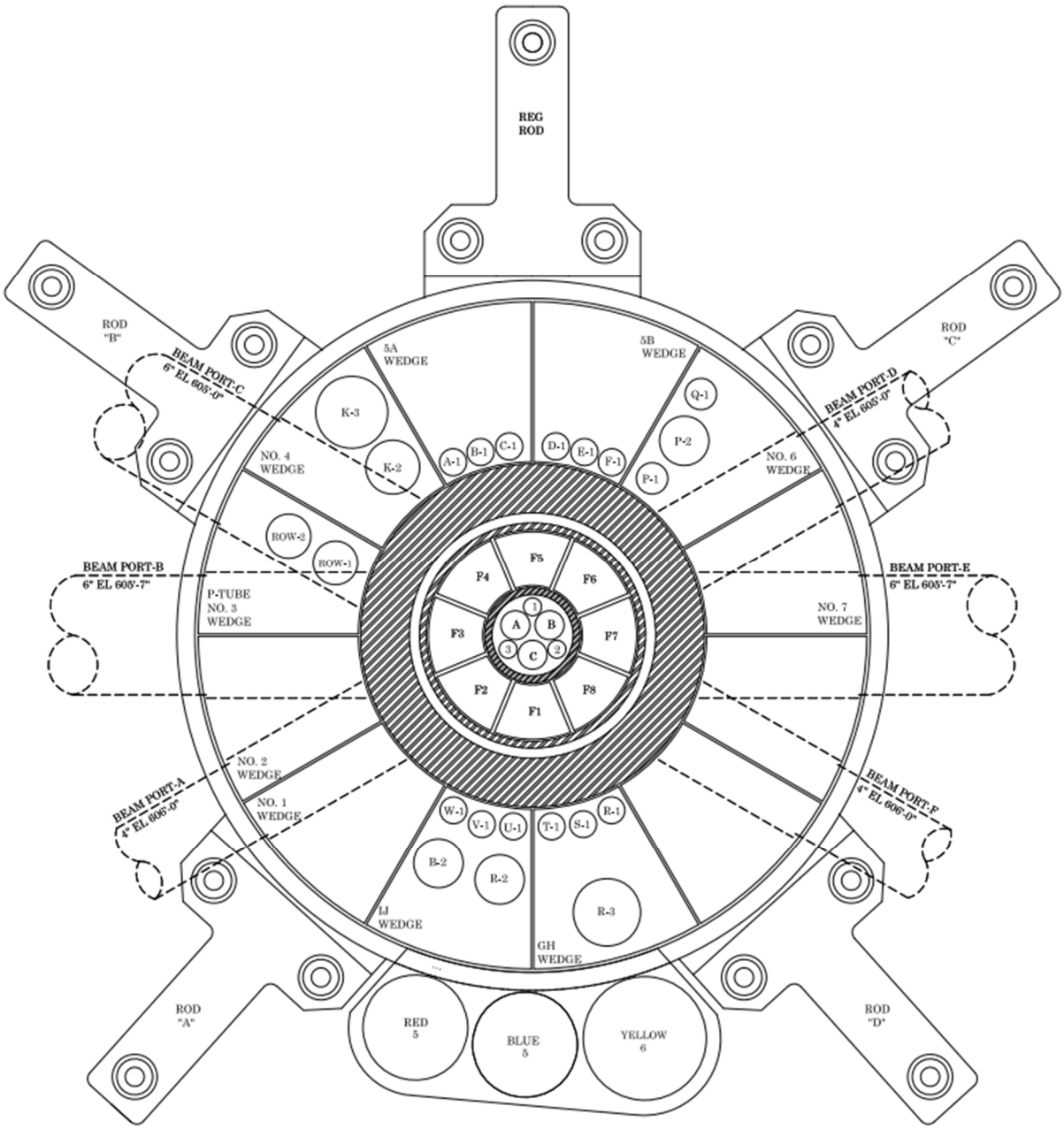
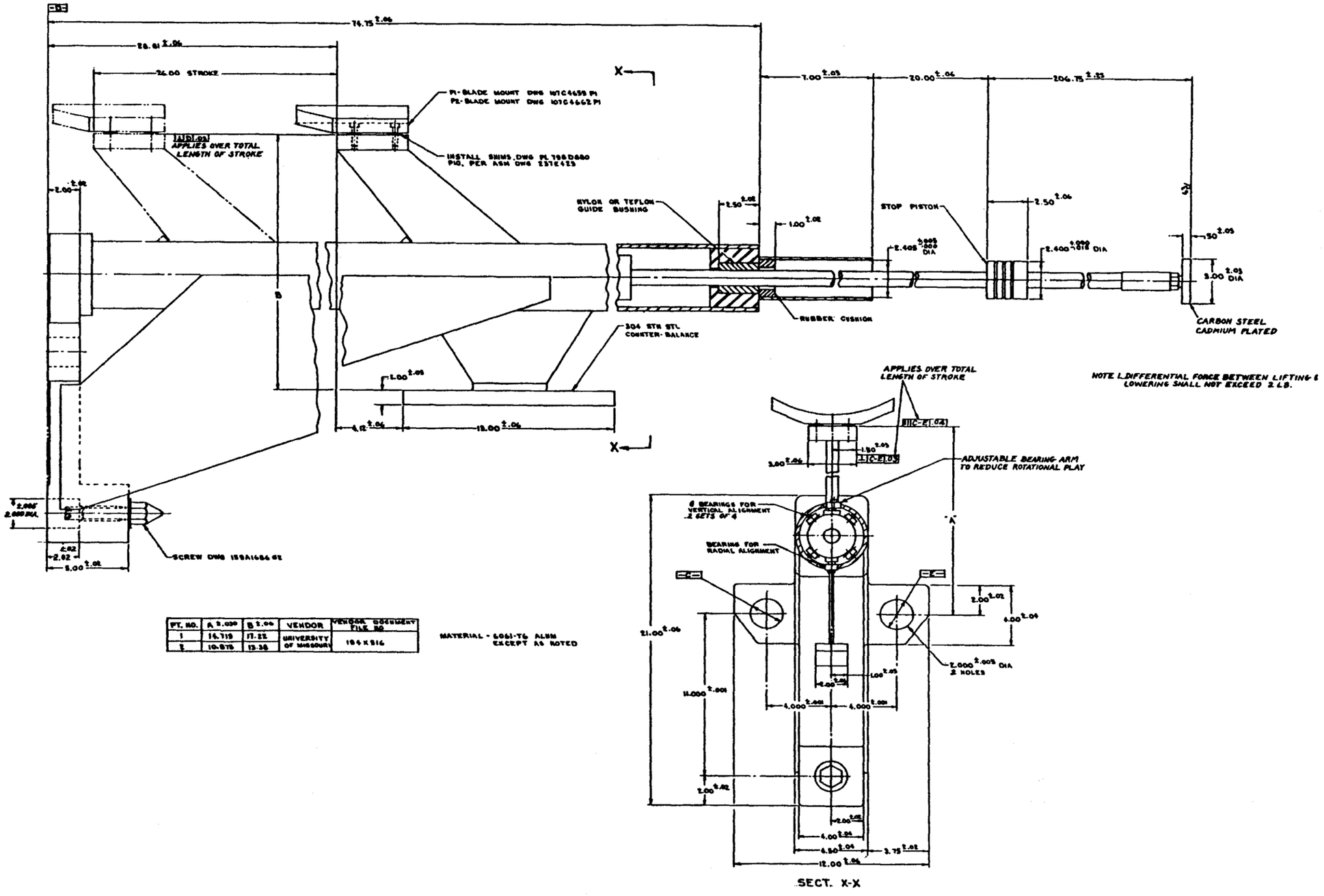


FIGURE 4.4 – MURR CORE LOADING



NOTE 1. DIFFERENTIAL FORCE BETWEEN LIFTING & LOWERING SHALL NOT EXCEED 2 LB.

PT. NO.	A ±.030	B ±.06	VENDOR	VENDOR DOCUMENT FILE NO
1	14.719	17.22	UNIVERSITY OF MISSOURI	194 X 816
2	10.875	13.38		

MATERIAL - 6061-T6 ALUM EXCEPT AS NOTED

FIGURE 4.5 – OFFSET MECHANISM

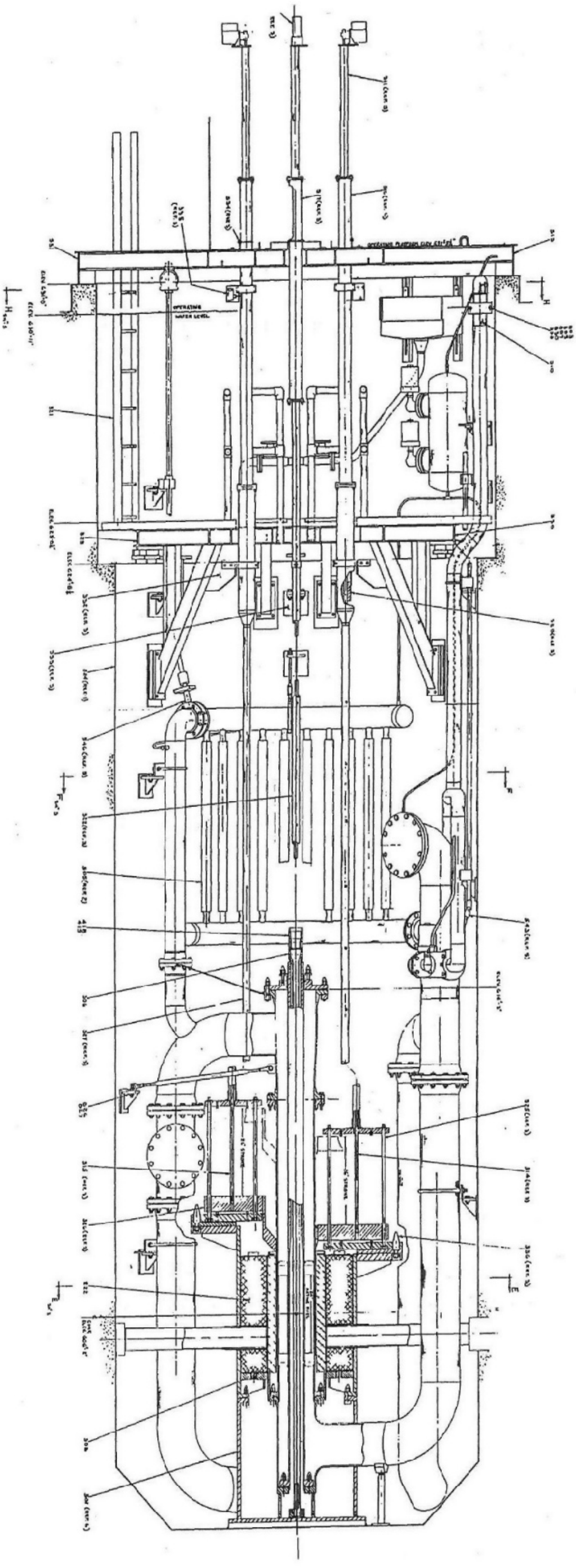


FIGURE 4.6 – REACTOR ASSEMBLY AND ARRANGEMENT

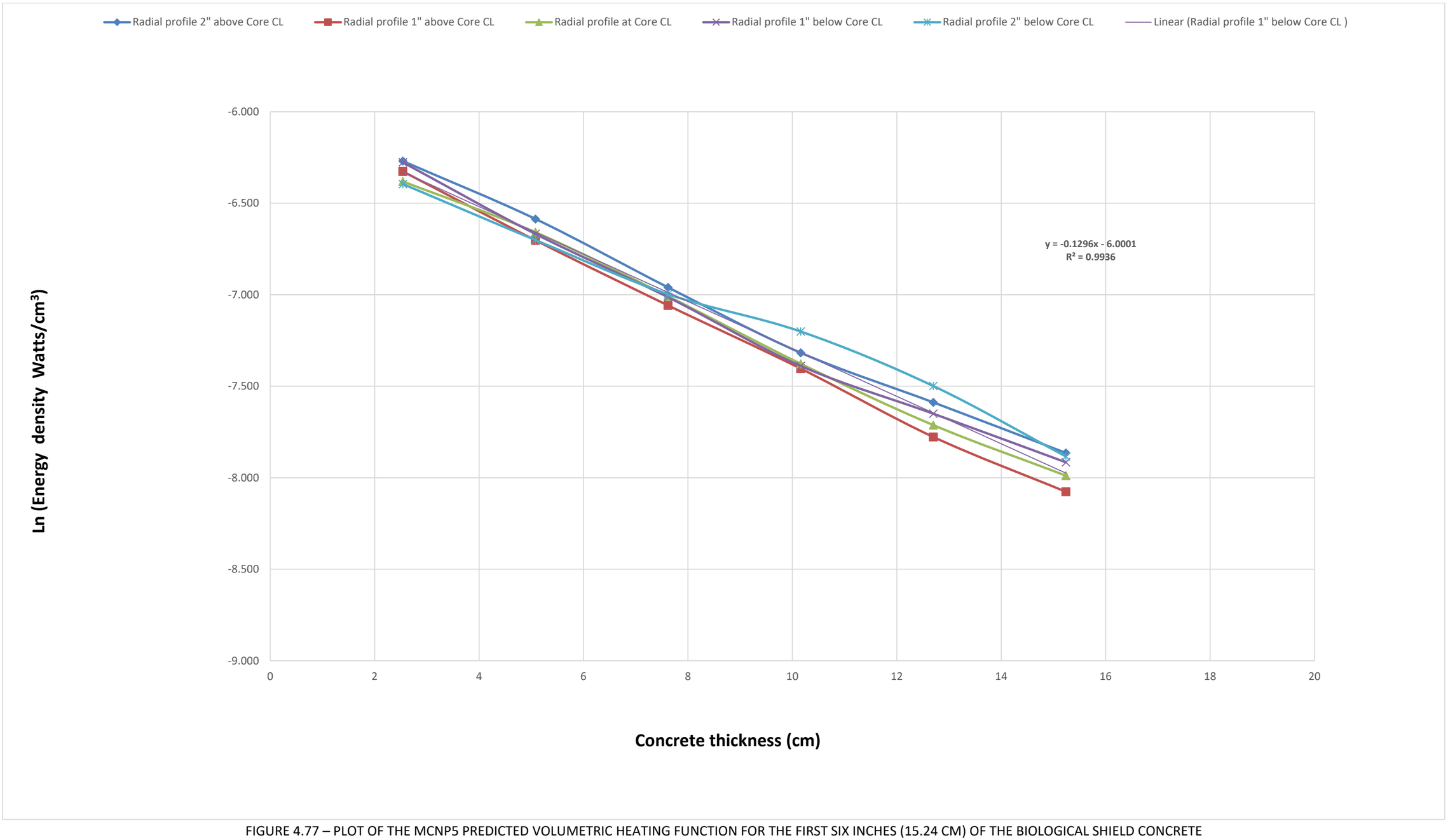


FIGURE 4.77 – PLOT OF THE MCNP5 PREDICTED VOLUMETRIC HEATING FUNCTION FOR THE FIRST SIX INCHES (15.24 CM) OF THE BIOLOGICAL SHIELD CONCRETE

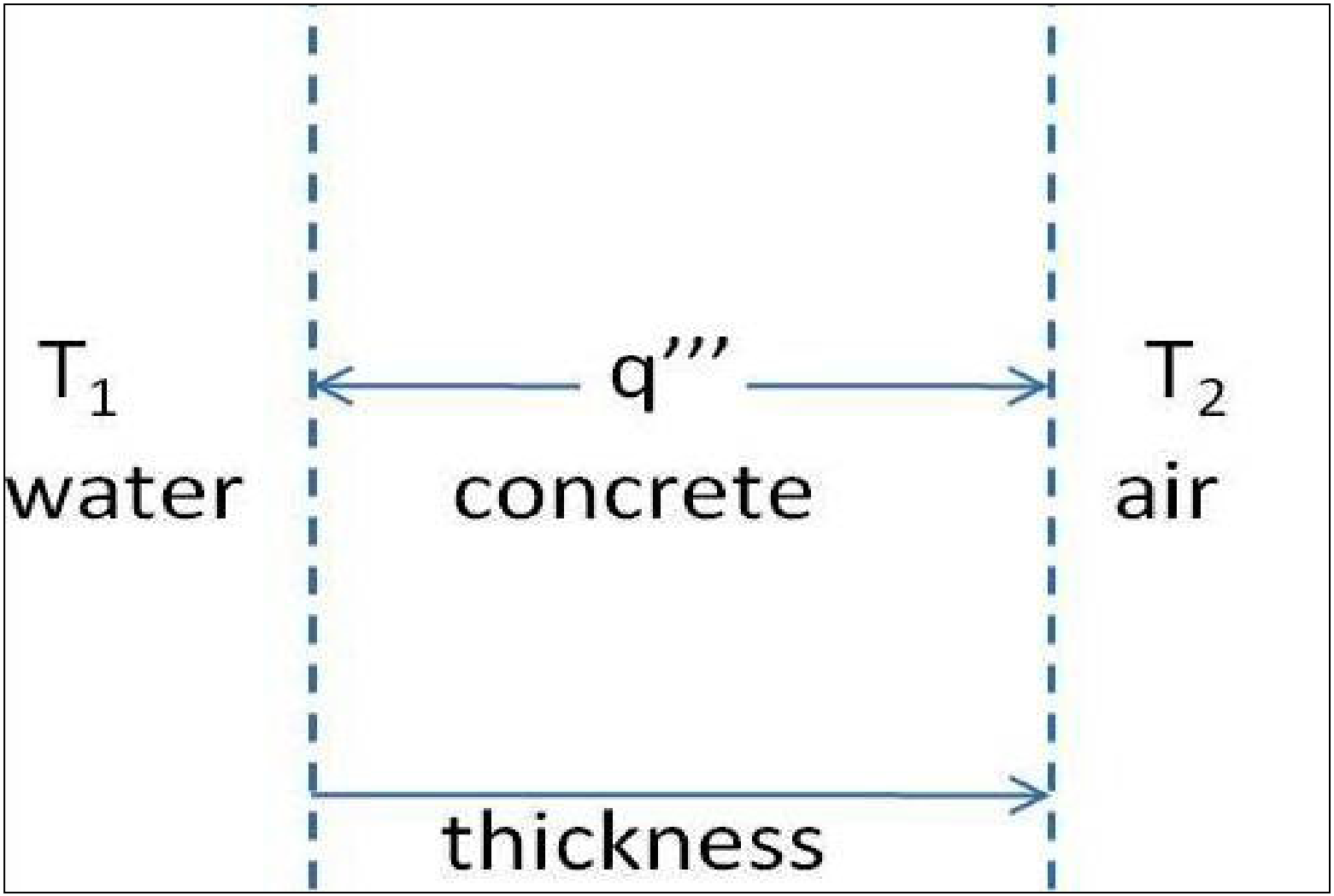


FIGURE 4.88 – SCHEMATIC OF THE ONE-DIMENSIONAL HEAT TRANSFER MODEL USED TO STUDY THE TEMPERATURE PROFILE IN THE SHIELD DUE TO GAMMA ENERGY DEPOSITION

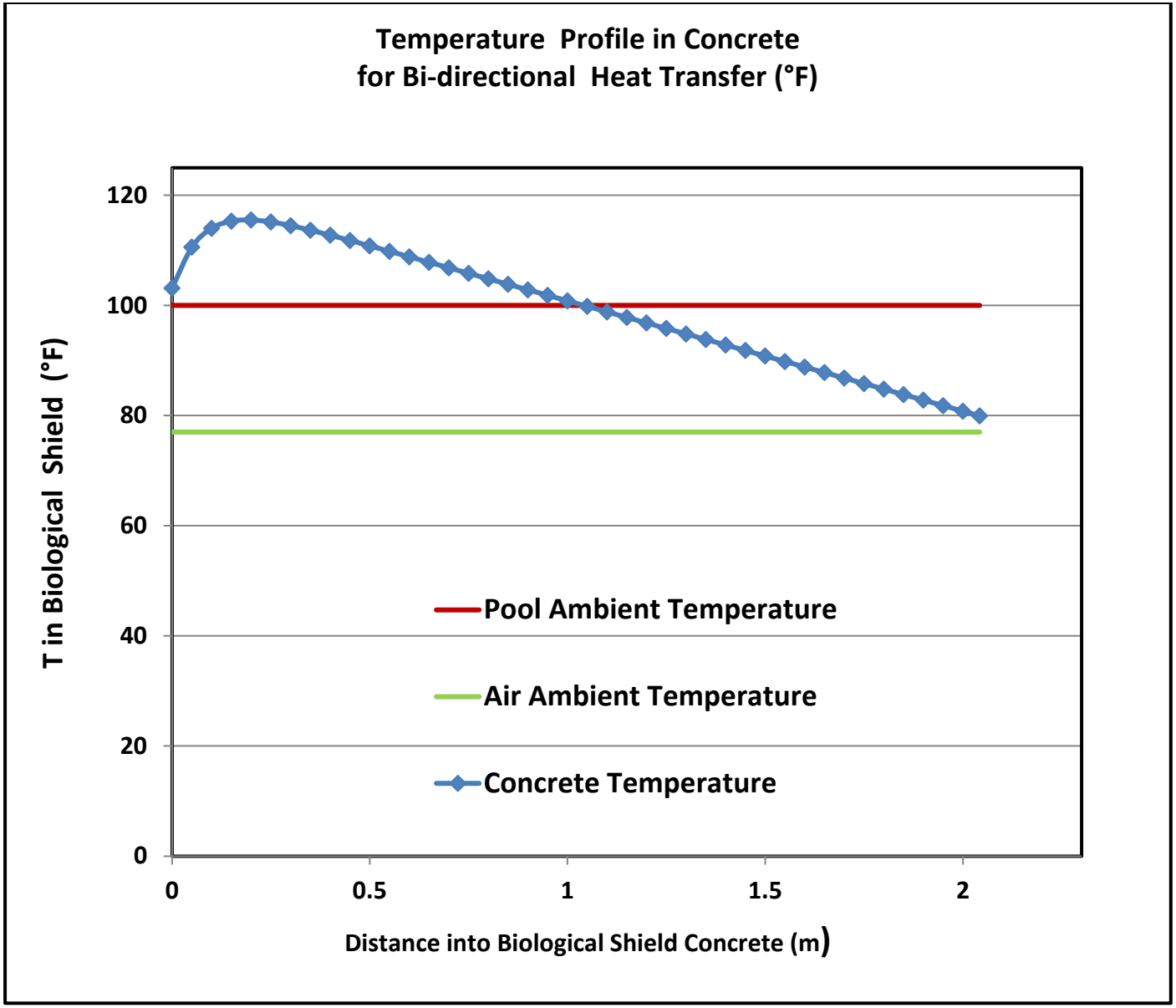


FIGURE 4.99 – TEMPERATURE PROFILE DUE TO GAMMA HEATING ACROSS THE BOLOGICAL SHIELD

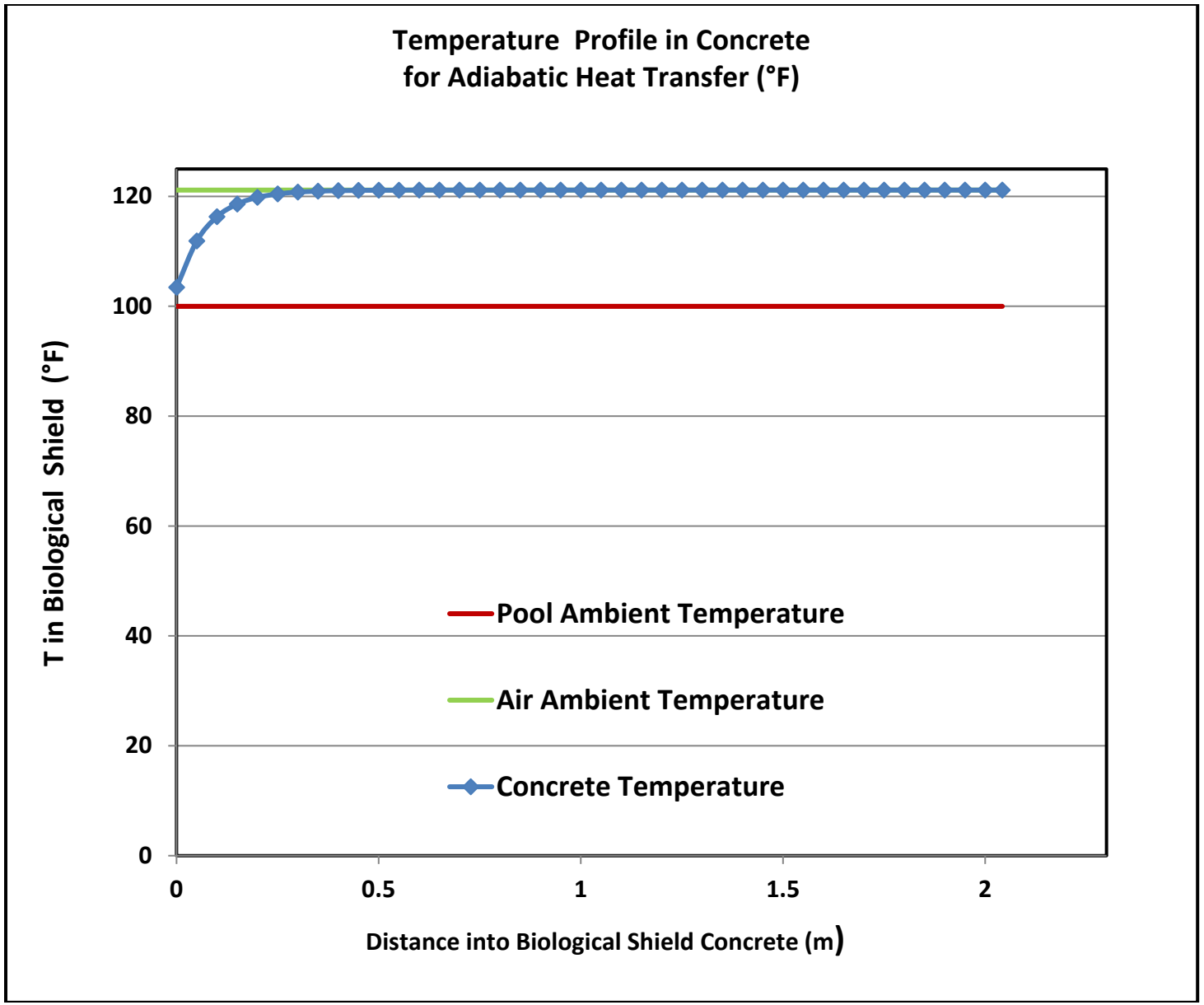


FIGURE 4.1010 – TEMPERATURE PROFILE FOR ADIABATIC HEAT TRANSFER TO THE AIR OUTSIDE THE BIOLOGICAL SHIELD

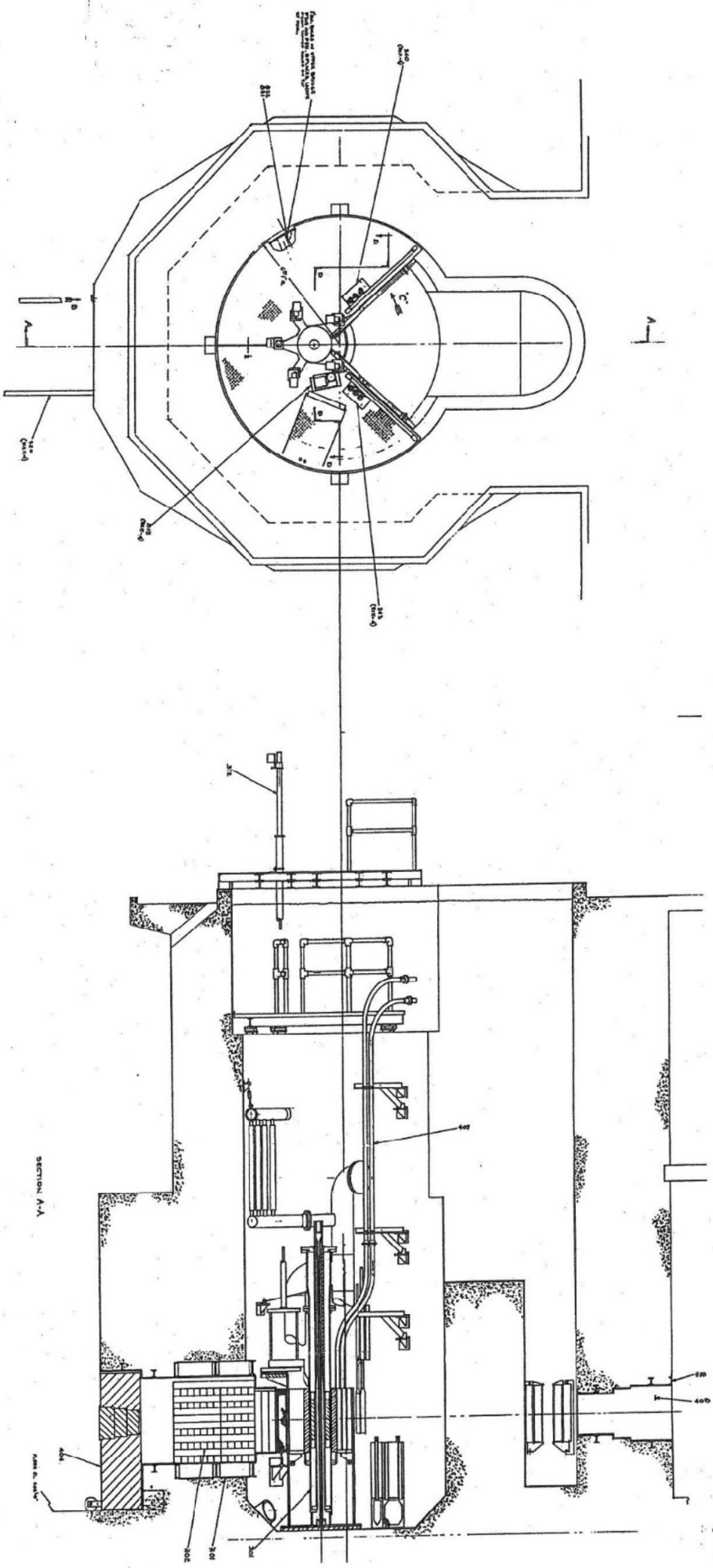


FIGURE 4.11 – REACTOR AND BIOLOGICAL SHIELD - PLAN AND SECTION VIEW

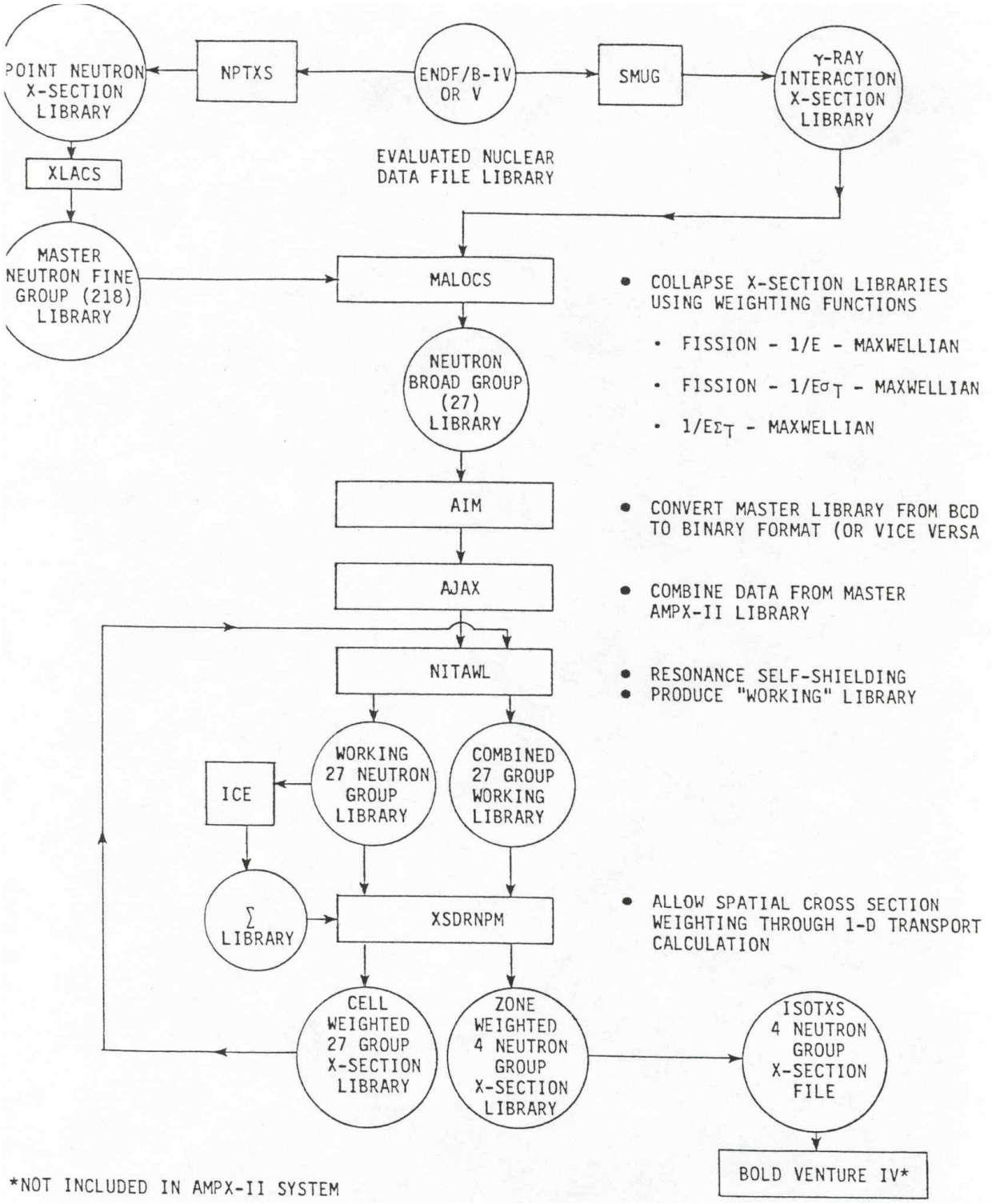


FIGURE 4.12 – CALCULATIONAL SEQUENCE OF AMPX-II

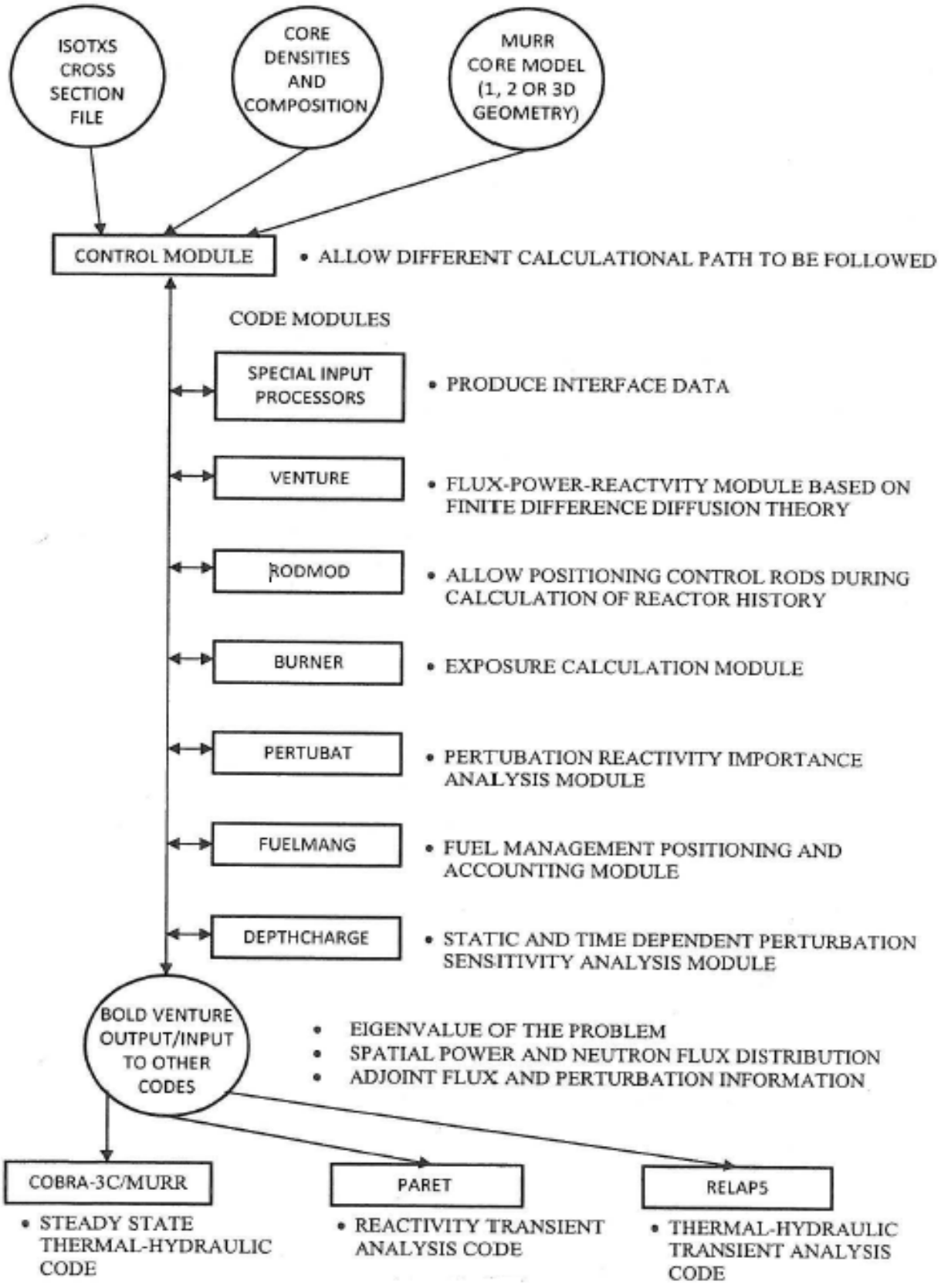


FIGURE 4.13 – COMPUTATIONAL MODULES OF BOLD VENTURE

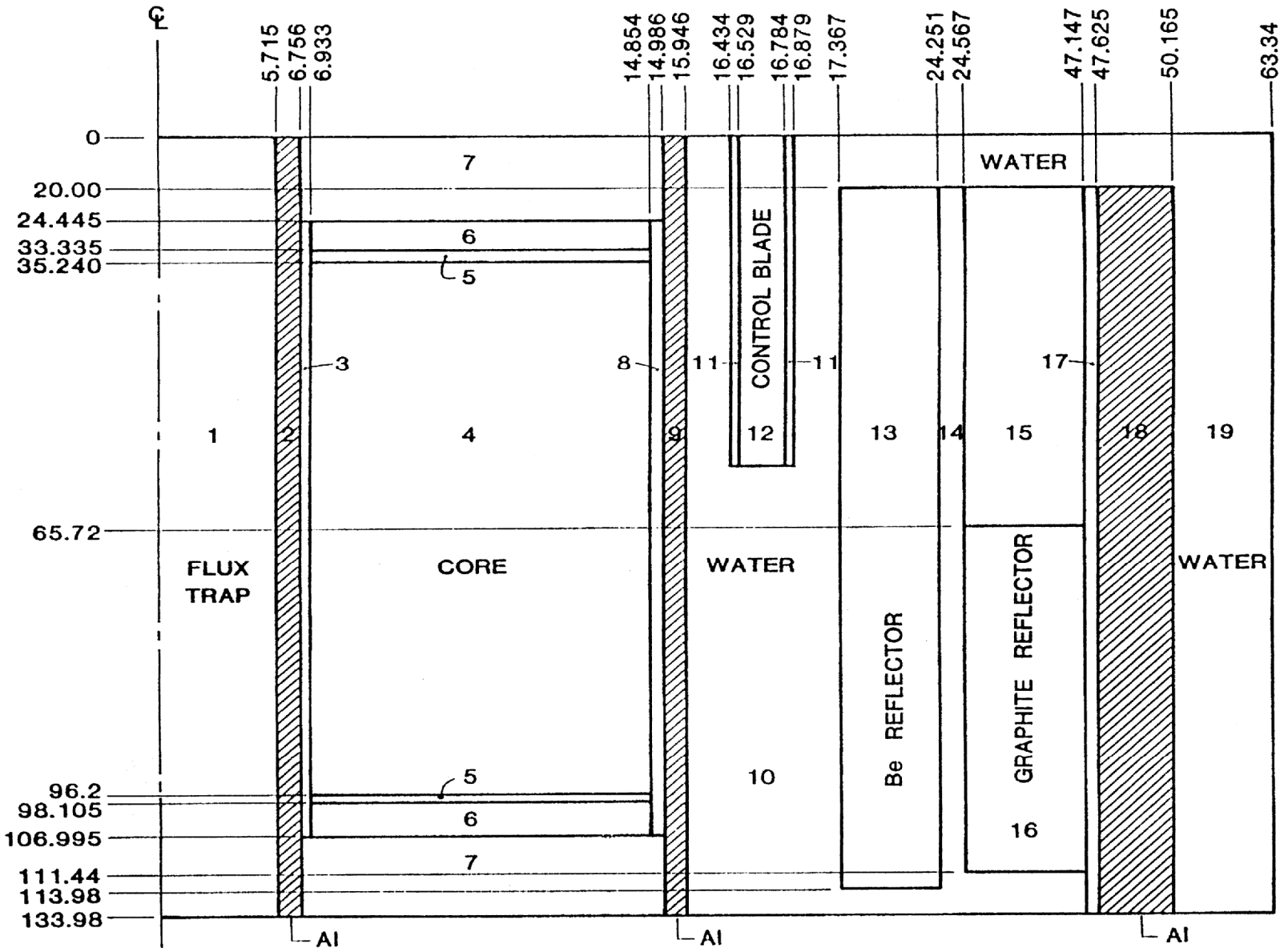


FIGURE 4.14 – R-Z MODEL OF THE MURR REACTOR

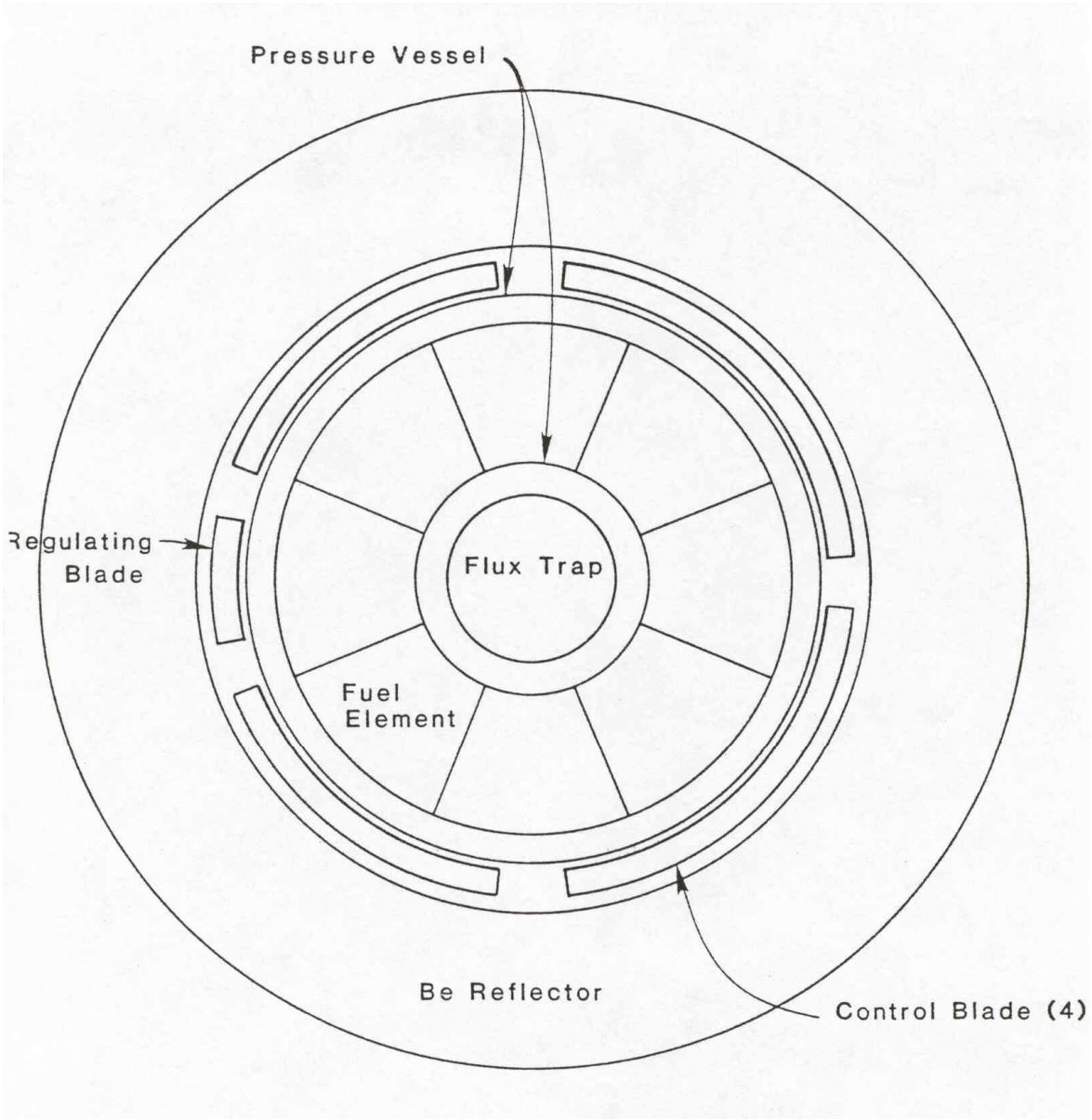


FIGURE 4.15 – Θ-R MODEL OF THE MURR CORE AND BERYLLIUM REFLECTOR

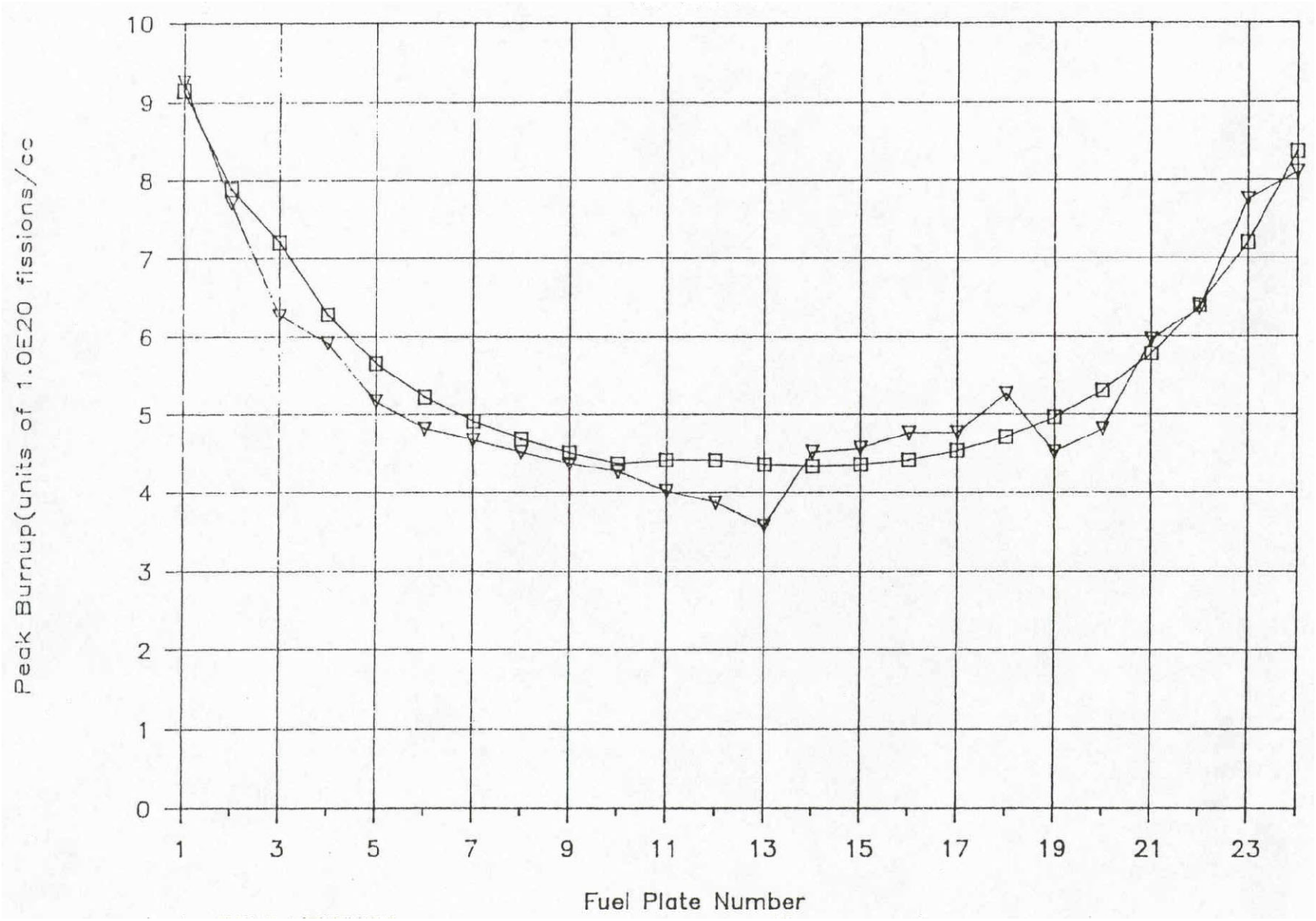


FIGURE 4.16 – COMPARISON OF PEAK BURNUP IN FUEL PLATES OF ELEMENT 775F3

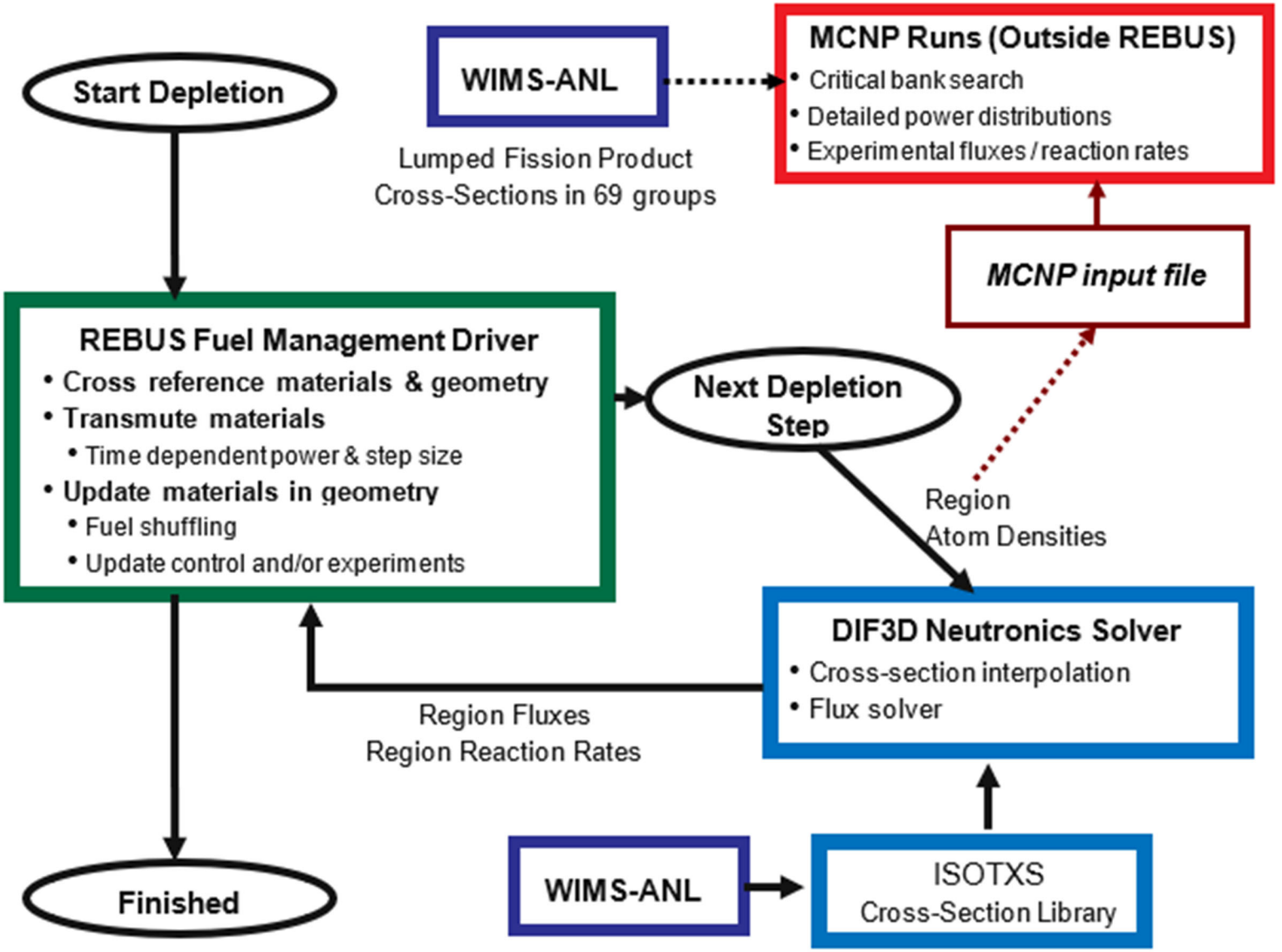


FIGURE 4.17 – LINKAGE OF CODES USED IN THE ANALYSIS SUITE

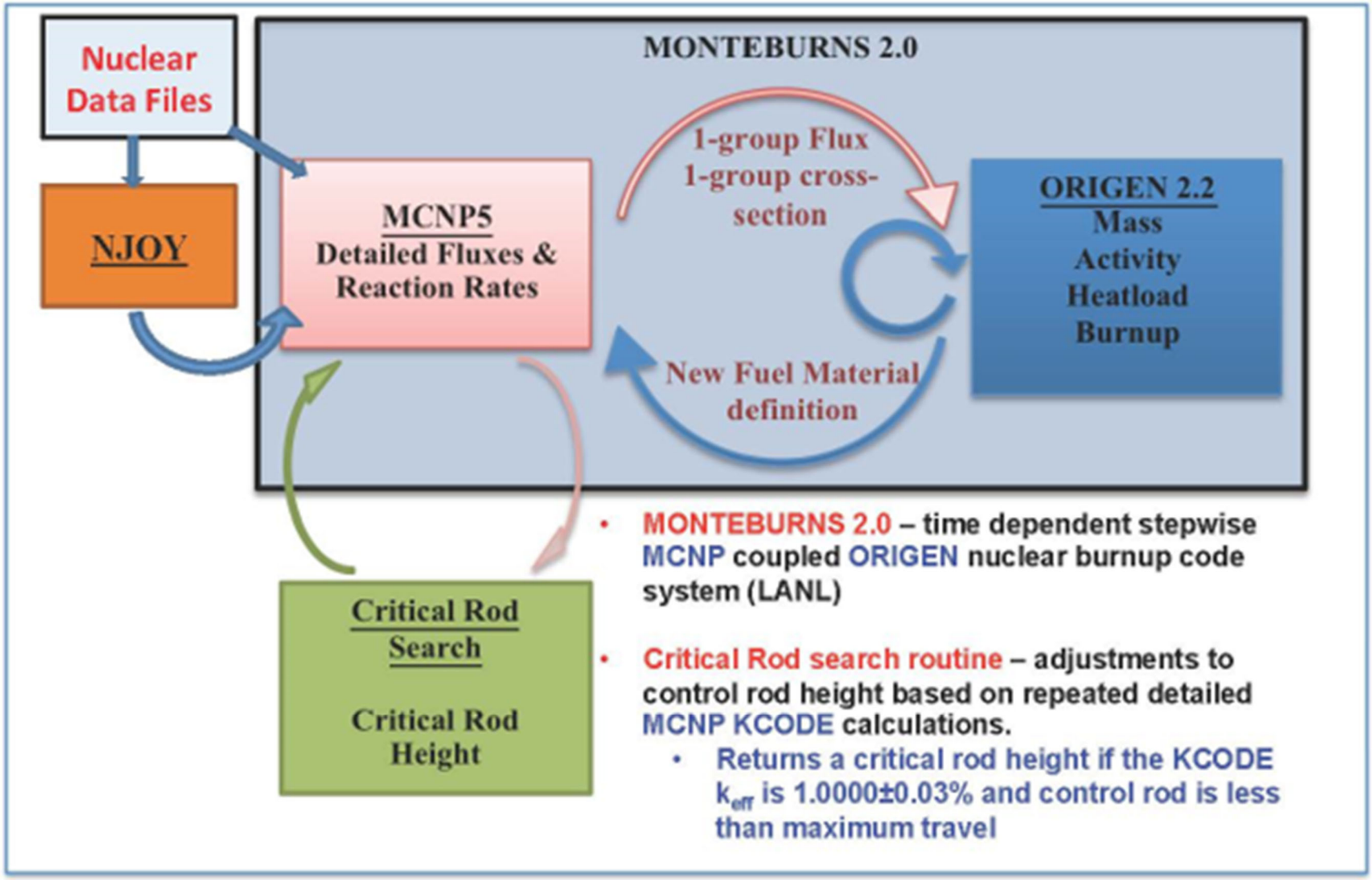


FIGURE 4.18 – CODE SUITE FLOW DIAGRAM IMPLEMENTED AT MURR FOR ROUTINE CORE PHYSICS ANALYSIS

CHAPTER 4

ATTACHMENT A

CURRENT SAFETY LIMIT ANALYSIS FOR MURR

APPENDIX F

SAFETY LIMIT ANALYSIS FOR MURR

TABLE OF CONTENTS

F.0 SAFETY LIMIT ANALYSIS FOR THE MURR 3
F.1 Introduction 3
F.2 Safety Limit Criteria 4
F.3 Discussion of Results 9
F.4 Method of Analysis 9
F.5 Determination of Worst-Case Safety Limit Peaking Factors 10
F.6 Method of Performing Safety Limit Calculations 13
F.7 Output 17

LIST OF TABLES

Table F-1 MAXIMUM ALLOWABLE CORE POWER LEVEL (MW) WITH PRESSURIZER PRESSURE AT 60 PSIA (413.7 kPa) 6
Table F-2 MAXIMUM ALLOWABLE CORE POWER LEVEL (MW) WITH PRESSURIZER PRESSURE AT 75 PSIA (517.1 kPa) 7
Table F-3 MAXIMUM ALLOWABLE CORE POWER LEVEL (MW) WITH PRESSURIZER PRESSURE AT 85 PSIA (586.1 kPa) 8
Table F-4 SUMMARY OF MURR HOT CHANNEL FACTORS 12
Table F-5 REFERENCE PRESSURE DROP DATA 15

LIST OF FIGURES

Figure F.1 – MURR SAFETY LIMIT CURVES FOR PRESSURIZER PRESSURE AT 60 PSIA 18
Figure F.2 – MURR SAFETY LIMIT CURVES FOR PRESSURIZER PRESSURE AT 75 PSIA 19
Figure F.3 – MURR SAFETY LIMIT CURVES FOR PRESSURIZER PRESSURE AT 85 PSIA 20
Figure F.4 – SECTION OF PRIMARY COOLAND SYSTEM FOR FLOW MODEL 21

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12. "Flow of Fluids Through Valves, Fittings, and Pipe," Technical Paper No. 410, Crane Co., 1991.

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15. Written communication dated July 8, 2013, from the Nuclear Regulatory Commission providing the Safety Evaluation Report regarding issuance of Amendment #36 to Amended Facility License N0. R-103.

F.0 SAFETY LIMIT ANALYSIS FOR THE MURR

F.1 Introduction

The University of Missouri Research Reactor (MURR) Safety Limits (SLs) establish the maximum allowable reactor power limits, the dependent variable, for safe operation under different combinations of three (3) measurable independent operating parameters – primary coolant flow, reactor inlet water temperature and primary coolant pressurizer pressure. The limits provide the basis for determining the Limiting Safety System Settings (LSSSs) and operating limits for 5- and 10-MW operations (also known as Mode II and Mode I operation, respectively).

The previous Mode I and II SL curves were developed in 1973 by the NUS Corporation for the 1974 uprate in power from 5 to 10 MW. Appendix F of Addendum 4 to the MURR Hazards Summary Report (HSR) (Reference 1) provided the SL Analysis for Mode I and II operation. This analysis generated two (2) SL curves corresponding to primary coolant pressurizer pressures of 60 and 75 psia. Section 6.0 of Addendum 5 to the MURR HSR (Reference 2) extended the original analysis to include a third SL curve for a pressurizer pressure of 85 psia, i.e., the nominal operating pressure. The safety analysis for natural convective cooling of the core, or Mode III operation, is provided in Section 5.5.3 of the MURR HSR (Reference 3). The power peaking factors used in the determination of the MURR SLs for Mode I and II operation are provided in Section 3.3 of Addendum 3 to the MURR HSR (Reference 4). Appendix H of Addendum 4 to the MURR HSR (Reference 5) provided the bases for determining the LSSSs for Mode I and Mode II operation.

While answering a relicensing Request for Additional Information (RAI) question, an error was discovered in the following document: “Safety Limit Analysis for the MURR Facility,” NUS Corporation, NUS-TM-EC-9, May 1973 (Reference 6). NUS-TM-EC-9 was the base document that was used in preparing the original Appendix F of Addendum 4 to the MURR HSR. The error was a discrepancy between the “diameter of heated surface,” known as the variable D_i, as it is defined by Bernath (Reference 7) and a more commonly used “heat diameter” definition that was inadvertently used by the NUS Corporation when developing the SLs.

A reanalysis of the MURR SLs resulted in the generation of new reactor power limits as shown in Tables F-1, F-2, and F-3, and three new sets of SL curves corresponding to primary coolant pressurizer pressures of 60, 75, and 85 psia (413.7, 517.1, and 586.1 kPa). The SL curves are shown in Figures F.1, F.2, and F.3, respectively. This new analysis is a replacement for the original analysis performed by the NUS Corporation (References 13, 14 and 15). The results are the bases for the changes to the SLs for Mode I and II operation; however, no changes to the LSSSs for Mode I, II, and III operation are required. This revised Appendix F of Addendum 4 to the MURR HSR combines and replaces the current versions of the following three documents:

1. Section 3.3 of Addendum 3 to the MURR HSR;

2. Appendix F of Addendum 4 to the MURR HSR; and

3. Section 6.0 of Addendum 5 to the MURR HSR.

Section 5.5.3, "Analysis of Natural Convective Cooling of the Core," of the MURR HSR assumed a combination of power peaking factors that exceed the combined power peaking factors for the current uranium-aluminide fuel core. The 1965 natural convective cooling analysis was based on the original 5.2 Kg U-235 alloy fuel core and assumed an overall power peaking factor of 4.653 for the hot spot heat flux. Mode III operation would only be used to perform reactor physics measurements on a 0 MWD core (eight fresh fuel assemblies). The 0 MWD uranium-aluminide fuel core has an overall power peaking factor of 3.315 for the hot spot heat flux. Including the engineering peaking factors, the overall hot spot heat flux peaking factor is 3.927. As provided later in Table F-4, the worst-case overall hot spot heat flux peaking factor for a mixed burnup uranium-aluminide fuel core is 4.116. Therefore, the analysis in Section 5.5.3 still conservatively envelops Mode III operation of MURR.

F.2 Safety Limit Criteria

The study objective was to determine core power limits for safe operation at specified combinations of possible core operating conditions. Safe operation is defined as avoiding excessive fuel or clad temperature that could cause clad failure and thereby release fission products into the primary coolant. To accomplish this, the power level that ensures safe margins to both flow instability and critical heat flux are determined for the designated combination of primary coolant parameters of primary coolant flow, reactor inlet water temperature and pressurizer pressure. The following discussion presents the basis for specifying criteria that defines the safe operating margin.

In subcooled flowing water, nucleate boiling at a heated surface can substantially enhance heat transfer. However, when the heat flux is sufficiently high, a layer of vapor, which is a poor conductor of heat, can form on the heated surface and impede heat transfer to the flowing liquid from the core. This is called "critical heat flux" (CHF) or "departure from nucleate boiling" (DNB). Heat flux above CHF can cause dramatic increases in the surface temperature and severely damage the heated surface. For this reason, CHF is sometimes referred to as "burnout."

When coolant is flowing in a series of parallel heated channels, such as in the MURR fuel assemblies, flow instability is possible. If the power of the reactor is sufficiently increased beyond the onset of nucleate boiling, boiling in the limiting channel(s) can increase the resistance to flow in that location. This added resistance reduces the flow in the limiting channel(s) by diverting flow to the other cooler channels. The loss of flow in the limiting channel(s), in turn, makes the boiling more severe and further reduces the flow, resulting in a sudden flow excursion in the limiting channel(s). This is called "Ledinegg flow instability." The abnormally low flow can cause CHF to occur in that location. When flow instability leads to CHF, the cause of the failure is attributed to flow instability rather than to CHF. Correlations that predict CHF typically assume that the flow rate is maintained and that flow instability is not present.

Since 1974, the MURR SLs have been based on two groups of flow instability tests performed for the Advanced Test Reactor (ATR). The Croft (Reference 8) tests were performed at Argonne National Laboratory (ANL) in 1963 on three channel thicknesses (0.054, 0.072, and 0.094 inches (0.137, 0.183,

and 0.239 cm)). The Waters (Reference 9) tests were performed at Pacific Northwest Laboratory (PNL) in Richland, Washington on a channel thickness of 0.070 inches (0.178 cm).

The conceptual design for MURR was performed by the Internuclear Company of Clayton, Missouri, while they were designing the ATR. A literature review demonstrates that the ATR closely compares to MURR in fuel design, although the ATR core is twice as long as the MURR core. Based on the Croft and Waters tests and criteria suggested by Croft for the burnout (caused by flow instability) of the 0.094-inch (0.239-cm) ATR test channel, MURR adapted the following two SL criteria (Reference 6):

1. 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 CHF Correlation at that point; and

2. The coolant exit temperature from the hot channel shall be less than the saturation temperature at the core exit pressure.

These criteria are designed to avoid flow instability. The Croft version of the first criterion, based on measurements of the 0.094-inch (0.239-cm) wide test channel, allowed the local heat flux at any point to be as large as 0.6 of the Bernath CHF. However, MURR conservatively reduced the allowed heat flux fraction to 0.5 of the Bernath CHF. Therefore, the MURR assumed flow instability point corresponds to a CHF ratio of 2.0 based on the Bernath Correlation because, by definition, the ratio of local heat flux to CHF is the reciprocal of the CHF ratio. When the second criterion above is the more limiting, the CHF ratio based on the Bernath Correlation is greater than 2.0.

The Bernath CHF Correlation equation (1) is given by the following:

$$
\begin{aligned}
&(\mathrm{Q/A})_{\mathrm{BO}} = \mathrm{h_{BO}}\left(\mathrm{T_{W_{BO}}} - \mathrm{T_b}\right) \\
&\mathrm{h_{BO}} = 10890\left(\frac{\mathrm{D_e}}{\mathrm{D_e} + \mathrm{D_i}}\right) + (\mathrm{slope})\mathrm{V} \\
&\mathrm{slope} = 48/\mathrm{D_e^{0.6}}, \text{if } \mathrm{D_e} \le 0.1\mathrm{ft} \\
&\mathrm{slope} = 90 + 10/\mathrm{D_e}, \text{if } \mathrm{D_e} > 0.1ft \\
&\mathrm{T_{W_{BO}}} = 57\ln \mathrm{P} - 54\left(\frac{\mathrm{P}}{\mathrm{P}+15}\right) - \frac{\mathrm{V}}{4}
\end{aligned}
\qquad (1)
$$

where:

$(Q/A)_{BO}$ = CHF in p.c.u./hr-ft^2 (BO stands for "burnout") (p.c.u. is a "pound centigrade unit" and 1 p.c.u. = 1.8 btu);

h_{BO} = heat transfer coefficient corresponding to the CHF in p.c.u./hr-ft^2-°C;

$T_{W_{BO}}$ = wall temperature at which CHF occurs in °C;

T_b = local bulk coolant temperature in °C;

D_e = hydraulic diameter of the coolant passage in feet, which equals 4 times the flow area divided by the *wetted* perimeter;

D_i = diameter of the heated surface, which equals the heated perimeter divided by π in feet;

P = pressure in psia; and

V = velocity of the coolant in ft/s.

The above two safety criteria used by MURR for the SLs since 1974 have been retained in the current safety analysis and were used in defining the current SL curves. The derived curves allow sufficient margin between the SLs and the actual predicted flow instability DNB and a significantly greater margin to CHF DNB. Additionally, the usual conservatism of worst-case power peaking and non-uniform fuel loading and appropriate hot channel factors are included, lending greater assurance that MURR will not approach fuel failure under the most severe operating conditions for the licensed steady-state power level of 10 MW.

The reanalysis of the MURR SLs for Mode I and Mode II resulted in the generation of new reactor power limits as shown in Tables F-1, F-2 and F-3 and three (3) new SL curves corresponding to primary coolant pressurizer pressures of 60, 75, and 85 psia (413.7, 517.1, and 586.1 kPa) as shown in Figures F.1, F.2, and F.3, respectively. This new analysis is a replacement of the original analysis performed by the NUS Corporation in 1973-1974. The results are the basis for the changes to the SLs.

TABLE F-1
MAXIMUM ALLOWABLE CORE POWER LEVEL (MW) WITH PRESSURIZER PRESSURE AT 60 PSIA (413.7 KPA)

REACTOR INLET WATER CONDITIONS										
Temp °F (°C)	**Flow Rate (GPM) (LPM)**									
	400 (1514)	800 (3028)	1200 (4542)	1600 (6057)	2000 (7571)	2400 (9085)	2800 (10599)	3200 (12113)	3600 (13627)	4000 (15142)
120 (48.9)	3.116	5.352	7.444	9.406	11.216	12.837	14.220	15.299	15.972	16.078
140 (60)	2.862	4.912	6.826	8.615	10.255	11.713	12.943	13.880	14.432	14.453
155 (68.3)	2.670	4.582	6.362	8.020	9.533	10.868	11.980	12.809	13.266	13.216
160 (71.1)	2.606	4.472	6.207	7.821	9.292	10.586	11.659	12.451	12.876	12.800
180 (82.2)	2.289	4.032	5.588	7.027	8.327	9.456	10.371	11.015	11.309	11.128
200 (93.3)	1.863	3.592	4.970	6.235	7.364	8.327	9.084	9.580	9.740	9.450

Notes:

For 400 gpm (1514 lpm), the lowest inlet water temperature with saturation limiting at exit: 172.7 °F (78.2 °C), 2.445 MW.
For 180 °F (82.2 °C), the highest flow rate with saturation limiting at exit: 449.0 gpm (1700 lpm) at 291.2 °F (144 °C), 2.567 MW.
For 200 °F (93.3 °C), the highest flow rate with saturation limiting at exit: 701.0 gpm (2654 lpm) at 290.4 °F (143.6 °C), 3.235 MW

TABLE F-2
MAXIMUM ALLOWABLE CORE POWER LEVEL (MW) WITH PRESSURIZER PRESSURE AT 75 PSIA (517.1 KPA)

REACTOR INLET WATER CONDITIONS										
Temp °F (°C)	**Flow Rate (GPM) (LPM)**									
	400 (1514)	800 (3028)	1200 (4542)	1600 (6057)	2000 (7571)	2400 (9085)	2800 (10599)	3200 (12113)	3600 (13627)	4000 (15142)
120 (48.9)	3.369	5.799	8.093	10.273	12.326	14.227	15.944	17.434	18.639	19.480
140 (60)	3.113	5.358	7.473	9.478	11.359	13.093	14.649	15.986	17.051	17.769
155 (68.3)	2.922	5.027	7.007	8.880	10.632	12.240	13.674	14.894	15.850	16.472
160 (71.1)	2.858	4.916	6.851	8.680	10.389	11.955	13.348	14.529	15.449	16.038
180 (82.2)	2.597	4.474	6.229	7.881	9.417	10.814	12.043	13.067	13.840	14.295
200 (93.3)	2.169	4.033	5.608	7.083	8.446	9.674	10.738	11.605	12.230	12.550

Notes:
For 400 gpm (1514 lpm), the lowest inlet water temperature with saturation limiting at exit: 179.4 °F (81.9 °C), 2.609 MW.
For 180 °F (82.2 °C), the highest flow rate with saturation limiting at exit: 403.2 gpm (1526.3 lpm) at 306.2 °F (152.3 °C), 2.617 MW
For 200 °F (93.3 °C), the highest flow rate with saturation limiting at exit: 571.0 gpm (2161.5 lpm) at 305.9 °F (152.2 °C), 3.085 MW

TABLE F-3
MAXIMUM ALLOWABLE CORE POWER LEVEL (MW) WITH PRESSURIZER PRESSURE AT 85 PSIA (586.1 KPA)

REACTOR INLET WATER CONDITIONS										
Temp °F (°C)	**Flow Rate (GPM) (LPM)**									
	400 (1514)	800 (3028)	1200 (4542)	1600 (6057)	2000 (7571)	2400 (9085)	2800 (10599)	3200 (12113)	3600 (13627)	4000 (15142)
120 (48.9)	3.514	6.054	8.461	10.762	12.947	14.997	16.884	18.576	20.028	21.183
140 (60)	3.257	5.612	7.839	9.965	11.977	13.858	15.582	17.116	18.420	19.439
155 (68.3)	3.065	5.280	7.372	9.365	11.247	13.001	14.600	16.015	17.206	18.121
160 (71.1)	3.001	5.169	7.217	9.165	11.004	12.714	14.273	15.647	16.800	17.680
180 (82.2)	2.744	4.726	6.593	8.364	10.028	11.568	12.960	14.174	15.173	15.910
200 (93.3)	2.349	4.284	5.970	7.563	9.053	10.422	11.647	12.700	13.545	14.138

Notes:
For 400 gpm (1514 lpm), the lowest inlet water temperature with saturation limiting at exit: 184.0 °F (84.4 °C), 2.693 MW
For 200 °F (93.3 °C), the highest flow rate with saturation limiting at exit: 516 gpm (1953 lpm) at 314.9 °F (157.2 °C), 3.025 MW

All values in Tables F-1, F-2 and F-3 are limited by mesh interval 18 except the 5 values in red, which are limited by (1) reaching saturation temperature at the exit, and (2) the 400 gpm (1514 lpm) and 180 °F (82.2 °C) point in Table F.3 which is based on mesh interval 19.

The red values in Tables F-1, F-2 and F-3 are the power limits as established by the second criterion of approaching saturation temperature at the hot channel outlet to avoid bulk boiling of the coolant. The remaining entries reflect the thermal limits established by the first criterion which requires that the local heat flux does not exceed 50% of the Bernath predicted DNB heat flux. In the present study, the power limits for coolant flow rates greater than 701 gpm (2654 lpm) are always dictated by the first criterion (50% of Bernath), while there are only five (5) data points for 400 gpm (1514 lpm) in the tables where approaching bulk boiling at the exit criterion dictates the safe power level. The transition points for going from the second criterion to the first criterion are given at the bottom of each table for increasing coolant flow rate or decreasing reactor inlet water temperature.

The three (3) sets of curves together define a four-dimensional SL envelope that prescribes limiting values for the reactor power dependent variable for the 180 different combinations of the three (3) independent variables: primary coolant flow, reactor inlet water temperature and primary coolant pressurizer pressure. Each set of curves is for a different pressurizer pressure with the independent axis being primary coolant flow and the dependent axis being the SL power level. Each of the curves

corresponds to a different reactor inlet water temperature. Operation of MURR within this safety envelope will prohibit fuel meltdown or cladding damage as a result of either flow instability or CHF DNB. To evaluate SLs for pressurizer pressures intermediate to the three cases considered, interpolation will be used. For pressurizer pressures below 60 psia (414 kPa), extrapolation will be used to determine the SLs.

F.3 Discussion of Results

Figures F.1, F.2, and F.3 illustrate the effects of core operating conditions on the maximum allowed core power for safe MURR operation. The increase in allowed core power with flow is provided by the Bernath Correlation, equation (1). For all lower flow rates, the Bernath allowed DNB heat flux increases with flow but the slope slightly decreases with increasing flow. This is more pronounced for lower primary coolant pressurizer pressures and higher reactor inlet water temperatures. For the 60 psia (414 kPa) curves, all but the 120 and 140 °F (48.9 and 60 °C) curves have a decrease in maximum allowed power for a core flow rate increasing from 3600 to 4000 gpm (12627 to 15142 lpm). This shows that the core flow rate caused drop in core outlet pressure has more of a negative effect than the higher flow rate has a positive effect. As is expected and dictated by equation (1), the allowed power decreases with increasing inlet water temperature. Similarly, the increase in allowed power with increasing pressurizer pressure is as expected because the saturation temperature of the coolant increases with pressure.

F.4 Method of Analysis

In 1974, when MURR, working with the NUS Corporation, developed the previous SLs, the peaking factors used were extremely conservative because peaking factors had to be determined utilizing three (3) different 2D diffusion code models, which was the only code method available in the early 1970's. These peaking factors were provided in the original Table F-2 of Appendix F of Addendum 4 to the MURR HSR (A detailed description is provided in Section 3.3 of Addendum 3 to the MURR HSR):

On Heat Flux
Power-Related Factors

Nuclear Peaking Factors:

Radial	= 2.220	[*from all fresh fuel R-Z model*]
Local (Circumferential)	= 1.040	[*from all fresh fuel R-Θ model*]
Non-uniform Burnup	= 1.112	[*from mixed burnup fuel R-Θ model*]
Axial	= 1.432	[*from all fresh fuel R-Z model*]

The product of the above four (4) nuclear peaking factors is 3.676. With the engineering hot channel factors on flux included, the overall product is 4.35. This overall peaking factor is extremely conservative because peaking factors from two different core arrangements were combined. For a core consisting of eight (8) fresh fuel elements (0 MWD), causing a lower control rod height, higher radial and axial

peaking factors are then combined with the smaller local (circumferential) peaking factor caused by no burnup. For a non-uniform burnup core with a higher control rod height, resulting in lower radial and axial peaking factors, the larger azimuthal (non-uniform burnup) factor was obtained. These four (4) factors, which cannot all occur in the same core, were then combined to provide this very conservative, overall peaking factor.

Current 3D nucleonics codes show the axial peaking factors in a mixed burnup core are lower than in an all fresh fuel core because of less excess reactivity and the corresponding higher critical control rod height position. Also, the 3D model provides an accurate average plate peaking factor, instead of the 2D approach of trying to approximate this by using the all fresh fuel radial factor (R-Z) times the non-uniform burnup azimuthal factor (R-Θ). This is a result of combining the high excess reactivity and lower critical rod height of an all fresh core with the MURR design, consisting of all the control blades external to the outer pressure vessel, which tends to push the power down and inward in the core increasing the axial and radial peaking factors for the hot fuel plate (fuel plate-1).

Since 2006, MURR has been actively collaborating with the Department of Energy's Reduced Enrichment for Research and Test Reactor (RERTR) Program on the conversion from high-enriched uranium (HEU) to low-enriched uranium (LEU) fuel. During this time, the Argonne National Laboratory (ANL)/MURR team has benchmarked the MURR HEU fuel and reactor core design performance, which is documented in MURR Technical Data Report TDR-0125, "Feasibility Analyses for HEU to LEU Fuel Conversion of the University of Missouri Research Reactor (MURR)" (Reference 10). The computer modeling completed in performing the feasibility analyses provides more accurate 3D peaking factors, which were used in determining the revised SLs. In addition to the feasibility analyses worst-case HEU peaking factors, an additional peaking factor of 1.062 is included to define the maximum allowable nuclear peaking factor. The maximum allowable nuclear peaking factors were combined with the engineering peaking factors, which are provided below in Table F-4, to establish the maximum allowable overall peaking factor for enthalpy rise and heat flux used to generate the revised SLs that meet the existing criteria.

F.5 Determination of Worst-Case Safety Limit Peaking Factors

MURR Technical Data Report TDR-0125 documents the neutronic analysis steps taken to develop the code system, modeling and benchmarking for the MURR HEU fuel performance. To compare the performance of the proposed LEU design to the current HEU operation, models were developed for the current reactor reflector configuration with typical experimental loadings. To properly model the current HEU core fuel utilization, shutdown margin and experimental performance, it was necessary to develop a computational shuffling that would accurately model the actual complex fuel cycle used at MURR. This is also described in detail in TDR-0125.

Also, in order to perform a comparison between HEU and LEU fuel, it was necessary to define reference cores that could be compared in order to establish feasibility of all major parameters: fuel cycle performance, shutdown margin, thermal-hydraulic steady-state safety margins, and experimental performance. It was decided that these reference cores should be close to limiting in order to provide additional confidence that the safety margin calculations treat the potentially different limiting power

shapes from the HEU and LEU fuel elements. The actual limiting peaking factor for the MURR uranium-aluminide fuel elements is modeled by multiplying the worst-case peaking factors by an additional 1.062 factor to determine the SLs.

Core configurations with potentially high-power peaking were identified from the fuel cycle simulation in order to analyze for thermal-hydraulic behavior. The fuel cycle simulations provided a collection of more than 325 HEU Core States (4 state points per week x 82 weeks or different core combinations). Each of these State points from the REBUS-DIF3D model was examined and sorted to rank the cases with the highest peak heat flux in the core. As expected, it was found that the highest heat flux always occurs in cases where a fresh element is loaded next to an element that is near its discharge burnup limit.

Based on the selection criteria, the core conditions defined as week 58 (out of 82) of the HEU fuel cycle simulation was chosen for more detailed evaluation. Depleted fuel material compositions were extracted from the REBUS-DIF3D results for each plate and axial depletion zone in the model; this consisted of 2,304 fuel compositions (8 elements x 24 plates x 12 axial zones) for each of the HEU cores. This data was utilized as material composition data for a detailed MCNP model for calculating power distributions and estimated critical positions.

The week 58 HEU core has fuel elements with the following power history: two elements each with 0, 65, 81 and 142 MWD. Of the four different states for the week 58 core, the beginning of the week xenon-free core with no samples or sample holder loaded in the center flux trap region produces the highest power peaking factors. This is because the highest peaking factors are always on the inner fuel plate, fuel plate-1, and initial critical rod height combined with the 100% water filled flux trap produces even higher peaking factors in fuel plate-1.

The detailed week 58 core MCNP run for the 3B Core State was used to extract the 3D power peaking factors. From these fuel plate heat flux peaking factors, the peaking factors for the hottest coolant channel (channel 2) were determined. The enthalpy rise in channel 2 is calculated based on the average heat flux for fuel plate-1 and -2. This is conservative because both fuel plate-1 and -2 transfer a little less than half their total heat to coolant channel 2 since the coolant channels on the opposite side of each plate operate at a lower temperature than does channel 2.

The peaking factors for channel 2 are provided in Table F-4. Fuel plate-1 and -2 average and the azimuthal in the channel are calculated from the MCNP results. To establish the maximum allowable overall peaking factor, an additional 1.062 factor is included in both the enthalpy rise and the heat flux. The Engineering Hot Channel Factors are based on the MURR aluminide fuel fabrication specifications and are as used in NUS-TM-EC-9 (Reference 6). The overall flow-related factor of 0.8197 is the product of the 0.9 (72-mils/80-mils) narrow channel thickness factor times the 0.9108 reduction of coolant velocity in the narrow channel.

TABLE F-4
SUMMARY OF MURR HOT CHANNEL FACTORS

On Enthalpy Rise in Channel 2			
Power-Related Factors			-
Nuclear Peaking Factors			-
Fuel Plate-1 and -2 Average			1.9843
Azimuthal in the Channel			1.0921
Additional Allowable Factor			1.0620
Engineering Hot Channel Factors			-
Fuel Content Variation			1.0300
Fuel Thickness/Width Variation			1.0300
Overall Product			2.4416
Flow Related Factors			-
Core/Loop Flow Fraction			1.0000
Assembly Minimum/Average Flow Fraction			1.0000
Channel Minimum/Average Flow Fraction			-
Inlet Variation			1.0000
Width Variation			1.0000
Thickness Variation			0.72/0.80
Within Channel Minimum Channel Thickness Effect On:			-
Velocity Factor			0.9108
Overall Factor on Flow Reduction			0.8197
-			
On Heat Flux from Fuel Plate-1			
Power-related Factors	Mesh interval between locations down the fuel plate[1]		
Mesh Interval Number	14 (13-14”) (33.02-35.56 cm)	18 (17-18”) (43.18-45.72 cm)	19 (18-19”) (45.72-48.26 cm)
Nuclear Peaking Factors	-	-	-
Fuel Plate (Hot Plate Average)	2.215	2.215	2.215
Azimuthal Within Plate	1.070	1.070	1.070
Axial Peak	1.3805	1.2958	1.2266
Additional Allowable Factor	1.062	1.062	1.062
Engineering Hot Channel Factors	-	-	-
Fuel Content Variation	1.030	1.030	1.030
Fuel Thickness/Width Variation	1.150	1.150	1.150
Overall Product	4.116	3.863	3.657
Percentage Enthalpy Rise at Hot Spot	52.3%	74.8%	79.9%
-			
Energy Fraction Generated in Fuel Plate			93%

Note 1: Mesh interval 14 has the highest heat flux but a lower enthalpy rise and is not the limiting point for any of the SLs. In the SL tables, out of the 180 data points, 174 are based on mesh interval 18 being the most limiting. The 85 psia (586 kPa), 400 gpm (1514 lpm) and 180 °F (82.2 °C) point is based on mesh interval 19 being the most limiting due to the higher enthalpy rise. The other 5 points, which are in red, are limited by saturation temperature at the channel exit due to being the lowest flow rate with the highest inlet temperature.

F.6 Method of Performing Safety Limit Calculations

A computer spreadsheet model (Reference 11) was developed to determine the margin to flow instability for any MURR core coolant channel and the outlet saturation temperature. The spreadsheet was used to determine the MURR SLs based on whichever was more limiting of the two previously stated criteria:

1. 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 CHF Correlation at that point; and

2. The coolant exit temperature from the hot channel shall be less than the saturation temperature at the core exit pressure.

Except for conditions of very low flow rates and high water inlet temperature, the first criterion is the more restrictive one.

The spreadsheet is arranged to calculate the reactor power level SL for any combination of the three (3) independent variables: primary coolant flow, reactor inlet water temperature and primary coolant pressurizer pressure.

The computer spreadsheet model does not consider the individual strips of each fuel plate. Instead, the power of each fuel plate is increased by its azimuthal peaking factor. This approach produces bounding values of local bulk coolant temperature because no credit is taken for azimuthal heat conduction in the fuel plates or azimuthal mixing in the coolant channels. The hottest strip of each of the two fuel plates on either side of a coolant channel is located directly across from each other. In the MURR core, a fairly significant fraction, 7%, of the recoverable fission power from the fuel escapes the core as gamma or neutron radiation without creating heat in the fuel plate or in the primary coolant by depositing this 7% of the fission energy in the reflector and pool coolant system and creating the heat at those locations. Therefore, only 93% of the power contributes to the fuel plate heat fluxes heating the primary coolant.

The computer spreadsheet model determines the core power at which the SL criteria of the most limiting axial interval combination of both heat flux and enthalpy rise results in the axial heat flux for that location equal to 0.5 times the Bernath Correlation. In the SL analysis, at each axial level the Bernath CHF value must be compared with the highest heat flux of the two fuel plates that form the coolant channel. In practice this can be accomplished by using the heat fluxes of the plate with the higher plate peaking factor. The CHF is strongly affected by the margin between saturation pressure and local pressure at each mesh point. This difference decreases with increasing coolant temperature and is a significant factor in determining which mesh interval the 0.5 ratio of local heat flux to Bernath CHF occurs. The location will be after the coolant flows past the peak heat flux mesh point and reaches the worst combination of lower heat flux with a reduced margin between mesh interval coolant saturation pressure and fluid pressure. Conservatively, the spreadsheet uses the channel outlet pressure in the Bernath Correlation to calculate the CHF for each mesh point. This results in a calculated smaller margin

between the mesh interval saturation pressure and the coolant pressure, which results in the SL corresponding to a mesh point closer to the peak heat flux point resulting in a lower SL power level.

The spreadsheet includes the calculations used to determine the absolute pressure (psia) at the core outlet for every combination of operating conditions in this study. This uses Table F-5, which is an improved version of the old model provided in the original Table F-3 of Appendix F of Addendum 4 to the MURR HSR. Besides the improved model for predicting pressure drops due to flow in the primary coolant system, the spreadsheet now corrects for the slightly higher elevation of the core outlet in relationship to the water level in the pressurizer, which is assumed to be at the low level SCRAM set point of 16 inches (40.64 cm) below center line for a total elevation difference of 23.25 (59.06) inches (Note: Center line of the pressurizer is at an elevation of 604.5 feet (184.3 m)). The primary coolant charging pump starts charging when pressurizer water level is 6 inches (15.24 cm) below center line. The core outlet pressure is also corrected for the velocity head loss due to the much higher coolant velocity in the core compared to primary coolant loop 'A' where the pressurizer surge line is connected.

To improve the values listed in the original Table F-3, a flow model of the primary coolant system from the point where the pressurizer surge line connects to primary coolant loop 'A' to primary coolant isolation valve 507A (hot leg) was developed using the methodology in Crane Co. Technical Paper No. 410, "Flow of Fluids Through Valves, Fittings, and Pipe" (Reference 12). The modeled portion of the primary coolant system is shown in Figure F.4 and includes the following pressure monitoring instrumentation:

1. PI-916 – pressure gauge for monitoring pressurizer pressure;

2. PT-943 – pressure transmitter for monitoring primary coolant heat exchanger outlet pressure (at the T-junction point combining primary coolant loops 'A' and B;

3. DPS-929 – differential pressure sensor for monitoring differential pressure across the reactor core (between primary coolant isolation valves 507A and 507B); and

4. PT-944 – pressure transmitter for monitoring reactor core outlet pressure (just before primary coolant isolation valve 507A).

There is an elevation difference of approximately 21 feet (6.4 m) in the primary coolant system over this part of the system. The elevation differences are included in the model prediction of the calculated pressure indication. The model also includes changes in the coolant pressure due to changes in velocity as can be represented by the Bernoulli equation.

To validate the modeling and the pressure drop across the fuel elements, the model was checked and refined using the following two different operating conditions: (1) with the reactor operating at 10 MW (March 27, 2011), and (2) with the primary coolant system operating with no fuel elements installed (March 28, 2011). The modeling includes the hydraulic pressure drops and pressure head difference due to elevation in calculating what the model would have as indication at PT-943, PT-944 and DPS-929.

The model calculated values agree with the recorded indication values for the condition of no fuel elements installed measurements conducted on March 28, 2011. In the fuel elements installed model, the pressure drop across the elements was increased by a factor of 1.04 to be conservative. With this adjustment, the model calculated values for the fuel elements installed agreed with PT-943 indication and over predicted DPS-929 indication slightly by 3.2%. The PT-944 indication was predicted to be 34.9 psi (240.6 kPa) compared to the actual reading of 35.0 psi (241.3 kPa).

Table F-5 groups the component pressure drops into four (4) frictional and four (4) non-frictional sets from the pressurizer to the core outlet. The path from the pressurizer to the core outlet is divided into 25 component pressure drops, ranging from 1 to 10 pressure drops per group. Primary coolant loop 'A' and loop B each have the same flow rate of 1825 gpm (6908 lpm), however 50 gpm (189 lpm) of demineralizer flow is diverted from loop 'A' just before the 8-inch/12-inch (20.32-cm/30.48-cm) expander. After loops 'A' and B combine at the T-junction, the combined flow is 3600 gpm (13627 lpm) in the rest of the flow path. Except for the core, the coolant temperature through all components is the inlet temperature. The core coolant temperature is taken to be the core average coolant temperature and is used to calculate the slightly higher average core flow rate as a result of the thermal expansion as indicated in the table. The pressurizer to core outlet pressure drop is the sum of the 25 component pressure drops corrected for elevation difference and coolant velocity between where the pressurizer attaches to loop 'A' and the core coolant channels.

TABLE F-5
REFERENCE PRESSURE DROP DATA

Component Group	**ΔP_0 (PSI) (kPa)**	**Q_0 (GPM) (LPM)**	**T_0 (°F) (°C)**	**Frictional**
1. 8-inch fine mesh strainer before expander (1)	4.5000 (31.0264)	1825 (6908)	120 (48.89)	No
2. Loop 'A' between pressurizer to expander (2-3)	0.0913 (0.6294)	1826 (6912)	120 (48.89)	Yes
3. Loop 'A' 8-inch/12-inch expander (4)	0.2640 (1.8202)	1775 (6719)	120 (48.89)	No
4. Expander through pressure vessel entrance (5-14)	3.3961 (23.4153)	3600 (13627)	120 (48.89)	Yes
5. In pressure vessel up to reactor core (15-19)	1.1590 (7.9910)	3600 (13627)	120 (48.89)	Yes
6. Step changes in pressure vessel (20-21)	0.0439 (0.3027)	3600 (13627)	120 (48.89)	No
7. Entrance into fuel element channels (22-24)	0.7569 (5.2186)	3600 (13627)	120 (48.89)	No
8. Reactor core - 25.5 inches of fuel channel (25)	13.8049 (95.1814)	3608.3 (13568.9)	128.68 (53.71)	Yes

Description of Component Group:

1. 8-inch fine mesh strainer before the expander (1)

2. Primary coolant loop ‘A’ between pressurizer to expander:
   a. Pressurizer surge line to primary coolant loop ‘A’ centerline (2)
   b. 4.5 feet (1.37 m) of 8-inch (20.32-cm) pipe (3)
3. Primary coolant loop ‘A’ 8-inch/12-inch (20.32-cm/30.48-cm) expander (4)
4. Expander through pressure vessel entrance:
   a. 1 foot (0.3 m) of 12-inch (30.48-cm) pipe (5)
   b. T-junction combining primary coolant loop ‘A’ & loop B (6)
   c. 65.5 feet (19.96 m) of 12-inch (30.48-cm) pipe (7)
   d. Two 12-inch (30.48-cm) 90° elbows (8)
   e. 12-inch (30.48-cm) butterfly valve (9)
   f. 31.575 feet (9.62 m) of 12-inch pipe (30.48-cm) (10)
   g. Two 12-inch (30.48-cm) 90° elbows (11)
   h. Three 12-inch (30.48-cm) 45° elbows (12)
   i. 12-inch (30.48-cm) swing check valve (13)
   j. Entrance to pressure vessel annulus (14)
5. In pressure vessel up to reactor core:
   a. Junction of 12-inch (30.48-cm) inlet pipe through fuel element upper end fitting (15)
   b. 17.75-inch (45.09 cm) long annulus (16)
   c. 34.25-inch (87 cm) long annulus (17)
   d. 3.75-inch (9.53 cm) long annulus (18)
   e. Flow in fuel element upper end fitting (19)
6. Step changes in pressure vessel:
   a. Gentle reduction in annulus (20)
   b. Two step changes in annulus (21)
7. Entrance into fuel element channels:
   a. Entrance to fuel element end fitting (22)
   b. Exit from fuel element end fitting (23)
   c. Entrance into fuel element channels (24)
8. Reactor Core - 25.5 inches (64.77 cm) of fuel channel (25)

For a known reactor inlet water temperature, T_0, of 120 °F (48.89 °C) and a known reactor core inlet flow rate (primary coolant flow), Q_0, of 3600 gpm (13627 lpm), known component pressure drops, ΔP_0, are provided, as shown. As Table F-5 indicates, one combined value of ΔP_0 is provided for the pressure drop across each of the eight sets of components. For set number 1, Q_0 is shown in Table F-5 as 1825 gpm (6908 lpm). This flow rate for primary coolant loop ‘A’ is the same for loop B, however 50 gpm (189 lpm) of demineralizer flow is diverted from loop ‘A’ prior to the 8-inch/12-inch (20.32-cm/30.48-cm) expander giving the expander a flow rate of 1775 gpm (6719 lpm). Thus, the total flow rate (or effective Q_0) through the rest of the components is 3600 gpm (13627 lpm). The method for correcting the reference ΔP components depends on the type of pressure drop involved. For non-frictional components, pressure drop is proportional to density and flow as follows:

$$\frac{\Delta P}{\Delta P_0} = \left[\frac{\rho(T)}{\rho(T_0)}\right]^{1.0} \left[\frac{Q}{Q_0}\right]^{2.0} \qquad (2)$$

In this equation, the subscript "0" denotes the reference conditions as provided in Table F-5. For the frictional loss components, the pressure drop is given by:

$$\frac{\Delta P}{\Delta P_0} = \left[\frac{\rho(T)}{\rho(T_0)}\right]^{0.8} \left[\frac{Q}{Q_0}\right]^{1.8} \left[\frac{\mu(T)}{\mu(T_0)}\right]^{0.2} \qquad (3)$$

where ρ is coolant density and μ is coolant dynamic viscosity. Equation (2) is for portions of the flow path that are characterized by non-frictional losses, such as changes in flow area and or direction. Equation (3) is for portions of the path that are characterized by wall friction. Component Groups 1, 3, 6 and 7 are shown in Table F-5 to be non-frictional. The rest, as indicated in the table, are frictional. Both equation (2) and equation (3) can be derived from well-known hydraulic relationships for turbulent flow. In the derivation of equation (3) it is assumed that the friction factor is proportional to the Reynolds number to the −0.2 power.

F.7 Output

The core power levels limited by the criterion of ≤ 50% of the Bernath CHF were the result of iteration in the MURR SL Program. For each set of primary coolant flow, reactor inlet water temperature and primary coolant pressurizer pressure, the program iterated on core power level to determine when the most limiting 1-inch (2.54-cm) axial mesh interval with fuel plate-1 heat flux and the corresponding coolant channel 2 enthalpy rise reached this criterion. The power levels in Tables F-1, F-2 and F-3 correspond to the iteration terminating when the Bernath CHF Correlation value for the most limiting axial mesh interval in the hot channel converges to 0.499999. At all three pressurizer pressures, the lowest flow rate of 400 gpm (1514 lpm) combined with the highest reactor inlet water temperature of 200 °F (93.3 °C) results in the power level being limited by the channel outlet temperature reaching the saturation temperature. This is also the case at 180 °F (82.2 °C) with 400 gpm (1514 lpm) flow for pressurizer pressures of 60 and 75 psia (413.7 and 517.1 kPa). These five points are highlighted in red with the power levels given being limited by the hot stripe exit temperature reaching the saturation temperature for the channel exit pressure.

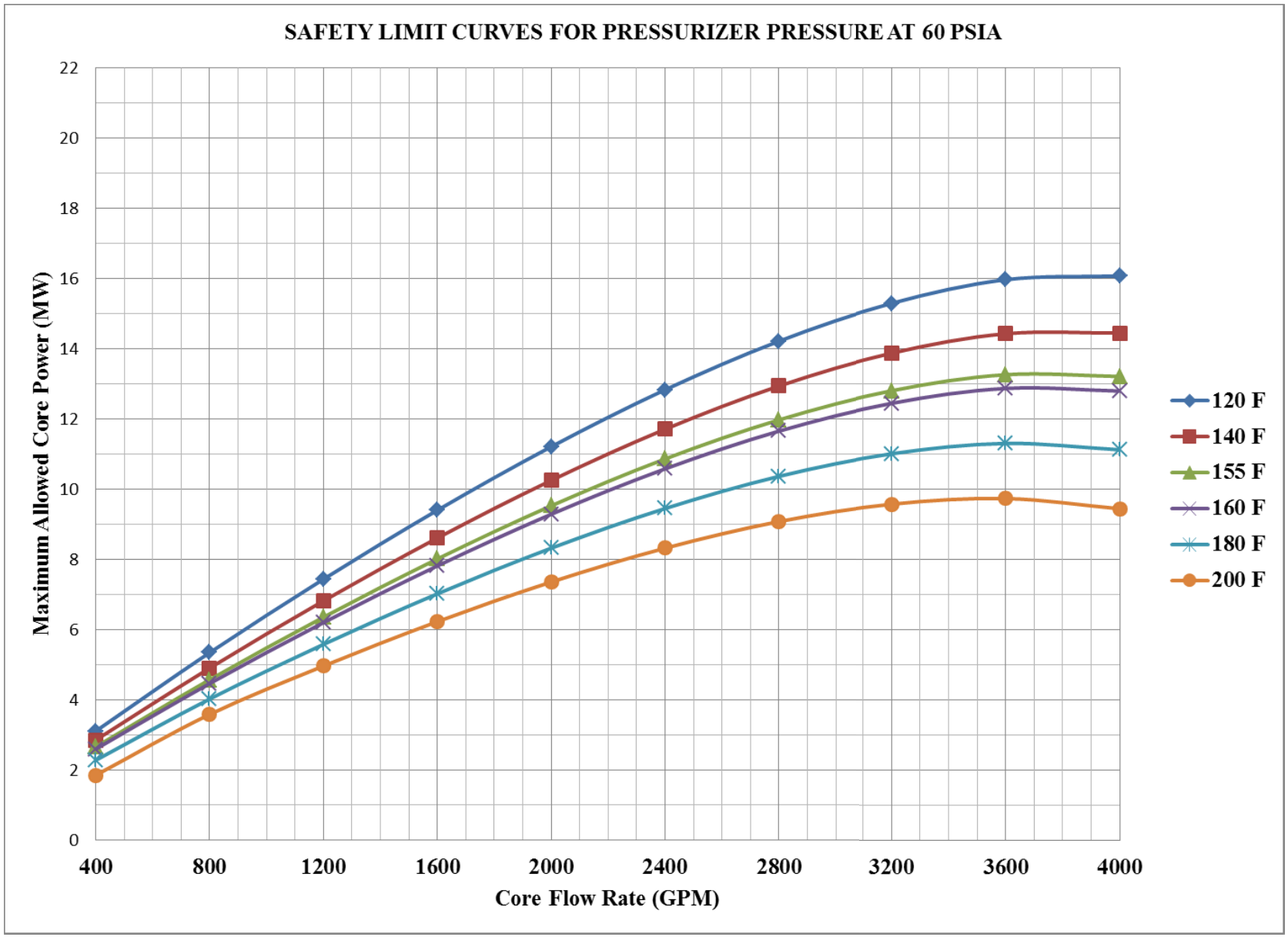


FIGURE F.1 – MURR SAFETY LIMIT CURVES FOR PRESSURIZER PRESSURE AT 60 PSIA

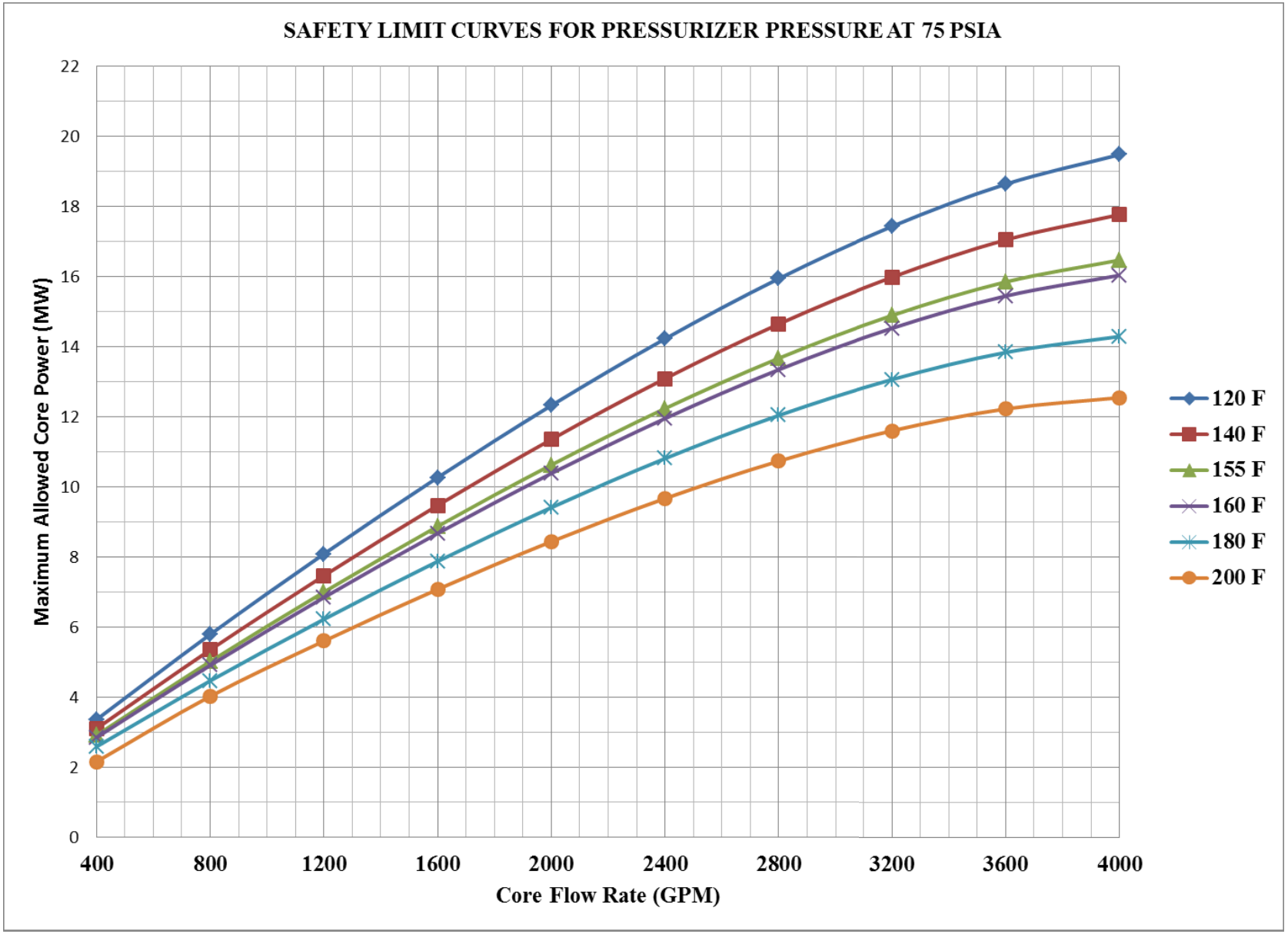


FIGURE F.2 – MURR SAFETY LIMIT CURVES FOR PRESSURIZER PRESSURE AT 75 PSIA

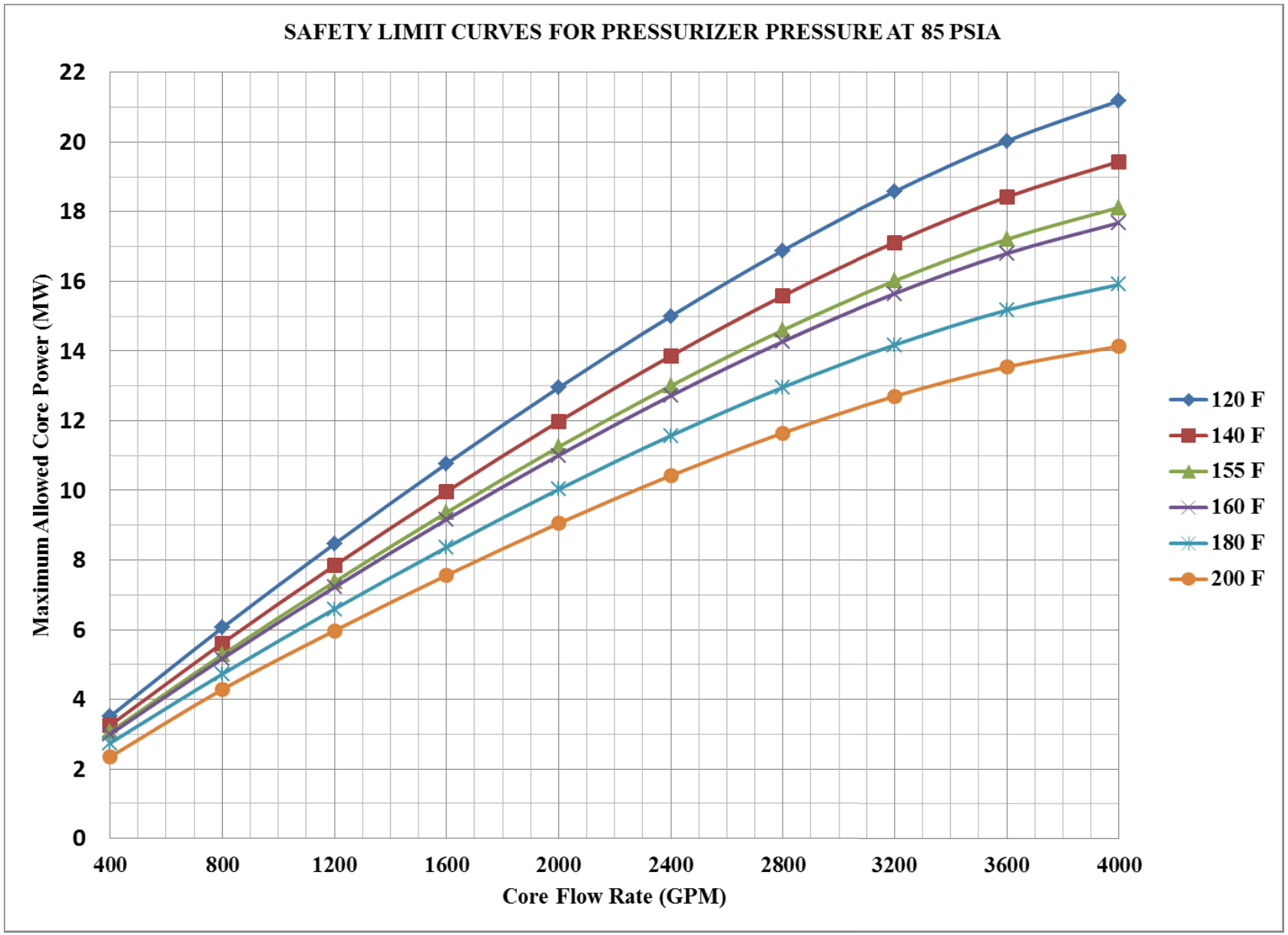


FIGURE F.3 – MURR SAFETY LIMIT CURVES FOR PRESSURIZER PRESSURE AT 85 PSIA

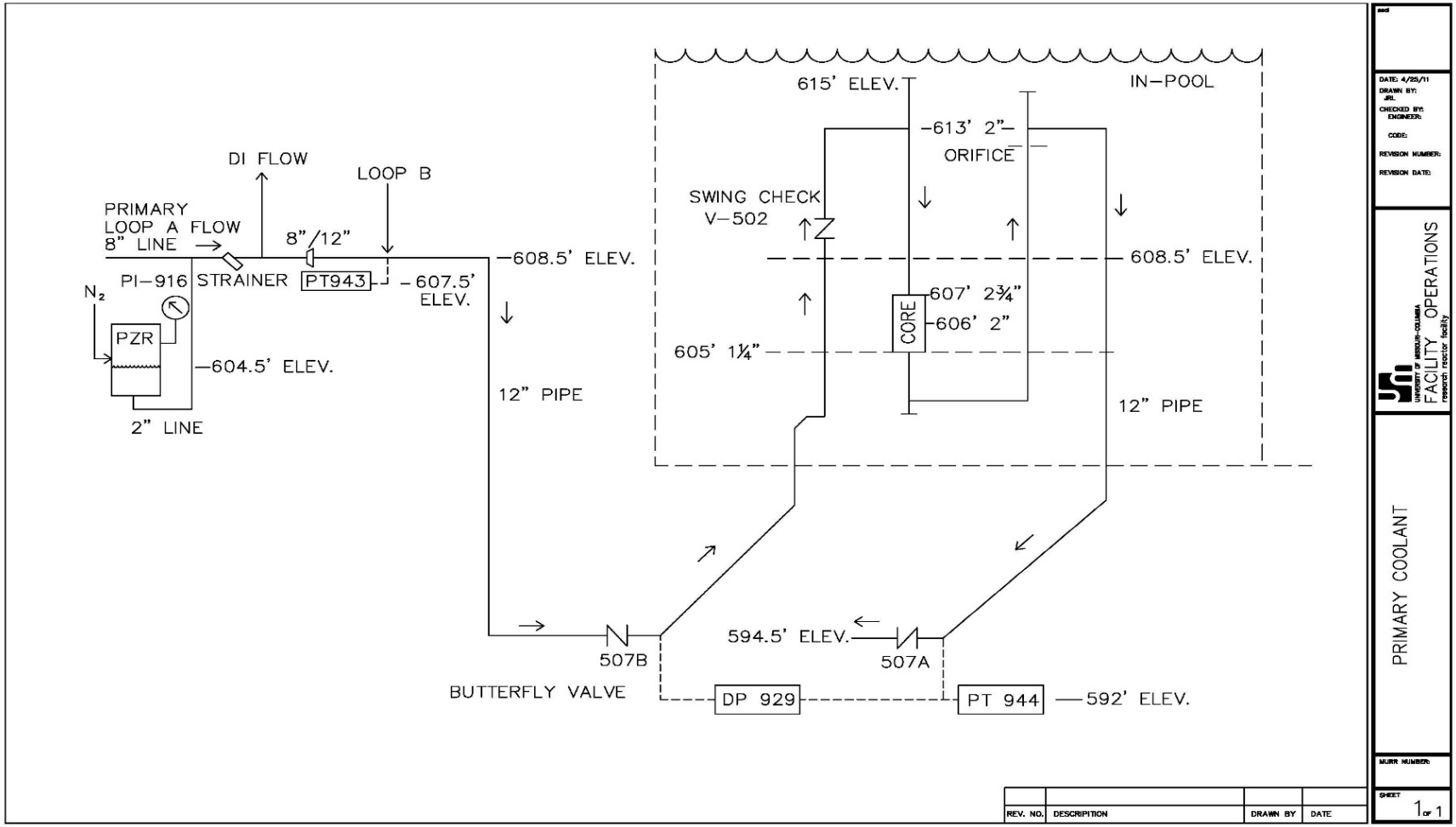


FIGURE F.4 – SECTION OF PRIMARY COOLAND SYSTEM FOR FLOW MODEL

CHAPTER 4

ATTACHMENT B

CURRENT BASES FOR LIMITING SAFETY SYSTEM SETTINGS

APPENDIX H

BASES FOR LIMITING SAFETY SYSTEM SETTINGS FOR MODES I AND II OPERATION

TABLE OF CONTENTS

H.0 BASES FOR LIMITING SAFETY SYSTEM SETTINGS FOR MODES I AND II OPERATION 2

LIST OF FIGURES

Figure H.1 – MURR CHF Curve Mode I Operation for Pressurizer Pressure at 75 PSIA (517 kPA) 4
Figure H.2 – MURR CHF Curve Mode II Operation for Pressurizer Pressure at 75 PSIA (517 KPA) 5

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2. Written communication dated July 30, 2012, regarding responses to “University of Missouri at Columbia – Request for Additional Information Regarding Amendment Application No. 36 [ML12214A310].

3. Written communication dated July 8, 2013, from the Nuclear Regulatory Commission providing the Safety Evaluation Report regarding issuance of Amendment #36 to Amended Facility License N0. R-103.

H.0 BASES FOR LIMITING SAFETY SYSTEM SETTINGS FOR MODES I AND II OPERATION

The Limiting Safety System Settings (LSSSs) for Modes I and II, i.e., 10 MW and 5 MW operation respectively, of MURR are as follows. For reactor power level the LSSS is 125 % of full power for both modes, thus, the highest powers obtainable before a reactor scram would be 12.5 MW (1.25 x 10 MW) in Mode I and 6.25 MW (1.25 x 5 MW) in Mode II. For both modes, the LSSS on pressure is a minimum of 75 psia (517 kPa) in the primary coolant 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 1625 gpm (6151 lpm) in either of the parallel coolant loops. The same LSSS of 1625 gpm (6151 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 cleanup (demineralizer) system, the actual core flow rates at the LSSS are 3200 gpm (12113 lpm) in Mode I and 1575 gpm (5962 lpm) in Mode II, respectively.

The August 2011 Amendment application to the U.S. Nuclear Regulatory Commission to revise the "Safety Limit Analysis for the MURR," presented in Appendix F of Addendum 4 to the MURR Hazards Summary Report, presents parametric curves for the conditions which would lead to reaching the conservatively defined critical heat flux (CHF), corresponding to 0.5 times the Bernath CHF Correlation. From the analysis, Figure H.1 depicts the CHF conditions for the LSSS on a pressurizer pressure of 75 psia (517 kPa). From the curves one can predict the safety margin for several anticipated transients.

Case 1 postulates a severe power transient with primary coolant flow and pressure reduced to their LSSS values in Mode I operation. Figure H.1 predicts that the temperature LSSS of 155 °F (68.3 °C) could not be reached until the power has risen to 14.896 MW, or 2.396 MW above the reactor power scram set point, thus, an ample safety margin exists for safety system reaction time required to prevent reaching the CHF threshold.

Case 2 postulates steady-state Mode I operation of the reactor with flow and pressure again reduced to their LSSS and reactor power at the LSSS of 12.5 MW. Figure H.1 predicts that CHF would not occur until a reactor inlet water temperature of 187.7 °F (86.5 °C) was obtained. The safety margin is thus 32.7 °F (18.2 °C) above the LSSS of 155 °F on reactor inlet water temperature. 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 +/- 2 °F (1.1 °C); therefore, there is excess safety margin for a temperature transient of this type.

Case 3 postulates Mode I operation with pressurizer pressure reduced to the LSSS of 75 psia (517 kPa), reactor power and reactor water inlet temperature raised to their LSSS of 12.5 MW and 155 °F (68.3 °C), respectively. Figure H.1 predicts that the primary coolant flow rate could be reduced to 2469 gpm (9346 lpm) before CHF would occur, implying a safety margin of 731 gpm (3767 lpm) below the LSSS of 3200 gpm (12113 lpm) on coolant flow through the core. Operating history has shown that the true value of primary coolant flow does not vary from the measured value by more than +/- 50 gpm (189 lpm), thus there is excess margin for safety system reaction time to scram the reactor before CHF occurs.

Appendix D to the MURR HSR provides a detailed analysis of the results of the most severe loss of flow accident for MURR.

Consideration of the same transients for Mode II (5 MW) operation yields even greater safety margins. Figure H.2 presents the results for Mode II operation of the same transients discussed for Mode I. Case 1 predicts CHF at 8.766 MW, i.e. 2.516 MW above the LSSS of 6.25 MW for Mode II. Case 2 results indicate that the reactor could be operated with a reactor inlet water temperature of 219 °F (103.9 °C) for Mode II before reaching CHF. Case 3 shows CHF occurring only with total core flow reduced to 1045 gpm (3956 lpm) or 530 gpm (2006 lpm) less than a core flow rate of 1575 gpm (5962 lpm). Thus, the safety margin is 34% of the 1575 gpm (5962 lpm) core flow rate corresponding to the LSSS flow value for Mode II operation as compared to 23% for Mode I operation.

The LSSS for pressurizer pressure is 75 psia (517 kPa); a margin of 17 psi (117 kPa) above the 58 psia (400 kpa) where CHF is predicted to occur in Mode I with reactor power raised to the LSSS and reactor inlet water temperature and flow rate reduced to their LSSS values. For Mode II, there is a 36 psi (248 kPa) margin above the 39 psia (269 kPa) where CHF is predicted to occur. Past operating experience has shown the pressurizer sensors to be accurate within +/- 2 psi (14 kpa). Additionally, there are four independent sensors capable of causing a reactor scram in the event of a loss of pressure transient, thus, there is sufficient margin to ensure that the low-pressure safety limit will not be violated.

Therefore, the LSSSs on the four (4) important parameters of reactor power, primary coolant pressurizer pressure, primary coolant flow rate, and reactor inlet water temperature are easily capable of causing the reactor to scram and prevent a violation of the safety limit envelope.

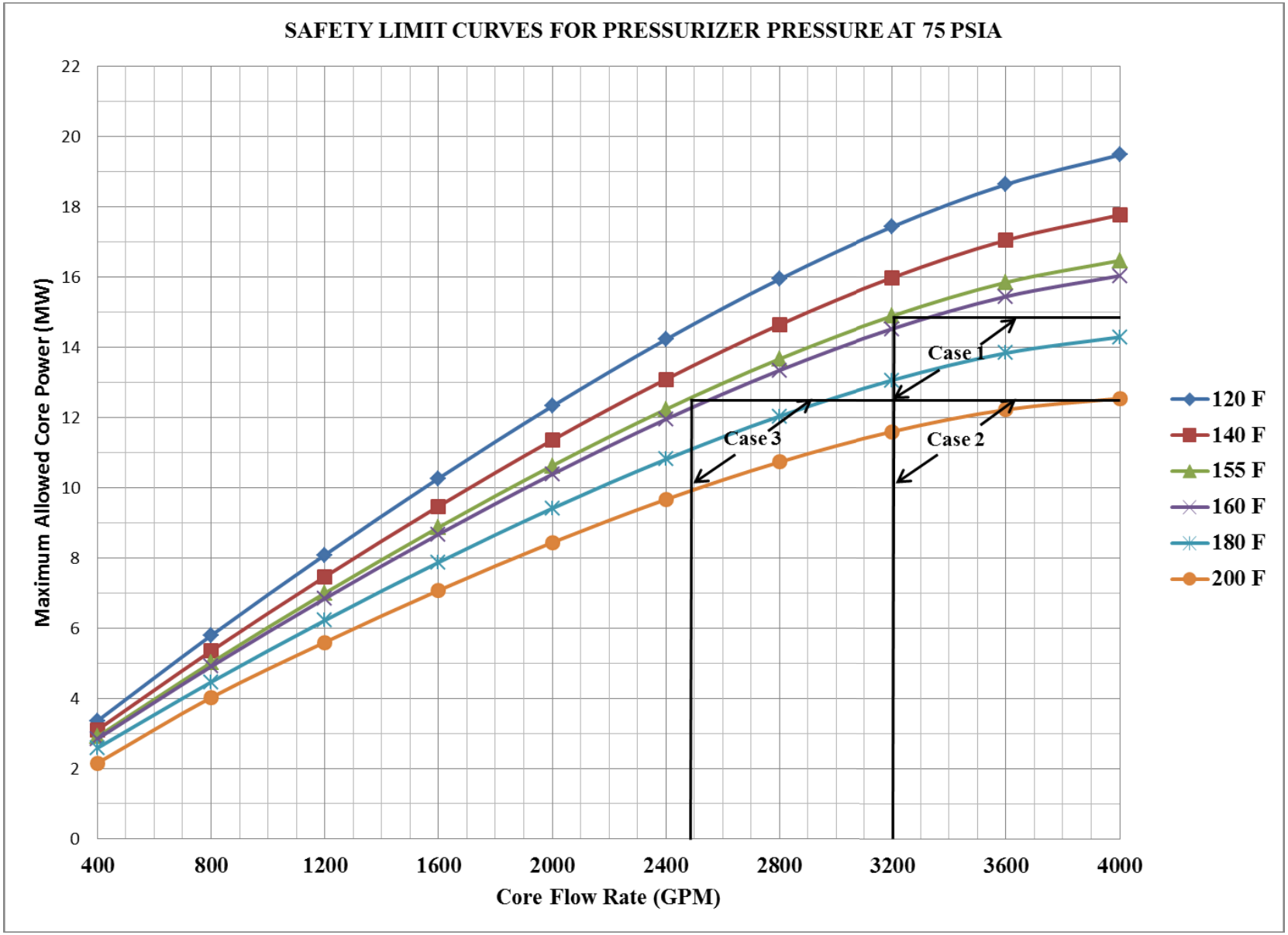


FIGURE H.1 – MURR CHF CURVE MODE I OPERATION FOR PRESSURIZER PRESSURE AT 75 PSIA (517 KPA)

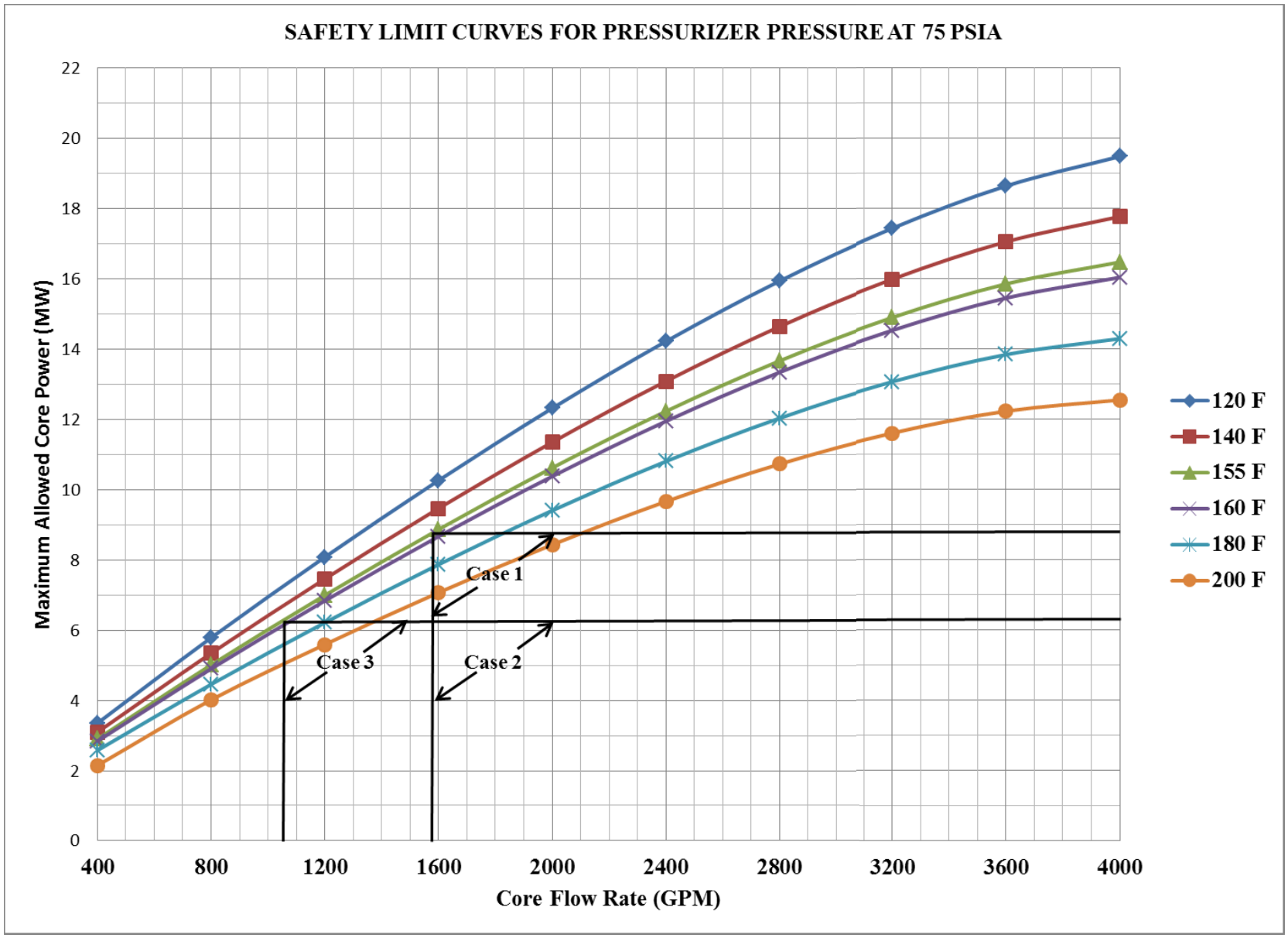


FIGURE H.2 – MURR CHF CURVE MODE II OPERATION FOR PRESSURIZER PRESSURE AT 75 PSIA (517 KPA)

CHAPTER 4

ATTACHMENT C

MURR SAFETY LIMIT ANALYSIS PRIOR TO AMENDMENT #36

TABLE OF CONTENTS

4.C.0 MURR SAFETY LIMIT ANALYSIS PRIOR TO AMENDMENT #36 3
4.C.1 Safety Limit Analysis 9
4.C.2 Limiting Safety System Settings 13
4.C.2.1 Introduction 13
4.C.2.2 Bases 14
4.C.2.3 Conclusions 15

LIST OF TABLES

Table 4-C-1 SUMMARY OF HEAT TRANSFER DATA 5
Table 4-C-2 SUMMARY OF MURR HOT CHANNEL FACTORS 7
Table 4-C-3 REFERENCE PRESSURE DROP DATA* 8
Table 4-C-4 SAFETY LIMITS FOR MURR OPERATIONS 10
Table 4-C-5 MODE I LIMITING SAFETY SYSTEM SETTINGS 13
Table 4-C-6 MODE II LIMITING SAFETY SYSTEM SETTINGS 13
Table 4-C-7 MODE III LIMITING SAFETY SYSTEM SETTINGS 14

LIST OF FIGURES

Figure 4.C.1 – MURR SAFETY LIMIT CURVES (PRESSURIZER AT 60 PSIA (413.7 kPa)) 16
Figure 4.C.2 – MURR SAFETY LIMIT CURVES (PRESSURIZER AT 75 PSIA (517.1 kPa)) 17
Figure 4.C.3 – MURR SAFETY LIMIT CURVES (PRESSURIZER AT 85 PSIA (586.1 kPa)) 18

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4.C.0 MURR SAFETY LIMIT ANALYSIS PRIOR TO AMENDMENT #36

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 (Reference 4.C.1). All data used in the determination of the MURR safety limits was obtained from the MURR Hazards Summary Report (References 4.C.2, 4.C.3, 4.C.4), the MURR Design Data report (Reference 4.C.5), and the MURR hydraulic analysis (Reference 4.C.6). A summary of heat transfer data is provided in Table 4-C-1.

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 (References 4.C.7, 4.C.8). Since the MURR fuel channel length [25.5 inches (64.8 cm)] is about one half that of ATR, the use of ATR test results can, in fact, provide conservatism for the MURR because investigators (Reference 4.C.9) 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 (Reference 4.C.7). Tests were performed at Argonne in 1963 on three channel thicknesses, 0.054, 0.072, and 0.094 inches (0.137, 0.183, and 0.239 cm), 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 (0.239-cm) 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 (Reference 4.C.10):

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

where:

h_{bo} = $10,890 [D_e/(D_e + D_i)] + [(48/D_e^{0.6}) \times V]$;

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

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_t = heated hydraulic diameter, in ft;

V = coolant velocity in fps; and

P = system pressure, in psia.

Subsequent full-scale ATR testing (Reference 4.C.8) 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 the 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-C-1
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 U-235
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	0.86×10^5	1.72×10^5	Btu/ft^2-h
Fuel Assembly Average Outlet Temp.	159.3 (70.7)	159.3 (70.7)	°F (°C)
$\frac{\text{Ave. Power Density in Hot Channel}}{\text{Ave. Core Power Density}}$	2.469[a]	2.469[a]	
$\frac{\text{Max. Power Density in Hot Channel}}{\text{Ave. Power Density in Hot Channel}}$	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	
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:	2.0 (0.6)	4.0 (1.2)	ft/sec (m/sec)
a. Island Region			
b. Unobstructed Test Hole	4.7 (1.4)	9.4 (2.9)	ft/sec (m/sec)
c. Control Blade Gaps	2.2 to 2.9 (0.7 to 0.9)	4.5 to 5.9 (1.4 to 1.8)	ft/sec (m/sec)
d. Reflectors	variable	variable	
Design Temperature Rise in Island	5.8	5	°F

a. This value assumes the worst-case nonuniform core loading.

The BOLERO program was used to perform calculations which determine local conditions of enthalpy, heat flux, and DNB heat flux for the core hot channel (Reference 4.C.11). Table 4-C-2 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 (Reference 4.C.12).

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 delta-p components as given in Reference 4.C.6 to new flow, temperature, and core power conditions (See Table 4-C-3). The new delta-p 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 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.

TABLE 4-C-2
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	1.030
Variation Overall Product	2.724
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	0.3231/0.3505
Area Overall Product	0.813
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	1.150
Variation Overall Product	4.355
Energy Fraction Generated in Fuel Plate	0.930

TABLE 4-C-3
REFERENCE PRESSURE DROP DATA*

Component**	Delta-P_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.C.6

** Component description using notation of Reference 4.C.6

1. Across pressurizer surge line to pressurizer outlet
2. Across 5 feet of 8-inch (20.32-cm) pipe
3. Across 8-inch (20.32-cm) Y-strainer
4. Across 8-inch/12-inch (20.32-cm/30.48-cm) expansion
5. Across 80 feet of 12-inch (30.48-cm) pipe
6. Across four 12-inch (30.48-cm) 90-degree elbows
7. Across three 12-inch (30.48-cm) 45-degree elbows
8. Across one 12-inch (30.48-cm) butterfly valve (507B)
9. Across one 12-inch (30.48-cm) swing check valve (502)
10. Across entrance to annular pressure vessel
11. Across 6 feet (1.8 m) of annular pressure vessel
12. Across entrance to fuel element plates
13. Across Core ... 25.5 inches (64.77 cm) of fuel element plates ... to core exit

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-C-2. 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 (Reference 4.C.5). 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 includes 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.C.1 Safety Limit Analysis

The NUS study as described in Section 4.C.0 resulted in the generation of reactor power limits shown in Table 4-C-4 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.C.1, 4.C.2, and 4.C.3. 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-C-4 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

$$DNBR = (\Phi\ DNB/\Phi\ LOCAL) > 1.0$$

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-C-4
SAFETY LIMITS FOR MURR OPERATIONS

Maximum Allowable Core Power Level (MW) with Pressurizer at 60 PSIA (413.7 kPa)										
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	3.011	5.870	7.980	9.843	11.574	13.099	14.426	15.450	16.217	16.654
140	2.650	5.262	7.299	9.035	10.582	11.960	13.155	14.071	14.729	15.075
160	2.292	4.546	6.675	8.202	9.600	10.822	11.877	12.669	13.228	13.501
180	1.935	3.834	5.667	7.409	8.612	9.685	10.603	11.267	11.715	11.906
200	1.583	3.131	4.615	6.009	7.282	8.400	9.301	9.863	10.204	10.267
Maximum Allowable Core Power Level (MW) with Pressurizer at 75 PSIA (517.1 kPa)										
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	3.278	6.334	8.647	10.742	12.668	14.435	16.050	17.394	18.532	19.438
140	2.916	5.798	7.939	9.906	11.667	13.282	14.746	15.967	16.993	17.787
160	2.556	5.080	7.317	9.067	10.676	12.138	13.458	14.534	15.437	16.139
180	2.197	4.363	6.474	8.236	9.680	10.988	12.152	13.104	13.892	14.467
200	1.843	3.656	5.415	7.099	8.686	9.845	10.868	11.689	12.339	12.810
Maximum Allowable Core Power Level (MW) with Pressurizer at 85 PSIA (586.1 kPa)										
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	3.292	6.584	9.097	11.299	13.336	15.227	16.878	18.356	19.662	20.776
140	2.930	5.860	8.421	10.452	12.326	14.062	15.568	16.920	18.108	19.102
160	2.570	5.139	7.709	9.607	11.326	12.908	14.271	15.488	16.549	17.434
180	2.211	4.421	6.632	8.766	10.319	11.749	12.969	14.051	14.980	15.741
200	1.856	3.712	5.568	7.424	9.280	10.593	11.673	12.617	13.418	14.069

Note: Underlined power levels are limited by bulk boiling.

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-C-4 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 (Reference 4.C.11) and MURRPGM (Reference 4.C.12) 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 (References 4.C.7, 4.C.8) 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 (Reference 4.C.11). 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.C.1, 4.C.2, and 4.C.3 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.C.1, 4.C.2, and 4.C.3. 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.C.1, 4.C.2, and 4.C.3.

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 (Reference 4.C.13). 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 (Reference 4.C.14) was purchased from ANL for the MURR fuel upgrade safety limit analysis. This is a modified version of COBRA-3C/MIT (Reference 4.C.15); 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 (Reference 4.C.16), COBRA-3C/RERTR has been extensively used for steady-state thermal hydraulic analysis for various research reactors.

4.C.2 Limiting Safety System Settings

4.C.2.1 Introduction

Limiting Safety System Settings (LSSS) were established from the results of the safety limit analysis as presented in Section 4.C.1. 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-C-5 and 4-C-6, 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-C-7. 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-C-5
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-C-6
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-C-7
MODE III LIMITING SAFETY SYSTEM SETTINGS

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

4.C.2.2 Bases

The safety limit analysis as described in Section 4.C.1 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.C.2 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.C.2 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:

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

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.C.2 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.C.2 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.C.2.3 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.

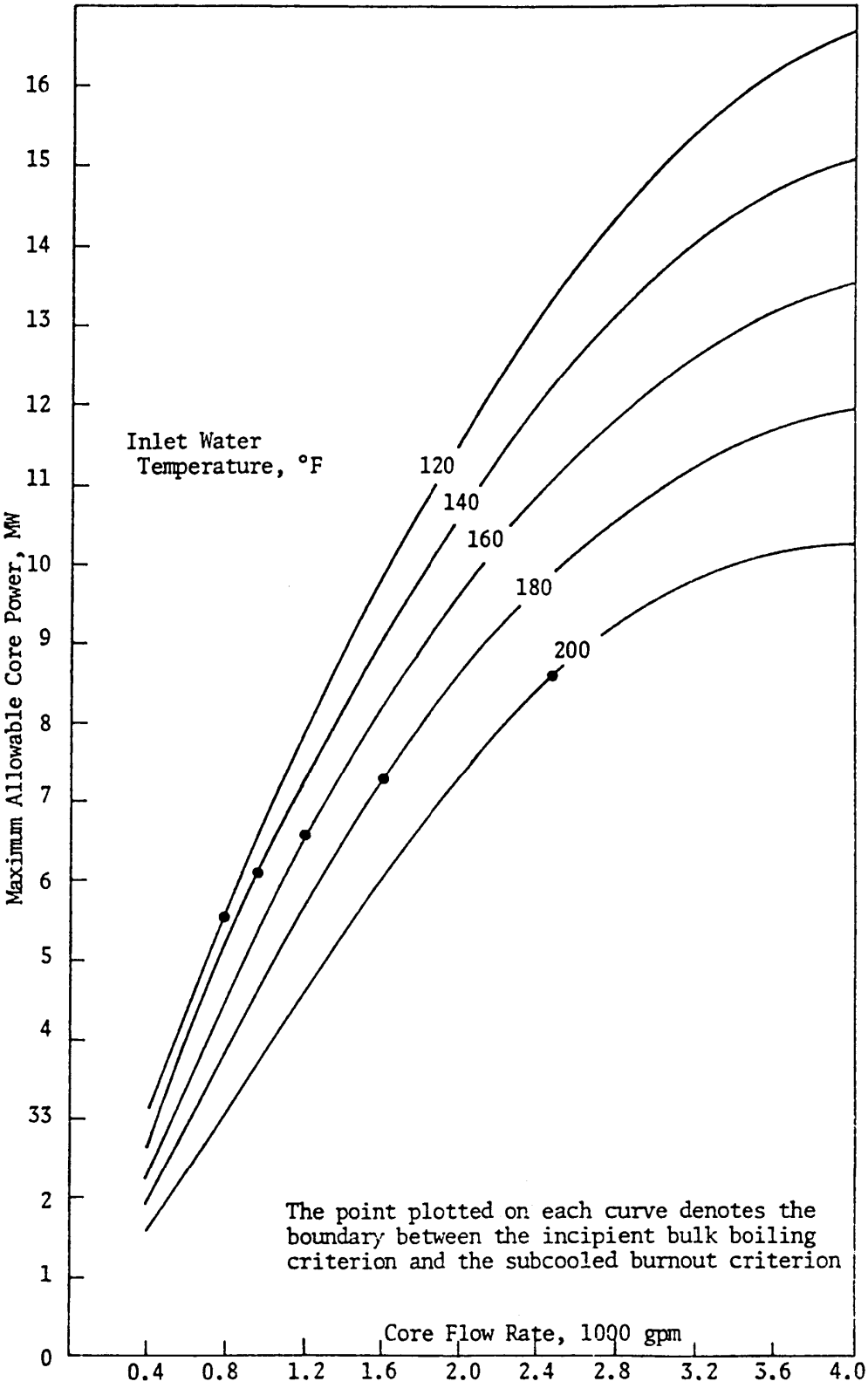


FIGURE 4.C.1 – MURR SAFETY LIMIT CURVES (PRESSURIZER AT 60 PSIA (413.7 KPA))

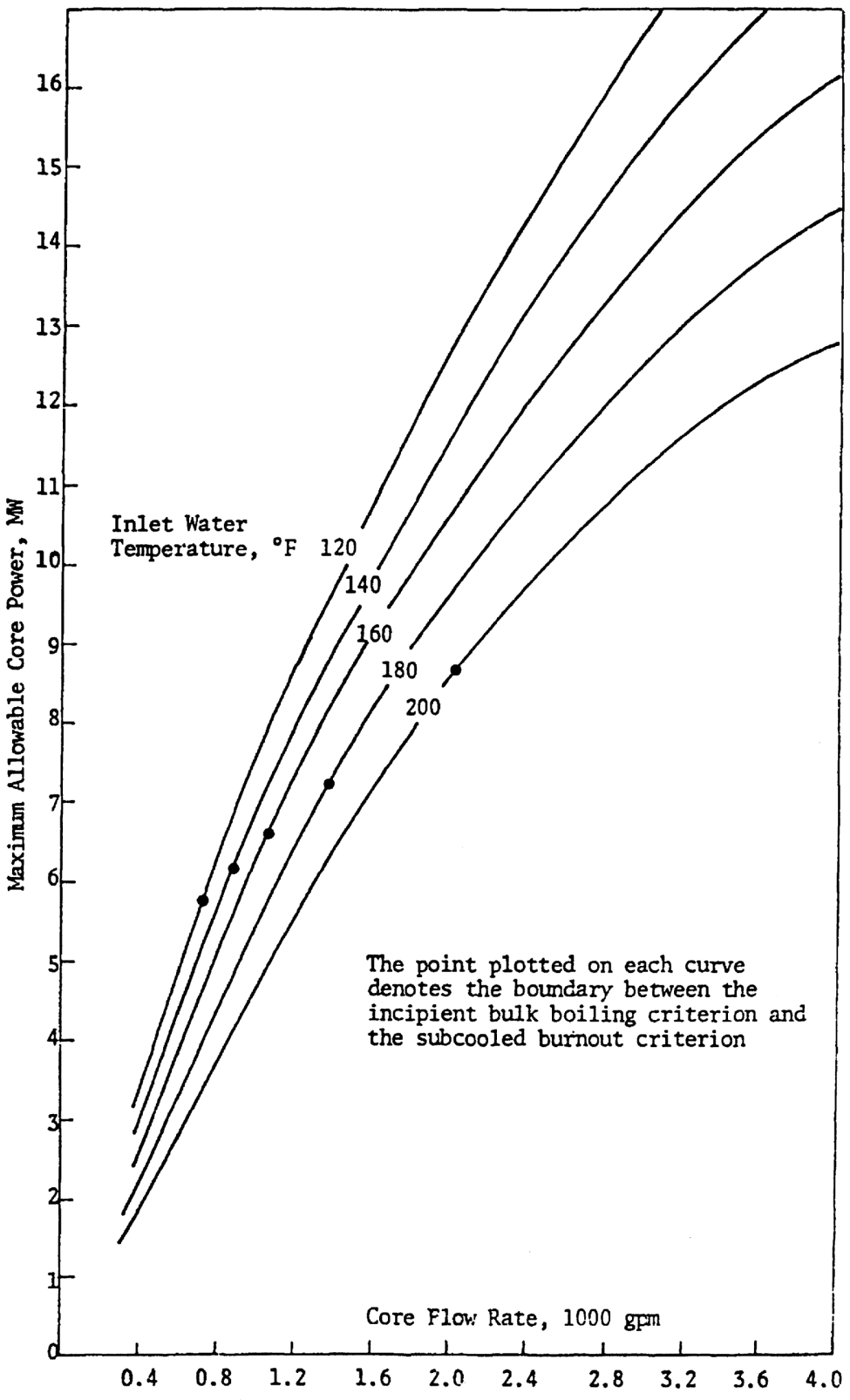


FIGURE 4.C.2 – MURR SAFETY LIMIT CURVES (PRESSURIZER AT 75 PSIA (517.1 KPA))

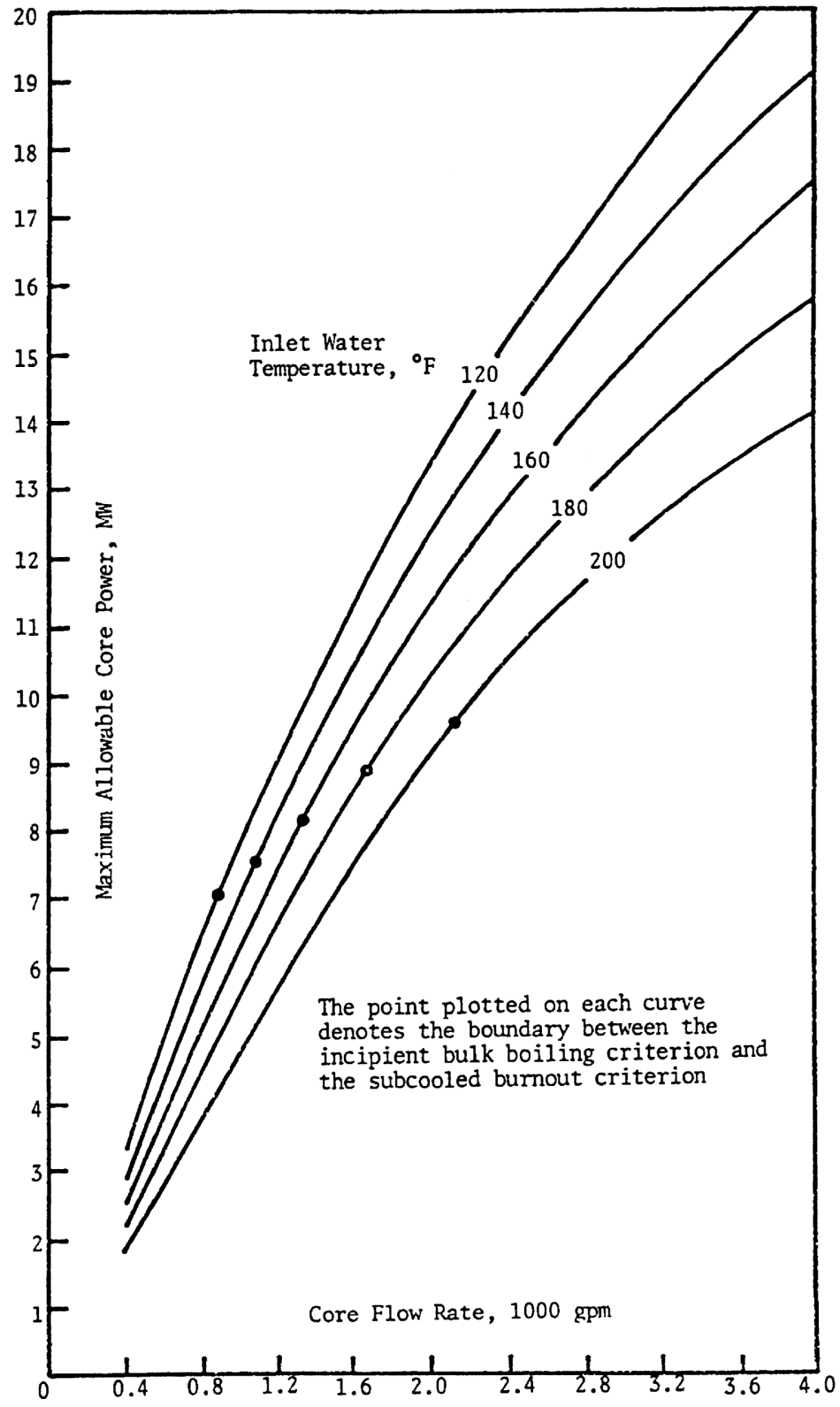


FIGURE 4.C.3 – MURR SAFETY LIMIT CURVES (PRESSURIZER AT 85 PSIA (586.1 KPA))

CHAPTER 5

REACTOR COOLANT SYSTEMS

TABLE OF CONTENTS

5.0 REACTOR COOLANT SYSTEMS 4
5.1 Introduction 4
5.2 Primary Coolant System 4
5.2.1 Introduction 4
5.2.2 General Operating Conditions 5
5.2.3 Circulation Pumps 5
5.2.4 Heat Exchangers 5
5.2.5 Isolation Valves 6
5.2.6 Pressurizer System 6
5.2.7 Reactor Convective Cooling Loop 6
5.2.8 Anti-Siphon System 6
5.2.9 Fuel Element Failure Monitoring System 6
5.2.10 Primary Coolant Demineralizer Loop 7
5.2.11 Primary Coolant System Water Quality 7
5.2.12 Primary Coolant System Technical Specifications 7
5.3 Pool Coolant System 8
5.3.1 Introduction 8
5.3.2 Circulation Pumps 8
5.3.4 Heat Exchanger 8
5.3.5 Isolation Valve 9
5.3.6 Reflector Plenum Natural Convection Valve 9
5.3.7 Pool Water Hold-Up Tank 9
5.3.8 Pool Water Return Diffuser 9
5.3.9 Pool Coolant Demineralizer Loop 9
5.3.10 Pool Coolant System Water Quality 10
5.3.11 Pool Coolant System Technical Specifications 10
5.4 Secondary Coolant System 11
5.4.1 Introduction 11
5.4.2 Circulation Pumps 11
5.4.3 Heat Exchangers 11
5.4.5 Automatic Temperature Control Valves 11
5.4.6 Water Treatment System 11
5.4.7 Instrumentation 12
5.4.8 Cooling Tower 12
5.4.9 Secondary Coolant Monitoring System 12
5.4.10 Secondary Coolant System Technical Specifications 12
5.5 Reactor Coolant Cleanup System 13
5.5.1 Introduction 13

5.5.2 Demineralizers 13
5.5.3 Filters 13
5.5.4 Primary Coolant Demineralizer Loop 13
5.5.5 Pool Coolant Demineralizer Loop 14
5.5.6 General Description 14
5.5.7 Description of Operation 15
5.5.8 Instrumentation 15
5.5.9 Reactor Coolant Cleanup System Technical Specifications 15
5.6 Primary Coolant Make-Up Water System 15
5.6.1 Introduction 15
5.6.2 Pressurizer Tank 16
5.6.3 Primary Coolant Charging Pump 16
5.6.4 Automatic Control Valves 16
5.6.5 Nitrogen Supply System 16
5.6.6 Primary Grade Water Supply System 16
5.6.7 Description of Operation 17
5.7 Nitrogen-16 and Oxygen-19 Control System 17
5.7.1 Introduction 17
5.7.2 Primary Coolant Demineralizer Loop Hold-Up Tank 17
5.7.3 Pool Coolant Hold-Up Tank 18
5.7.4 Description of Operation 18
5.8 Decay Heat Removal System 18
5.8.1 Introduction 18
5.8.2 In-Pool Heat Exchanger 18
5.8.3 Automatic Isolation Valves 19
5.8.4 Description of Operation 19
5.9 Radiolysis of Oxygen and Hydrogen 19
5.9.1 Introduction 19
5.9.2 Radiolysis Potential at MURR 20

LIST OF TABLES

Table 5-1 MURR POOL AND PRIMARY COOLANT SYSTEM WATER QUALITY (2024 AVERAGE AND STANDARD DEVIATION FROM WEEKLY DATA) 20
Table 5-2 HYDROGEN (H2) AND OXYGEN (O2) GAS SOLUBILITY IN MURR POOL AND PRIMARY WATER AT ATMOSPHERIC PRESSURE 21

LIST OF FIGURES

Figure 5.1 – PIPING AND INSTRUMENT DIAGRAM 23
Figure 5.2 – SECONDARY SYSTEM DIAGRAM 24

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

- Mode I - At power levels of up to 10 MW with the primary coolant system pressurized and at a flow rate of approximately 3,800 gpm (14,384 lpm), and a pool coolant flow rate of approximately 1,200 gpm (4,542 lpm); used when all heat exchange and pumping capacity is available;
- Mode II - At power levels of up to 5 MW with the primary coolant system pressurized and at a flow rate of approximately 1,900 gpm (7,192 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
- 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 primary and pool coolant systems and the instrumentation that accommodates these three operating modes are shown in Figure 5.1. The secondary system is shown in Figure 5.2.

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.

All major components of the reactor coolant systems in contact with pool or primary water shall be constructed principally of aluminum alloys or stainless steel.

5.2 Primary Coolant System

5.2.1 Introduction

The primary coolant system consists of the reactor pressure vessels, two (2) main circulating pumps, two (2) heat exchangers, two (2) 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.

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 1,900 gpm (7,192 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 3,800 gpm (14,384 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 (93 kW) drive units through flexible couplings. The pump and driving unit are mounted on a common base. One pump will provide approximately 1,900 gpm (7,192 lpm) to the reactor, and two pumps will supply 3,800 gpm (14,384 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 water-to-water plate-type with all surfaces in contact with the primary coolant constructed of stainless steel. 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 16.9×10^6 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.

The design of the plate-type primary coolant heat exchangers makes a primary-to-secondary coolant leak extremely unlikely. The plate design is such that the most likely leak path is past a plate-to-plate gasket and into the mechanical equipment room (Room 114). To leak into the adjacent system, a path would have to be created through one of the plates, which is minimized by a design that has no flow stagnation points. However, should such a leak occur in the primary coolant heat exchangers, a primary-to-secondary leak would be detected by the secondary coolant monitor located in the coolant leg leaving the heat exchangers. The secondary coolant monitor is discussed in further detail in Chapter 7, "Instrumentation and Control Systems."

5.2.5 Isolation Valves

Two (2) 12-inch (30.48-cm), 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 (594 kPa) for both 1,900- and 3,800-gpm (7,192- and 14,384-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.

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, and is discussed in greater detail in Section 5.8.

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. This 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. This 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.

5.2.11 Primary Coolant System Water Quality

MURR preventative maintenance surveillance system, sampling and analysis procedures, and conductivity and pH limits are more than sufficient to meet the intent of ANSI/ANS 15.1 regarding water chemistry requirements for the primary coolant system and ensures an acceptable level of corrosion control to essential components.

The primary coolant system is sampled weekly as required by preventative maintenance surveillance check R4-W1, "Primary Water Analysis," and in accordance with operating procedure OP-RO-531, "Primary and Pool Sample Station."

The primary coolant sample is then analyzed in accordance with operating procedure OP-RO-420, "Primary and Pool Water Analysis." The analysis includes measuring pH and conductivity, and performing a radio-chemical analysis for iodine-131 and other radioisotopes using a High Purity Germanium (HPGe) detector. A monthly tritium analysis is also performed using a Liquid Scintillation Counter (LSC).

In addition to the weekly sample, primary coolant conductivity is measured continuously with an online monitoring system. The monitoring system remote reading is displayed in the Control Room and its value recorded every two hours by the Operators. Primary coolant conductivity and pH are maintained at less than 2.0 µS and in a range between 5.0 and 6.0, respectively, by a water purification loop consisting of filters and a mixed bed resin demineralizer. This system is described in greater detail in Section 5.5.

5.2.12 Primary Coolant System Technical Specifications

The objective of the Limiting Conditions for Operations (LCO) technical specifications on the reactor coolant systems are to protect the integrity of the reactor fuel and to prevent the release of fission product isotopes. The following is a list of the associated LCOs for the primary coolant system:

3.3.a; 3.3.b; 3.3.c; 3.3.e; 3.3.f; 3.3.g; 3.3.h; 3.5.b.1; 3.5.b.2; and 3.9.a.

The objective of the surveillance technical specifications on the reactor coolant systems are to reasonably assure proper operation of the primary coolant system. The following is a list of the associated surveillance technical specifications for the primary coolant system:

4.3.a; 4.3.b; 4.3.c; 4.3.e; 4.3.f; 4.3.h; and 4.9.a.

The objective of the design features technical specifications on the reactor coolant systems are to ensure proper coolant for safe operation. The following is a list of the associated design features technical specifications for the primary coolant system:

5.2.a; 5.2.c; 5.2.e; 5.2.f; 5.2.g; 5.2.h; 5.2.i; 5.2.j; and 5.2.k.

5.3 Pool Coolant System

5.3.1 Introduction

The pool coolant system consists of two (2) 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 3.6×10^6 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 (45 kW) drive units through flexible couplings. The pumps are designed to overcome system pressure drops at variable flow rates to a maximum flow of 1,200 gpm (4,452 lpm) with one pump operating and 1,400 gpm (5,299 lpm) when both pumps are operating. The pool coolant circulation pumps are designated as P508A and P508B.

5.3.4 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.

The design of the plate-type pool coolant heat exchangers makes a pool-to-secondary coolant leak extremely unlikely. The plate design is such that the most likely leak path is past a plate-to-plate gasket and into the mechanical equipment room (Room 114). To leak into the adjacent system, a path would have to be created through one of the plates, which is minimized by a design that has no flow stagnation points. In the event of this leak path in the pool coolant heat exchanger, pressures are higher on the secondary coolant side under all operating conditions. This would result in a secondary-to-pool coolant

leak and be detected by Operations Staff during sample handling evolutions or while on their periodic routine patrols.

5.3.5 Isolation Valve

The pool coolant system outlet line contains a 6-inch (15.24-cm), 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.

5.3.6 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 ensure 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.

5.3.7 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 and is described in greater detail in Section 5.7.

5.3.8 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 5/8-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.

5.3.9 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.

5.3.10 Pool Coolant System Water Quality

MURR preventative maintenance surveillance system, sampling and analysis procedures, and conductivity and pH limits are more than sufficient to meet the intent of ANSI/ANS 15.1 regarding water chemistry requirements for the pool coolant system and ensures an acceptable level of corrosion control to essential components.

The pool coolant system is sampled weekly as required by preventative maintenance surveillance check P5-W1, "Pool Water Analysis," and in accordance with operating procedure OP-RO-531, "Primary and Pool Sample Station."

The pool coolant sample is then analyzed in accordance with operating procedure OP-RO-420, "Primary and Pool Water Analysis." The analysis includes measuring pH and conductivity and performing a radio-chemical analysis for iodine-131 and other radioisotopes using a High Purity Germanium (HPGe) detector. A monthly tritium analysis is also performed using a Liquid Scintillation Counter (LSC).

In addition to the weekly sample, pool coolant conductivity is measured continuously with an online monitoring system. The monitoring system remote reading is displayed in the Control Room and its value recorded every two hours by the Operators. Pool coolant conductivity and pH are maintained at less than 2.0 µS and in a range between 5.0 and 6.5, respectively, by a water purification loop consisting of filters and a mixed bed resin demineralizer. This system is described in greater detail in Section 5.5.

5.3.11 Pool Coolant System Technical Specifications

The objective of the LCO technical specifications on the pool coolant systems are to protect the integrity of the reactor fuel and to prevent the release of fission product isotopes. The following is a list of the associated LCOs for the pool coolant system:

3.3.d; 3.3.i; and 3.5.b.3.

The objective of the surveillance technical specifications on the reactor coolant systems are to reasonably assure proper operation of the pool coolant system. The following is a list of the associated surveillance technical specifications for the pool coolant system:

4.3.d and 4.3.g.

The objective of the design features technical specifications on the reactor coolant systems are to assure proper coolant for safe operation. The following is a list of the associated design features technical specifications for the pool coolant system:

5.2.a; 5.2.d; 5.2.e; 5.2.f; and 5.2.g.

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 system consists of three (3) circulation pumps, a pool coolant heat exchanger, two (2) primary coolant heat exchangers, two (2) 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 (112-kW), variable speed, centrifugal 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. The secondary coolant circulating pumps are designated SP-1, SP-2, and SP-3.

5.4.3 Heat Exchangers

The secondary water flow makes a single pass on the opposite plate side of the primary coolant in the primary coolant system heat exchangers and also on the opposite plate side of the pool coolant in the pool coolant system heat exchanger. In both the primary and pool heat exchangers, the secondary water flows in a crossflow configuration to the process water being cooled. The primary and pool coolant system heat exchangers are further described in Sections 5.2.4 and 5.3.3.

5.4.5 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.

5.4.6 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 volumetric pump feeds concentrated sulfuric acid into the secondary coolant system to maintain pH within specification. Conductivity is

maintained by the removal of water from the secondary coolant 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.

5.4.7 Instrumentation

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

5.4.8 Cooling Tower

The cooling tower is a stainless steel, induced-draft, crossflow type, with three cells and two-speed fan assemblies for each cell. The tower is designed to cool 5,400 gallons (20,440 l) of water per minute to a temperature of 85 °F (29.4 °C) from an initial temperature of 115 °F (46.1 °C) at a maximum wet bulb temperature of 80 °F (26.7 °C). Vibration switches are mounted on each fan assembly to secure the associated fan motor to prevent damage to the fan or cooling tower structure should an imbalance condition develop. The number of fans and fan speed is configured as required to provide sufficient cooling for 10-MW operation.

5.4.9 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 leaving the heat exchangers. It is operational and responsive over a wide range of energies which allows it to detect any activity present in the secondary coolant. The detector is located in the secondary coolant line such that it views an effective sample volume of 7.5 liters moving past the detector. 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.”

5.4.10 Secondary Coolant System Technical Specifications

The objective of the LCO technical specifications on the reactor coolant systems are to protect the integrity of the reactor fuel and to prevent the release of fission product isotopes. The following is a list of the associated LCOs for the secondary coolant system:

3.3.e; 3.3.f; and 5.2.b

The objective of the surveillance technical specifications on the reactor coolant systems are to reasonably assure proper operation of the primary coolant system. Technical specification 4.3.e provides this assurance for the secondary coolant system.

The objective of the design features technical specifications on the reactor coolant systems are to ensure proper coolant for safe operation. Technical specification 5.2.b provides this assurance for the secondary coolant system.

5.5 Reactor Coolant Cleanup System

5.5.1 Introduction

There are two (2) 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 main purposes of the cleanup system are to reduce the inventory of radioactive nuclides present in the coolant and 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 (3) stainless steel demineralizer tanks sized to hold up to 25.6 cubic feet (0.72 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 head losses, 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 experiences a five-minute delay in the hold-up tank, 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, reducing the mixing of bulk pool water near the surface and reducing 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 would be detected either by the liquid waste disposal system (as described in Chapter 9, “Auxiliary Systems”) and/or Operations Staff while on their periodic routine patrols. This would lead to an investigation of the potential source of the water and subsequent radiochemical analysis. This coolant would be sent to the Liquid Waste Disposal System where it will be analyzed prior to release into the sanitary sewage system. Any personnel exposed to these liquids would be no different from what is encountered during much of the day-to-day operation of the reactor. Therefore, any malfunction 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 and reduces 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.

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 previously 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 five 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 Reactor Coolant Cleanup System Technical Specifications

The objective of the design features technical specifications on the reactor coolant systems are to ensure proper coolant for safe operation. Technical specification 5.2.f provides this assurance for the reactor coolant cleanup system.

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), 3/4-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 heat up 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:

- Nitrogen Addition Valve: V526;
- Nitrogen Vent Valve: V545;
- Water Addition Valve: V527B;
- Water Drain Valve: V527A;
- Primary System Drain Valve: V527D; and
- 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 (2) 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, lowering system pressure.

Water level is maintained in the pressurizer by the primary coolant charging pump, P533. If the 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 the 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 and Oxygen-19 Control System

5.7.1 Introduction

Radiation levels due to nitrogen-16 (N-16) and oxygen-19 (O-19) activity are reduced by the use of hold-up tanks in both the primary coolant demineralizer loop and the pool coolant system. N-16, 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. O-19, beta and gamma emitter with a half-life of 26.9 seconds, is also produced when the coolant water is irradiated with thermal neutrons. 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 N-16 and O-19, to decay, thereby reducing radiation levels and exposure to personnel. The design of the nitrogen-16 and oxygen-19 control system is an integral part of the coolant systems. The accident scenarios, as discussed in Chapter 13, "Accident Analyses," 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 (1034 kPa 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 1/4-inch (6.35-mm) thick aluminum with a design pressure of 50 psig (0.34 MPa above atmosphere). The tank contains seven internal 1/4-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 (2) 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 (2) 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 Radiolysis of Oxygen and Hydrogen

5.9.1 Introduction

In radiolysis of pure water, ionizing radiation produces hydrogen atoms (H) and hydroxyl radicals by reaction 1 and hydrated electrons (e^-_{aq}) by reaction 2.

[1] $HOH \rightarrow H + OH$

[2] $2HOH \rightarrow H_3O^+ + OH + e^-_{aq}$

Comparison of the reaction rates of H and e^-_{aq} with hydrogen peroxide (a stable product of hydrolysis) for reactions 3 and 4, results in a ratio of rate constants $k_{e^-_{aq}}(H_2O_2)/k_H(H_2O_2) = 500$ demonstrating the hydrated electron to be a much stronger reducing agent than hydrogen atoms.

[3] $H + H_2O_2 \rightarrow H_2O + OH$

[4] $e^-_{aq} + H_2O_2 \rightarrow OH^- + OH$

Depending on the type of ionizing radiation (alpha, beta and/or gamma), the purity of the water, and, to a lesser extent, the pH and temperature, numerous secondary free-radical reactions take place. Thirty-two (32) such reactions have recently been identified by Ershov and Gordeev (Reference 5.5).

The stable product signatures resulting from the radiolysis of water are hydrogen gas ($H_{2(g)}$) and hydrogen peroxide (H_2O_2) and their respective g-values[3] at 54 °C are 0.5 and 0.6. The low g-values for these products are the result of the fast recombination rates of the precursor radicals on timescales of picoseconds to microseconds in water.

5.9.2 Radiolysis Potential at MURR

MURR operates at 10 MW approximately 150 hours per week. Consequently, the total gamma-ray activity resulting from neutron capture, fission products and activation products is substantial. Any resulting γ-radiolysis would be maximized in the reactor core and fuel storage areas in the reactor pool. In both locations, the water is continuously purified via ion exchange maintaining a quality standard for purified water that meets the applicable standards of ISO 3696 (Reference 5.6). The water quality data for the MURR pool and primary coolant systems are given in Table 5-1.

TABLE 5-11
MURR POOL AND PRIMARY COOLANT SYSTEM WATER QUALITY
(2024 AVERAGE AND STANDARD DEVIATION FROM WEEKLY DATA)

System	pH	Conductivity (µS/cm)
Pool	6.0 ± 0.4	1.7 ± 0.7
Primary	5.8 ± 0.2	1.0 ± 0.3

The eight (8) element reactor core is cooled by a closed circulating water system (primary coolant system) containing approximately 2,000 gallons (7,571 l) of deionized water with a flow rate of approximately 3,750 gpm (14,195 lpm). At 10 MW, the reactor inlet and outlet temperatures are 120 °F (49 °C) and 136 °F (58 °C), respectively, giving an average operating temperature of 128 °F (54 °C). The primary coolant system also includes two (2) heat exchangers, an anti-siphon tank, and a vent tank having a volume of 3,391 in^3. During reactor operation, the vent tank contains water, water vapor, air and any collected gases resulting from radiolysis.

The fuel storage areas can accommodate up to 88 irradiated fuel elements at various stages in the fuel cycle. These elements are cooled by the pool water moving through the fuel storage areas by pool circulation and convection. The pool water volume is 28,000 gallons (105,992 l) and is pumped at a flow rate of approximately 1,200 gpm (5,542 lpm). During 10 MW operation, the pool water temperature is 100 °F (38 °C).

Under routine MURR operating conditions, radiolysis of water in the pool and primary coolant systems will occur; however, given the high water quality, recombination of the initial radiolysis species (H

atoms, hydroxyl radicals and hydrated electrons) occurs on picosecond timescales resulting in negligible production of the stable radiolysis products (H_2, H_2O_2 and O_2). Observation of hydrogen (H_2) or oxygen (O_2) gases in the reactor pool or primary coolant system vent tank requires these gases to be present at concentrations exceeding their solubility at the respective temperatures, 38 °C (pool water) and 54 °C (primary water). These solubility values are summarized in Table 5-2.

TABLE 5-22
HYDROGEN (H2) AND OXYGEN (O2) GAS SOLUBILITY
IN MURR POOL AND PRIMARY WATER AT ATMOSPHERIC PRESSURE

	Pool Water at 38 °C		**Primary Water at 54 °C**	
Gas	g/L	g/106,000 L	g/L	g/7571 L
$H_{2(g)}$	0.0014	148.4	0.0012	9.1
$O_{2(g)}$	0.0066	699.6	0.0054	40.9

Supporting evidence exists for the conclusion that radiolysis products are negligible in the MURR pool and primary coolant system based on unobserved hydrogen gas (H_2), a signature radiolysis product.

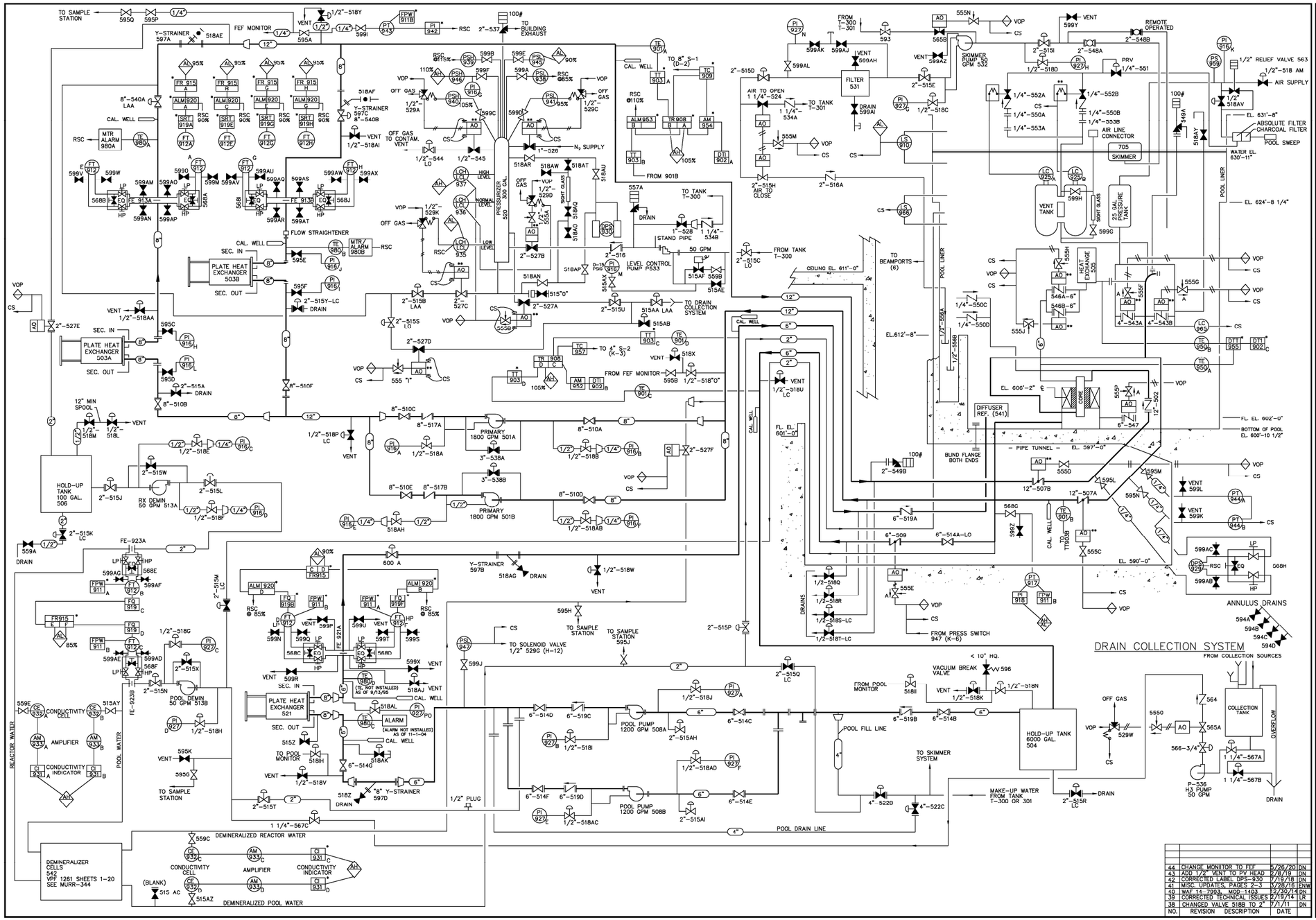
If radiolysis without recombination, in excess of negligible quantities, were to occur on the surfaces of the irradiated fuel elements stored in the pool, hydrogen gas would be constantly produced beyond its solubility limit and ultimately emanate as a gas, rise to the pool surface and be exhausted from the containment building by the pool-sweep system. However, the routine emanation (continuous or intermittent) of gas from the irradiated fuel elements stored in the MURR pool is NOT observed supporting the conclusion that radiolysis, without recombination, of MURR pool water is negligible.

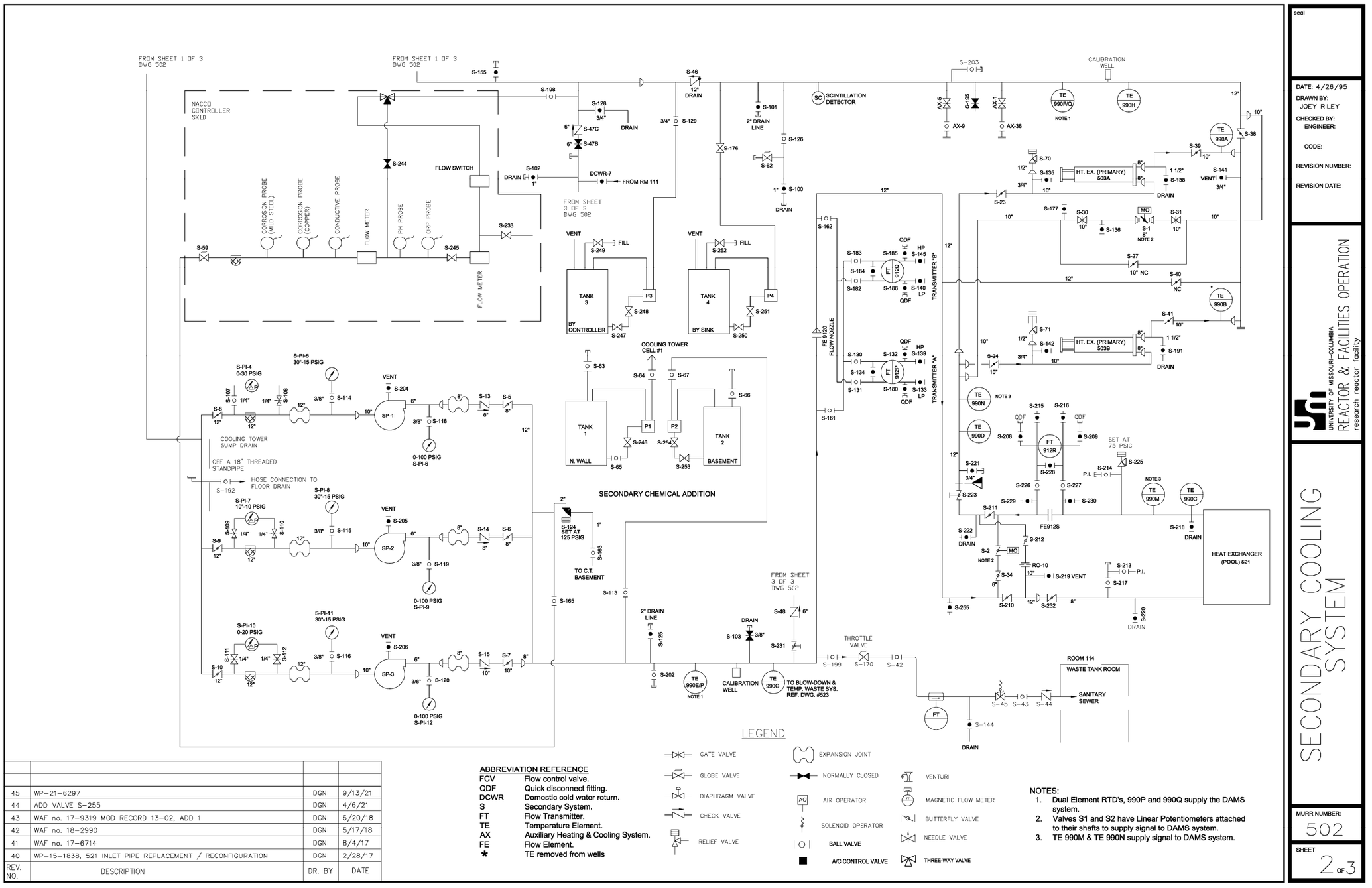
If radiolysis without recombination, in excess of negligible quantities, were to occur in the primary coolant system, hydrogen gas would be constantly produced beyond its solubility limit and ultimately be collected in the primary coolant system vent tank causing it to pressurize and sporadically off-gas to the pool and be exhausted from the containment building by the pool sweep system. However, off-gassing of the vent tank is NOT observed supporting the conclusion that radiolysis, without recombination, of primary coolant system water is negligible.

The same arguments apply for O_2 gas produced through radiolysis in the pool and primary coolant systems. However, some O_2 will already be present in pool and primary water from dissolved air. Consequently, any O_2 produced by radiolysis would be additive and induce gas emanation at that point when the sum of O_2 from the two sources (dissolved air and radiolysis) exceed the solubility limit for O_2. As stated in the previous two bullets, off-gassing in the pool or the primary coolant system vent tank is NOT observed supporting the conclusion that radiolysis, without recombination, in these systems is negligible.

Neither hydrogen or oxygen gas emanation is observed in the irradiated fuel storage areas in the reactor pool nor from the primary coolant system vent tank indicating that radiolysis, without recombination, of

MURR pool and primary water is negligible. Any gases emanating from the pool and primary coolant systems are captured by the pool sweep system and exhausted from the containment building without being recirculated and with large dilution factors.


PIPING AND INSTRUMENT DIAGRAM
DRAIN COLLECTION SYSTEM
REACTOR WATER
POOL WATER
PRESSURIZER 520 GAL.
PLATE HEAT EXCHANGER 503B
PLATE HEAT EXCHANGER 503A
PLATE HEAT EXCHANGER 521
HOLD-UP TANK 100 GAL. 506
HOLD-UP TANK 6000 GAL. 504
RX DEMIN 50 GPM 513A
POOL DEMIN 50 GPM 513B
DEMINERALIZER CELLS 542 VPF 1261 SHEETS 1-20 SEE MURR-344
PRIMARY 1800 GPM 501A
PRIMARY 1800 GPM 501B
POOL PUMP 1200 GPM 508A
POOL PUMP 1200 GPM 508B
LEVEL CONTROL PUMP P533
SKIMMER PUMP 50 GPM 532
FILTER 531
VENT TANK
25 GAL PRESSURE TANK
HEAT EXCHANGER 520
COLLECTION TANK
P-536 H.3 PUMP 50 GPM
CORE
POOL LINER
SKIMMER
DIFFUSER REF. (541)
BLIND FLANGE BOTH ENDS
PIPE TUNNEL
CAL. WELL
ABSOLUTE FILTER CHARCOAL FILTER
POOL SWEEP
WATER EL 630'-11"
EL. 631'-8"
EL. 624'-8 1/4"
EL. 612'-8"
EL. 600'-2"
EL. 597'-0"
EL. 590'-0"
FL. EL. 602'-0"
BOTTOM OF POOL EL. 600'-10 1/2"
CEILING EL. 611'-0"
ANNULUS DRAINS
TO BEAMPORTS (6)
TO SAMPLE STATION
TO SOLENOID VALVE 1/2" 529G (H-12)
TO POOL MONITOR
TO SKIMMER SYSTEM
TO DRAIN COLLECTION SYSTEM
FROM COLLECTION SOURCES
FROM PRESS SWITCH 947 (K-6)
MAKE-UP WATER FROM TANK T-300 OR 301
VACUUM BREAK VALVE
DEMINERALIZED REACTOR WATER
DEMINERALIZED POOL WATER
POOL FILL LINE
POOL DRAIN LINE
1/2" PLUG
ALARM (ALARM NOT INSTALLED AS OF 11-1-04)
(DC NOT INSTALLED AS OF 9/13/85)
(BLANK)
FROM T-300 T-301
TO 8" S-1 (D-2)
FROM 901B
TO T-300
TO T-301
TO TANK T-300
FROM TANK T-300
FROM FEF MONITOR 595B
TO 4" S-2 (K-3)
TO BUILDING EXHAUST
REMOTE OPERATED
AIR SUPPLY
1/2" RELIEF VALVE 563
AIR LINE CONNECTOR
SIGHT GLASS
OVERFLOW
DRAINS
FROM POOL MONITOR
TO TR908B
TO TANK T-301
AIR TO OPEN 1 1/4"-524
1 1/4"-534A
2"-515H AIR TO CLOSE
2"-516A
2"-515C LO
2"-515P
2"-515Q LC
2"-515R LC
2"-515T
2"-515V
2"-515W
2"-515X
2"-515N
2"-515L
2"-515J
2"-515K LC
2"-515D
2"-515E
2"-515I
2"-518C
2"-518D
2"-518Q
2"-518R
2"-518S-LC
2"-518T-LC
2"-518U LC
2"-518V
2"-518W
2"-518Z
2"-518AA
2"-518AB
2"-518AC
2"-518AD
2"-518AE
2"-518AF
2"-518AG
2"-518AH
2"-518AI
2"-518AJ
2"-518AK
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2"-518AT
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2"-518BB
2"-518BC
2"-518BD
2"-518BE
2"-518BF
2"-518BG
2"-518BH
2"-518BI
2"-518BJ
2"-518BK
2"-518BL
2"-518BM
2"-518BN
2"-518BP
2"-518BQ
2"-518BR
2"-518BS-LC
2"-518BT-LC
2"-518BU LC
2"-518BV
2"-518BW
2"-518BX
2"-518BY
2"-518BZ
2"-518CA
2"-518CB
2"-518CC
2"-518CD
2"-518CE
2"-518CF
2"-518CG
2"-518CH
2"-518CI
2"-518CJ
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2"-518ED
2"-518EE
2"-518EF
2"-518EG
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2"-518EJ
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2"-518EL
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2"-518EN
2"-518EP
2"-518EQ
2"-518ER
2"-518ES
2"-518ET
2"-518EU
2"-518EV
2"-518EW
2"-518EX
2"-518EY
2"-518EZ
2"-518FA
2"-518FB
2"-518FC
2"-518FD
2"-518FE
2"-518FF
2"-518FG
2"-518FH
2"-518FI
2"-518FJ
2"-518FK
2"-518FL
2"-518FM
2"-518FN
2"-518FP
2"-518FQ
2"-518FR
2"-518FS
2"-518FT
2"-518FU
2"-518FV
2"-518FW
2"-518FX
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2"-518FZ
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2"-518GF
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2"-518HI
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2"-518IF
2"-518IG
2"-518IH
2"-518II
2"-518IJ
2"-518IK
2"-518IL
2"-518IM
2"-518IN
2"-518IP
2"-518IQ
2"-518IR
2"-518IS
2"-518IT
2"-518IU
2"-518IV
2"-518IW
2"-518IX
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2"-518LD
2"-518LE
2"-518LF
2"-518LG
2"-518LH
2"-518LI
2"-518LJ
2"-518LK
2"-518LL
2"-518LM
2"-518LN
2"-518LP
2"-518LQ
2"-518LR
2"-518LS
2"-518LT
2"-518LU
2"-518LV
2"-518LW
2"-518LX
2"-518LY
2"-518LZ
2"-518MA
2"-518MB
2"-518MC
2"-518MD
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45	WP-21-6297	DGN	9/13/21
44	ADD VALVE S-255	DGN	4/6/21
43	WAF no. 17-9319 MOD RECORD 13-02, ADD 1	DGN	6/20/18
42	WAF no. 18-2990	DGN	5/17/18
41	WAF no. 17-6714	DGN	8/4/17
40	WP-15-1838, 521 INLET PIPE REPLACEMENT / RECONFIGURATION	DGN	2/28/17

seal

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UNIVERSITY OF MISSOURI-COLUMBIA
REACTOR & FACILITIES OPERATION
research reactor facility

SECONDARY COOLING SYSTEM

MURR NUMBER: 502

SHEET 2 OF 3

FIGURE 5.22 – SECONDARY SYSTEM DIAGRAM

Revision 2
6/30/2025

CHAPTER 6

ENGINEERED SAFETY FEATURES

TABLE OF CONTENTS

6.0 ENGINEERED SAFETY FEATURES 3
6.1 Introduction 3
6.2 Containment System 4
6.2.1 Introduction 4
6.2.2 Reactor Containment Building 4
6.2.3 Penetrations and Closures 6
6.2.4 Backup Doors 12
6.2.5 Compressed Air and Inflatable Gaskets 12
6.2.6 Description of Operation 13
6.2.7 System Redundancy and Reliability 13
6.2.8 Reactor Containment Building Technical Specifications 14
6.3 Anti-Siphon System 15
6.3.1 Introduction 15
6.3.2 Design Criteria 15
6.3.3 Pressure Tank 16
6.3.4 Anti-Siphon Isolation Valves 16
6.3.5 Level Controller 16
6.3.6 Description of Operation 16
6.3.7 System Redundancy and Reliability 17
6.3.8 Design Analysis 17
6.3.9 Anti-Siphon System Technical Specifications 19

LIST OF TABLES

Table 6-1 REACTOR CONTAINMENT BUILDING PENETRATIONS 8
Table 6-2 REACTOR CONTAINMENT BUILDING ISOLATION SET POINTS 13
Table 6-3 PRESSURE TRANSMITTER RESPONSE DELAY TIME OF 0.2 SECONDS 18
Table 6-4 TRANSMITTER RESPONSE DELAY TIME VS. COOLANT LEVEL ABOVE THE CORE 19

LIST OF FIGURES

Figure 6.1 – BACKUP DOOR 21
Figure 6.2 – SIMPLIFIED DIAGRAM OF THE PRIMARY COOLANT SYSTEM 22

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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 University of Missouri Research Reactor (MURR), two systems were identified as required ESFs: the containment system and the anti-siphon system. (The original safety evaluation of the MURR is documented in the Preliminary Hazards Report (Reference 6.1), the Hazards Summary Report (Reference 6.2), and Hazards Summary Report, Addenda 1-5 (References 6.3-6.7).) 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 four accident scenarios which require an engineered safety feature are the Maximum Hypothetical Accident (MHA), the Fuel Handling Accident (FHA), the Fuel Failure During Reactor Operation Accident (FFA), and the Loss of Coolant Accident (LOCA).

The MHA is an accident condition which postulates the complete failure of a fueled experiment and the subsequent release of fission products to the pool coolant system. 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 initial barrier of protection against the release of fission products to the atmosphere, the target encapsulation, has failed. The containment system will perform a complete isolation of the reactor containment building, providing a final barrier against the 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 ensure 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, breaking any potential siphon which may have been created by the pipe rupture.

The MHA, the Fuel Handling Accident, the Fuel Failure During Reactor Operation, and the LOCA analyses are discussed in detail in Chapter 13, “Accident Analyses.”

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. 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 210). Actuation of the Room 210 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.

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 dumbbell 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 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 supporting 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 (Figure 1.1). The walls and ceiling of this space are also 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 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) (Reference 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). 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. The calculation includes the following assumptions:

- No heat sinks;
- Initial containment building air temperature: 75 °F (24 °C) with 50% relative humidity;
- Initial pool water temperature: 100 °F (38 °C);
- Containment building volume: 240,000 ft3 (6,796 m3); and
- 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. (The design basis assumptions used a containment building volume of 240,000 ft^3; however, the current technical specification minimum free volume limit within containment is 225,000 ft^3. Using the technical specification minimum value, the amount of steam that must be released from the pool to cause a 2.0-psi (13.8 kPa) increase within the containment structure corresponds to 981 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.[3]

However, the possibility exists that a severe fuel meltdown could cause a molten aluminum water (Al-H_2O) reaction which releases a considerable amount of energy. The specific conditions under which such a reaction might occur are widely discussed in literature pertaining to reactor hazards. 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 (8) 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 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 conservative approach is to assume that only 1.3% of the aluminum would react. This assumption is consistent with the Fuel Failure During Reactor Operation accident analysis which postulates the melting of four number-1 fuel plates in the reactor core. The corresponding amount of aluminum which would react with the coolant would be 0.3822 kg (1.3% of 29.4 kg). This equates 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. One line used for containment DP instrumentation exits the containment structure through one of the penetration plates. (Reference 6.12)

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. The water 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.

The following services or lines enter or exit the reactor containment building through the seal trench: (Reference 6.11)

- 6-inch fire protection water line;
- 6-inch emergency pool fill line;
- 1½-inch natural gas line (purged and blank flanged - no longer in use);
- 2-inch compressed air line;
- 2-inch domestic cold water line;
- 2-inch domestic hot water line;
- ¾-inch domestic hot water return line;
- 1 ½-inch containment sump discharge;
- 1 ½-inch sanitary sump discharge;
- ½-inch demineralized water line;
- 1¼-inch vacuum line;
- ¾-inch PVC alternate air supply line to the supply and exhaust plenum backup doors from the emergency air system;
- ¾-inch PVC (capped - no longer in use); and
- ¾-inch PVC (capped - no longer in use).

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. DP Instrument Line 2. Pressurization Line 3. Depressurization Line	 Manual Valve (Reference 6.12) 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.

6.2.3.2 Pedestrian Entry

The pedestrian entry for the reactor containment building is at grade level (Figure 6.2) and consists of two (2) 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 (2) 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 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 floor. This entry, serviced by the laboratory building 15-ton (13,608-kg) freight elevator, provides access for heavy and large equipment to be moved into and out of the containment building at below-grade level (Figure 1.1). 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 (2) 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 close. 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 and 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 (2) electric-motor-driven horizontal sliding doors located in the ventilation plenums on the fifth level of the reactor containment building (Figure 1.5). 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 0.25-inch (0.64-cm) thick metal plate welded to a steel frame. Each door is suspended from a 12-1/2-foot (3.8-m) long, 6-inch (15.2-cm) I-beam by two (2) adjustable one-half-ton (453-kg) trolleys. To ensure 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, 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 210), 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, 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, 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 1.5-inch (3.8-cm) sample carrier tubes and four 1.5-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, the two carrier tubes on the south side terminate within the graphite reflector elements surrounding the reactor core and are currently in use. The two carrier tubes on the north side have been terminated immediately after entering the reactor pool and removed from current use.

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 (40.64-cm) 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 (40.64-cm) 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 (40.64-cm) pipe is cast into the containment building wall. The 16-inch (40.64-cm) pipe is welded to this collar. Two (2) quick-closing 16-inch (40.64-cm) 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 210) 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 (2) 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 (2) valves, installed in series, control the air supply to the “open side” of the actuator, and two (2) valves, installed in parallel, control the air supply to the “close side” of the actuator. A closure signal will de-energize all four (4) 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) (Figure 6.1). 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 via a point-of-use air dryer to prevent moisture accumulation (Reference 6.12). 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. An automatic 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. The 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 (Figure 1.5). 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, 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:

- Personnel Airlock Door 276 Sealing Gasket;
- Personnel Airlock Door 277 Sealing Gasket;
- Motorized Ventilation Isolation Door 504 Sealing Gasket;
- Motorized Ventilation Isolation Door 505 Sealing Gasket;
- Truck Entry Door (Door 101) Sealing Gasket;
- Ventilation Exhaust Valve 16A Actuator;

- Ventilation Exhaust Valve 16B Actuator; and
- 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.

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 (1.65 m), 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 set points associated with the automatic initiation of the reactor containment building isolation are contained in Table 6.2. 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 210).

TABLE 6-22
REACTOR CONTAINMENT BUILDING ISOLATION SET POINTS

Detector Location	**Current Set Point**	**Set Point Basis**
Containment Building Exhaust Plenum No. 1	3.0 mR/hr	10 x normal operating background
Reactor Pool Upper Bridge – ALARA	10K mR/hr	As determined by Health Physics
Reactor Pool Upper Bridge	50 mR/hr	10 x normal operating background
Containment Building Exhaust Plenum No. 2	3.0 mR/hr	10 x normal operating background

When controlled evolutions such as experiment transfers or minor maintenance are conducted in the pool area, the “Reactor Pool Upper Bridge” detector is bypassed, as allowed by TS 3.7.a, however, the “Reactor Upper Bridge-ALARA” will provide a containment isolation if its set point of 10 R/hr is exceeded. (Reference 6.9)

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:

- 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;
- All relays which de-energize to perform their intended function are assumed to be fail-safe;
- 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;
- 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;
- 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. The doors will close in case of simultaneous containment isolation and loss of normal electrical power; and
- 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. Reactor 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 Reactor Containment Building Technical Specifications

The objective of the Technical Specification Limiting Condition for Operation (LCO) on the reactor containment building is to ensure that containment integrity is maintained when required so that the health and safety of the general public is not endangered as a result of reactor operation. Technical Specification 3.4 lists the associated LCOs for the reactor containment building.

The objective of the surveillance technical specifications on the reactor containment building are to reasonably assure proper operations of the containment system. Technical Specification 4.4 lists the associated surveillances for the reactor containment building.

The objective of the Design Feature Technical Specification on the reactor containment building is to assure adequate restriction to the accidental release of radioactivity to the environment. Technical

Specification 5.5 lists the associated design features technical specification for the reactor containment building.

6.3 Anti-Siphon System

6.3.1 Introduction

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 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.

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:

- 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
- 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.

The MURR HSR used the above two criteria to impose 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 5 feet (1.52 m) of water.

The thermal-hydraulic transient code RELAP5 was used to update and validate the original design analysis of the anti-siphon system.

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 (1034 kPa above atmosphere). A 0.75-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 (2) 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, 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 (30.48 cm) primary coolant system piping, a loss of system pressure will be detected by four (4) 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 P501A and P501B will stop; the primary coolant isolation valves V507A and V507B will close; and the anti-siphon 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, 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 ensure 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:

- All relays which de-energize to perform their intended function are assumed to be fail-safe;
- Redundant primary coolant system low pressure signals (2), either of which is capable of initiating operation of the anti-siphon system, are provided;
- 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
- 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.

The RELAP5 code analysis was based on benchmarked data provided by a high speed digital chart recorder used to perform precise response time measurements for the various transmitters and sensors

in the reactor coolant system that provide the signals for valve action and for valve actuation (time from the fully closed to fully open or from fully open to fully closed). Response time measurements were performed for anti-siphon isolations valves V543A and V543B, in-pool heat exchanger isolation valves V546A and V546B, and primary coolant isolation valves V507A and V507B. Anti-siphon isolation valves V543A and V543B start to open from their fully closed positions 0.2 seconds after reactor core outlet pressure transmitter PT-944A and/or PT-944B sends a signal to the valve actuators. The valves are fully open 0.4 seconds after the valve action signal (open signal) is received. Including the measured pressure transmitter response time of 0.2 seconds, it will take 0.6 seconds for valves V543A and V543B to reach their fully open positions. In-pool heat exchanger valves V546A and V546B are fully open 0.8 seconds after core inlet flow rate decreases to 3,200 gpm (12,113 lpm); the trip set point of the differential pressure across the reactor core sensor DPS-929.

The anti-siphon system analysis simulating a primary coolant system piping shear on the reactor core side of each primary coolant isolation valve was performed with benchmarked opening times for valves V543A, V543B, V546A, and V546B. The initiating event was followed by a low-pressure trip which actuated the opening of valves V543A and V543B and also caused primary coolant circulation pumps P501A and P501B to secure (relays 2K13 and 2K28 de-energize). When primary coolant flow through the core reduces to 3,200 gpm (12,113 lpm), differential pressure across the reactor core sensor DPS-929 sends a signal to open in-pool heat exchanger isolation valves V546A and V546B. It was determined that an anti-siphon system pressure of 0 psig (101.3 kPa) would be sufficient to ensure that at least 5 feet (1.52 m) of water remained above the top of the reactor core after a primary coolant piping rupture. Primary coolant temperature was assumed to be 80 °F (27 °C) for these runs. As shown in Figure 6.2, the centerline of the horizontal primary coolant system inlet piping after core inlet check valve V502 is 5.937 feet (1.81 m) above the core where it enters the pressure vessel.

Table 6-3 provides a comparison of the amount of coolant that will remain above the reactor core based on a measured pressure transmitter response delay time of 0.2 seconds – time for a valve action signal (open signal) to be sent to anti-siphon isolation valves V543A and V543B – for the following four assumed anti-siphon system pressures: 0, 10, 20 and 26 psig (101.3, 170.3, 239.2 and 280.6 kPa).

TABLE 6-33
PRESSURE TRANSMITTER RESPONSE DELAY TIME OF 0.2 SECONDS

Anti-Siphon System Pressure	**Coolant Above the Core**
0 psig (101.3 kPa)	5.110 feet (1.556 m)
10 psig (170.3 kPa)	5.443 feet (1.659 m)
20 psig (239.2 kPa)	5.939 feet (1.810 m)
26 psig (280.6 kPa)	5.900 feet (1.798 m)

As stated above, the horizontal primary coolant system inlet piping centerline after core inlet check valve V502 is 5.937 feet (1.81 m) above the core where it enters the pressure vessel. With 26 psig

(280.6 kPa) of air pressure in the anti-siphon system (1.0 psi (6.9 kPa) below Technical Specification 3.5.b.2), Table 2 provides the coolant levels above the top of the core for the given time interval (assumed transmitter response delay time) between when a low pressure trip from reactor core outlet pressure transmitter PT-944A and/or PT-944B occurs due to a primary coolant system shear on the reactor side of both isolation valves and when the valve action signal is sent to open the anti-siphon isolation valves from their fully closed positions.

TABLE 6-44
TRANSMITTER RESPONSE DELAY TIME VS. COOLANT LEVEL ABOVE THE CORE

Time Interval (seconds)	**Coolant Level Above the Core (feet)**
0.20	5.900 (1.798 m)
1.20	6.009 (1.832 m)
2.20	5.636 (1.718 m)
4.20	5.430 (1.655 m)
6.20	5.574 (1.699 m)
8.20	4.556 (1.389 m)
10.20	4.400 (1.341 m)

A delay in the opening action time of the anti-siphon isolation valves would need to be greater than 6 seconds to result in a coolant level of less than 5 feet (1.52 m) above the top of the core during a loss of coolant accident. An increase in the normal valve opening response time of 0.6 seconds would have to be caused by either a malfunction of both V543A and V543B valve actuators or some sort of electronic failure in both pressure transmitters PT-944A and PT-944B. As stated in Section 6.3.4, the operation of either valve will perform the intended function of the anti-siphon system. Additionally, two (2) pressure transmitters are installed for redundancy, either of which will cause the anti-siphon isolation valves to open. 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. (Reference 6.10)

6.3.9 Anti-Siphon System Technical Specifications

The objective of the Limiting Conditions for Operations (LCO) technical specifications on the Anti-Siphon System are to:

- Reasonably assure proper operation of the reactor control system, thus avoiding conditions which could jeopardize the integrity of the fuel element cladding or endanger personnel health and safety, and to specify minimum number of reactor safety system instrument channels that must be operable for safe reactor operation. Technical specification 3.2.f(6) lists the associated LCO for the anti-siphon system;

- Protect the integrity of the reactor fuel and to prevent the release of fission product radioisotopes. Technical specification 3.3.a(1) lists the associated LCO for the Anti-Siphon System; and
- Ensure that sufficient reliable information is presented to the operator to ensure safe operation of the reactor. Technical specification 3.5.b(2) lists the associated LCO for the anti-siphon system.

The objective of the surveillance technical specification on the Anti-Siphon System are to reasonably assure proper operation of the anti-siphon system and its associated instrumentation system. The following is a list of the associated surveillance technical specifications for the anti-siphon system;

4.2.h, 4.3.a(1) and 4.5.b.

The objective of the design features technical specification on the Anti-Siphon System are to assure proper coolant for safe operation. Technical specification 5.2.h provides this assurance for the anti-siphon system.

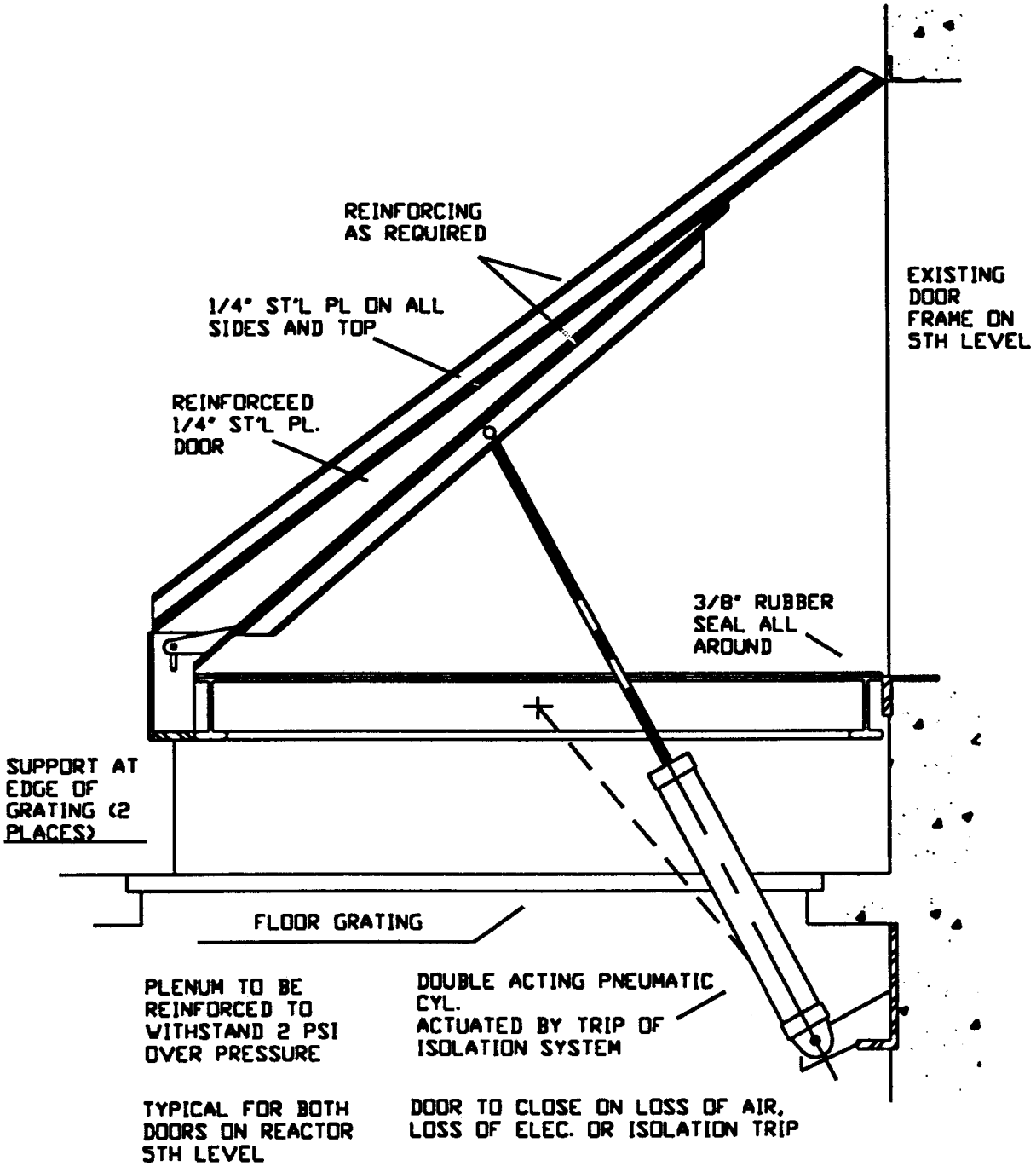


FIGURE 6.1 – BACKUP DOOR

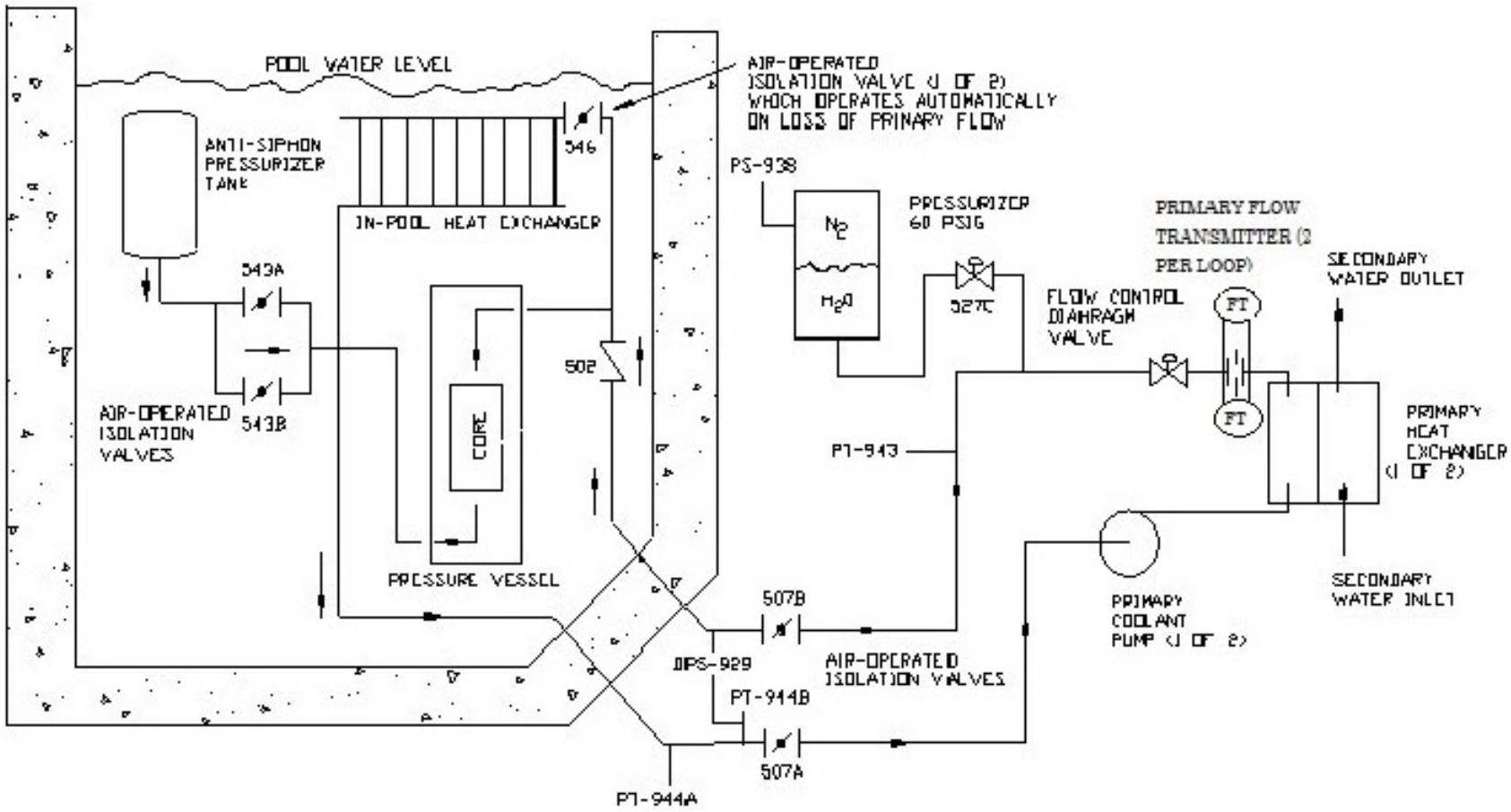


FIGURE 6.2 – SIMPLIFIED DIAGRAM OF THE PRIMARY COOLANT SYSTEM

CHAPTER 7

INSTRUMENTATION AND CONTROL SYSTEMS

TABLE OF CONTENTS

7.0 INSTRUMENTATION AND CONTROL SYSTEMS 5
7.1 Summary Description 5
7.2 Design of Instrumentation and Control Systems 6
7.2.1 Design Criteria 6
7.2.2 Design-Basis Requirements 7
7.2.3 Conclusion 9
7.3 Control Console and Display Instruments 9
7.3.1 Annunciator 17
7.4. Nuclear Instrumentation 20
7.4.1 Introduction 20
7.4.2 Neutron Flux Monitors 20
7.4.2.1 Source Range - Channel 1 22
7.4.2.2 Intermediate Range - Channels 2 & 3 22
7.4.2.3 Power Range - Channels 4 & 5 (Fission Chamber Flux Monitro) 22
7.4.3 Power Range Monitor (UIC Flux Monitor) 23
7.4.4 Wide Range Neutron Flux Monitor 23
7.4.5 Multiscaler 24
7.4.6 Nuclear Instrumentation Technical Specifications 25
7.5 Rod Control System 25
7.5.1 Introduction 25
7.5.2 Control Rods 26
7.5.2.1 Control Rod Technical Specifications 27
7.5.3 Manual Control 27
7.5.3.1 Rod Withdrawal Prohibit 27
7.5.3.2 Rod Withdrawal Prohibit Circuit Technical Specifications 28
7.5.4 Automatic Control 28
7.5.4.1 Automatic Shim Control 29
7.5.4.2 Automatic Control Prohibit Technical Specifications 30
7.5.5 Rod Run-In System 30
7.5.5.1 Rod Run-In System Technical Specifications 31
7.5.6 Control Rod Position Indication 32
7.6 Process Instrumentation and Control System 33
7.6.1 Introduction 33
7.6.2 Temperature Measurement and Control 33
7.6.2.1 Primary Coolant System 33
7.6.2.2 Pool Coolant System 34
7.6.3 Pressure Measurement and Control 35
7.6.3.1 Primary Coolant System 35
7.6.4 Flow Measurement and Control 36
7.6.4.1 Primary Coolant System 36

7.6.4.2 Pool Coolant System 38
7.6.5 Pressurizer Pressure and Liquid Level Control 39
7.6.6 Power Level Interlock 40
7.6.7 Process Instrumentation and Control System Technical Specifications 41
7.7 Reactor Scram System 41
7.7.1 Introduction 41
7.7.2 Description 42
7.7.2.1 Non-Coincidence Logic Units 43
7.7.2.2 Trip Actuator Amplifiers 44
7.7.3 Reactor Scram System Technical Specifications 44
7.8 Engineered Safety Features Actuation Systems 45
7.8.1 Introduction 45
7.8.2 Containment Actuation (Reactor Isolation) System 45
7.8.2.1 Automatic Initiation 46
7.8.2.2 Manual Initiation 47
7.8.2.3 Containment Actuation System Technical Specifications 47
7.8.3 Anti-Siphon Actuation System 48
7.8.3.1 Anti-Siphon System Technical Specifications 48
7.9 Radiation Monitoring Systems 49
7.9.1 Introduction 49
7.9.2 Area Radiation Monitoring System 50
7.9.2.1 Description 50
7.9.2.2 Area Radiation Monitoring System Technical Specifications 51
7.9.3 Fuel Element Failure Monitoring System 52
7.9.3.1 Description 52
7.9.3.2 Fuel Element Failure Monitoring System Technical Specifications 52
7.9.4 Secondary Coolant Monitoring System 53
7.9.4.1 Description 53
7.9.4.2 Secondary Coolant Monitor Technical Specifications 54
7.9.5 Off-Gas Radiation Monitoring System 54
7.9.5.1 Description 54
7.9.5.2 Off-Gas Radiation Monitoring System Technical Specifications 54
7.10 Center Test Hole Instrument Channel 55
7.10.1 Introduction 55
7.10.2 Description 55
7.10.3 Center Test Hole Instrument Channel Technical Specifications 60

LIST OF TABLES

Table 7-1 REACTOR CONTROL CONSOLE DISPLAY INSTRUMENTS 11
Table 7-2 REACTOR CONTROL CONSOLE CONTROL EQUIPMENT 12
Table 7-3 INSTRUMENT PANEL INSTRUMENTATION (UPDATED BY REFERENCE 7.17) 14

Table 7-4 ANNUNCIATOR ALARM SEQUENCE 18
Table 7-5 ANNUNCIATOR ALARM CONDITIONS 19
Table 7-6 ROD RUN-INS 31
Table 7-7 ROD RUN-IN NON-COINCIDENCE LOGIC UNIT SIGNAL INPUTS 31
Table 7-8 REACTOR SCRAMS 41
Table 7-9 REACTOR SCRAM NON-COINCIDENCE LOGIC UNIT SIGNAL INPUTS 43
Table 7-10 AREA RADIATION MONITORING SYSTEM 51

LIST OF FIGURES

Figure 7.1 – REACTOR CONTROL CONSOLE LAYOUT 61
Figure 7.2 – INSTRUMENT PANEL LAYOUT 62
Figure 7.3 – ANNUNCIATOR CONTROL 63
Figure 7.4 – ROD CONTROL SYSTEM 64
Figure 7.5 – ROD RUN-IN SYSTEM 65
Figure 7.6 – PROCESS INSTRUMENTATION CONTROL AND INTERLOCK (Reference 7.13) 66
Figure 7.7 – PROCESS INSTRUMENTATION CONTROL AND INTERLOCK 67
Figure 7.8 – PROCESS INSTRUMENTATION CONTROL AND INTERLOCK 68
Figure 7.9 – REACTOR SCRAM SYSTEM 69
Figure 7.10 – AREA RADIATION MONITORING SYSTEM 70

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7.0 INSTRUMENTATION AND CONTROL SYSTEMS

This chapter describes and discusses the operating characteristics of the University of Missouri Research Reactor (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 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 I&C 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 I&C System;
- Reactor Scram 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 I&C 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 Scram System in addition to control interlocks for the Primary and Pool Coolant System Circulation Pumps and Automatic Isolation Valves.

The Reactor Scram 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. Inputs which govern the Reactor Scram 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, Standards for Protection Against Radiation.

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 I&C Systems are designed 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 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 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 assumes that the graphite region exhibits its maximum possible reflectivity, i.e., a solid ring.)

Since the purpose of the University of Missouri Research Reactor (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/k to the reactor core.

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.

The University of Missouri Research Reactor (MURR) Reactor Safety System includes all of the sensing devices, electronic circuits and equipment, signal conditioning equipment, and electro-mechanical devices that serve to affect 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 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 (Reference 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" (Reference 7.8). 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 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 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 I&C Systems all meet the single failure criterion of IEEE-279.

The channels which provide input signals to the Reactor Scram 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 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 no single component or circuit failure will render any portion of the system inoperative. Characteristics of the Engineered Safety Features Actuation System which provide redundancy and reliability are discussed in Section 7.8.

Bypass switches are used in the Reactor Scram System to change the protective system lineup to correspond to the three (3) 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 Reactor 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 Reactor Instrument Panel.

The prudent operational limits imposed upon the parameters monitored by the Reactor Scram 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.

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 University of Missouri Research Reactor (MURR), these occurrences would be such that more than sufficient time would exist for prompt operator action. 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.

Safe shutdown of the reactor is not dependent on the availability of forced cooling, decay heat removal (see Section 13.2.9.3), coolant charging systems, or emergency electrical power (see Section 13.2.7). Power availability and drive train thermal delay protection and/or fire suppression for these systems is not required from the standpoint of fire protection.

All physical components involved in shutting down the reactor are below the pool water line. They are not vulnerable to the typical weaknesses, embrittlement, and deformation failures associated with a fire. Thermal delay protection and/or fire suppression for these components is not required from the standpoint of fire protection.

All remaining components involved in safely shutting down the reactor are essentially electrical in nature and are utilized in a non-coincidence logic system. Under fire conditions, these control components, circuits and instrument channels may fail, but will fail in either a grounded or an open circuit state. Either and any combination of these failure states in any of the circuits significant to safety will still result in the desired loss of power, and a resultant safe shutdown of the reactor. The failure state where ungrounded continuity could be maintained due to fire conditions would have to occur in multiple locations simultaneously with no grounding and no open circuit failures. Thermal delay protection and/or fire suppression for these instrument channels, circuits, and control components are not required from the standpoint of fire protection. (Reference 7.10)

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.

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 Reactor Control Console have been designed for checking operability, inserting test signals, performing calibrations, and verifying trip settings. The availability and

use of these features ensure that the Reactor Control Console devices and subsystems and display instruments will operate as designed. Reactor Control Console locking devices reasonably ensure that the facility is only operated by authorized personnel. The Reactor 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. A 0.75-inch (19 mm) lead sheet encapsulated in aluminum 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 within Figure 7.3.

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 7.1.

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 7.2.

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. These numbers refer to the position of the display instruments on the Reactor Control Console (Figure 7.1).

TABLE 7-2
REACTOR CONTROL CONSOLE CONTROL EQUIPMENT

No.[a]	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 Position Spring Return
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 Position Spring Return
36	Regulating Rod Operate	“IN-NORMAL-OUT”	3 Position Spring Return
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

TABLE 7-2 (CONT.)
REACTOR CONTROL CONSOLE CONTROL EQUIPMENT

No.[a]	Function	Positions	Type of Switch
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
52	Anti-Siphon High Level	"OFF-BYPASS"	2 Position - Key Lock
53	Intrusion Alarm	"OFF-ON"	2 Position
54	Airlock Door Security	"NORMAL-CLOSED-OUTER OPEN"	3 Position
55	Thermal Column Shutter	"CLOSED-OPEN"	2 Position
56	Pneumatic Tube Blowers	"OFF-ON"	2 Position
57	Airlock Door Open	"OUTER-INNER"	3 Position Spring Return
58	Range Switch	N/A	18-position
59	Intercommunications Box	N/A	Light
60	Bridge Door Unlock	N/A	Push Button
61	FIRST	"OFF-BYPASS"	2 Position - Key Lock
62	FIRST	"OFF-BYPASS"	2 Position - Key Lock

a. These numbers refer to the position of the control equipment on the Reactor Control Console (Figure 7.1).

TABLE 7-3
INSTRUMENT PANEL INSTRUMENTATION (UPDATED BY REFERENCE 7.17)

No.	Function
1	Annunciator
2	Spare
3	Intermediate Range Monitor Level Recorder
4	Source and Wide Range Monitors Level Recorder
5	Power Range Level Monitor Recorder
6	Multiscaler
7	Neutron Flux Monitor – Signal Processor Drawer No. 1
8	Neutron Flux Monitor – Signal Processor Drawer No. 2
9	Neutron Flux Monitor – Channel No. 6 Drawer
10	Neutron Flux Monitor – Wide Range Drawer
11	Annunciator & Interlock Relay Drawer (K-1 Relays)
12	Servo Amplifer Drawer
13	Auxiliary Annunciator - Panalarm
14	Non-Coincidence Logic Unit – Reactor Scram System “Yellow Leg”
15	Trip Actuator Amplifier – Reactor Scram 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 Scram System “Green Leg”
19	Trip Actuator Amplifier - Reactor Scram 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 Scram System Relay Drawer (K-2 Relays)
25	Primary Coolant System Pressure Meter - PT 943
26	Primary & Pool Coolant Demineralizer Flow Recorder and Pressurizer Level
27	Dual Alarm Unit (EP-920 E/F) Primary Low Flow Scram & Low Pressure Scram (PT-944A)
28	Dual Alarm Unit (EP 953B) - Primary Coolant High Temperature Scram
29	RTD Transmitter (EP 903B) - Primary Coolant T_h
30	Isolated Power Supply - Reactor Scram System “Yellow Leg”
31	Square Root Transmitter (EP 919A) - Primary Flow “A” Loop
32	Dual Alarm Unit (EP 920A/B) - Primary & Pool Low Flow Scrams
33	Square Root Transmitter (EP 919F) - Pool Flow “B” Loop
34	RTD Transmitter (EP 903A) - Primary Coolant T_c
35	Dual Alarm Unit (EP 920L/M) Reflector Hi/Lo DP
36	RTD Transmitter (EP 955) - In-Pool Heat Exchanger Differential Temperature
37	RTD Transmitter (EP 903C) - Pool Coolant T_c
38	Spare
39	RTD Transmitter (EP 903D) - Pool Coolant T_h
40	Square Root Transmitter (EP 919C) - Primary Demineralizer Flow
41	Square Root Transmitter (EP 919D) - Pool Demineralizer Flow
42	Isolated Power Supply - Reactor Scram System “Green Leg”

TABLE 7-3

INSTRUMENT PANEL INSTRUMENTATION (UPDATED BY REFERENCE 7.17) (CONT.)

No.	Function
43	Square Root Transmitter (EP 919E) - Primary Flow “A” Loop
44	Dual Alarm Unit (EP 920C/D) - Primary & Pool Low Flow Scrams
45	Square Root Transmitter (EP 919B) - Pool Flow “A” Loop
46	Square Root Transmitter (EP 919G) - Primary Flow “B” Loop
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-NORM”
76	Valve 527D Open Indication Light
77	Valve 527D Closed Indication Light
78	Valve 527D Control Switch - 2 Position - “OPEN-NORM”
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[e] - 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

TABLE 7-3

INSTRUMENT PANEL INSTRUMENTATION (UPDATED BY REFERENCE 7.17) (CONT.)

No.	Function
86	Primary Coolant System Temperature Controller (S-1)
87	Pool Coolant T_h - T_c Recorder
88	Pool Coolant System Temperature Controller (S-2)
89	Primary Coolant System Flow Recorder
90	Pool Coolant System Flow Recorder and Reflector Region Differential Pressure (Reference 7.14)
91	Primary & Pool Coolant Demineralizer Flow Recorder
92	Closed Circuit Television Monitor
93	Spare
94	Spare
95	Spare
96	In-Pool Heat Exchanger Differential Temperature Meter
97	Spare
98	Reactor Core Outlet Pressure Meter - PT 944A
99	Reactor Core Outlet Pressure Meter - PT 944B
100	Primary Coolant HX503A Outlet Temperature Meter – TE 980A
101	Primary Coolant HX503B Outlet Temperature Meter - TE 980B
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
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 Recorder
115	Reactor Scram System Monitoring Circuit
116	Pool Coolant Flow Bypass Switch (2S40) - 4 Position
117	Primary Coolant Flow Bypass
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

TABLE 7-3
INSTRUMENT PANEL INSTRUMENTATION (UPDATED BY REFERENCE 7.17) (CONT.)

No.	Function
128	Off-Gas Radiation Monitor Recorder
129	Off-Gas Radiation Monitor Recorder
130	Off-Gas Radiation Monitor Recorder
131	Secondary Coolant System Flow Transmitter Selector Switch - 2 Position - “912P-912Q”
132	Emergency Diesel Generator Elapsed Time Run Meter
133	Dual Alarm Unit (EP 920G/H) Primary Low Flow Scram
134	Square Root Transmitter (EP 919H) - Primary Flow “B” Loop
135	Containment Differential Pressure Indication/Alarm
136	Fire Protection System Alarm Panel
137	Isolated Power Supply (EP 911C) - TE-980 A/B
138	Rod Run-In System Monitoring Circuit
139	T-300 & T-301 Water Level Display
140	Dual Alarm Unit (EP 920 J/K) Primary Low-Pressure Scram (PT-943)

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). 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.

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 3.625-inch (9.2-cm) by 3.375-inch (8.6-cm) 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 7.3. 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 7-4
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

TABLE 7-5
ANNUNCIATOR ALARM CONDITIONS

No.	Condition
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 two (2) Fission Chamber Neutron Flux Monitors, one (1) Uncompensated Ion Chamber Power Range Flux Monitor, one (1) Compensated Ion Chamber Wide Range Neutron Flux Monitor, and one (1) 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 Power Range Monitor is an Uncompensated Ion Chamber (UIC) based system providing reliable neutron flux measurement from 0% to 125% full power. The system provides inputs to the Reactor Safety and Rod Run-In Systems on sensing a high-power level.

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 Fission Chamber Neutron Flux Monitors provide the following six (6) channels of neutron flux measurement from source level (shutdown) through 100% of full power operation: one (1) Source Range, two (2) Intermediate Ranges, and two (2) Power Ranges. Each fission chamber 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 (6) 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 (2) 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 (2) 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 watertight 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:

- Initiation of the “NUCLEAR INSTRUMENT ANOMALY” annunciator alarm; and
- Initiation of a reactor scram.

The inoperative signal indicates that one or more of the following conditions exists:

- The ±15-VDC supply has failed;
- Magnitude of the detector excitation high voltage supply is low;
- Detector signal is low; or
- One of the front panel switches (S1 thru S5) on the signal processor drawer has been actuated. (Reference 7.11)

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 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 (1) Source Range, one (1) Intermediate Range, and one (1) Power Range Channel. However, only six (6) of the available nine (9) 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 Nuclear Instrumentation (NI) monitors count rate over a six (6)-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 Yokogawa paperless recorder mounted on the instrument panel records Channel 1 source range level and transfers the data to a server for archiving. (Reference 7.17)

If source range level indication decreases to less than 20 counts per second (cps), a trip circuit will de-energize relay K10 and cause the following actions to occur:

- Initiation of the "CHANNEL 1 LO COUNT RATE" annunciator alarm; and
- 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 7.4.5.

7.4.2.2 Intermediate Range - Channels 2 & 3

The Intermediate Range Channels of the Nuclear Instrumentation (NI) monitor count rate over a ten (10)-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 Yokogawa paperless recorder mounted on the Instrument Panel records Channel 2 and 3 intermediate range levels and transfers the data to a server for archiving. (Reference 7.17)

The Intermediate Range Channels also provide the following reactor protective and interlock functions:

- Reactor Scram: A reactor scram and a "CHANNEL 2 OR 3 PERIOD SCRAM" annunciator alarm initiation if reactor period decreases below a predetermined value;
- Rod Run-In: A rod run-in and a "CHANNEL 2 OR 3 SHORT PERIOD ROD RUN-IN" annunciator alarm initiation if reactor period decreases below a predetermined value; and
- 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 (Fission Chamber Flux Monitro)

The Power Range Channels of the Nuclear Instrumentation (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 Yokogawa paperless recorder mounted on the Instrument Panel records Channel 4, 5, and 6 power range levels and transfers the data to a server for archiving. (Reference 7.17)

The Power Range Channels also provide the following reactor protective and alarm functions:

- Reactor Scram: A reactor scram and a "CHANNEL 4, 5 OR 6 HI POWER SCRAM" annunciator alarm initiation if reactor power increases above a pre-determined value;
- Rod Run-In: A rod run-in and a "CHANNEL 4, 5 OR 6 HI POWER ROD RUN-IN" annunciator alarm initiation if reactor power increases above a pre-determined value; and
- 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 Power Range Monitor (UIC Flux Monitor)

Power Range Monitor 6 (PRM 6) provides an indication of neutron flux level from 0 to 125% full rated power. PRM 6 level is displayed by analog meters on the signal processor and Reactor Control Console and on the same Yokogawa paperless recorder as Power Range Channels 4 and 5.

The Uncompensated Ion Chamber provides an input current to the DC amplifier.

PRM 6 provides the same reactor protective and alarm functions as the other two Power Range Channels.

7.4.4 Wide Range Neutron Flux Monitor

The Wide Range Neutron Flux Monitor (WRM) provides continuous neutron flux monitoring over a ten (10)-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, Automatic Control. The WRM 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 (2) main components of the flux level measuring section are the amplifier unit and the Range Switch (1S2). Switch 1S2 has a primary function of allowing the reactor operator to observe reactor power over ten decades via selection of the eighteen-position switch. Various positions and groups of positions provide mode interlocks associated with Mode I, Mode II, and Mode III operation. Switch 1S2 works in conjunction with other interlocks, such as the Auto Control Prohibit Interlock and the Power Level Interlock. (Reference 7.11)

Switch 1S2 selects feedback elements of the amplifier unit, which have a value commensurate with the desired full-scale deflection input current. Essentially, switch 1S2 selects a different value feedback element each time it is set to a new position. The 18-position switch provides greater accuracy over the

ten (10) decades of input current. The first nine (9) decades are divided into two (2) 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 watertight 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 (2) separate dual trip units with each trip unit containing two (2) separate independent trip circuits. Presently, only one (1) of the four (4) trip circuits is used for alarm annunciation. If one (1) 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 module is removed from the instrument drawer;
- Drawer Selector Switch (S1) is placed in any position other than "OPERATE"; or
- The high voltage power supply output drops below a predetermined minimum voltage. (Reference 7.11)

The remaining three (3) trip circuits, one (1) downscale and two (2) upscale, can all be independently set to trip at different input voltages, if desired. There are no Safety System trips associated with the WRM and therefore, the Reactor Scram System satisfies the intent of IEEE-279 and 10 CFR 50, Appendix A, Criterion 24, 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 (2) separate independent high voltage power supplies which supply the positive and negative high voltage potentials used as polarizing voltages for ion chambers.

7.4.5 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 (3) 85-MHZ counting channels and a crystal controller time base. The timer can be used as a fourth 85-MHZ counting channel. The data from all channels are shown on two (2) seven (7)-decade LED displays. The selection of the channels to be displayed is made via two (2) pushbutton indicators.

The two (2) seven (7)-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 ($1 \times 10^7 - 1$) counts can be displayed, however, counting capacity can be expanded to $1 \times 10^{28} - 1$ by use of its overflow capability.

7.4.6 Nuclear Instrumentation Technical Specifications

The objective of the Limiting Conditions for Operations (LCO) Technical Specifications on Nuclear Instrumentation is to ensure that sufficient reliable information is present to the Operator to assure safe operation of the reactor. Technical Specifications 3.5.a.(1-3) provide this assurance for the Nuclear Instrumentation.

The objective of the Surveillance Technical Specifications on Nuclear Instruments is to reasonably assure proper operation of the reactor instrumentation. Technical Specification 4.5.a and 4.5.c provide this assurance for the Nuclear Instrumentation.

The objective of the Design Features Technical Specification on the Nuclear Instruments is to ensure that, during a reactor startup, the reactor power level is continuously monitored over the entire range. Technical Specification 5.3.m provides this assurance for the Nuclear Instrumentation.

7.5 Rod Control System

7.5.1 Introduction

The reactivity of the reactor is controlled by five (5) 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 (4) 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 (1) inch per minute (2.54 cm per minute) in the outward direction and two (2) inches per minute (5.08 cm 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 Reactor Control Console in either of two (2) 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 7.4). The Rod Control System is a relay and switch logic system used to prohibit accidental or incorrect operation which could result in unsafe conditions. 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 six (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 (4) 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 (14.9 W), 115-volt, one-amp, single-phase, 60-cycle motor connected to a ball-bearing lead screw assembly through a reduction gear box and adapter. (Reference 7.17) 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 limit wear on the rotating member and produces a uniform rotation. 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:

- “POWER ON” - Power is available to the electromagnets;
- “DRIVE FULL-IN”- Control rod drive mechanism is fully inserted;
- “DRIVE FULL-OUT” - Control rod drive mechanism is fully withdrawn;
- “MAGNET ENGAGED” - Electromagnet engaged to the anvil; and
- “BLADE 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 and 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.

7.5.2.1 Control Rod Technical Specifications

The objective of the Limiting Conditions for Operations (LCO) Technical Specifications on control rods is to reasonably assure proper operation of the Reactor Control System, avoiding conditions which could jeopardize the integrity of the fuel element cladding or endanger personnel health and safety. Technical Specifications 3.2.a-e provide this assurance for the control rods.

The objective of the Surveillance Technical Specifications on the control rods is to reasonably assure proper operation of the Reactor Control System. Technical Specifications 4.2.a-g and 4.2.m provide this assurance for the control rods.

The objective of the Design Features Technical Specification on the control rods is to specify the general reactor core configuration. Technical Specifications 5.3.d and 5.3.k provide the core configuration for the control rods.

7.5.3 Manual Control

A three (3)-position ("OFF-TEST-ON") keylock Master Control Switch (1S1) and a two (2)-position ("OFF-ON") Magnet Current Switch (1S14) located on the Reactor Control Console controls power to the entirety of the Rod Control System. Switch 1S1 and 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.

Switch 1S1 provides perhaps the most wide-ranging interlock function for the I&C Systems, including both Rod Control and Process Instrumentation. This includes inputs to the Reactor Safety System, Rod-Run-In System, Rod Withdrawal Prohibit circuit, Control Rod Drive Mechanisms, Auto Control Prohibit circuit and Range Switch 1S2. (Reference 7.11)

The shim and regulating blades are withdrawn or inserted manually by three (3)-position ("IN-NORMAL-OUT") switches located on the Reactor Control Console. The switches are spring return to the 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:

- Master Control Switch (1S1) in the "ON" position;

- No Nuclear Instrument anomaly;
- Shim rods bottomed and in contact with their electromagnets;
- Source range level indication greater than 20 cps or intermediate range level recorder indication greater than 2 x 10^{-5}% power; and
- Thermal Column door closed.

When all of the foregoing conditions have been satisfied, the Reactor Scram System and Rod Run-In System Trip Actuator Amplifiers (TAAs) may be reset and the shim blades withdrawn. There is no interlock or prohibit circuit which prevents the shim 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:

- Switch (1S1) is in the "TEST" position; and
- Power to the control rod electromagnets is secured (relays 2K20 and 2K21 are de-energized). (Reference 7.11)

7.5.3.2 Rod Withdrawal Prohibit Circuit Technical Specifications

The objective of the Limiting Conditions for Operations (LCO) Technical Specifications on Rod Withdrawal Prohibit Circuit is to reasonably assure proper operation of the Reactor Control System, avoiding conditions which could jeopardize the integrity of the fuel element cladding or endanger personnel health and safety. Technical Specification 3.2.h.1 provides this assurance for the Rod Withdrawal Prohibit Circuit.

The Surveillance Technical Specifications on the Rod Withdrawal Prohibit Circuit is to reasonably assure proper operation of the Reactor Control System. Technical Specification 4.2.k provides this assurance for the Rod Withdrawal Prohibit Circuit.

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 (WRM) 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 WRM is described in greater detail in Section 7.4.4, Wide Range Neutron Flux Monitor.

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 zero (0) to + ten (10) VDC signal, corresponding to zero (0) to 125% reactor power level (+8.0 VDC = 100% power). The output from the power schedule potentiometer is a zero (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 (4) 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 WRM recorder;
3. The regulating blade position greater than 60% withdrawn ("REG BLADE 60% WITHDRAWN" annunciator lit); and
4. The Range Selector Switch (1S2) in the 5-kW red scale position or above.

After the above four (4) conditions have been met, reactor control is placed in the automatic mode by depressing a push button switch (S2-2) 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:

- Control mode transferred to manual (push button S1-2 depressed);
- Regulating Rod Control Switch (1S5) actuated;
- Regulating rod inserted to the 10% withdrawn position;
- A reactor scram or rod run-in; or
- Indicated reactor power reaching a level less than the "Auto Control Prohibit" set point on the WRM recorder. (Reference 7.11)

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.

7.5.4.2 Automatic Control Prohibit Technical Specifications

The objective of the Limiting Conditions for Operations (LCO) Technical Specifications on Automatic Control Prohibit Circuit is to reasonably assure proper operation of the Reactor Control System, avoiding conditions which could jeopardize the integrity of the fuel element cladding or endanger personnel health and safety. Technical Specification 3.2.h.2 provides this assurance for the Automatic Control Prohibit Circuit.

The Surveillance Technical Specifications on the Automatic Control Prohibit Circuit is to reasonably assure proper operation of the Reactor Control System. Technical Specification 4.2.k provides this assurance for the Automatic Control Prohibit Circuit.

7.5.5 Rod Run-In System

The Rod Run-In System is designed to initiate the automatic insertion of the shim blades at a controlled rate should a monitored parameter exceed a predetermined value. This system is not part of the Reactor Scram System, however, it does provide a protective function by introducing shim blade insertion to terminate a transient prior to actuating a Reactor Scram 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 the "ROD RUN-IN 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 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 (1) of the four (4) devices in Board 'A' (1-out-of-4) or any one of the three (3) devices in Board B (1-out-of-3) 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 two (2) inches per minute (5.08 cm per minute). The Rod Run-In System and all of the conditions which will initiate a rod run-in are shown in Figure 7.5.

TABLE 7-6
ROD RUN-INS

1	Manual	6	Regulating Rod <= 10% Withdrawal
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

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-out-of-4)		**Non-Coincidence Logic Board B (1-out-of-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	-	-

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 Scram System TAAs must be reset prior to resetting the rod run-in TAA (relays 2K20 and 2K21 must be energized). (Reference 7.11)

The rod run-in NCLU and TAA are similar to the ones used in the Reactor Scram System. Their operation is described in Sections 7.7.2.1, "Non-Coincidence Logic Units" and 7.7.2.2, "Trip Actuator Amplifiers."

7.5.5.1 Rod Run-In System Technical Specifications

The objective of the Limiting Conditions for Operations (LCO) Technical Specifications on the Rod Run-In System is to reasonably assure proper operation of the Reactor Control System, avoiding conditions which could jeopardize the integrity of the fuel element cladding or endanger personnel health and safety, and to specify the minimum number of Reactor Scram System instrument channels that must be operable for safe reactor operation. Technical Specification 3.2.f provides this assurance for the Rod Run-In System.

The objective of the Surveillance Technical Specifications on the Rod Run-In System is to reasonably assure proper operation of the Reactor Control System instrument channels. Technical Specification 4.2.h provides this assurance for the Rod Run-In 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; and
- Security against tampering or inadvertent alteration by use of tiered password control.

The RPI System consists of the following major components: five (5) encoder transducers, one (1) mounted on each control rod drive mechanism; a chassis with installed hardware; and a remote Operator Display Assembly (ODA).

Signals from the encoders are converted to relative rod positions by the Programmable Logic Controller (PLC) in the RPI chassis. The chassis, mounted in the Instrument Panel, contains an AC to DC power supply, circuit breaker, a network switch for module communication, a Human Machine Interface (HMI), and a PLC consisting of the following components: a fieldbus logic controller, five (5) synchronous serial interface encoder input modules, one (1) digital input module, one (1) digital output module, and an end module. All inputs for the system are through either the HMI on the front panel of the chassis, or the HMI serving as the ODA.

The ODA is a second HMI identical in every way to the front panel of the chassis. The ODA is connected to the chassis through a communication cable and a power cable. The power for the ODA is from the power supply in the main rack mounted chassis.

The RPI System consists of three (3) main screens. Along the bottom of each screen are three (3) menu navigation buttons:

1. Alarms: The alarms screen shows a log of all faults and alarm situations. This log requires a user password to clear the alarm log.
2. Auto Run: The default screen is the "AUTO RUN" screen which displays the five (5) rod positions. It will also display calculated rod travel speed when enabled.
3. Setup: The setup screen requires a user login to access. A User login will allow the zero (0) position of the control rods to be set. The remainder of the settings require an Admin login.

The system defaults to the "AUTO RUN" screen after a timeout of five (5) minutes in the setup menu. (Reference 7.17)

7.6 Process Instrumentation and Control System

7.6.1 Introduction

The Process I&C 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 Scram System. Generally, the Process I&C 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 I&C System is shown in Figures 7.7, 7.8, and 7.9.

Master Control Switch (1S1) provides a wide-ranging interlock function for the I&C Systems, including Process Instrumentation. In the Process Instrumentation System, switch 1S1 provides process control interlocks in the form of a "test permissive" function for nearly all remotely operated process system valves. A typical valve control scheme allows the valve to be manually operated when switch 1S1 is in the "Test" position and automatically operated when switch 1S1 is in the "Off" or "On" position. (Reference 7.11)

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):

- Reactor Inlet (T_c) - TE 901A;
- Reactor Outlet (T_h) - TE 901B;
- Primary Coolant Heat Exchanger Outlet (HX503A) - TE 980A;
- Primary Coolant Heat Exchanger Outlet (HX503B) - TE 980B;
- In-Pool Heat Exchanger (HX505) Inlet - TE 950A; and
- In-Pool Heat Exchanger (HX505) Outlet - TE 950B.

Reactor inlet and outlet temperatures are recorded on a Yokogawa paperless recorder mounted on the Instrument Panel which transfers the data in a server for archiving. (Reference 7.17) The low-level DC millivolt output signal produced by the RTD is converted to a proportional DC milliampere signal (10 to 50 *m*A) by an RTD Transmitter. The output signal from the RTD Transmitter for TE 901A is directed to an Adder-Subtractor Module (ASM) and Yokogawa paperless recorder. (Reference 7.17) 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 Yokogawa paperless recorder mounted on the Instrument Panel. The recorder uses an analog computer circuit to provide the Reactor Operator with a digital indication of reactor power level in megawatts (MW). The chart recorder will display an

additional trend line on a loss of electrical power to the primary coolant circulation pumps. A pair of relays are connected to the control circuits for primary coolant circulation pumps P501A and P501B that will provide a real time indication of electrical power to the pumps. This feature assists in data collection for a primary coolant system flow coast down. (Reference 7.15) 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 Scram 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 Scram 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 (20.32-cm) 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):

- Pool Coolant Heat Exchanger (HX521) Inlet (T_h) - TE 901C; and
- Pool Coolant Heat Exchanger (HX521) Outlet (T_c) - TE 901D.

Pool Coolant heat exchanger inlet and outlet temperatures are recorded on a Yokogawa paperless recorder mounted on the Instrument Panel which transfers the data to a server for archiving. (Reference 7.17) 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 Yokogawa paperless recorder. (Reference 7.17) The ASM provides a differential temperature input signal to the Yokogawa paperless recorder displaying reactor power. 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 (15.24-cm) 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.

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):

- Reactor Core Outlet - PT 944A & PT 944B; and
- 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 a Dual Alarm Unit and meter. If Primary Coolant pressure decreases to 90% of the normal operating value, a reactor scram is initiated. The Dual Alarm Unit for PT 944A (920F) de-energizes relay 2K13 which opens a contact in the process input string to E4A of the Reactor Scram System NCLUs, thereby interrupting power to the control blade electromagnets. The Dual Alarm Unit for PT 944B (920H) 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:

- Primary Coolant Circulation Pumps 501A and 501B will stop;
- Primary Coolant Isolation Valves V507A and V507B will close;
- Anti-Siphon System Isolation Valves V543A and V543B will open; and
- "REATOR LOOP LO PRESS SCRAM" annunciator alarm will be initiated.

Separate meter units 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 a dual alarm unit. If Primary Coolant pressure decreases to 90% of the normal operating value, a reactor scram and a "REACTOR LOOP LO PRESS SCRAM" annunciator alarm is initiated. The Dual Alarm Unit for PT 943 (EP920J) opens a contact in the process input string to E3B of the Reactor Scram System NCLUs, thereby interrupting power to the control blade electromagnets. (Reference 7.18) The separate meter unit 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):

- Downstream of Primary Coolant Heat Exchanger (HX503A) - FT 912A and FT 912E;
- Downstream of Primary Coolant Heat Exchanger (HX503B) - FT 912G and FT 912H; and
- Differential Pressure across the Reactor Core - DPS 929.

Primary Coolant System flow is recorded on two (2) Yokogawa paperless recorders mounted on the Instrument Panel which transfers the data to a server for archiving. (Reference 7.17) Differential pressure across orifice plates (Flow Element 913A and 913B) located downstream of HX503A and HX503B respectively are dual tapped. Flow of HX503A is monitored by flow element 913A, flow transmitters 912A and 912E, square root transmitters 919A and 919E, electronic alarms 920A and 920C, and recorder 915A/B. Flow of heat exchanger 503B is monitored by flow element 913B, flow transmitters 912G and 912H, square root transmitters 919G and 919H, electronic alarms 920E and 920G, and recorder 915G/H. The output signal (10 to 50 *m*A) generated by each flow transmitter is directed to a Square Root Transmitter which provides a linear output signal for the Yokogawa paperless recorder and a Dual Alarm Unit. (Reference 7.17) 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 is initiated. The Dual Alarm Unit for FT 912A opens a contact (K30-2) in the process input string to E4A of the Reactor Scram 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. The Dual Alarm Unit for FT 912G opens a contact in the process input string to E4A of the NCLUs. The Dual Alarm Unit for FT 912H opens a contact in the process input string to E3B of the NCLUs. The Alarm Units for FT 912A and 912H will also de-energize relays 2K24 and 2K26, which will cause the following actions to occur:

- ”REACTOR LOOP LO FLOW SCRAM” annunciator alarm is initiated; and
- In-Pool Heat Exchanger Isolation Valves V546A and V546B open.

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 Coolant low flow scrams. The output signal (5 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 Scram 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:

- “PLI OR FIRST SCRAM” annunciator alarm will be initiated; and
- 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, 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:

- Reactor will scram (relays 1K13 and 1K26 de-energize);
- Primary Coolant Circulation Pumps 501A and 501B will stop;
- In-Pool Heat Exchanger Isolation Valves V546A and V546B will open;
- Anti-Siphon System Isolation Valves V543A and V543B will open;
- Surge Line Isolation Valve V527C will close;
- “PLI OR FIRST SCRAM” annunciator alarm will be initiated; and
- “PUMP 501A/B ON VALVE 507A/B CLOSED” annunciator alarm will be initiated. (Reference 7.11)

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 Yokogawa paperless recorder mounted on the Instrument Panel which transfers the data to a server for archiving. (Reference 7.17) 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 Yokogawa paperless recorder. (Reference 7.17) In addition to providing flow indication, the recorder initiates a “REACTOR OR POOL LOOP DEMIN 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):

- Downstream of Pool Coolant Heat Exchanger (HX521) - FT 912D & FT 912F; and
- Differential Pressure across the Reactor Pool Reflector - PT 917.

Pool Coolant System Flow is recorded on a Yokogawa paperless recorder mounted on the Instrument Panel which transfers the data to a server for archiving (Ref. 7.17). 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 recorder. (Reference 7.17) 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 is initiated. The Dual Alarm Unit for FT 912D opens a contact (K37-2) in the process input string to E3B of the Reactor Scram 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 a Yokogawa paperless recorder and Dual Alarm 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 is initiated. The Dual Alarm Unit opens contacts in the process input string to E3B of the Reactor Scram System NCLUs, thereby interrupting power to the control blade electromagnets. The Yokogawa paperless recorder also provides reactor pool reflector differential pressure indication on the Instrument Panel. (Reference 7. 14)

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:

- Reactor will scram (a contact in the process input string to E4A opens);
- Pool Coolant Circulation Pumps 508A and 508B will stop;
- Pool Coolant Demineralizer Pump 513B will stop; and
- "POOL LOOP VALVE 509 OFF OPEN SCRAM" annunciator alarm will be initiated. (Reference 7.11)

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 Yokogawa paperless recorder mounted on the Instrument Panel which transfers the data t a server for archiving (Ref. 7.17). 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 Transmitter which provides a linear output signal for the Yokogawa paperless recorder. In addition to providing flow indication, the recorder initiates a "REACTOR OR POOL LOOP DEMIN 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 (6) 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:

- At 105% of normal pressure, PS 940 provides the signal to open Nitrogen Vent Valve V545 and vent the Pressurizer to the Exhaust System;
- At 110% of normal pressure, PS 946 initiates a "PRESSURIZER HI PRESS" annunciator alarm; and
- 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 Scram 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:

- At 95% of normal pressure, PS 941 provides the signal to open Nitrogen Addition Valve V526 and emit nitrogen into the Pressurizer;
- At 90% of normal pressure, PS 945 initiates a "PRESSURIZER LO PRESS" annunciator alarm and PS 938 initiates a reactor scram by opening a contact (K25-2) in the process input string to E4A of the Reactor Scram 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 (3) Level Controllers (LCs) and one (1) 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:

- At approximately six (6) inches (15.24 cm) above center line, LC 936 signals Water Drain Valve V527A to open and drain water from the Pressurizer; and

- 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:

- At approximately seven (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;
- At approximately 11 inches below center line, LC 935 initiates a "PRESSURIZER WATER LO LEVEL" annunciator alarm; and
- 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 Scram System NCLUs, thereby interrupting power to the control blade electromagnets. In addition, a "PRESSURIZER WATER LO LEVEL" 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 Yokogawa paperless recorder 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.

7.6.6 Power Level Interlock

The Power Level Interlock (PLI) circuit ensures proper alignment of Primary and Pool Coolant Systems to the associated mode of operation (50KW/5MW/10MW) as selected by the Power Level Selector Switch (1S8). The Power Level Interlock scram provides assurance that the reactor cannot be operated with a power level greater than that authorized for the mode of operation selected on PLI switch 1S8. The PLI scram also provides the interlocks to ensure that the reactor cannot be operated in Mode I with a primary or pool coolant low flow scram bypassed (2S40 and 2S41). Switch 1S8 is interlocked with the bypass switched (2S40 and 2S41) to prevent reactor operation with a bypass function unless the reactor is lined up for the proper mode of operation.

If core differential pressure decreases to approximately 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 PLI 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:

- Reactor will scram;
- "PLI OR FIRST SCRAM" annunciator alarm will be initiated; and
- In-Pool Heat Exchanger Isolation Valves V546A and V546B will open. (Reference 7.11)

7.6.7 Process Instrumentation and Control System Technical Specifications

The objective of the Limiting Conditions for Operations (LCO) on the Process I&C System is to ensure that sufficient reliable information is present to the Operator to assure safe operation of the reactor. Technical Specification 3.5.b provides this assurance for the Process I&C System.

The objective of the Surveillance Technical Specification on the Process I&C System is to reasonably assure proper operation of the Reactor Instrumentation System. Technical Specification 4.5.b provides this assurance for the Process I&C System.

7.7 Reactor Scram System

7.7.1 Introduction

The Reactor Scram 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 the "SCRAM" push button (1S10) on the reactor control console. The reactor scrams are listed in Table 7-8.

TABLE 7-8
REACTOR SCRAMS

1	Manual	10	Reflector High-Low Differential Pressure
2	Channel 2 & 3 Short Period	11	Pool Valve 509 Off Open
3	Reactor Loop Low Flow	12	Building Plenum & Bridge High Activity
4	Reactor Loop High Temperature	13	Evacuation or Isolation
5	Pressurizer Low Level	14	Reactor Loop Low Pressure
6	Reactor Pool Below Refuel Level	15	Pool Loop Low Flow
7	Channel 4, 5, & 6 High Power	16	FIRST
8	Power Level Interlock	17	Primary Coolant Valve Off Open
9	Pressurizer High Pressure		

The protection channels and protective responses are sufficient to ensure that no Safety Limit, Limiting Safety System Settings, or Reactor Scram 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 Scram 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.

Table 7-8, "Reactor Scrams," is a general description of the scrams associated with the reactor safety system and the terminology designated on the annunciator panel.

Each scram is itemized in TS 3.2.g with the exception of Building Plenum & Bridge High Activity. Building Plenum & Bridge High Activity is included in TS 3.7.a as required reactor instrumentation, but it is also an input into the reactor isolation system, and functions to input a reactor isolation scram.

Table 7-8, Reactor Scrams	Associated TS 3.2.g
Manual	Manual Scram
Channel 2 & 3 Short Period	Reactor Period
Reactor Loop Low Flow	Primary Coolant Flow
Reactor Loop High Temp	Reactor Out Water Temperature
Pressurizer Low Level	Pressurizer Low Water Level
Reactor Pool Below Refuel Level	Pool Low Water Level
Channel 4, 5, & 6 High Power	High Power Level
Power Level Interlock	Power Level Interlock
Pressurizer High Pressure	Pressurizer High Pressure
Reflector High-Low Differential Pressure	Differential Pressure Across the Reflector
Pool Valve 509 Off Open	Pool Coolant Isolation Valve 509 Off Open Position
Evacuation or Isolation	Reactor Isolation and Facility Evacuation
Reactor Loop Low Pressure	Primary Coolant Low Pressure
Pool Loop Low Flow	Pool Coolant Flow
Primary Valve Off Open	Primary Coolant Valves Off Open Position
Center Test Hole Device (FIRST)	Center Test Hole

Table 7-8, Reactor Scrams	Associated TS 3.7.a
Bldg Plenum & Bridge High Activity	Reactor Containment Building Exhaust Plenum Radiation Monitor

7.7.2 Description

The Reactor Scram System consists of two (2) NCLUs and two (2) 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 (2) control blade electromagnets. Should an input signal to either NCLU 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 switch (1S15) or the facility evacuation switch (1S16) (Section 7.8). Figure 7.9 shows the Reactor Scram System, and the conditions which initiate a reactor scram.

The signal inputs from the process instruments are discussed in detail in Section 7.6.

7.7.2.1 Non-Coincidence Logic Units

Each NCLU consists of three (3) interconnected component boards: two (2) non-coincidence logic (A and B) and one (1) 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 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 zero (0) volts, the base-emitter circuits of the logic unit transistors are reverse-biased, causing the output voltage to approach zero (0) volts. This output is applied to one (1) of the two (2) 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 two (2) volts (safe condition). This is accomplished by the redundancy designed into the circuit.

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-out-of-4)		**Non-Coincidence Logic Board B (1-out-of-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-out-of-4)		**Non-Coincidence Logic Board B (1-out-of-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 NCLU is also referred to as the "yellow leg" of the Reactor Scram System.
b. This process instrumentation input string to the NCLU is also referred to as the "green leg" of the Reactor Scram System.

7.7.2.2 Trip Actuator Amplifiers

Each TAA consists of one (1) power supply and two (2) 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 (2) 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, scramming the reactor.

In addition to interrupting current to the electromagnets, a loss of TAA output current (approximately zero (0) amperes) will de-energize relays 2K20 and 2K21. De-energizing either relay causes the following actions to occur:

- The Rod Run-In System is activated (supply power to the rod run-in TAA is interrupted); and
- 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.

The reactor safety system TAAs must be reset prior to resetting the rod-run-in TAA (relays 2K20 and 2K21 must be energized.)

A malfunction in one (1) TAA could result in, at most, the failure to turn off the current to two (2) control blade electromagnets. The other two (2) control blades would drop and successfully shut down the reactor.

7.7.3 Reactor Scram System Technical Specifications

The objective of Limiting Safety System Settings Technical Specification on the Reactor Scram System is to assure that automatic protective action is initiated to prevent a Safety Limit from being exceeded. Technical Specification 2.2 provides this assurance for the Reactor Scram System.

The objective of the Limiting Conditions for Operations (LCO) Technical Specifications on the Reactor Scram System is to specify the minimum number of Reactor Scram System instrument channels that must be operable for safe reactor operation. Technical Specification 3.2.g provides this assurance for the Reactor Scram System.

The Surveillance Technical Specifications on the Reactor Scram System is to reasonably assure proper operation of the Reactor Scram System instrument channels. Technical Specifications 4.2.i, 4.2.j and 4.2.m provide this assurance for the Reactor Scram 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), the Fuel Handling Accident FHA), the Fuel Failure Accident during Reactor Operation (FFA), 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, Standards for Protection Against Radiation. The Engineered Safety Features Actuation Systems' designs give reasonable assurance of reliable operation, if required. The two (2) systems which are identified as ESFs for the University of Missouri Research Reactor (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, the FHA, the FFA, and the 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 210). The radiation monitors are described in detail in Section 7.9, Radiation Monitoring Systems.

Manual or automatic actuation of the CAS causes the following actions to occur:

- Reactor will scram;
- All normally open Reactor Containment Building penetrations with automatic sealable closures will close;
- An audible alarm will sound throughout the Containment Building;
- A flashing light at the entrance to the Containment Building Personnel Airlock will illuminate; and
- "EVACUATION OR ISOLATION" annunciator alarm will be initiated.

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:

- All relays which energize to perform their intended function are paralleled by a second relay such that if one (1) fails to energize, the unaffected relay will operate as required;
- All relays which de-energize to perform their intended function are assumed to be fail-safe;
- Four (4) independent radiation detectors can initiate automatic isolation of the Reactor Containment Building, two (2) at each location (Reactor Pool Upper Bridge and Containment Building Ventilation Exhaust Plenum);
- The radiation monitors are isolated from each other such that the operation of one (1) does not affect the operation of another;
- The radiation monitors are powered by independent low voltage power supplies and are not subject to a single failure event; and
- 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 (2) 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 (1) 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:

- A reactor scram is initiated by the “green leg” of the Reactor Scram System by a process input string to E3B of the NCLUs;
- Parallel R2 series relays (R2A – R2F) and relay 2K2 in the CAS will de-energize; (Reference 7.15)
- Containment Building Ventilation Exhaust Plenum Backup Doors will close; and
- “BLDG AIR PLENUM AND HI ACTIVITY SCRAM” annunciator alarm will be initiated.

De-energizing the R2 series relays of the Containment Actuation System (CAS) will cause the following actions to occur: (Reference 7.15)

- Containment Building Ventilation Supply and Exhaust Duct Electric-Motor-Driven Isolation Doors 504 and 505 will close;
- Containment Building Hot Exhaust Line Isolation Valves 16A and 16B will close;
- The Reactor Isolation Horns will activate; and
- 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 (1) minute). This failure will de-energize relays 2K1A and 2K1B, preventing reactor operation with a failed radiation monitor.

The reactor containment building supply (SF-2) and return (RF-2) fans are interlocked such that securing SF-2 will automatically secure RF-2, preventing a negative overpressure condition from occurring within the containment building. However, operation of Sf-2 and RF-2 is completely independent of reactor operations. (Reference 7.11)

7.8.2.2 Manual Initiation

Isolation of the Reactor Containment Building can be manually initiated from either of two (2) locations: The Reactor Control Room or the Facility Lobby (Room 210). Manual initiation from the Reactor Control Room is performed by the actuation of either of two (2) switches located on the Reactor Control Console: Reactor Isolation Switch (1S15) and Facility Evacuation Switch (1S16).

Actuation of switch 1S15 will cause the following actions to occur:

- Relay 2K2 will de-energize and cause:
    - A reactor scram from the "yellow leg" of the Reactor Scram System by a process input string to E4A of the NCLUs; and
    - Initiation of the "EVACUATION OR ISOLATION SCRAM" annunciator alarm; and
- All actions caused by the R2 series relays de-energizing. (Reference 7.15)

Actuation of Switch 1S16 will cause the following actions to occur:

- Parallel relays R3A and R3B will de-energize and open contacts in the Containment Actuation System (CAS) thereby de-energizing relay 2K2 and the R2 series relays; and (Reference 7.15)
- 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 210).

Postulated emergencies necessitating the evacuation of all personnel from the Reactor Facility do not require instantaneous action and therefore, the actuation of the Facility Evacuation System is an administrative decision with no automatic trip function required.

7.8.2.3 Containment Actuation System Technical Specifications

The objective of the LCO Technical Specifications on the CAS is to ensure that Containment Integrity is maintained when required so that the health and safety of the general public is not endangered as a result of reactor operation. Technical Specification 3.4.c provides this assurance for the CAS.

The Surveillance Technical Specifications on the CAS is to reasonably assure proper operation of the Containment System. Technical Specification 4.4.c provides this assurance for the CAS.

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 Loss of Coolant Accident (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 (2) electronic Pressure Transmitters (PT 944A and PT 944B) located on the 12-inch (30.48-cm) 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:

- 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 Scram System NCLUs;
- Primary Coolant Circulation Pumps 501A and 501B will stop;
- Primary Coolant Isolation Valves 507A and 507B will close; and
- 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. The characteristics of this system providing redundancy and reliability are:

- All relays which de-energize to perform their intended function are assumed to be fail-safe;
- Redundant Primary Coolant System pressure monitors PT 944A and PT 944B, either of which will initiate operation of this system are provided; and
- Redundant relays 2K13 and 2K28, either of which will initiate operation of this system are provided.

At least two (2) major failures must occur before the Anti-Siphon Actuation System would be unable to perform its intended function. This is not a credible assumption. (Reference 7.11)

7.8.3.1 Anti-Siphon System Technical Specifications

The objective of the Limiting Conditions for Operations (LCO) Technical Specifications on the Anti-Siphon System is to protect the integrity of the fuel and to prevent the release of fission product radioisotopes and to ensure that sufficient reliable information is presented to the Operator to assure safe operation

of the reactor. Technical Specifications 3.3.a(1) and 3.5.b.2 provide this assurance for the Anti-Siphon System.

The Surveillance Technical Specifications on the Anti-Siphon System are to reasonably assure proper operation of the Anti-Siphon System. Technical Specifications 4.3.a(1) and 4.5.b provide this assurance for the Anti-Siphon System.

The objective of the Design Features Technical Specification on the Anti-Siphon System is to assure proper coolant for safe operation. Technical Specification 5.2.h provides this assurance for the Anti-Siphon System.

7.9 Radiation Monitoring Systems

7.9.1 Introduction

The installed systems at the University of Missouri Research Reactor (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.

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.

This section discusses the operating principles, designs, and the functional performance of the 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 (4)- or five (5)-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 7.10.(Reference 7.19)

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 Sr-90/Y-90 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 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 (6) channels of analog radiation meters and is the interconnecting tie point for those six (6) channels in addition to supplying the AC line voltage. Two (2) rack units are presently installed.

A remote indicator is mounted next to each detector assembly except for the two (2) 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 Area Radiation Monitoring System Technical Specifications

The objective of the Limiting Conditions for Operations (LCO) Technical Specifications on the Area Radiation Monitoring System (ARMS) is to assure that sufficient radiation monitoring information is available to the Reactor Operator during reactor operations and exposure to the public resulting from the radioactivity released from the Reactor Facility to the unrestricted environment will not exceed the

limits of 10 CFR 20, Standards for Protection Against Radiation. Technical Specification 3.7.a.3 provides this assurance for the ARMS.

The Surveillance Technical Specifications on the ARMS is to reasonably assure proper operation of the Radiation Monitoring Instrumentation. Technical Specifications 4.7.a and 4.7.b provide this assurance for the ARMS.

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 Area Radiation Monitoring System (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.

The Fuel Element Failure Monitoring System's high radiation annunciator alarm is tested prior to each reactor startup in accordance with the "Process Radiation Monitor" portion of the Long Form Startup Checksheet (FM-57). This checksheet implements operating procedure OP-RO-710, "Radiation Monitoring – Area Monitors" Section 4.3, where the "Reactor Coolant Loop Hi Activity" annunciator and associated circuitry is tested and the alarm set point is verified. (Reference 7.10)

7.9.3.2 Fuel Element Failure Monitoring System Technical Specifications

The objective of the Limiting Conditions for Operations (LCO) Technical Specifications on the Fuel Element Failure Monitoring System is to protect the integrity of the fuel and to prevent release of fission product radioisotopes. Technical Specification 3.3.b and 3.3.c provides this protection for the Primary Coolant System.

The Surveillance Technical Specifications on the Fuel Element Failure Monitoring System is to reasonably assure proper operation of the Primary Coolant System. Technical Specifications 4.3.b and 4.3.c provide this assurance for the Primary Coolant System.

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 (1) of the electronics channels of the Area Radiation Monitoring System (ARMS) where the signal is processed and displayed on a logarithmic scale analog meter in counts per minute (cpm). 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 sodium iodide detector 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 heat exchanger. It is operational and responsive for a wide range of energies which allows it to detect any activity present in the secondary coolant. The detector is roughly 2.5% efficient over the range of energies that would be expected during a pool/primary to secondary leak. The detector is situated in the coolant line such that it views an effective sample volume of 7.5 liters moving past the detector.

To detect an increase in Secondary Coolant Monitor levels would require these levels to increase by roughly 7000 cpm to reach the alarm set point. A maximum interval of two (2) hours is possible between log readings if the alarm set point is not reached prior to the log readings. Using the total volume of the Secondary Coolant System and the average total activity concentration of the primary coolant, a 7000 cpm rise in two (2) hours would occur at a leak rate of approximately 2.2 liters (0.58 gallons) per hour. At this leak rate, an increase of 100 cpm over the baseline would be reached in approximately 17 minutes and an increase of 2500 cpm would be reached in 43 minutes. This increase would likely be identified by the Control Room staff at some point between the two (2)-hour log data window.

The environmental consequences of this hypothetical leak would be less than 50% of the 10 CFR 20, Appendix B, Table 2 Effluent Limits. This determination is based upon a leak duration of the longest interval between logs (2 hours) and the concentrations of the three (3) most dose significant isotopes associated with primary coolant. Upon the discovery of the leak with the secondary coolant alarm, the activity concentration of the individual constituents of the contaminated secondary coolant water would be near the uncontrolled release limits for those isotopes and would not pose a dose threat to the general public.

7.9.4.2 Secondary Coolant Monitor Technical Specifications

The objective of the Limiting Conditions for Operations (LCO) Technical Specifications on the Secondary Coolant Monitoring System is to protect the integrity of the fuel and to prevent release of fission product radioisotopes. Technical Specification 3.3.e and 3.3.f provides this protection for the Secondary Coolant Monitoring System.

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 the 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, Standards for Protection Against Radiation.

The monitoring equipment of this system consists of a three (3)-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 (3) inches (7.62 cm) of lead (4π configuration). One (1)-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 Yokogawa paperless recorder mounted on the instrument panel in the Reactor Control Room which transfers the data to a server for archiving. (Reference 7.17) An audible and visual alarm alerts the Reactor Operator to high activity or abnormal air flow through the radiation detection equipment.

7.9.5.2 Off-Gas Radiation Monitoring System Technical Specifications

The objective of the Limiting Conditions for Operations (LCO) Technical Specifications on the Off-Gas Radiation Monitoring System is to assure that sufficient radiation monitoring information is available to the Reactor Operator during reactor operations and exposure to the public resulting from the radioactivity released from the Reactor Facility to the unrestricted environment will not exceed the limits of 10 CFR 20, Standards for Protection Against Radiation. Technical Specification 3.7.a.3 provides this assurance for the Off-Gas Radiation Monitoring System.

The Surveillance Technical Specifications on the Off-Gas Radiation Monitoring System is to reasonably assure proper operation of the Radiation Monitoring Instrumentation. Technical Specifications 4.7.a and 4.7.b provide this assurance for the Off-Gas Radiation Monitoring System.

7.10 Center Test Hole Instrument Channel

7.10.1 Introduction

The center test hole instrument channels, called the Flux-Trap Irradiations Reactivity Safety Trip (FIRST), detect the position of the center test hole canister, or strainer, when installed. If these channels do not detect that the center test hole canister, or strainer, is in the secured and latched position, a scram signal is initiated. The FIRST provides protection against an excessive reactivity step insertion in the flux trap region.

7.10.2 Description

The FIRST is two (2) instrument channels which detect the positions of the center test hole canister and convert that position into a logic input to the Reactor Scram System. The FIRST is comprised of the following components: center test hole canister wear ring, sensor switches, support rig, relays and power supply, Reactor Scram System interface, and FIRST bypass switches.

The wear ring is the top-most component of the center test hole canister, and its position is fixed with respect to both the canister and the inner pressure vessel when the center test hole canister is installed. This serves as the reference point for contact with the sensor switches. The sensor switches are waterproofed plunger-type switches with molded cables to ensure all connections are made above the water line. The height of the switches are set to allow roughly 0.125 inches (3.18 mm) of vertical travel prior to initiation of a scram signal. The reactivity effect of this amount of removal prior to initiation of the scram is much less than the reactivity associated with the Rapid Step Insertion of Positive Reactivity accident analyzed in Chapter 13, “Accident Analysis”. The sensor switches are supported and aligned by the support rig which is held in place by the lifting ears of the pressure vessel head. The original design and safety evaluation of the FIRST is documented in Reference 7.12.

The switches are activated by movement of the center test hole canister. The switch contacts provide power to the coils of relays K-61 and K-62. The normally open contacts of these K-relays provide an additional series contact in each of the Yellow and Green Legs of the Reactor Scram System. An additional normally open contact in the isolation relay provides input to the “PLI OR FIRST Scram” annunciation.

The FIRST also consists of two (2) bypass switches. These switches provide two (2) functions: (1) allow testing and troubleshooting capability, and (2) provide the use of either of two (2) methodologies (canister worth included or canister worth excluded, TS 3.8.b) for calculating the reactivity contribution of experiments in the flux trap region. The bypass switches are located on the Reactor Console and directly jumper both FIRST sensor switches to energize relay coils on K-61 and K-62. Operation of these switches are administratively controlled by the Reactor Manager.

The following provides MURR’s design evaluation with regard to IEEE Standard 279-1971, “Criteria for Protection Systems for Nuclear Power Generating Stations, clarifying meeting the applicable criteria or

meeting the intent of IEEE Standard 279-1971. Some of the requirements are not applicable to any part of MURR's reactor safety system, whereas others have limited applicability. For each of the design requirements, a short discussion clarifies the applicability of the requirement and provides the justification for said applicability. (Reference 7.9)

The descriptions were compared to the updated standard contained in IEEE 603-2009, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations" (the replacement for IEEE 279). With respect to single failure criteria, completion of protective action, appropriateness for the environment in which the device is operated, capability for testing, and operation and independence, the FIRST device meets the guidance of IEEE 603-2009.

IEEE Standard 279-1971 proposes the following twenty-two (22) requirements:

1. General Functional Requirement

This requirement is satisfied by the combined physical channel arrangement and periodic testing. The channel arrangement uses precision position sensor switches to detect a small change in the position of the center test hole canister, and using standardized relays, converts that change to one or more logic inputs to the reactor safety system. Each component chosen has been tested in its intended working environment or has a proven reliable track record at MURR to meet the full range of transient and steady-state conditions anticipated. Credible malfunctions have been considered and are discussed in more detail below. Performance requirements ensure that the instrument channels provide an accurate indication of this small change in position to ensure automatic protective action is completed prior to exceeding step reactivity insertion limits.

2. Single Failure Criterion

This requirement is satisfied primarily by the provision of redundant instrument channels to provide the protective function. Efforts to provide channel independence, and consideration of any credible single failure, provide additional confidence that this requirement is met. The worst-case credible single failure of these instrument channels would be an unintended closed-circuit condition that masks a true detected condition. This single failure could occur in many different locations within the physical channel but would be minimized by efforts to provide channel independence. A more detailed discussion of those locations is discussed in the channel independence section.

3. Quality of Components

This requirement is satisfied by the component specifications with regard to materials compatibility, reliable operation, availability of spare parts, and access to components for inspection, testing and calibration.

4. Equipment Qualification

This requirement is satisfied by the test data gained through previous bench testing and the ongoing in-situ testing of the temporary mock instrument channels. With the exception of actuating the reactor safety system, the mock instrument channels and their components performed reliably in their intended working environment and continued to produce timely and accurate channel responses within the

proposed operating range. The means of actuating the safety system uses existing proven components with a high degree of operational reliability.

5. Channel Integrity

This requirement is satisfied by the consideration of the extremes of applicable conditions. The instrument channels are susceptible only to the same extremes of conditions as the remainder of the reactor safety system. These extremes, if produced by the reactor systems, or if produced by external environmental factors would initiate and complete a safe reactor shutdown by other means (including Manual SCRAM) prior to impacting the operability of these instrument channels.

6. Channel Independence

This requirement is effectively satisfied by the provision of two independent instrument channels from detection through reactor safety system actuation. While the two instrument channels are fully independent electrically, physical independence is only provided through effective alternate means.

Starting at detection, the switches, though separate, share a common support rig. No credible single failure could be envisioned that would allow both switches to close. Two separate cable bundles are routed together through the pool from the support rig to the pool bridge mezzanine level. This arrangement reduces the likelihood of physically interfering with the cable bundle during routine pool operations. Two separate cable bundles are routed together through a dedicated conduit through the Instrument Cabinet and Reactor Console to the bypass switch. This co-routing of safety system instrument channel cable runs is not a common practice at MURR but has been accepted where the conduits are physically protected by a concrete chase or other structural support.

The bypass switch provides an effective mechanical separation of the channels by its proven reliable track record. Each contact block is a separate physical component of the key switch, so electrical independence remains intact. The common failure point then is the mechanical operation of the key tumbler. These standardized key switches have a long track record at MURR, with no record of failure or replacement due to reliability concerns. Mechanically, they are of sound construction and not prone to unintended movement without the key properly fitted. A high level of administrative control is utilized in the operation of this bypass switch, such that the key will not be present or accessible unless the Reactor Manager has granted permission for its use and directed the channels to be placed in bypass. No credible single event, physical or administrative, could result in a reduction of channel independence with regard to the bypass switch.

Two separate cable bundles are routed together from the bypass switch to the two independent relays in the safety system K-relay drawer. The co-routing of cable runs is accepted in this case due to the cable routing being physically protected.

The independent relays provide the channel inputs to the reactor safety system via a single series contact in each of the “Yellow” and “Green” input legs to the NCLUs. This system arrangement and the remainder of the safety system is in accordance with the MURR convention for safety system channels.

7. Control and Protection System Interaction

This requirement is not applicable to these instrument channels. The instrument channels provide only protective system inputs and annunciator indication of those inputs. No control function is provided by these instrument channels. Therefore, none of the remaining criteria of this requirement (isolation devices, single random failure, or multiple failures from a credible single event) are applicable to these instrument channels.

8. Derivation of System Inputs

This requirement is effectively satisfied by directly measuring the position of the center test hole canister. While reactivity is the variable to be controlled, only indirect means are available to measure this variable. The proposed instrument channels provide a positive protective action in the event that a very small amount of motion is measured. This tight range between normal conditions and actuation conditions ensures that a timely response is generated in the safety system before any significant reactivity change can occur in the core due to unintended movement of the center test hole canister.

9. Capability for Sensor Checks

This requirement is not applicable to these instrument channels. The proposed instrument channels produce a logical state output, which through the 1/N logic system, will either continue to operate or SCRAM the reactor. Thus, only a single state of the instrument channel operability can be verified during reactor operation.

10. Capability for Test and Calibration

This requirement is satisfied by the performance of the semiannual surveillance Compliance Check Procedure CP-36, quarterly physical inspection Preventive Maintenance Procedure RX-Q2, and weekly operability checks. Each of these tests provide specific detailed instructions for their respective purposes and collectively ensure that the entire system can perform its intended safety function in the intended manner. No testing interval is less than the normal time interval between shutdowns, so no provision will be allowed for testing during reactor operation.

11. Channel Bypass or Removal from Operation

This requirement is not applicable to these instrument channels. The instrument channels are either in service or in bypass during reactor operation. This condition is determined administratively by the Reactor Manager. No change in the bypass status of these instrument channels is allowed during reactor operation.

12. Operating Bypasses

This requirement is not applicable to these instrument channels. The proposed instrument channels are either in service or in bypass during reactor operation. This condition is determined administratively by the Reactor Manager. No permissive conditions are implemented for these instrument channels.

13. Indication of Bypasses

This requirement is satisfied by the combination of the key-type bypass switch, its location on the Reactor Console, and the high level of administrative control required to operate the bypass switch. The

bypass switch for these instrument channels is located in the Reactor Console adjacent to the existing bypass switches. The switch is a standardized key-type switch. The presence of the key in the switch provides the operator a direct and unmistakable indication that these instrument channels are bypassed. Due to the high level of administrative control required for use, key presence and not key position is the first indication necessary to positively inform the operator of the bypass condition.

14. Access to Means for Bypassing

This requirement is satisfied by the administrative control of the key used to operate the bypass switch. Reactor Manager's permission is required to operate the key switch. The key is kept in a locked key box, which is accessible only by licensed operators. In accordance with MURR conventions, the key is clearly identified as a key requiring Reactor Manager's permission to remove from the locked key box.

15. Multiple Set Points

This requirement is not applicable to these instrument channels. The instrument channels do not offer different modes of operation or sets of operating conditions. There are no less-restrictive set points associated with these instrument channels.

16. Completion of Protective Action Once Initiated

This requirement is satisfied at the system level by the existing NCLUs and the TAAs. The instrument channels provide a logic input to these safety system major components.

17. Manual Initiation

This requirement is satisfied at the system level by the existing Manual SCRAM Switch. This switch remains independent of all other safety system actuation signals.

18. Access to Set Point Adjustments

Set point adjustment, calibration adjustment and test points are combined in these instrument channels and represented by the sensor switches. Access to these switches is administratively controlled via performance of semiannual surveillance Compliance Check Procedure, quarterly physical inspection Preventive Maintenance Procedure, and weekly operability checks.

19. Identification of Protective Actions

This requirement is satisfied by the Annunciator window and the Safety System Monitoring Circuit. Actuation of either of these instrument channels will result in an Annunciator window illumination, and a Safety System Monitoring Circuit indication that provide unmistakable indication and identification of the responsible instrument channel and the resulting SCRAM to the operator.

20. Information Read-Out

Actuation of either of these instrument channels results in an Annunciator window illumination, and a Safety System Monitoring Circuit indication that provide unmistakable indication without providing anomalous or confusing indications to the operator.

21. System Repair

Provisions for identifying malfunctioning components are provided by the semiannual surveillance Compliance Check Procedure, quarterly physical inspection Preventive Maintenance Procedure, and weekly operability checks. The mechanical components of this system can be removed from the reactor pressure vessel head and relocated to the pool bridge deck for inspection and repair. The electrical components of this system can be readily accessed for inspection and repair. Replacement parts are maintained in Reactor Operations Spare Parts, including a complete support rig with switches installed.

22. Identification

The visible channel components including cable runs and conduits will be identified in accordance with the existing MURR convention for reactor safety system channels. Safety System Yellow Leg channel components are identified with yellow tape or other indelible marking. Safety System Green Leg channel components are identified with green tape or other indelible markings.

7.10.3 Center Test Hole Instrument Channel Technical Specifications

The objective of the Limiting Conditions for Operations (LCO) Technical Specification on the Center Test Hole Instrument Channel is to reasonable assure proper operation of the Reactor Control System, avoiding conditions which could jeopardize the integrity of the fuel element cladding or endanger personnel health and safety, and to specify the minimum number of Reactor Scram System instrument channels that must be operable for safe reactor operation. Technical Specifications 3.2.g.21, 3.8.b and 3.8.u provide this assurance for the Center Test Hole Instrument Channel.

The objective of the Surveillance Technical Specifications on the Center Test Hole Instrument Channel (3.2.g) is to reasonably assure proper operations of the Reactor Control System and the Reactor Scram System instrument channels. Technical Specification 4.2.j provides this assurance for the Center Test Hole Instrument Channel:

The objective of the Surveillance Technical Specifications (3.8.b and 3.8.u) is to prevent the conduct of experiments which may damage the reactor or release excessive amounts of radioactive materials as a result of experiment failure. Technical Specification 4.8 provides this prevention.

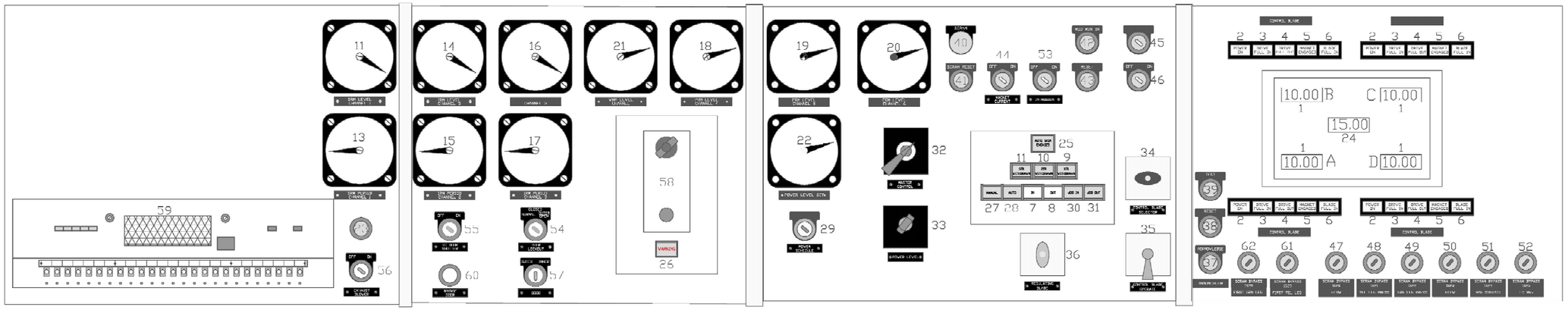


FIGURE 7.11 – REACTOR CONTROL CONSOLE LAYOUT

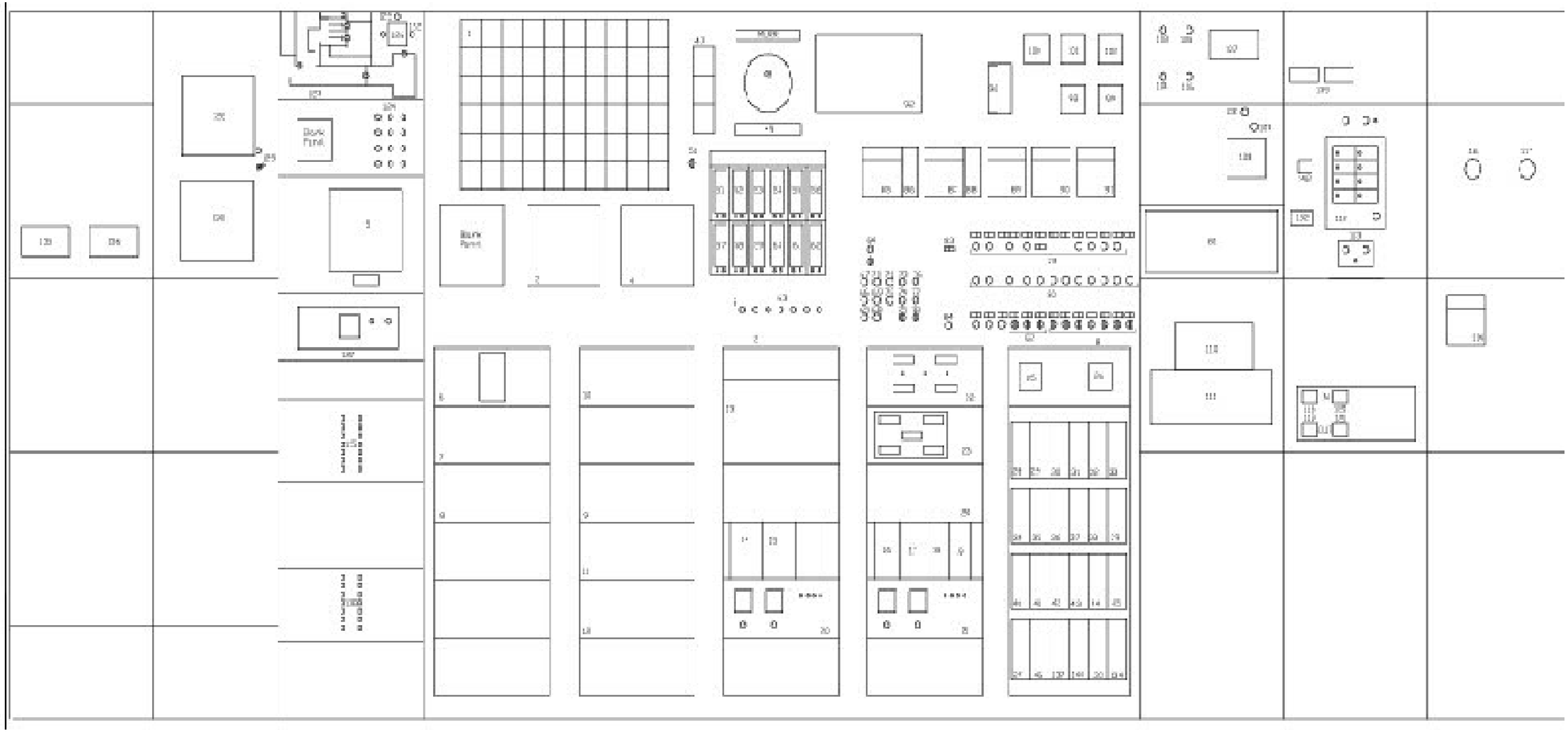
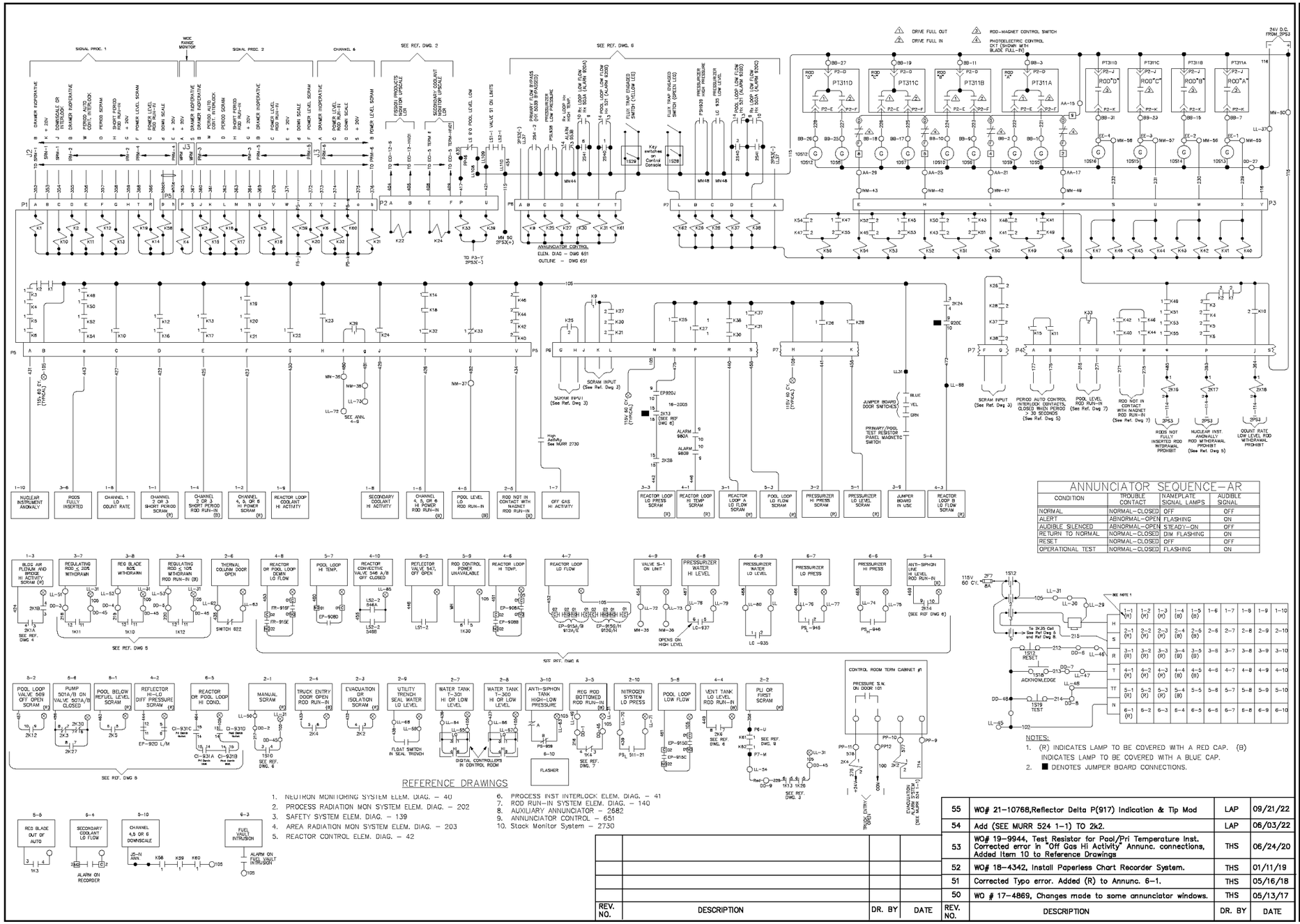


FIGURE 7.22 – INSTRUMENT PANEL LAYOUT



ANNUNCIATOR SEQUENCE–AR

CONDITION	TROUBLE CONTACT	NAMEPLATE SIGNAL LAMPS	AUDIBLE SIGNAL
NORMAL	NORMAL–CLOSED	OFF	OFF
ALERT	ABNORMAL–OPEN	FLASHING	ON
AUDIBLE SILENCED	ABNORMAL–OPEN	STEADY–ON	OFF
RETURN TO NORMAL	NORMAL–CLOSED	DIM FLASHING	ON
RESET	NORMAL–CLOSED	OFF	OFF
OPERATIONAL TEST	NORMAL–CLOSED	FLASHING	ON

NOTES:
1. (R) INDICATES LAMP TO BE COVERED WITH A RED CAP. (B) INDICATES LAMP TO BE COVERED WITH A BLUE CAP.
2. ■ DENOTES JUMPER BOARD CONNECTIONS.

REFERENCE DRAWINGS

1. NEUTRON MONITORING SYSTEM ELEM. DIAG. – 40
2. PROCESS RADIATION MON SYSTEM ELEM. DIAG. – 202
3. SAFETY SYSTEM ELEM. DIAG. – 139
4. AREA RADIATION MON SYSTEM ELEM. DIAG. – 203
5. REACTOR CONTROL ELEM. DIAG. – 42
6. PROCESS INST INTERLOCK ELEM. DIAG. – 41
7. ROD RUN–IN SYSTEM ELEM. DIAG. – 140
8. AUXILIARY ANNUNCIATOR – 2682
9. ANNUNCIATOR CONTROL – 651
10. Stack Monitor System – 2730

REV. NO.	DESCRIPTION	DR. BY	DATE

REV. NO.	DESCRIPTION	DR. BY	DATE
55	WO# 21–10768,Reflector Delta P(917) Indication & Tip Mod	LAP	09/21/22
54	Add (SEE MURR 524 1–1) TO 2k2.	LAP	06/03/22
53	WO# 19–9944, Test Resistor for Pool/Pri Temperature Inst. Corrected error in "Off Gas Hi Activity" Annunc. connections, Added Item 10 to Reference Drawings	THS	06/24/20
52	WO# 18–4342, Install Paperless Chart Recorder System.	THS	01/11/19
51	Corrected Typo error. Added (R) to Annunc. 6–1.	THS	05/16/18
50	WO # 17–4869, Changes made to some annunciator windows.	THS	05/13/17

seal

DATE: 5/18/95
DRAWN BY: J RILEY
CHECKED BY: ENGINEER:
CODE:
REVISION NUMBER:
REVISION DATE:

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research reactor facility

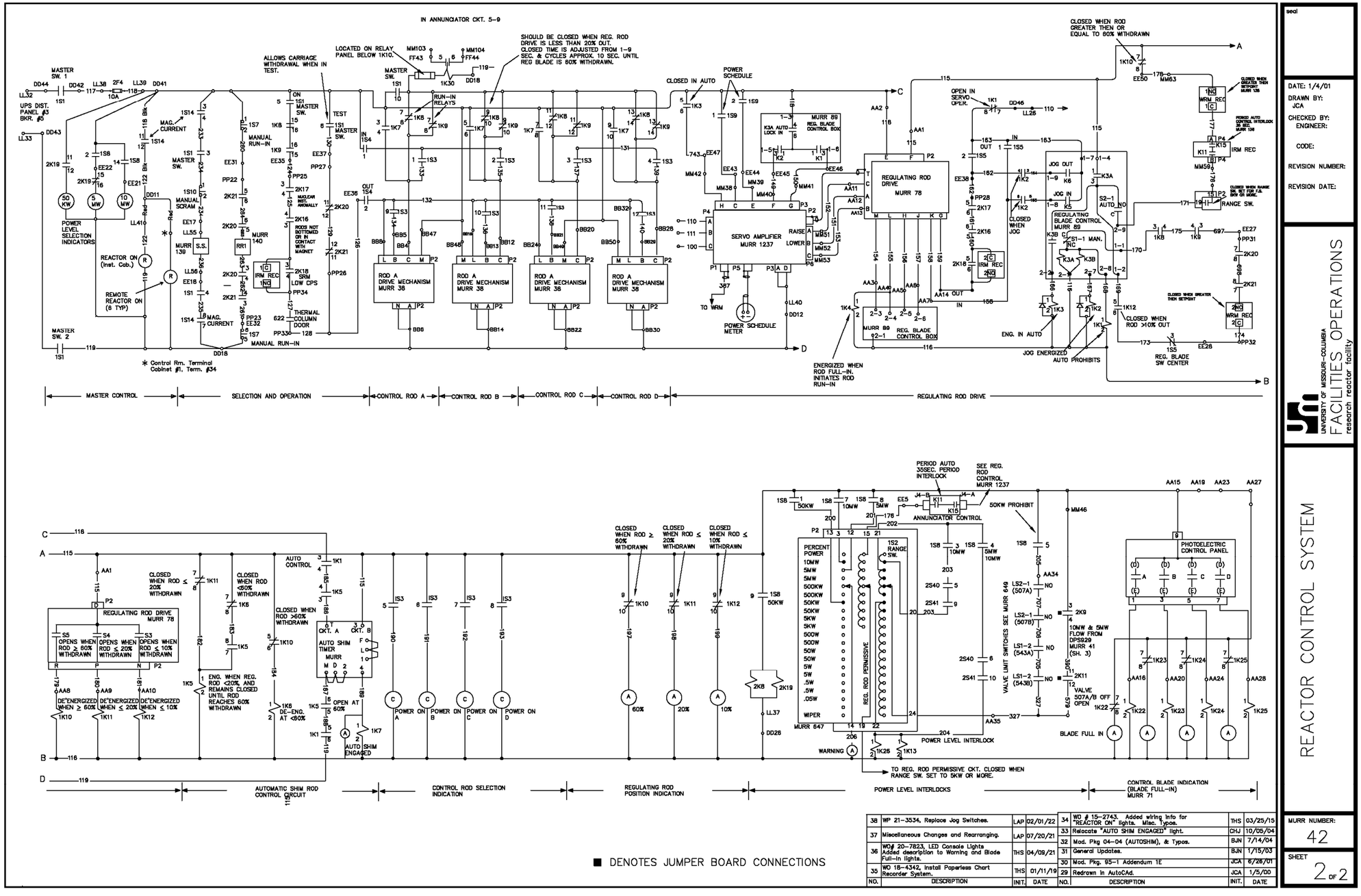
ANNUNCIATOR CONTROL
10 MW

MURR NUMBER: 138

SHEET 1 OF 1

FIGURE 7.33 – ANNUNCIATOR CONTROL

Revision 2
6/30/2025



NO.	DESCRIPTION	INIT.	DATE	NO.	DESCRIPTION	INIT.	DATE
38	WP 21-3534, Replace Jog Switches.	LAP	02/01/22	34	WO # 15-2743. Added wiring info for "REACTOR ON" lights. Misc. Typos.	THS	03/25/15
37	Miscellaneous Changes and Rearranging.	LAP	07/20/21	33	Relocate "AUTO SHIM ENGAGED" light.	CHJ	10/05/04
36	WO# 20-7823, LED Console Lights Added description to Warning and Blade Full-In lights.	THS	04/09/21	32	Mod. Pkg 04-04 (AUTOSHIM), & Typos.	BJN	7/14/04
				31	General Updates.	BJN	1/15/03
35	WO 18-4342, Install Paperless Chart Recorder System.	THS	01/11/19	30	Mod. Pkg. 95-1 Addendum 1E	JCA	6/28/01
				29	Redrawn in AutoCAd.	JCA	1/5/00

seal

DATE: 1/4/01
DRAWN BY: JCA
CHECKED BY: ENGINEER:
CODE:
REVISION NUMBER:
REVISION DATE:

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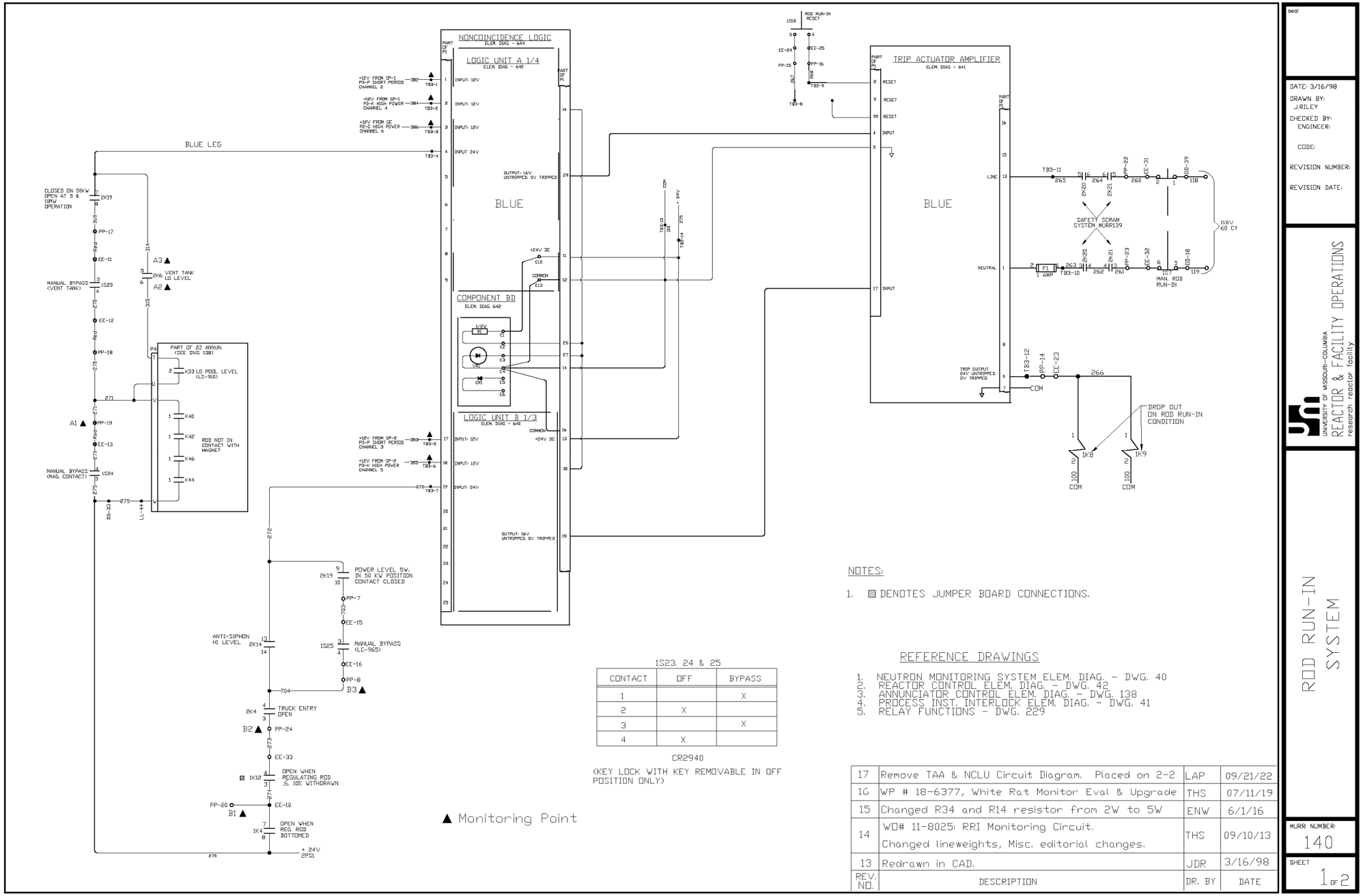
REACTOR CONTROL SYSTEM

MURR NUMBER: 42

SHEET 2 OF 2

FIGURE 7.44 – ROD CONTROL SYSTEM

Revision 2
6/30/2025



1S23, 24 & 25

CONTACT	OFF	BYPASS
1		X
2	X	
3		X
4	X	

CR2940

(KEY LOCK WITH KEY REMOVABLE IN OFF POSITION ONLY)

NOTES:

1. ▨ DENOTES JUMPER BOARD CONNECTIONS.

REFERENCE DRAWINGS

1. NEUTRON MONITORING SYSTEM ELEM. DIAG. - DWG. 40
2. REACTOR CONTROL ELEM. DIAG. - DWG. 42
3. ANNUNCIATOR CONTROL ELEM. DIAG. - DWG. 138
4. PROCESS INST. INTERLOCK ELEM. DIAG. - DWG. 41
5. RELAY FUNCTIONS - DWG. 229

REV. NO.	DESCRIPTION	DR. BY	DATE
17	Remove TAA & NCLU Circuit Diagram. Placed on 2-2	LAP	09/21/22
16	WP # 18-6377, White Rat Monitor Eval & Upgrade	THS	07/11/19
15	Changed R34 and R14 resistor from 2W to 5W	ENW	6/1/16
14	WO# 11-8025: RRI Monitoring Circuit. Changed lineweights, Misc. editorial changes.	THS	09/10/13
13	Redrawn in CAD.	JDR	3/16/98

seal

DATE: 3/16/98
DRAWN BY: J.RILEY
CHECKED BY: ENGINEER:
CODE:
REVISION NUMBER:
REVISION DATE:

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research reactor facility

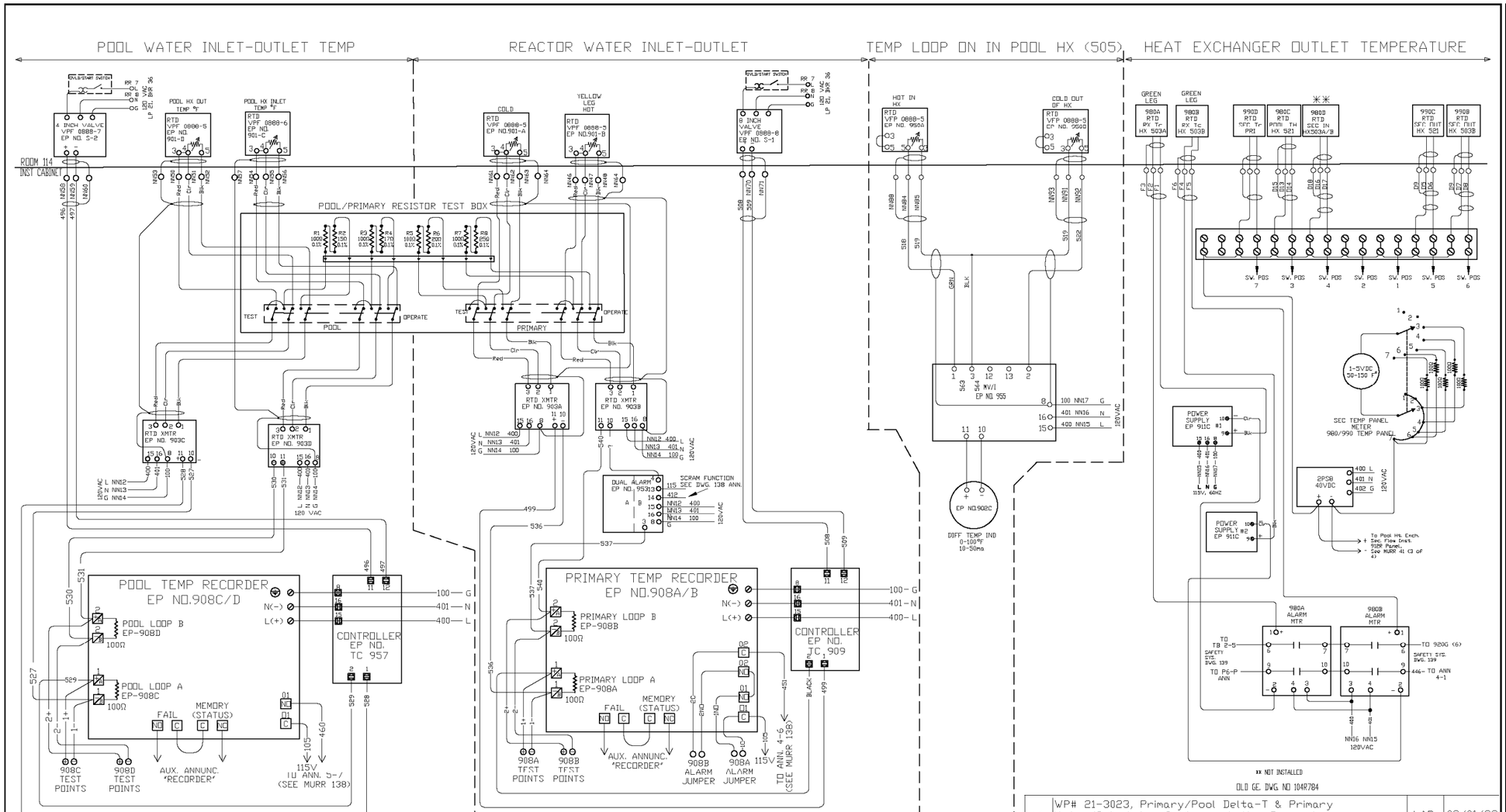
ROD RUN-IN SYSTEM

MURR NUMBER: 140

SHEET 1 OF 2

FIGURE 7.55 – ROD RUN-IN SYSTEM

Revision 2
6/30/2025



REV. NO	DESCRIPTION	DR. BY	DATE
31	WP# 21-3023, Primary/Pool Delta-T & Primary Meter/Calculator Instrumentation Removal WP# 22-2196, Secondary Coolant Piping Replacement.	LAP	03/01/23
30	WP# 20-1671, DCT System Upgrade.	THS	01/14/21
29	WO# 19-3864, Relocate 990G/H Readout to Secondary Temp. Flow/Paperless Recorder. WO# 19-9944, Test resistor for Pool & Pri Temp. Instruments	THS	06/24/20
28	WP# 18-4342, Install Paperless Chart Reorder System.	THS	01/10/19
27	WP# 14-11953, 990G & 990H Upgrade	THS	03/08/17
26	WP# 12-10366, 980A/B Power Supply Modification	THS	08/22/13
25	CT UPGRADE: 11-11232, 912R addition to 2PS8.	THS	05/04/12
24	MOD PKG. 05-8 Addendum 1. Misc. typo changes.	THS	12/26/06
23	REPLACED REACTOR AND POOL DELTA TEMP. INDICATORS. GENERAL CHANGES	THS	12/20/05

seal

DATE: 2/18/96
DRAWN BY: JCG
CHECKED BY: ENGINEER:
CODE:
REVISION NUMBER:
REVISION DATE:

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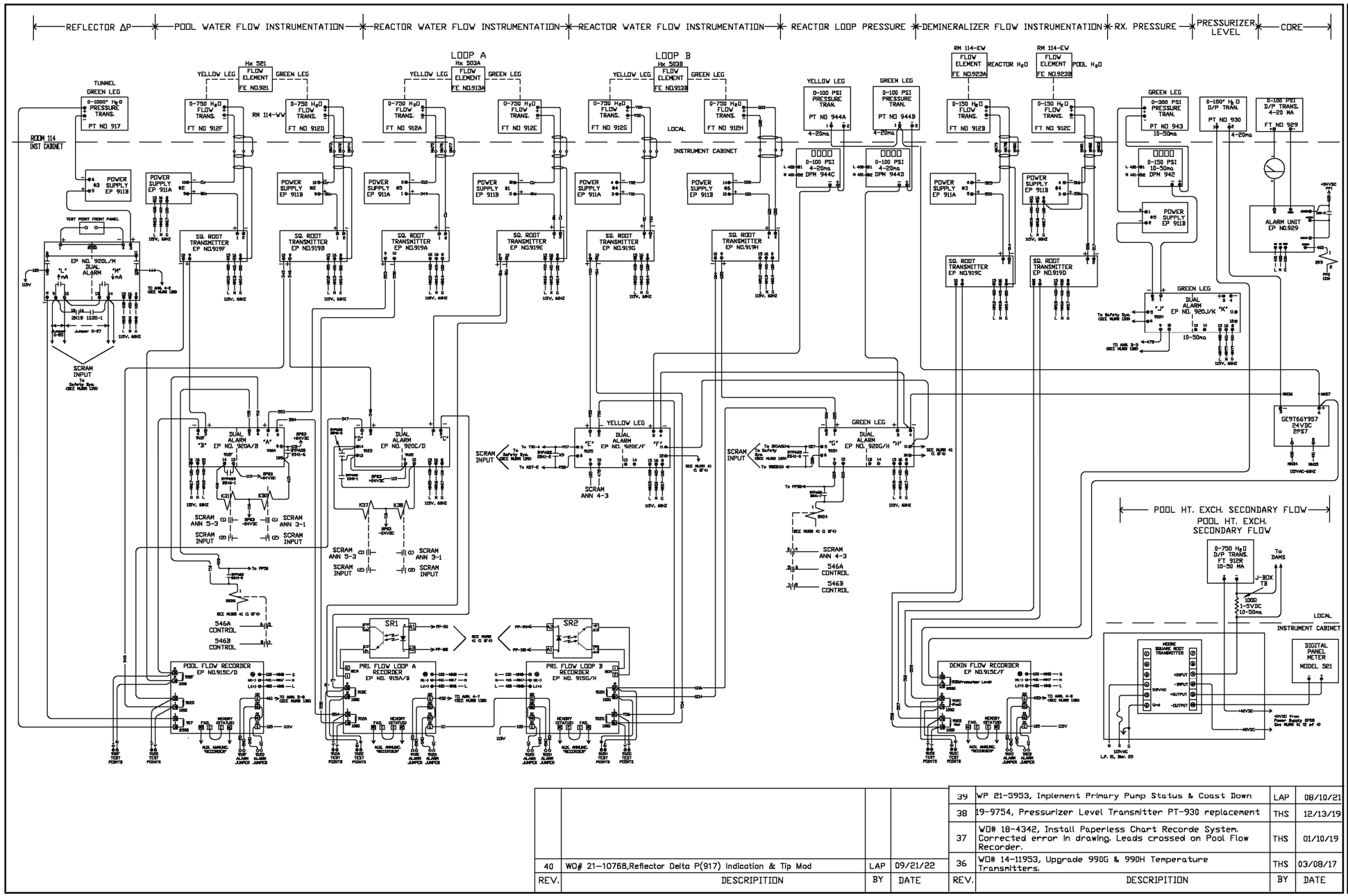
10MW PROCESS INSTRUMENTATION
CONTROL & INTER_OCK
(TEMPERATURE INST.)

MURR NUMBER: 41

SHEET 2 OF 4

FIGURE 7.66 – PROCESS INSTRUMENTATION CONTROL AND INTERLOCK (REFERENCE 7.13)

Revision 2
6/30/2025



REV.	DESCRIPTION	BY	DATE
40	WO# 21-10768,Reflector Delta P(917) Indication & Tip Mod	LAP	09/21/22
39	WP 21-5953, Implement Primary Pump Status & Coast Down	LAP	08/10/21
38	19-9754, Pressurizer Level Transmitter PT-930 replacement	THS	12/13/19
37	WO# 18-4342, Install Paperless Chart Recorde System. Corrected error in drawing. Leads crossed on Pool Flow Recorder.	THS	01/10/19
36	WO# 14-11953, Upgrade 990G & 990H Temperature Transmitters.	THS	03/08/17
REV.	DESCRIPTION	BY	DATE

neci

DATE:
DRAWN BY: TOM SEEGER
CHECKED BY: ENGINEER:
CODE:
REVISION NUMBER:
REVISION DATE:

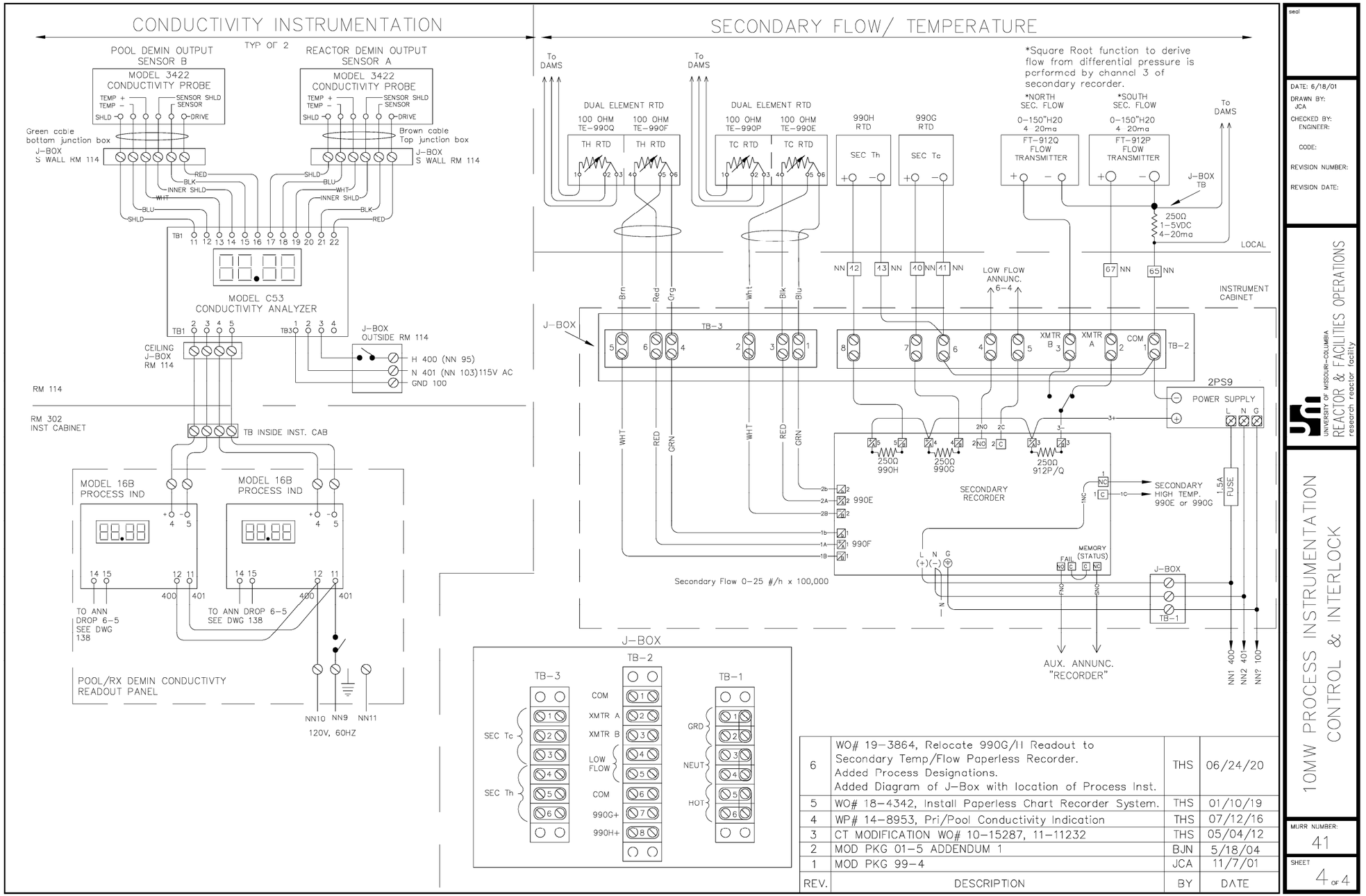
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PROCESS INSTRUMENTATION CONTROL & INTERLOCK

MURR NUMBER: 41

SHEET 3 OF 4

FIGURE 7.77 – PROCESS INSTRUMENTATION CONTROL AND INTERLOCK



REV.	DESCRIPTION	BY	DATE
6	WO# 19–3864, Relocate 990G/H Readout to Secondary Temp/Flow Paperless Recorder. Added Process Designations. Added Diagram of J–Box with location of Process Inst.	THS	06/24/20
5	WO# 18–4342, Install Paperless Chart Recorder System.	THS	01/10/19
4	WP# 14–8953, Pri/Pool Conductivity Indication	THS	07/12/16
3	CT MODIFICATION WO# 10–15287, 11–11232	THS	05/04/12
2	MOD PKG 01–5 ADDENDUM 1	BJN	5/18/04
1	MOD PKG 99–4	JCA	11/7/01

seal

DATE: 6/18/01
DRAWN BY: JCA
CHECKED BY: ENGINEER:
CODE:
REVISION NUMBER:
REVISION DATE:

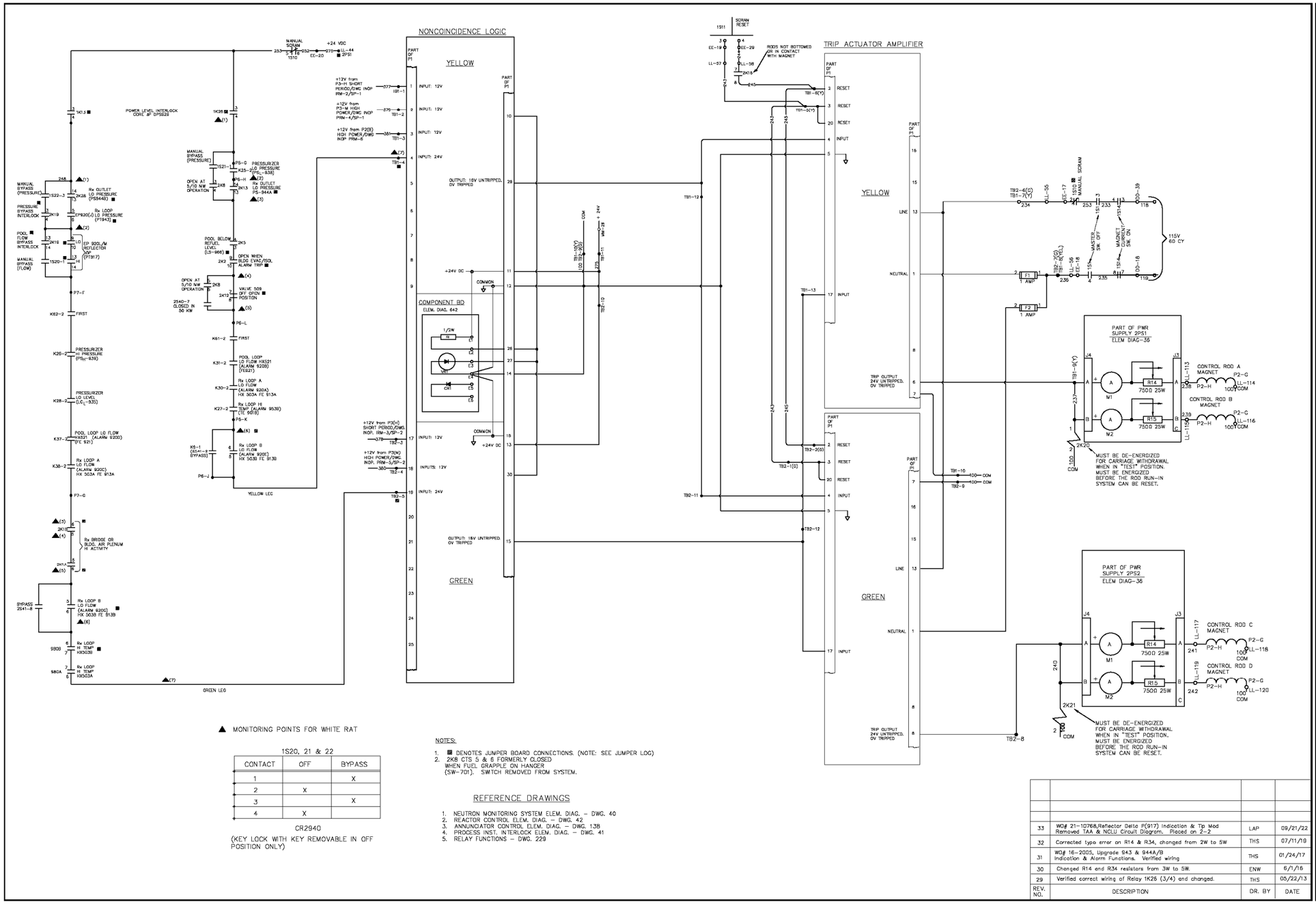
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10MW PROCESS INSTRUMENTATION CONTROL & INTERLOCK

MURR NUMBER: 41

SHEET 4 OF 4

FIGURE 7.88 – PROCESS INSTRUMENTATION CONTROL AND INTERLOCK



▲ MONITORING POINTS FOR WHITE RAT

1S20, 21 & 22

CONTACT	OFF	BYPASS
1		X
2	X	
3		X
4	X	

CR2940
(KEY LOCK WITH KEY REMOVABLE IN OFF POSITION ONLY)

NOTES:

1. ■ DENOTES JUMPER BOARD CONNECTIONS. (NOTE: SEE JUMPER LOG)
2. 2K8 CTS 5 & 6 FORMERLY CLOSED WHEN FUEL GRAPPLE ON HANGER (SW-701). SWITCH REMOVED FROM SYSTEM.

REFERENCE DRAWINGS

1. NEUTRON MONITORING SYSTEM ELEM. DIAG. – DWG. 40
2. REACTOR CONTROL ELEM. DIAG. – DWG. 42
3. ANNUNCIATOR CONTROL ELEM. DIAG. – DWG. 13B
4. PROCESS INST. INTERLOCK ELEM. DIAG. – DWG. 41
5. RELAY FUNCTIONS – DWG. 229

REV. NO.	DESCRIPTION	DR. BY	DATE
33	WO# 21-10768,Reflector Delta P(917) Indication & Tp Mod Removed TAA & NCLU Circuit Diagram. Placed on 2-2	LAP	09/21/22
32	Corrected typo error on R14 & R34, changed from 2W to 5W	THS	07/11/19
31	WO# 16-2005, Upgrade 943 & 944A/B Indication & Alarm Functions. Verified wiring	THS	01/24/17
30	Changed R14 and R34 resistors from 3W to 5W.	ENW	6/1/16
29	Verified correct wiring of Relay 1K26 (3/4) and changed.	THS	05/22/13

seal

DATE: 3/3/98
DRAWN BY: J.RII FY
CHECKED BY: ENGINEER:
CODE:
REVISION NUMBER:
REVISION DATE:

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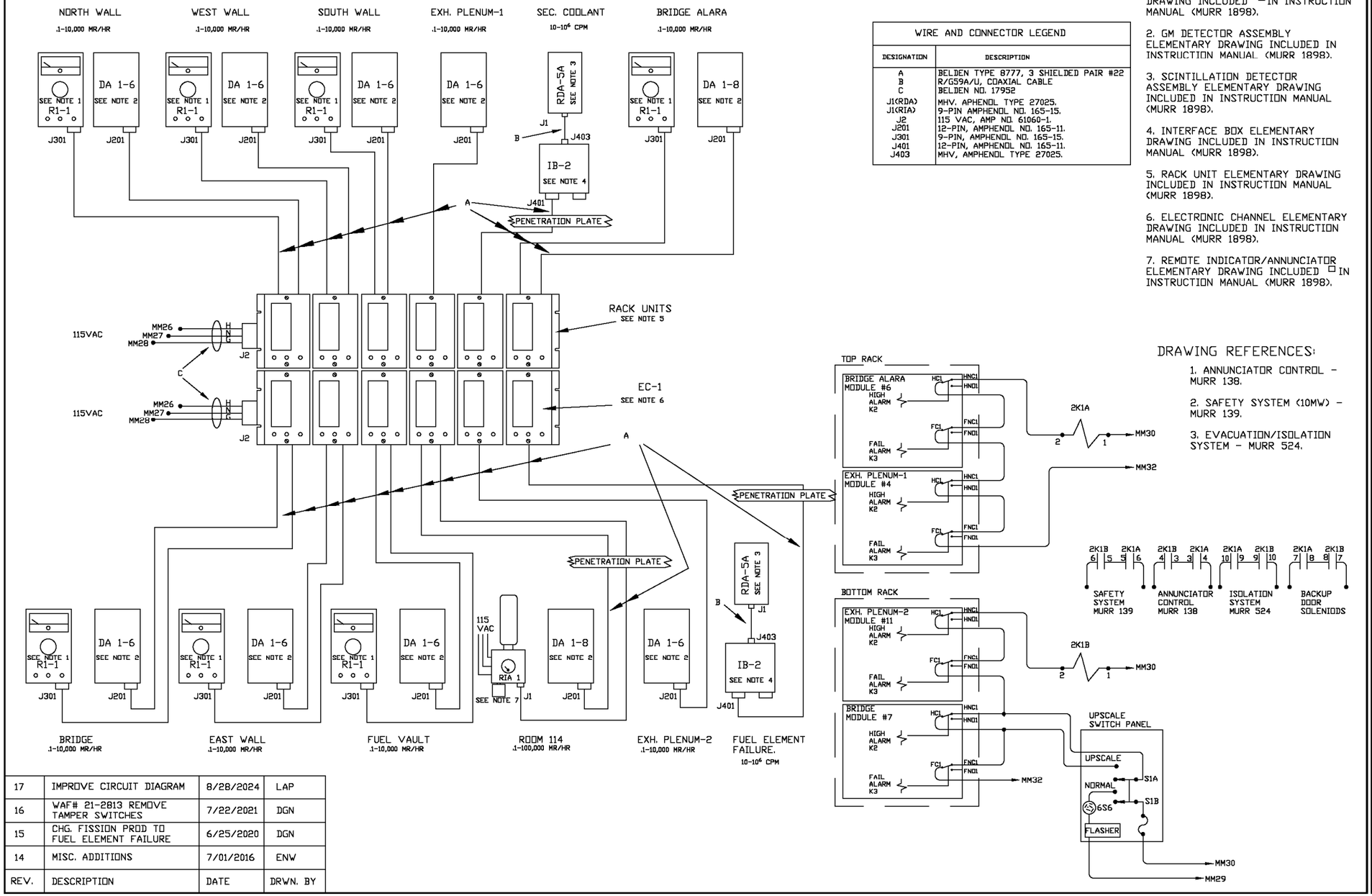
SAFETY SYSTEM

MURR NUMBER: 139

SHEET 1 OF 2

FIGURE 7.99 – REACTOR SCRAM SYSTEM

Revision 2
6/30/2025



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 TYPE 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).

DRAWING REFERENCES:

1. ANNUNCIATOR CONTROL - MURR 138.
2. SAFETY SYSTEM (10MW) - MURR 139.
3. EVACUATION/ISOLATION SYSTEM - MURR 524.

REV.	DESCRIPTION	DATE	DRWN. BY
17	IMPROVE CIRCUIT DIAGRAM	8/28/2024	LAP
16	WAF# 21-2813 REMOVE TAMPER SWITCHES	7/22/2021	DGN
15	CHG. FISSION PROD TO FUEL ELEMENT FAILURE	6/25/2020	DGN
14	MISC. ADDITIONS	7/01/2016	ENW

eecl

DATE: 11/30/00
DRAWN BY: JCA
CHECKED BY: ENGINEER:
CODE:
REVISION NUMBER:
REVISION DATE:

UNIVERSITY OF MISSOURI-COLUMBIA
FACILITIES OPERATIONS
research reactor facility

AREA MONITORING SYSTEM

MURR NUMBER: 203

SHEET 1 OF 1

FIGURE 7.1010 – AREA RADIATION MONITORING SYSTEM

CHAPTER 8

ELECTRICAL POWER SYSTEMS

TABLE OF CONTENTS

8.0 ELECTRICAL POWER SYSTEMS 3
8.1 Normal Electrical Power System 3
8.1.1 Introduction 3
8.1.2 Description of System 3
8.2 Emergency Electrical Power System 5
8.2.1 Introduction 5
8.2.2 Emergency Power Generator 5
8.2.3 Automatic Transfer Switch 5
8.2.4 Emergency Power Loads 6
8.2.5 Description of Operation 7
8.2.6 Technical Specifications 8

LIST OF FIGURES

Figure 8.1 – UNINTERRUPTIBLE POWER SUPPLY 9
Figure 8.2 – MURR NORMAL ELECTRICAL POWER SYSTEM 10
Figure 8.3 – MURR EMERGENCY ELECTRICAL POWER SYSTEM 11

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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.

8.9 Written communication to NRC dated February 25, 2016 regarding the Annual Report for the period January 1, 2015 to December 31, 2015, [ML16060A211].

8.0 ELECTRICAL POWER SYSTEMS

This chapter discusses and describes the electrical power systems designed to support reactor operation at the facility. (The original safety evaluation of MURR is documented in the Preliminary Hazards Report (Reference 8.1), the Hazards Summary Report (Reference 8.2), and Hazards Summary Report, Addenda 1-5 (References 8.3-8.7).) 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 University of Missouri Research Reactor (MURR) by the University of Missouri-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 network.

The electrical distribution system for MURR was originally designed and constructed according to the following codes and standards (Reference 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.

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

MURR receives its electrical power from the Research Park Substation located south of the facility. Two (2) independent primary transmission lines, routed through underground duct banks, supply 13.8 kV, 3-phase, 60 Hz electrical power to the MURR facility 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 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 connected in series. These feeder breakers provide electrical power to Motor Control Center 1 (MCC-1), also located in the electrical equipment room.

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, to Substation B, and high voltage panels (HVP-1 and HVP-3) through 1600-amp, 400-amp, and 1200-amp breakers.

HVP-1, which is located in the electrical equipment 299B of the MURR Industrial Building (MIB), houses feeder breakers for various electrical loads used in MIB and associated laboratories. HVP-3 is located in equipment room 231D of MIB and houses feeder breakers for various electrical loads used in the temporary office buildings, laboratory building and MIB.

Substation B, which is located in the inner corridor of the laboratory building, houses five (5) main feeder breakers. The feeder breakers provide electrical power to three (3) Motor Control Centers (MCC-3, MCC-4, MCC-5), a 120/208 V Distribution Center, and two (2) Emergency Distribution Centers (EDC and EDC-2) through an Automatic Transfer Switch (ATS). In the event of a loss of normal electrical power, the ATS transfers the emergency electrical load between normal and emergency power. Operation of the emergency electrical power system, including the ATS, is described in Section 8.2.

To reduce spurious reactor scrams caused by voltage fluctuations or a momentary interruption in electrical power, a 15-kVA Uninterruptible Power Supply (UPS) is installed to provide regulated 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 (EDC) is supplied by the emergency power generator. This allows the shutdown condition of the reactor to be monitored continuously by the reactor operator.

The UPS system (Figure 8.1) consists of a static inverter, a rectifier charger, 15KVA Isolation Line Conditioner, 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 (EDC) supplies power to the static Inverter and rectifier charger. The rectifier charger provides regulated DC power to maintain the battery bank in a fully charged condition. The inverter converts the DC power from the battery bank into regulated AC power. Upon loss of the primary AC source (EDC), the inverter automatically supplies power. If primary AC source (EDC) does not return (i.e., complete loss of electrical power to the facility and generator), the UPS will automatically secure when the discharge limit of the battery bank is reached (rated battery bank capacity 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 system is not required for safe reactor shutdown or to maintain an acceptable shutdown condition. However, the UPS system does supply power to the necessary reactor instrumentation (e. g.,

rod position indication, power level, etc.) to verify a complete reactor shutdown. Should a failure of the UPS occur, a reactor shutdown may also be verified upon visual observation that the control blades are fully inserted and that reactor power has been reduced by noting a reduction in the intensity of the Cherenkov 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 normal 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.

8.2 Emergency Electrical Power System

8.2.1 Introduction

The emergency electrical power system is designed to provide electrical power to essential systems in order to monitor reactor parameters and assure personnel safety should the facility suffer a loss of normal electrical power. The system design is simple 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 MIB (Room 231E). Electrical power is supplied to the selected loads through an ATS which transfers the power source from the normal electrical power system to the emergency electrical power system.

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 a four stroke, turbocharged, inline, 6-cyclinder diesel engine with a rated output of 275 kW (set at 0.8 Power Factor, 344 kVA) at 480 VAC (277 VAC Line Voltage) and 60 Hz at 1800 RPM. 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 16 hours under full load. Two (2)parallel sets of two 12-volt lead-acid batteries connected in series generate 24 volts for the starting system. An automatic, trickle/float battery charger maintains the startup batteries fully-charged.

8.2.3 Automatic Transfer Switch

The ATS transfers the emergency electrical load between normal and emergency power. The transfer switch is equipped with the following time delays as described.

Start Time Delay

Adjustable from 0 to 15 seconds. This delay prevents the diesel generator from starting during momentary power interruptions. The timing starts the moment the normal power source is interrupted.

Stop Time Delay

Adjustable from 0 to 10 minutes. This 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.

Transfer Time Delay

Adjustable from 0 to 120 seconds. This 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.

Retransfer Time Delay

Adjustable from 0 to 30 minutes. This 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 to verify its operability. The EPG can also be started manually to verify operability.

8.2.4 Emergency Power Loads

The emergency power system is routed through the ATS to two (2) Emergency Distribution Centers as shown in Figure 8.3. These distribution centers feed the following emergency electrical loads:

1. Emergency Distribution Center (EDC):
    a. Facility Exhaust Fan EF-13;
    b. Facility Exhaust Fan EF-14;
    c. Diesel Generator Room Panel;
    d. Emergency Power Panel (EPP):
        1) Emergency Air Compressor;
        2) Electric-motor-driven Containment Isolation Doors 504 and 505;
        3) West Pneumatic Tube Blower;
        4) Emergency Lighting Panel 2 (ELP-2);
        5) Waste Heat Pump No. 2;
        6) Motors for Personnel Entry Doors;
        7) Motor for Truck Door 101;

8) Emergency Lighting Panel (ELP):
    - Exit Lights;
    - Stairway Lights;
    - Fan Failure Alarm;
    - Intercommunications System;
    - Off-Gas Stack Monitor;
    - Emergency Lighting Panel 1 (ELP-1);
    - Hotcell Valve Position Indication;
    - Locker Room Lights;

e. Uninterruptible Power Supply (UPS) Panels:
    1) Area Radiation Monitoring System (ARMS);
    2) Annunciator Panel;
    3) Process Instrumentation;
    4) Evacuation & Isolation System;
    5) Off-Gas Stack Monitor;
    6) Containment Ventilation Back-up Doors
    7) Sample Information System
    8) Radioactive Sump and Drain Alarms
    9) Rod Control
    10) Chart Recorders
    11) Auxiliary Annunciator
f. North Office Addition Emergency Lighting Panel (NOA ELP)
g. MURR Industrial Building systems

2. Emergency Distribution Center 2 (EDC-2):
    a. Emergency Lighting Panel 3 (ELP-3)

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 Technical Specifications

Technical specification 3.6.a requires that the reactor not be operated unless the emergency electrical system is operable. The objective of this technical specification is to provide emergency electrical power to essential systems in order to monitor reactor parameters and assure personnel safety should the facility suffer a loss of normal electrical power.

Technical specifications 4.6.a and 4.6.b verify the operability of the emergency electrical system. The objective of these technical specifications is to ensure that the emergency power generator is operable and that the emergency electrical system is capable of assuming load.

Technical specification 5.6.a requires that MURR have an emergency power generator capable of providing emergency electrical power to essential systems in order to monitor reactor parameters and assure personnel safety should the facility suffer a loss of normal electrical power.

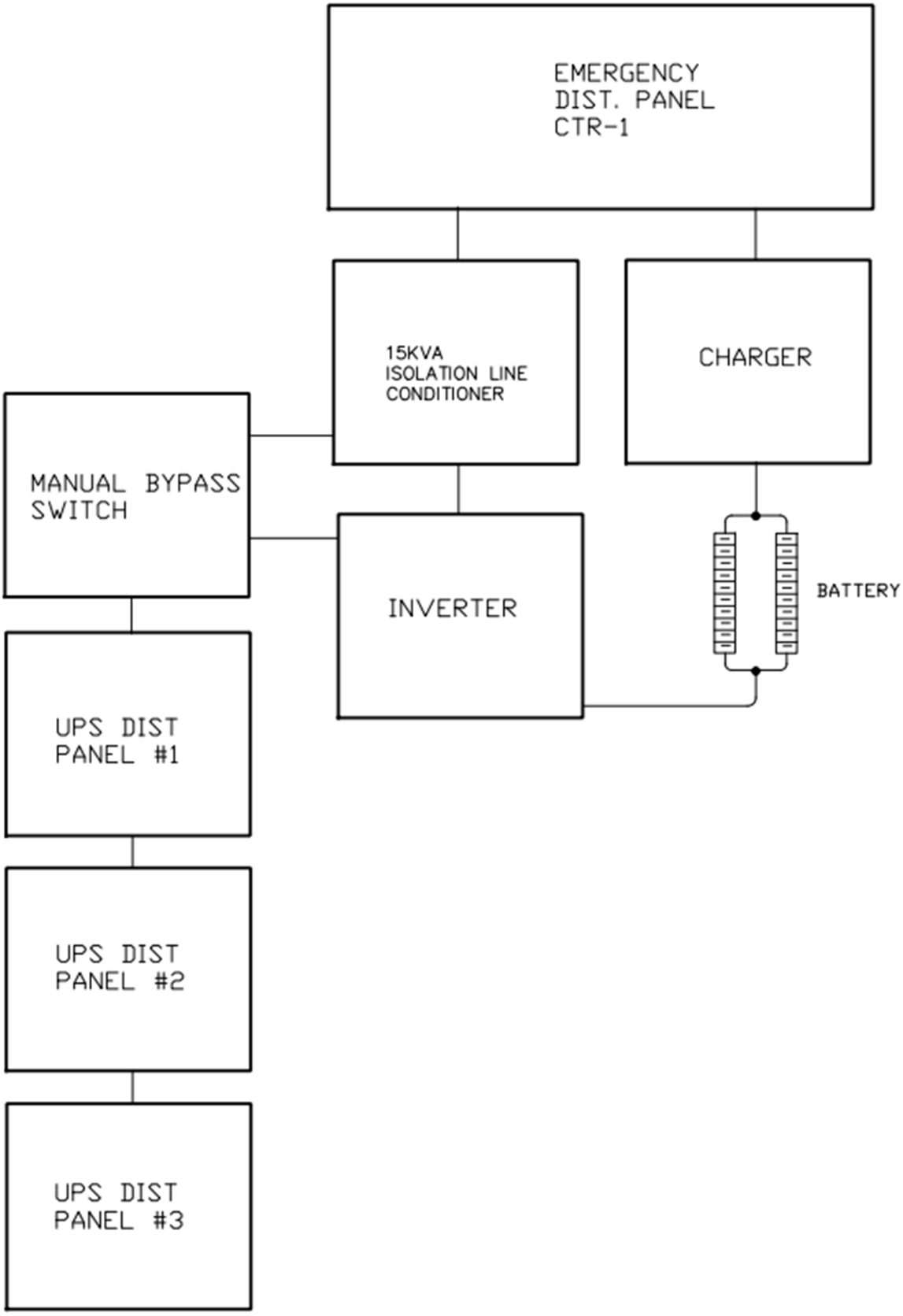


FIGURE 8.1 – UNINTERRUPTIBLE POWER SUPPLY

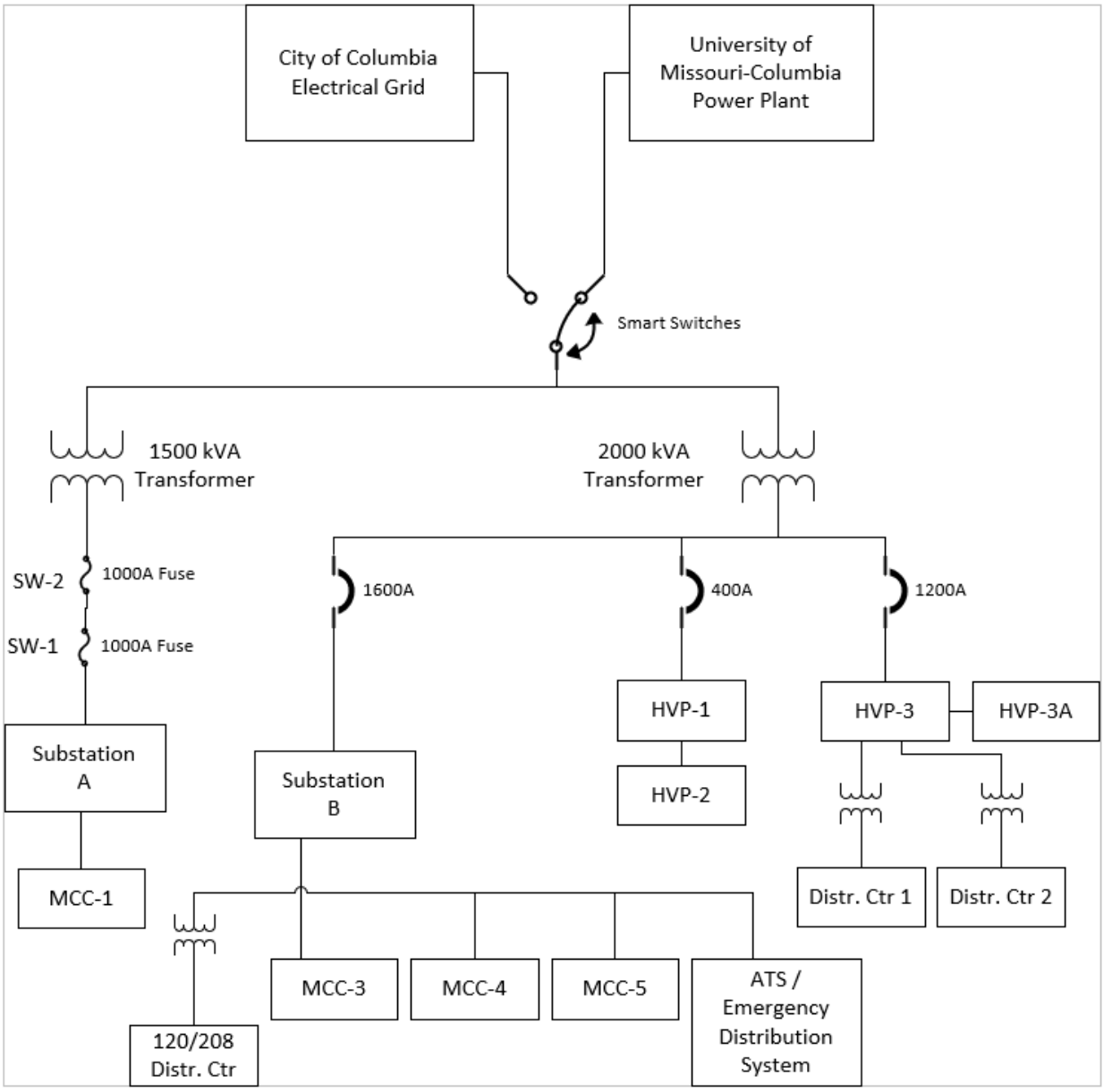


FIGURE 8.22 – MURR NORMAL ELECTRICAL POWER SYSTEM

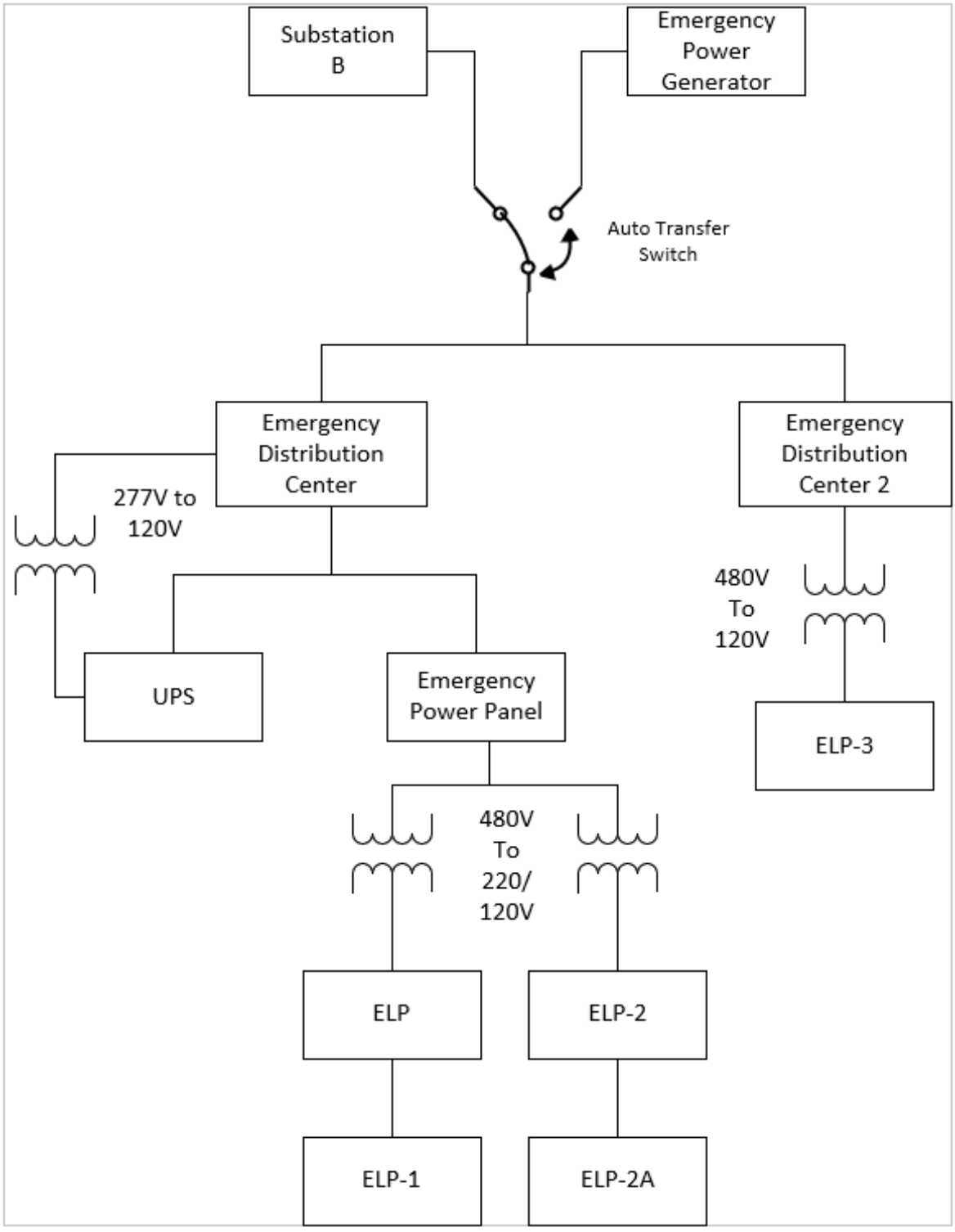


FIGURE 8.33 – MURR EMERGENCY ELECTRICAL POWER SYSTEM

CHAPTER 9

AUXILIARY SYSTEMS

TABLE OF CONTENTS

9.0 AUXILIARY SYSTEMS 4
9.1 Heating, Ventilation, and Air-Conditioning Systems 4
9.1.1 Introduction 4
9.1.2 Description of Operation 4
9.1.2.1 Facility Ventilation Supply System 4
9.1.2.2 Facility Ventilation Exhaust System 5
9.1.3 Facility Ventilation Exhaust System Technical Specifications 6
9.2 Handling and Storage of Reactor Fuel 7
9.2.1 Introduction 7
9.2.2 Methods of Storage and Transfer 8
9.2.3 Fuel Handling and Storage Technical Specifications 10
9.3 Fire Protection Systems and Programs 10
9.4 Communication Systems 11
9.5 Possession and Use of Byproduct, Source, and Special Nuclear Material 12
9.5.1 Types and Quantities of Radionuclides 13
9.5.2 Rooms, Spaces, and Equipment 14
9.5.3 Procedures 15
9.5.4 Uses of Radioactive Materials 15
9.6 Nitrogen Supply System 16
9.6.1 Introduction 16
9.6.2 Description of System 16
9.7 Emergency Pool Fill System 16
9.7.1 Introduction 16
9.7.2 Description of System 17
9.7.3 Water Sources 17
9.7.4 Basis 17
9.7.5 Emergency Pool Fill System Technical Specifications 18
9.8 Pool Skimmer System 18
9.8.1 Introduction 18
9.8.2 Skimmer Surface Box 18
9.8.3 Pump 18
9.8.4 Filter 18
9.8.5 Description of Operation 19
9.9 Closed Circuit Television System 20
9.10 Battery-Operated Emergency Lights 20
9.11 Radioactive Waste Disposal Systems 20
9.11.1 Introduction 20
9.11.2 Gaseous Waste Disposal 20
9.11.2.1 Airborne Effluent Technical Specifications 20
9.11.3 Solid Waste Disposal 21

9.11.4 Liquid Waste Disposal 21
9.12 Demineralized Water Supply System 21
9.12.1 Introduction 21
9.12.2 Water Treatment System 21
9.12.3 Reactor Plant Make-Up Water Storage Tanks 22
9.12.4 Facility Demineralized Water Storage Tanks 22
9.12.5 Primary Coolant Make-Up Water System Technical Specifications 23
9.13 Reactor Loop Vent System 23
9.13.1 Introduction 23
9.13.2 Vent Tank 23
9.13.3 Description of Operation 24
9.14 Compressed Air Systems 24
9.14.1 Introduction 24
9.14.2 Main Air System 24
9.14.3 Emergency Air System 25
9.14.4 Valve Operation Air System 25
9.14.5 Instrument Air System 25

LIST OF TABLES

Table 9-1 INTERCOMMUNICATION SYSTEM MASTER AND STAFF STATION LOCATIONS 12
Table 9-2 TYPICAL BYPRODUCT MATERIAL THAT MURR MAY POSSESS AND USE UNDER THE BROAD SCOPE MATERIAL LICENSE 14

LIST OF FIGURES

Figure 9.1 – MURR FACILITY VENTILATION SYSTEM 27
Figure 9.2 – MURR CONTAINMENT AIRFLOW PATHS 28

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9.12 Written communication to the NRC dated February 24, 2023, regarding the Annual Report for the period January 1, 2022 through December 31, 2022, [ML23055B137].

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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 University of Missouri 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 (The original safety evaluation of MURR is documented in the Preliminary Hazards Report (Reference 9.1), the Hazards Summary Report (Reference 9.2), and Hazards Summary Report, Addenda 1-5 (References 9.3-9.7).) 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.”

9.1.2 Description of Operation

9.1.2.1 Facility Ventilation 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 airflow 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 roof top air handlers (RTAHs). Two (2) of these RTAHs are 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. A fourth intake fan and preconditioning system services the MURR industrial building (MIB) eastward expansion including the iodine-131 (I-131) processing hot cells.

Ventilation for the reactor containment building is primarily recirculation with only a small amount of air being exhausted through a 16-inch (40.64-cm) 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) (495.5 m^3/min) by the reactor containment building return fan (RF-2). Approximately 2,500 cfm (70.8 m^3/min) 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 (566.3 m^3/min). 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 (2) isolation doors. One set of doors is air-operated-to-open, gravity-to-close, and is designated as a “back-up door.” The other set of doors is driven open and closed by a 3-phase motor connected to a chain drive 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 Facility Ventilation 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 (4) 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 the 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. 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 cells, and the pneumatic tube system passes through separate filtration systems before discharging into the exhaust plenum. Exhaust air from the iodine-131 processing hot cells is discussed in greater detail in Chapter 10, "Experimental Facilities and Utilization."

Reactor containment building exhaust air discharges to the exhaust plenum at approximately 2,500 cfm (70.8 m^3/min) through a 16-inch (40.64-cm) line which penetrates the west containment building wall. This line contains two (2) 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 (40.64-cm) 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."

Monthly Fan Failure Alarm Panel testing includes testing of the standby fan feature and verification of the audible low flow alarm. The Fan Failure Alarm Panel serves as an operator aid designed to alert the operator that an abnormal condition may exist in the facility ventilation exhaust system and prompt the on-duty reactor operator to investigate the cause of the alarm. The actuation of the alarm in the reactor control room does not initiate any protective action and there is no LCO associated with the Fan Failure Alarm Panel. Therefore, there are no Technical Specifications required for the surveillance of the Fan Failure Alarm Panel. (Reference 9.10)

9.1.3 Facility Ventilation Exhaust System Technical Specifications

The technical specifications on the Facility Ventilation Exhaust System are segregated into two sections: (1) operation of the reactor and (2) operation of the I-131 processing hot cells.

(1) The objective of the Limiting Conditions for Operations (LCO) technical specifications on the reactor portion of the facility ventilation exhaust system is to assure the containment integrity is maintained when required so that the health and safety of the general public is not endangered as a result of reactor operation. Technical specifications 3.4.a(4) and 3.4.a(6) provide this assurance for the reactor portion of the facility ventilation exhaust system.

The objective of the surveillance technical specification on the reactor portion of the facility ventilation exhaust system is to reasonably assure proper operation of the containment system. Surveillance technical specification 4.4.d provides this assurance for the reactor portion of the facility ventilation exhaust system.

The objective of the design features technical specifications on the reactor portion of the facility exhaust ventilation system is to assure adequate restriction to the accidental release of radioactivity to the environment. Technical specification 5.5.b provides this assurance for the reactor portion of the facility exhaust ventilation system.

(2) The objective of the LCO technical specifications on the iodine-131 processing hot cells portion of the facility ventilation exhaust system is to reasonably assure that the health and safety of the staff and public is not endangered as a result of processing iodine-131. Technical specifications 3.10.a and 3.10.b provide this assurance for the iodine-131 processing hot cells portion of the facility ventilation exhaust system.

The objective of the surveillance technical specifications on the iodine-131 processing hot cells portion of the facility ventilation exhaust system is to reasonably assure proper operation of the equipment needed to safely process iodine-131. Surveillance technical specifications 4.10.a and 4.10.b provide this assurance for the iodine-131 processing hot cells portion of the facility ventilation exhaust system.

There are no design feature technical specifications associated with the iodine-131 processing hot cells portion of the facility ventilation exhaust system.

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. All fuel elements that do not comply with 10 CFR 73.6(b), self-protection (radiation level in excess of 1 gray (100 rad)/hr at a distance of 1 meter) of highly enriched uranium (HEU) fuel, are stored within Controlled Access Areas as described in the MURR Physical Security Plan. A MURR fuel element that has operated at 10 MW for 5.3 minutes is self-protecting for 13 days. A fuel element that has operated a total of 20 MWDs is self-protecting for 10 years (Reference 9.8). 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 situated in three (3) areas within the reactor pool and are designated as the "X," "Y" and "Z" storage baskets. The "Z" storage basket contains 48 fuel element storage locations; consisting of two (2) levels, referred to as "upper" and "lower," of 24 locations per level. The "X" and "Y" storage baskets each contain 20 fuel element storage locations. There are eight (8) storage locations for new, fresh fuel elements in the fuel vault. These storage locations are designed to the following specifications:

- 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 (Reference 9.9);
- Sufficient natural convection cooling to prevent a fuel element from exceeding its design temperature;
- Location within the reactor pool at sufficient depth to provide adequate radiation shielding;
- Arrangement in the reactor pool to permit efficient handling during the insertion, removal, or interchange of the fuel elements; and
- 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 is removed from the shipping containers by hand and placed in a storage vault. 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 any time 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 the 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 mrem/hr 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 (13,608-kg) 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 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 spent fuel shipping cask lifting equipment, including the 15-ton (13,608-kg) capacity overhead rectilinear crane, is rigorously maintained, including preventive maintenance as appropriate. 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 their designee 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 Fuel Handling and Storage Technical Specifications

The objective of the design features technical specifications on the Fuel Storage System are to assure that fuel which is stored shall not become critical and will not reach an unsafe temperature. Design features technical specifications 5.4.a and 5.4.b provide this assurance for the storage of reactor fuel when not in the reactor core.

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. 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 sensors, flow switches, and manual pull stations. A central control station, located in Room 210, and repeater stations, located in the reactor control room and Room 204, monitor 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 different types of systems: 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. 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. 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 viewpoint of fire safety. Only one exit is required for these spaces provided that the travel distance to the exit is adequate. NFPA 101 (updated 2000), Table A.7.6.1, 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. (Reference 9.11)

9.4 Communication Systems

MURR utilizes two (2) 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").

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 216	Laboratory
Room 218	Laboratory
Room 227	Laboratory
Room 228	Laboratory
Room 271B	Reactor Health Physics Office
Room 278	Mechanical Equipment Room
Room 282	Electronic Shop
Room 285	Containment Building Lobby
Room 286	Personnel Airlock
Room 287	Laboratory Building Central Corridor
Room 307	West Tower Third Level
Room 503	Containment Building Fifth Level
Room 507	West Tower Fifth Level
Corridor C2002	Cyclotron Suite
Corridor C2003	NOA Laboratory Area Exit
Room 299T	Iodine Suite

9.5 Possession and Use of Byproduct, Source, and Special Nuclear Material

Renewed R-103 facility operating license, issued January 4, 2017, Section 2.B., describes the limits for receipt, possession and use of Special Nuclear Material. The physical protection system and the security organization which will detect the attempted theft or diversion of Special Nuclear Material (SNM) at

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.

In addition to the renewed 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 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. Also housed at the reactor facility is a 16 MeV cyclotron, owned and operated by a MURR partner. All associated research from the accelerator-based isotopes in conducted under a separate Material License: No. 24-32762-01MD.

Licensed material is segregated between the Reactor License and the Broad Scope Material License via the use of research or production specific Project Authorizations which fall under the purview of either the Radiation Safety Committee (RSC), for the Broad Scope Material License, or the Isotope Use Subcommittee (IUS), for the Reactor License. These Project Authorizations provide the administrative controls for the use of radioactive materials throughout the facility as well as provide the administrative controls necessary to segregate the Reactor License radioactive materials from other materials used and stored in the facility. Transfers of materials are not a regular occurrence between the two licenses but do occur periodically. However, when a material transfer occurs a dedicated transfer form is used which documents the transfer between one project and another. This is typically completed by the Health Physics Staff in concert with the authorized supervisor of the project or projects involved in the transfer. Records of all transfers are retained and available for inspection. (Reference 9.11)

9.5.1 Types and Quantities of Radionuclides

Typical of a research reactor of its size, MURR handles or produces a spectrum of byproduct materials. MURR has been licensed to handle isotopes ranging from tritium through transuranics. 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. 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 9-22
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, 20 Ci total
Isotopes with atomic number 84-96	Any	200 mCi each, 2 Ci total
Hydrogen-3 (H-3)	Any	6 Ci
Nickel-63 (Ni-63)	Encapsulated metallic powder	20 Ci
Samarium-153 (Sm-153)	Any	40 Ci
Uranium-238 (U-238) (depleted)	Any	500 grams
Neptunium-237 (Np-237)	Any	5 mCi
Plutonium-239 (Pu-239)	Any	0.5 grams
Plutonium-240 (Pu-240)	Any	0.5 grams
Americium-241 (Am-241)	Any	500 mCi
Cesium-137 (Cs-137)	Sealed source	4 Ci
Yttrium-90 (Y-90)	Any	25 Ci
Polonium-210 (Po-210)	Activation product	1 mCi
Lutetium-177 (Lu-177)	Any	120 Ci
Technicium-99m (Tc-99m)	Any	10 Ci
Fluorine-18 (F-18)	Any	20 Ci

9.5.2 Rooms, Spaces, and Equipment

Radioactive materials may be used in any room or area designated by MURR management and approved by Reactor 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 Reactor Health Physics staff to ensure that they are a safe environment for the work planned or conducted in that area. The areas that are under the jurisdiction of the Reactor License include the Laboratory Building, which includes the Reactor Containment Building, and all the attached structures to the Laboratory Building except the North Office Addition. (Reference 9.11)

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, Reactor 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 ensures the proposed use of materials and the accompanying procedure(s) are reviewed to ensure the physical safety of the reactor and its components. Reactor 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.

Reactor 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 Reactor 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, Reactor 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 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.

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. 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 gooseneck extending over the ledge of the reactor pool. The opening of the gooseneck 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 (10.16-cm) 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 pressure-indicating transmitter located in the basement of the laboratory building. 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 (25.4-cm) 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 1.0-million-gallon (3.8-million l) reservoir near capacity. Three (3) pumps, each with a 1,000-gpm (3,785-lpm) capacity, take suction from this reservoir and discharge into the 10-inch (25.4-cm) main to provide a portion of the campus water supply. The 8-inch (20.32-cm) wet fire line that provides the flow path for the emergency pool fill system is connected to this 10-inch (25.4-cm) 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 1.0-million-gallon (3.8-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 Emergency Pool Fill System Technical Specifications

The objective of the LCO technical specification on the Emergency Pool Fill System is to ensure that the water level in the reactor pool will remain above the reflector in case a 6-inch beamport or 6-inch pool coolant line is sheared. Technical specification 3.9.b provides this assurance for the emergency pool fill system.

The objective of the surveillance technical specification on the Emergency Pool Fill System is to reasonably assure proper operation of the emergency pool fill system. Technical specification 4.9.b provides this assurance for the emergency pool fill system.

9.8 Pool Skimmer System

9.8.1 Introduction

The pool skimmer system is designed 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.

9.8.2 Skimmer Surface Box

The pool skimmer surface box is an aluminum box, 29.75 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 (P532) 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).

9.8.4 Filter

The pool skimmer filter (F531) 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.

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 (5.08-cm) aluminum header that encircles the reactor pool within the biological shield. A 0.5-inch (1.27-cm) line allows this return water to flow from the 2-inch (5.08-cm) 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 (2) 0.5-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.

A loss of pool water caused by a failure of the pool skimmer system would most likely require the reactor to be shutdown if it occurred during operation. A leak in the pool skimmer system would be similar to a leak in the pool coolant system, but less severe because the skimmer system piping is only 2 inches (5.08 cm) in diameter whereas the pool coolant system piping is 6 inches in diameter. Additionally, the skimmer pump is only periodically operated to remove surface debris. Furthermore, use of the system to lower pool water level is only performed with the reactor shutdown.

Should a leak in a section of the pool skimmer system occur, the emergency actions are covered by Section 13.2.9.1. (Reference 9.11)

9.9 Closed Circuit Television System

A closed-circuit television (CCTV) system is installed in the reactor facility which allows the reactor control room operators to monitor the MURR facility. Security cameras are strategically located throughout the facility, both internally and externally, for the purpose of surveillance. The MURR Physical Security Plan, which describes the security requirements and security measures for the facility, outlines specific requirements of the CCTV system.

9.10 Battery-Operated Emergency Lights

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.

9.11 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 Reactor 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 (airborne 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 mixing the potentially contaminated air with uncontaminated air.

9.11.2.1 Airborne Effluent Technical Specifications

The objective of the LCO technical specification on the Radiation Monitoring Systems and Airborne Effluents is to ensure that exposure to the public resulting from radioactivity released from the reactor

facility to the unrestricted environment will not exceed the limits of 10 CFR 20. Technical specification 3.7.b provides this assurance for the Airborne Effluent Disposal System.

The objective of the surveillance technical specification on the Radiation Monitoring Systems and Airborne Effluents is to reasonably assure proper operation of the radiation monitoring instrumentation. Technical specification 4.7.b provides this assurance for the Radiation Monitoring Systems and Airborne Effluents.

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 until a sufficient volume has accumulated to be packaged in sealed containers. The containers are then processed by a member of the Reactor 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 (4) tanks, three (3) with a capacity of approximately 5,000 gallons (18,927 l) and one (1) with a capacity of 550 gallons (2,082 l), three (3) transfer pumps, three (3) 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 and facility operations. This system consists of three principal 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/cm.

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.

Demineralized water from the RO unit may be directed to the water storage systems for the facility or the reactor plant. (Reference 9.13)

9.12.3 Reactor Plant Make-Up Water Storage Tanks

Two (2) stainless 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 (5.08-cm) 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 1.25-inch (3.175-cm) line and into the reactor pool. A 4-inch (10.16 cm) 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 (52,996 l) for reactor pool water level lowering operations.

To provide local and remote water level indication, pressure transmitters are connected to the bottoms of both T300 and T301. In addition to the level indication, the pressure transmitters also provide inputs to the T-300 and T-301 Annunciator level alarms and power an auxiliary relay for interlocks in the Pool Water Movement System. Individual pressure gauges attached to the bottom of each tank also provide a backup system for tank water level indication.

9.12.4 Facility Demineralized Water Storage Tanks

Three (3) 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 sight glass 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 Primary Coolant Make-Up Water System Technical Specifications

The objective of the LCO technical specification on the Primary Coolant Make-Up Water System is to provide an adequate supply of primary grade water for reactor plant make-up during all modes of operation. Technical specification 3.9.a provides this assurance for the primary coolant make-up water system.

The objective of the surveillance technical specification on the Primary Coolant Make-Up Water System is to reasonably assure proper operation of the Primary Coolant Make-Up Water System. Technical specification 4.9.a provides this assurance for the primary coolant make-up water system.

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. Analysis has shown that radiolysis of primary coolant does not contribute to a significant buildup of hydrogen and oxygen gases at MURR. (Analysis presented in question 5.2.b of letter dated July 16, 2010, “Responses to the U.S. Nuclear Regulatory Commission’s (NRC) request for additional information, dated June 1, 2010, regarding our renewal request for Amended Facility Operating License R-103, which was submitted to the NRC on August 31. 2006, as supplemented.”)

9.13.2 Vent Tank

The vent tank is constructed of two lengths of 6-inch (15.24-cm) diameter aluminum pipe connected near the top and bottom by 3-inch (7.62-cm) 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 0.5-inch (1.27-cm) line into the vent tank. Any gases collected in the vent tank are released through an exhaust line which diverges into two parallel 0.25-inch (0.635-cm) 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 recedes. 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

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 (2.83 m^3/min). The two compressors automatically alternate roles between lead and lag unit each time the lead compressor cycles in order to equalize wear. When in the lead role, the compressor will automatically start when main air pressure reduces to a pre-established low setpoint and secure when main air pressure reaches a pre-established upper setpoint. If main air system pressure cannot be kept within this band and continues to lower, the lag compressor will start at

a pre-established lower setpoint to aid the lead compressor. This condition also initiates the lag alarm circuit, which generates a local alarm and illuminates a local warning lamp. The lag compressor will receive a signal to stop at a lower setpoint than the lead as pressure rises. The control system also allows for a manual mode of control for either compressor. When in manual control, the selected compressor will cycle within a pre-established pressure band." (Reference 9.12)

In the event of a failure of the compressors or other component within the system, each subsystem is isolated from the Main Air System by in-line check valves and has a backup compressor to provide service for that subsystem.

9.14.3 Emergency Air System

The emergency air system supplies compressed air to the hot exhaust isolation valves and backup doors and to supply the inflatable gaskets which maintain the reactor containment building closures. Included in this system is a backup air compressor and associated check valve to prevent subsystem failure if the Main Air System is compromised. The hot exhaust line isolation valves are in series and of different types—Valve 16A is spring-to-close and air-to-open which is designed fail-safe, while valve 16B is air-to-open and air-to-close. Valve 16B has an additional independent compressor should all the previous compressors fail. In the event that all air pressure is lost, Valve 16A would spring-to-close and isolate containment from the hot exhaust system. (Reference 9.11) Operation of the emergency air system is described in Sections 6.2.3.8 and 6.2.5.

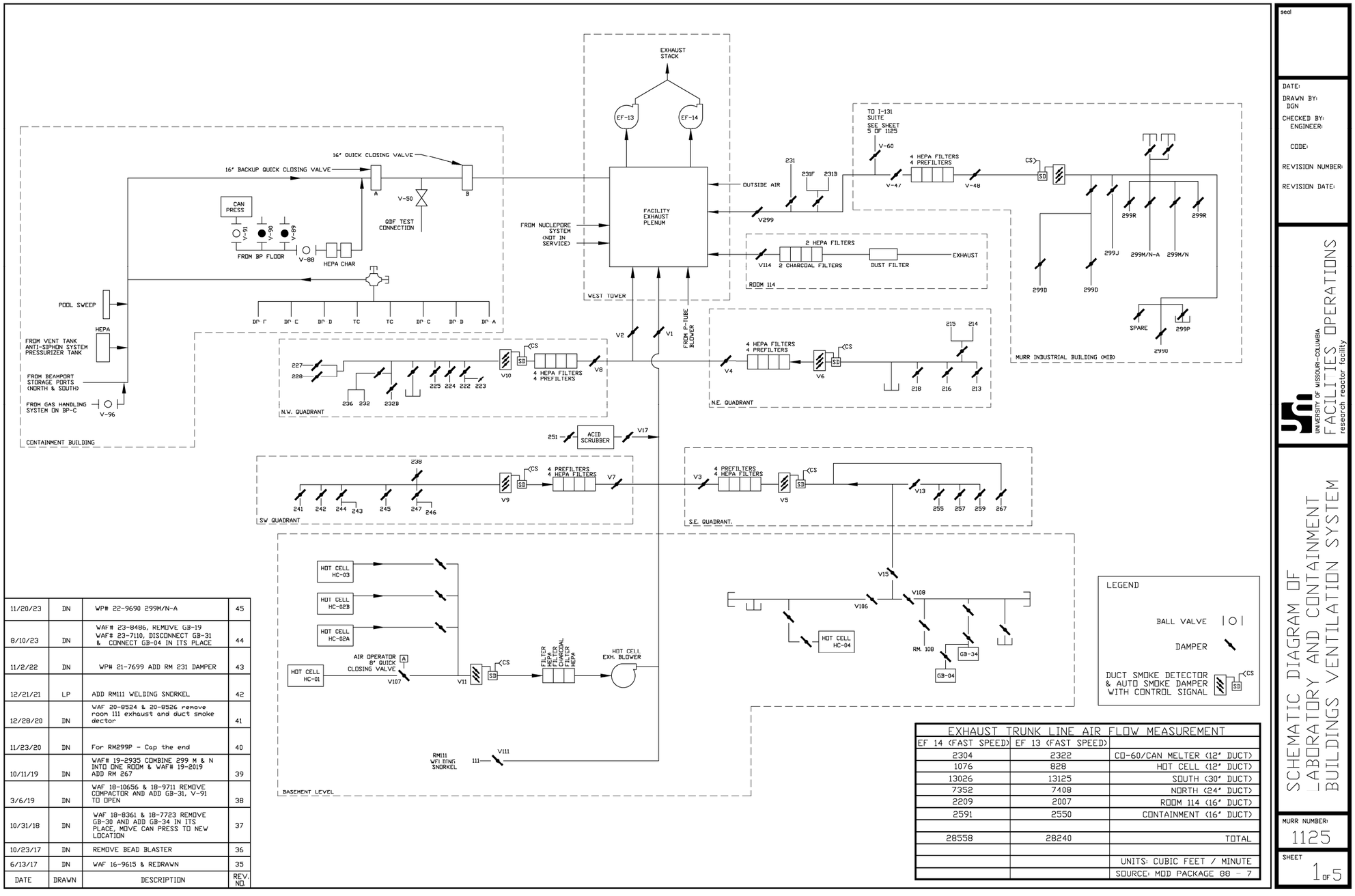
9.14.4 Valve Operation Air System

The valve operation air system is comprised of an air compressor which supplies compressed air to remotely operated primary and pool coolant valves. Each of these valves is air-operated to the appropriate operating position and spring-operated to the fail-safe condition upon a loss of compressed air. The valve operation air compressor will automatically start when the main air 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. It is normally divorced from the other subsystems by isolation valves. These controls include thermostats, humidistats, 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. No reactor safety features are associated with the facility heating and air conditioning system.



11/20/23	DN	WP# 22-9690 299M/N-A	45
8/10/23	DN	WAF# 23-8486, REMOVE GB-19 WAF# 23-7110, DISCONNECT GB-31 & CONNECT GB-04 IN ITS PLACE	44
11/2/22	DN	WP# 21-7699 ADD RM 231 DAMPER	43
12/21/21	LP	ADD RM111 WELDING SNORKEL	42
12/28/20	DN	WAF 20-8524 & 20-8526 remove room 111 exhaust and duct smoke dector	41
11/23/20	DN	For RM299P - Cap the end	40
10/11/19	DN	WAF# 19-2935 COMBINE 299 M & N INTO ONE ROOM & WAF# 19-2019 ADD RM 267	39
3/6/19	DN	WAF 18-10656 & 18-9711 REMOVE COMPACTOR AND ADD GB-31, V-91 TO OPEN	38
10/31/18	DN	WAF 18-8361 & 18-7723 REMOVE GB-30 AND ADD GB-34 IN ITS PLACE, MOVE CAN PRESS TO NEW LOCATION	37
10/23/17	DN	REMOVE BEAD BLASTER	36
6/13/17	DN	WAF 16-9615 & REDRAWN	35
DATE	DRAWN	DESCRIPTION	REV. NO.

EXHAUST TRUNK LINE AIR FLOW MEASUREMENT		
EF 14 (FAST SPEED)	EF 13 (FAST SPEED)	
2304	2322	CO-60/CAN MELTER (12" DUCT)
1076	828	HOT CELL (12" DUCT)
13026	13125	SOUTH (30" DUCT)
7352	7408	NORTH (24" DUCT)
2209	2007	ROOM 114 (16" DUCT)
2591	2550	CONTAINMENT (16" DUCT)
28558	28240	TOTAL
		UNITS: CUBIC FEET / MINUTE
		SOURCE: MOD PACKAGE 80 - 7

seal

DATE:
DRAWN BY: DGN
CHECKED BY: ENGINEER:
CODE:
REVISION NUMBER:
REVISION DATE:

UNIVERSITY OF MISSOURI-COLUMBIA
FACILITIES OPERATIONS
research reactor facility

SCHEMATIC DIAGRAM OF LABORATORY AND CONTAINMENT BUILDINGS VENTILATION SYSTEM

MURR NUMBER: 1125

SHEET 1 OF 5

FIGURE 9.11 – MURR FACILITY VENTILATION SYSTEM

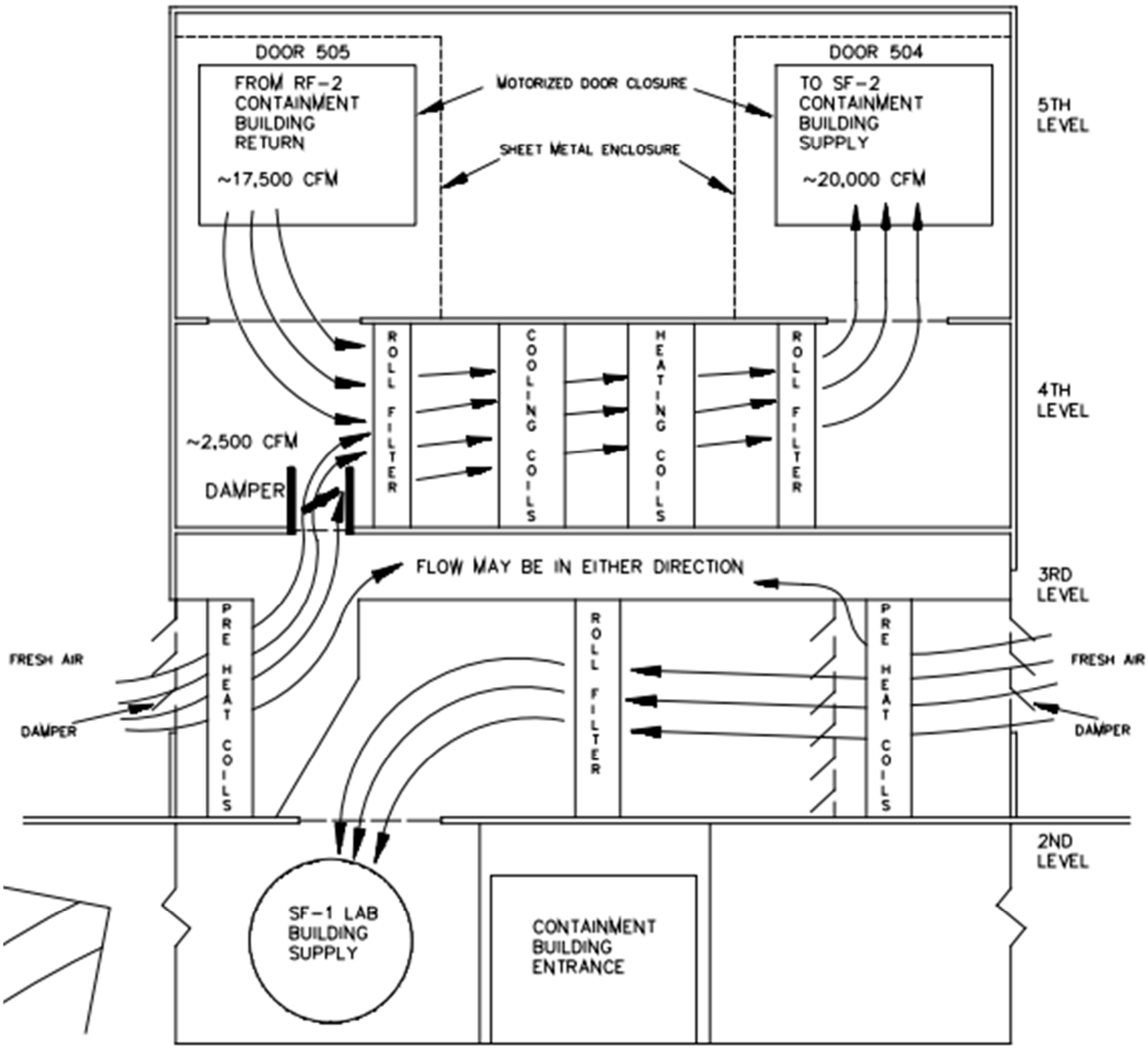


FIGURE 9.2 – MURR CONTAINMENT AIRFLOW PATHS

CHAPTER 10

EXPERIMENTAL FACILITIES AND UTILIZATION

TABLE OF CONTENTS

10.0 EXPERIMENTAL FACILITIES AND UTILIZATION 5
10.1 Introduction 5
10.2 Design 6
10.3 Experimental Facilities 6
10.3.1 Center Test Hole 7
10.3.1.1 Introduction 7
10.3.1.2 Methodology for Measuring the Reactivity Worth of Experiments in the Center Test Hole 7
10.3.1.3 Center Test Hole Canister 9
10.3.1.4 Center Test Hole Canister Latching Mechanism 9
10.3.1.5 Center Test Hole Strainer 10
10.3.1.6 Flux-Trap Irradiations Reactivity Safety Trip (FIRST) 11
10.3.1.7 Safety Analysis 11
10.3.2 Beamports 13
10.3.2.1 Introduction 13
10.3.2.2 Description 13
10.3.2.3 General Requirements 15
10.3.3 Thermal Column 16
10.3.3.1 Introduction 16
10.3.3.2 Description 16
10.3.4 Pneumatic Tube System 18
10.3.4.1 Introduction 18
10.3.4.2 Description 18
10.3.4.3 General Requirements 19
10.3.5 Graphite Reflector Irradiation Positions 20
10.3.6 Bulk Pool Irradiation Positions 20
10.4 Experiment Review 21
10.4.1 Introduction 21
10.4.2 Reactor Utilization Request 21
10.4.3 Review Process 22
10.4.4 Reactor Advisory Committee 24
10.4.5 Reactor Safety Subcommittee 24
10.5 Experimental Facilities Technical Specifications 24
10.6 Hot Cells, Glove Boxes, and Fume Hoods 25
10.6.1 Control of Hot Cells, Glove Boxes, and Fume Hoods 25
10.6.2 General Use Hot Cells, Glove Boxes, and Fume Hoods 26
10.6.3 Main Hot Cell 26
10.7 Iodine-131 Processing Hot Cells 28
10.7.1 Introduction 28
10.7.2 Laboratory Design 28

10.7.2.1 Handling Hot Cell Design 28
10.7.2.2 Processing Hot Cell 29
10.7.2.3 Dispensing Hot Cell 29
10.7.2.4 Ventilation System 29
10.7.3 Hot Cell Ventilation and Filtration Systems 30
10.7.4 Radiation Monitoring Equipment 31
10.7.5 Evaluation of I-131 Release and Dose Consequences 31
10.7.5.1 Assumptions 31
10.7.5.2 Dose Consequences in the Restricted Area 32
10.7.5.3 Dose Consequences in the Unrestricted Area 37
10.7.6 Iodine-131 Processing Hot Cells Technical Specifications 39

LIST OF TABLES

Table 10-1 BEAMPORTS - DESIGNATION, SIZE, ELEVATION, AND DESCRIPTION 13
Table 10-2 HOT CELLS, GLOVE BOXES, AND FUME HOODS 27
Table 10-3 I-131 PROCESSING LABORATORY VENTILATION AND FILTRATION FOR 150 CURIES OF I-131 33
Table 10-4 FILTER SHIELDING THICKNESS 33
Table 10-5 150 CURIE TARGET EXPOSURE RATES DUE TO SHINE FROM FILTERS AT 1 FOOT (IN MREM/H) 34
Table 10-6 HOT CELL TURNOVER TIME 35
Table 10-7 HOT CELL AIR CONCENTRATIONS 36
Table 10-8 VENTILATION EXHAUST FLOW RATE 36
Table 10-9 150 CURIE TARGET EXPOSURE RATES DUE TO SHINE FROM VENTILATION AT 1 FOOT 37
Table 10-10 OFFSITE DOSE CONSEQUENCES OF 150 CI I-131 RELEASE (IN MREM) 38

LIST OF FIGURES

Figure 10.1 – FIRST DEVICE 40
Figure 10.2 – THREE-TUBE CENTER TEST HOLE CANISTER 41
Figure 10.3 – SIX-TUBE CENTER TEST HOLE CANISTER 42
Figure 10.4 – CROSS-SECTIONAL VIEW OF A BEAMPORT ASSEMBLY 43
Figure 10.5 – THERMAL COLUMN ASSEMBLY 44
Figure 10.6 – THERMAL COLUMN DOOR 45
Figure 10.7 – PNEUMATIC TUBE SYSTEM 46
Figure 10.8 – GRADE LEVEL HOT CELLS, GLOVE BOXES, AND FUME HOODS 47
Figure 10.9 – BELOW GRADE LEVEL HOT CELLS, GLOVE BOXES, AND FUME HOODS 48
Figure 10.10 – I-131 PROCESSING LABORATORY VENTILATION AND FILTRATION 49

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10.2 Written communication dated August 18, 2006, SAR Chapters 10-18 as part of 2006 SAR submittal for reactor relicensing. [ML092110597]

10.3 Written communication dated August 31, 2010. Responses to NRC Requests for Additional Information regarding the FIRST amendment request. [ML120050315]

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10.11 Written communication dated February 19, 2021. Submittal of Annual Report for January 1, 2020 through December 31, 2020. [ML21050A251]

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10.13 Written communication dated July 31, 2015, regarding responses to "University of Missouri at Columbia – Request for Additional Information Regarding the Renewal of Facility Operating License No. R-103 for the University of Missouri at Columbia Research Reactor," dated June 18, 2015. [ML15216A122]

10.14 Written communication dated July 20, 2015, requesting U.S. Nuclear Regulatory Commission approval to amend the Technical Specifications appended to Amended Facility License No. R-103 in order to produce radiochemical sodium iodide (I-131). [ML15203A867]

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10.0 EXPERIMENTAL FACILITIES AND UTILIZATION

This chapter describes and discusses the experimental facilities at the reactor facility, their intended use, and the experiment program. (Refs. 10.1 and 10.2)

10.1 Introduction

The experiment program at the University of Missouri Research Reactor (MURR) provides a broad range of analytical, radiographic, and irradiation services for use by the research community and the commercial sector. MURR is designed to provide these services through the use of the following experimental facilities:

Center Test Hole (Flux Trap)
This region of the reactor allows for the placement of a removable sample canister in the peak thermal flux (approximately 6×10^{14} n/cm^2-sec) area of the core;

Beamports
Six (6) beamports penetrate the biological shield and terminate at the beryllium reflector. These hollow tubes provide a path for neutrons and gamma radiation between the reactor core and experimental equipment located on the below grade level of the reactor containment building;

Thermal Column
This experimental facility is designed for the purposes of performing neutron radiographs and large sample irradiations;

Pneumatic Tube System
This automatic system is designed to quickly transfer sample carriers, or "rabbits," into and out of the graphite reflector region from designated laboratories;

Graphite Reflector Region Irradiation Positions
The graphite reflector elements are designed to allow the placement of sample holders in a region of relatively high thermal flux (approximately 1×10^{14} n/cm^2-sec); and

Bulk Pool Region Irradiation Positions
The bulk pool provides an area for the placement of sample holders in a region of relatively low thermal flux (less than 5×10^{13} n/cm^2-sec).

An experiment, as defined in the Technical Specifications (TSs), "is any operation, hardware, or target (excluding devices such as detector or foils) which is designed to investigate non-routine reactor characteristics, or which is intended for irradiation within an irradiation facility. Hardware rigidly secured to a core or shield structure so as to be part of their design to carry out experiments is not normally considered an experiment." Experiments conducted at MURR are subdivided into two general classifications: neutron beam, and neutron irradiation and isotope production. The neutron beam experiments are those research projects which utilize one of the beamports. The neutron irradiation

and isotope production experimental facilities include the center test hole, the graphite reflector region, the pneumatic tube system, and in-pool locations external to the graphite reflector (bulk pool). Some of the irradiation services provided by these experimental facilities include isotope production for the development of radiopharmaceuticals and neutron activation analysis (e.g., archeological samples). The beamports are primarily used for neutron scattering useful in determining the structure of solids and liquids.

The reactor facility 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. MURR also provides support to institutions with reactors operating 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.

10.2 Design

The principal purpose of MURR is to provide neutrons to the experimental facilities, and the design effort has been directed toward accomplishing this end. In order to accomplish the goals of the experiment program, the design of the reactor and the experimental facilities must be rather unique, but must also emphasize safety as a paramount concern. This latter requirement further dictates design criteria. A major safety feature of the reactor can be found in the design of the beryllium reflector, which 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 neutronically decoupled from the reactor in this manner is the center test hole (flux trap). However, this experimental facility is subject to a high degree of administrative and engineering controls as discussed in the following section.

10.3 Experimental Facilities

The MURR experimental facilities are designed, operated, and utilized in a manner that will not exceed the reactor's Limiting Safety System Settings (LSSS) or Limiting Conditions for Operations (LCO) of the TS requirements during normal operations. The radiological controls and as low as is reasonably achievable (ALARA) programs ensure that personnel and public radiation doses do not exceed the requirements of Title 10, Chapter I of the Code of Federal Regulations, Part 20 (10 CFR 20) and are maintained ALARA. The experimental facilities contain sufficient control systems that the reactor is adequately protected. Failures of the experimental facilities do not subject the public or workers to exposures in excess of 10 CFR 20 requirements and do not compromise the ability to operate or shut down the reactor.

10.3.1 Center Test Hole

10.3.1.1 Introduction

The Center Test Hole (Flux Trap) is that portion of the reactor through the center of the core which is bounded by a 4.5-inch (11.43-cm) inside diameter inner reactor pressure vessel (island tube) and which extends 15 inches (38.1 cm) above and below the core centerline. A specially-designed center test hole canister is inserted into this region of peak thermal flux (6×10^{14} n/cm^2-sec) for the purpose of material irradiations.

The flux trap provides a rather large reactivity effect if utilized without proper restrictions and supervision. Consequently, use of this experimental facility is subject to both administrative and engineering controls in order to minimize the possibility of inadvertent insertion or removal of a high positive or negative reactivity worth sample during reactor operation. To eliminate the possibility of this occurring, reactor operating procedures require that the center test hole canister is inserted or removed only when the reactor is in a shutdown condition. Each time the Center Test Hole Canister is installed, proper latching is verified by two members of Reactor Operations staff, one being a senior Reactor Operator, both visually and physically. If the center test hole canister is not utilized during reactor operation, a strainer is installed to prevent foreign objects from entering the center test hole.

10.3.1.2 Methodology for Measuring the Reactivity Worth of Experiments in the Center Test Hole

The methodology used to determine the reactivity present in the Center Test Hole involves several independent measurements described as follows:

The reactivity neutral condition of the reactor, as far as the central flux trap region is concerned, assumes a center test hole filled with water at 110 °F (43.3 °C) and no individual experimental samples or the Center Test Hole Canister (also known as the removable experiment sample canister as defined in TS 1.3) installed.

The reactivity contribution of the Center Test Hole Canister is first determined by performing two back-to-back reactor startups; the first one with the Center Test Hole empty (only water present) and the second one with an empty Center Test Hole Canister, either the 3- or 6-barrel as described in Section 10.3.1.3. The difference in the control rod heights between the two startups in conjunction with the known integral rod worth curve is then used to determine the reactivity contribution from the empty Center Test Hole Canister. Any reactivity contribution due to a change in temperature is accounted for by the primary and pool coolant temperature coefficients of reactivity and the corresponding temperature difference between the two critical measurements.

The effects of fission product poisons are very minimal during reactivity worth measurements for the following reason: all reactivity measurements are performed during a scheduled reactor startup following a weekly maintenance day shutdown period. As part of the maintenance day activities, all eight (8) fuel elements are replaced with different elements of varying stages of burn-up, and

consequently, xenon effects are typically not present during the startup. When the reactivity worth of the loaded Center Test Hole Canister is being determined, the core will be loaded with only xenon-free fuel elements. Additionally, the reactivity measurements are conducted at very low power levels specifically to avoid fission product poison build up during multiple startups. Typically, these measurements are performed at or below 50 kW, or less than 0.5% of the MURR full core power level of 10 MW, which is sufficient to provide reliable nuclear instrumentation indication, yet not enough to produce a significant amount of poison in the core. Also, the amount of time spent at these low power levels are limited to only a few minutes, again for reducing the amount of poison build up in the core. Because of these precautions, effects from poisons are minimized and no poison corrections are required during reactivity worth estimations.

The same methodology is used to determine the total reactivity contribution from a fully loaded Center Test Hole Canister. That is, two back-to-back reactor startups are performed; one with just water in the central region and the second one with the loaded Center Test Hole Canister. The difference in control rod heights is used to determine the reactivity contribution from the loaded center test hole canister.

Individual sample materials are irradiated in the Center Test Hole by placing them inside tightly sealed standard sized aluminum cans. Materials that require multiple encapsulations are first placed inside smaller sealed quartz or aluminum vials and then placed within the welded aluminum cans. Typically, several of these individual sample cans are placed inside the tubes that make up the center test hole canister. The reactivity contribution from these individual sample targets are also measured in a very similar fashion. After first estimating the possible reactivity worth of a new sample material to ensure no reactivity limits will be exceeded, two back-to-back reactor startups are performed. First, a hollow thin-walled aluminum can-filled with water (referred to as water can) with the same dimensions as the individual sample can is placed within one of the Center Test Hole Canister tubes and a reactor startup is performed. Next the water can is removed and replaced with the sample can of unknown reactivity and a second reactor startup is performed. The rod worth difference in the control rod heights is then used to determine the reactivity worth of the unknown sample material.

Both analytical and experimental results demonstrate that the worth of a control rod is minimally influenced by the samples or the lack of samples (strainer) or the different types of sample holders in the central flux trap region. Because the control elements of MURR are situated entirely outside of the outer reactor pressure vessel, the eight (8) fuel elements, which make up the reactor core, are situated between the control rod water gap and the inner pressure vessel flux trap region where the sample filled center test hole canister is positioned. This arrangement reduces the coupling between the control rods and the effects of samples or sample holders in the center test hole region, minimizing the impact on rod worth characteristics whether the strainer, the 3-barrel, or the 6-barrel center test hole canister, are installed. (Refs. 10.3 and 10.4)

10.3.1.3 Center Test Hole Canister

The number and the volume of samples which can be irradiated in the center test hole are limited mechanically by the design of the center test hole canister. Three (3) center test hole canisters are designed and approved for use at MURR, those being a six-tube, a three-tube, and a single-tube.

The three-tube center test hole canister (Figure 10.1) consists of three (3) aluminum tubes, each being 10 feet 2 inches (3.1 m) long with an internal diameter (I.D.) of 1.334 inches (3.4 cm). The tubes are arranged in a clover leaf pattern and spot welded together to form a single assembly. The tubes are clearly identified as 'A', B, and C, both physically on the center test hole canister and in all sample loading documentation. Stainless steel bands wrapped around the aluminum tubes provide redundancy for the spot welds. A support rod and base piece are attached to the bottom of the tube assembly. The overall length of the center test hole canister is 14 feet 2.5 inches (4.3 m).

The six-tube center test hole canister (Figure 10.2) is similar in design to the three-tube test hole canister with a few exceptions. There are an additional four (4) vertical inches of irradiation capacity in the three 1.334-inch (3.4-cm) I.D. tubes and three (3) smaller diameter irradiation tubes [0.68-inch (1.7 cm) I.D.]. The small diameter tubes are designed to allow movable and unsecured experiments to be irradiated in the flux trap. The small diameter tubes are clearly identified as 1, 2, and 3, both physically on the center test hole canister and in all sample loading documentation.

The experiment volume of the larger diameter tubes of both the three- and six-tube center test hole canisters is filled at all times with aluminum or titanium spacers or experiment capsules (samples). The samples and spacers are maintained in position by a hold down rod assembly which is secured at the top of the canister by a head pin and pin head keeper. A 1/16-inch (1.59-mm) annulus exists between the samples and the internal wall of the sample tubes to provide cooling. The small diameter tubes of the six-tube center test hole canister may or may not contain experiment samples during operation.

All samples placed in the center test hole canister for irradiation are seal-welded, leak checked and have a negative buoyancy. Small diameter seal-welded samples may be placed in flooded carrier cans.

A single-tube center test hole canister is also designed for use in the center test hole. It was used during the initial operation of MURR and replaced by the three-tube assembly when an increase in material irradiation capacity was required.

10.3.1.4 Center Test Hole Canister Latching Mechanism

The following is a description of the center test hole canister latching mechanism: When installed the center test hole canister position is positively determined by a latching mechanism located at the top of the assembly. The latching mechanism consists of two Inconel metal fingers which secure the canister to the upper portion of the inner reactor pressure vessel. The fingers, which are attached to the upper section (head piece) of the canister by four (two per finger) countersunk, stainless steel screws, are positioned 180 degrees apart and are totally independent of one another. To provide additional vertical

alignment and support, the canister base piece engages into a test hole slot which is welded to the reflector tank base flange. The center test hole canister latching mechanism and method of vertical alignment and support have not been modified or altered from its original design and no failures have occurred in its operating history.

The condition of the latching fingers and stainless-steel screws are closely inspected at least twice a week; first, when the canister is removed after the reactor is shutdown to retrieve the irradiated samples and second, before it is reinserted into the inner pressure vessel after new samples have been loaded into the canister. In the extremely unlikely event of the failure of a single latching finger, the other finger can hold the canister in place under all operating conditions.

One latching finger fully meets the intent of the definition of a Secured Experiment (MURR TS 1.24), which states, "A secured experiment is any experiment which is rigidly held in place by mechanical means with sufficient restraint to withstand any anticipated forces to which the experiment might be subjected." Additionally, the overall length of the center test hole canister is 170.25 inches (4.32 m), and it is positioned vertically within the 4.5-inch (11.43-cm) inside diameter inner reactor pressure vessel. The weight of the center test hole canister, gravity and a downward pool coolant flow rate of approximately 100 gallons-per-minute (378.5 lpm) through the inner reactor pressure vessel collectively aid to provide additional downward force to maintain the canister in place under all operating conditions.

As described earlier, the center test hole canister latching fingers are positioned 180 degrees apart and are totally independent of one another. There is no common failure that could affect both latching fingers.

The latching mechanism, consisting of two separate, independent latching fingers, provides a redundant, positive latch of the center test hole canister to the inner reactor pressure vessel. Additionally, a special removal tool, with an overall length of approximately 20 feet (6.1 m), is required to unlatch and remove the center test hole canister from the inner pressure vessel. The head of the removal tool must be inserted into the center test hole canister wear ring to spread the latching fingers sufficiently to disengage the canister from the inner pressure vessel. Knowledge of the tool's operation, in addition to the skill that is required to operate the tool under nearly 17 feet (5.2 m) of pool water, also helps to prevent the inadvertent removal of the center test hole canister while the reactor is operating. (Reference 10.5)

10.3.1.5 Center Test Hole Strainer

The Strainer is installed to preclude the possibility of any foreign objects from entering the flux trap region if the reactor is operated without the Center Test Hole Canister inserted. The Strainer latching mechanism is of the exact same design as that of the Center Test Hole Canister. When installed, the Strainer is positively determined by a latching mechanism located at the top of the assembly. The Strainer latching mechanism has not been modified or altered from its original design and no failures have occurred the history of its of operation.

The overall length of the center test hole Strainer is 7 inches (17.8 cm).; it does not enter the flux trap region of the core and there is no reactivity contribution to the reactor when it is installed. [Ref 10.6]

10.3.1.6 Flux-Trap Irradiations Reactivity Safety Trip (FIRST)

The FIRST device and instrumentation were approved by Amendment No. 35 to Amended Facility License No. R-103 in a letter dated February 9, 2012. The FIRST device consists of equipment and electronics that will initiate a reactor scram if certain conditions are not met with the operation of the Center Test Hole. The FIRST device contains limit switches which indicate if the Center Test Hole Canister or strainer is secure in the Center Test Hole. This change to the Reactor Safety System effectively makes the Center Test Hole Canister an experimental facility and a normal part of the reactor. In addition to administrative controls, the FIRST device is designed to provide an input signal to the reactor safety system such that the reactor may not be operated unless the center test hole canister, or strainer, is properly installed – inserted and latched. When the FIRST device is in use, these stringent engineered and administrative controls allow the reactivity worth associated with the center test hole canister not be included with the total reactivity worth of the experiment capsules in the center test hole canister.

The FIRST system is tested for operability on a semiannual basis by surveillance procedure, “CP-36, FIRST scrams,” and it is channel tested as part of each reactor startup as part of FM-57, “Long Form Startup Checklist.”

The FIRST device is comprised of the following major components: Flux Trap Holder Wear Ring, Sensor Switches, Support Rig, Relays and Power Supply, safety System Interface, and FIRST Bypass switches.

Additional detail on the FIRST device and instrument channels are provided in Section 7.10.2.

The methodology that is used to determine the total reactivity present in the flux trap region, including the reactivity contribution of the center test hole canister itself, is the same methodology used when the FIRST system is in BYPASS. BYPASS operation requires Reactor Manager approval to perform.

The methodology used when the FIRST system is in use does not include the reactivity contribution of the Center Test Hole Canister but continues to sum the reactivity worth of the individual samples. The sum of the individual sample reactivity worth is subject to existing Technical Specification limits, as well as more restrictive administrative limits, and must be approved by the Assistant Reactor Manager-Physics and Lead Senior Reactor Operator. (Refs. 10.3, 10.4, 10.5, 10.7, 10.8, 10.9, 10.10)

10.3.1.7 Safety Analysis

At 10 MW operation, the peak unperturbed thermal flux in the center test hole is approximately 6×10^{14} n/cm^2-sec. This region is cooled by bulk pool water flowing downward through the island tube to the reflector plenum. The relatively stable bulk pool temperature prevents the center test hole temperature from responding quickly to core temperature changes and the core negative temperature coefficient adequately offsets any positive reactivity effects from the center test hole.

Because of its spatial importance, and the sensitivity of the reactor to reactivity changes in this region, the center test hole has particularly rigorous construction and administrative safety controls. Only movable experiments shall be installed or removed from the center test hole while the reactor is in operation. All other experiments shall be installed or removed with the reactor in a shutdown condition. Additionally, secured experiments shall be rigidly held in place and locked into position during reactor operation.

During low power testing for the conversion from the uranium alloy fuel core to the uranium-aluminide UAl_x dispersion fuel core, the temperature and void coefficients of the primary were carefully measured and found to be close to the original calculated values. Two independent measurements were performed to confirm these results. The final experimental values were -7.0×10^{-5} Δk/°F for the temperature coefficient and -2.51×10^{-3} Δk/% void for the void coefficient. The transient analysis for 10-MW operation to determine a safe step reactivity insertion in the center test hole using these values is described in Chapter 13, "Accident Analyses."

Each experiment is carefully reviewed to ensure safety and its reactivity worth is mathematically determined and/or measured. Prior to placing in the reactor, each proposed center test hole canister loading is reviewed and the reactivity worth of all samples is determined.

The most likely accident is the failure of a single experiment in the center test hole. The worst-case scenario is the sudden bursting of a sample can and the resulting discharge of its contents, with the possible damage to an adjacent sample can. Experiments shall be limited such that the failure of any single experiment cannot introduce a reactivity change of greater than 0.006 Δk/k. The limit for each individual experiment places utmost importance on a critical review by the Reactor Manager and, if required, the Reactor Advisory Committee (RAC).

Finally, the FIRST device will prevent the insertion of positive or negative reactivity from either the insertion or removal of the center test hole canister because (1) the reactor cannot be started up without the center test hole canister or strainer inserted and latched, and (2) removal of the center test hole canister while at power will cause a reactor scram. Because of the redundant FIRST device sensor switches, the reactor would immediately shutdown if the center test hole canister moved outward less than 1/4-inch (maximum travel of the sensor switches). It should be noted that because of the positive void coefficient of the flux trap region (approximately +0.865 x 10-5 Δ/K/cc void), removal of the center test hole canister would insert negative reactivity into the core.

Because of the FIRST device, insertion of positive or negative reactivity from either the insertion or removal of the center test hole canister while the reactor is operating is no longer considered a credible accident. The only credible accident is the failure of a single experiment in the center test hole. The worst- case scenario is the sudden bursting of a sample can and the resulting discharge of its contents, with the possible damage to an adjacent sample can. Experiments in the center test hole will continue to be limited by Technical Specification 3.1.h such that the failure of any single experiment cannot introduce a reactivity change of greater than 0.006 ΔK. The transient analysis for 10-MW operation to determine the maximum safe step reactivity insertion is described in Section 3.5 of Addendum 3 to the

Hazards Summary Report. The analysis shows that MURR can withstand a rapid step reactivity insertion of 0.007 ΔK with no core damage. (Reference 10.9)

10.3.2 Beamports

10.3.2.1 Introduction

The reactor is designed to accommodate six (6) horizontal beamports. These hollow tubes channel neutrons and gamma radiation from the reactor core with minimal scattering and attenuation between the beryllium reflector and the experimental equipment. Each beamport designation, size, elevation, and description is shown in Table 10-1.

10.3.2.2 Description

The six (6) beamports are arranged with three (3) on each side of the reactor core and are spaced at 30° intervals. The 4-inch (10.16-cm) beam tubes are the equivalent of 4-inch (10.16-cm) schedule 40 aluminum pipe and the 6-inch (beam tubes are the equivalent of 6-inch (15.24-cm) schedule 40 aluminum pipe. When fully inserted, each beam tube penetrates the graphite reflector region and terminates 1/8 of an inch (3.18-mm) +/- 1/16 of an inch (1.59 mm) from the beryllium reflector. The terminal (source) end of each beam tube is shaped to match the outer diameter of the beryllium ring. Both the 4- and 6-inch (10.16- and 15.24-cm) beam tubes increase to 8 inches (20.32 cm) approximately 40 inches (1.0 m) from the biological shield face prior to terminating in a recessed vestibule.

TABLE 10-1
BEAMPORTS - DESIGNATION, SIZE, ELEVATION, AND DESCRIPTION

Designation	Size (I.D.)	Elevation (with respect to Reactor Core Centerline)	Description
Beamport 'A'	4-inch (10.16-cm)	minus 2 inches (5.08 cm)	radial
Beamport B	6-inch (15.24-cm)	minus 7 inches (17.78 cm)	radial
Beamport C	6-inch (15.24-cm)	minus 14 inches (35.56 cm)	radial-tangential[a]
Beamport D	4-inch (10.16-cm)	minus 14 inches (35.56 cm)	radial-tangential[a]
Beamport E	6-inch (15.24-cm)	minus 7 inches (17.78 cm)	radial
Beamport F	4-inch (10.16-cm)	minus 2 inches (5.08 cm)	radial

a. The radial-tangential beamports are perpendicular to the beryllium reflector as are the radial beamports; however, their elevations are below the fuel region and they do not view the reactor core directly.

Each beamport assembly (Figure 10.3) consists of three (3) major components: a fixed beamport liner, a removable beamport liner (beam tube), and typically, a removable collimator liner. The fixed beamport liner provided the necessary form work during the pouring of the biological shield; it is the only component of the beamport assembly which is in direct contact with the magnetite concrete of the biological shield. The fixed liner is integrally welded to the reactor pool liner and serves as an extension

of the reactor pool into the beamport vestibule. The fixed beamport liner allows penetration of the removable liner through the biological shield from the recessed vestibule inward, penetrating the graphite reflector region and terminating adjacent to the beryllium reflector. The portion of the beam tube within the graphite reflector region is cooled by pool water flowing downward through the gaps between the graphite reflector elements and around the beam tube. When the removable liner is fully inserted, a gap exists between it and the fixed liner. A 0.5-inch (1.27-cm) line, which penetrates the fixed beamport liner, returns water from the pool skimmer system into this area and then back into the reactor pool volume. This flow path prevents the stagnation of pool water in the beamport, helping minimize corrosion to the fixed and removable liners. The removable liner is attached to the fixed liner by means of a bolt ring located in the recessed vestibule. A packing gland assembly seals the removable liner to the fixed liner. Typically, a removable collimator liner is located within the removable beamport liner. The removable beamport liner provides a path for the neutron beam to travel prior to entering the removable collimator liner, where the beam is further shaped and defined by the collimator for experimental equipment. The collimators are designed to accept different neutron filters. Crystal-sapphire and silicon are the predominant choices for filters utilized in the MURR neutron scattering experiment program. The removable collimator liner is attached and sealed to the removable beamport liner similarly to the way the removable liner is attached and sealed to the fixed liner. This allows filling the removable beamport liner with helium if a neutron beam is to be accessed, or demineralized water if not.

The recessed vestibule of each beamport is serviced by a 2-inch (5.08-cm) off-gas vent line, a 1-inch (2.54-cm) drain line, and two (2) 2-inch (5.08-cm) conduit sleeves. The conduit sleeves provide a path for instrumentation or equipment to be routed between the vestibule and the step-back (mezzanine level) of the biological shield. Utilities provided to each beamport include demineralized water, domestic cold water, vacuum, and 110-Vac electrical supply power.

Each beamport can be closed by a 3-inch (7.6-cm) thick lead-filled shield door located in the biological shield above the recessed vestibule. The shield door is opened and closed by the use of the reactor containment building 15-ton (13,607-kg) overhead rectilinear crane. When a neutron beam is accessed, the door is raised and two (2) pins are installed to secure it in the open position.

Two (2) concrete structures (hot storage ports) used to store contaminated beamport equipment and other activated components are located on the west side of the below-grade level of the reactor containment building, and in close proximity to the beamports. Each structure contains 17 storage ports, 10 feet 8 inches deep (3.3 m), with a 0.5-inch (1.27-cm) vent line connected to each port. The vent lines are ducted to the reactor containment building 16-inch (40.64-cm) hot exhaust line to ensure that any potential airborne contamination or radioactive gases which may accumulate within the ports are discharged through the facility ventilation exhaust stack. Shield plugs fabricated from steel pipe and filled with 6 to 10% antimonial lead alloy are used to cap the openings of the storage ports to reduce radiation levels in the area.

10.3.2.3 General Requirements

There are four (4) hazards of primary concern that are created with the utilization of a beamport for an experiment activity. They are as follows:

1. Changes in reactor reactivity due to beamport activities such as draining or flooding a beamport;

2. Exposure of personnel to radiation as a result of movements of shielding or inadequate shielding;

3. Release of radioactive gases such as argon-41 which are produced in the beamport; and

4. Production of explosive or toxic materials in the beamport.

The following limitations and operating guidelines have been established to minimize or eliminate these potential beamport hazards from occurring.

1. To ensure compliance with Section 50.54(j) of 10 CFR 50, which states "Apparatus and mechanisms other than controls, the operation of which may affect the reactivity or power level of a reactor shall be manipulated only with knowledge and consent of an Operator or Senior Operator licensed pursuant to Part 55 of this chapter present at the controls," all beamport evolutions such as draining, filling, or evacuating the beamport will be performed by licensed operators. Care must also be exercised in filling a drained beamport. During reactor operation, air within the port will become activated. This activated air is forced out of the port during the filling evolution. The activated gases present a radiation hazard to personnel in the reactor containment building and could result in a release of radioactive gases in excess of the license limit. This potential hazard also warrants that only licensed operators are permitted to drain, fill, or evacuate a beamport. Whenever practical, all changes in beamport status shall be made only after the reactor has been shut down for at least eight (8) hours.

2. All shielding movements (including temporary movements which will be returned to normal) shall be coordinated with Reactor Health Physics personnel. This restriction is necessary to ensure that no personnel radiation hazard is introduced by the movement of beamport shielding. Radiation surveys are performed on a regular basis to determine the adequacy of the beam stops and shielding barricades. In addition to the routine radiation surveys, permanently installed radiation detectors are strategically located on the below-grade level of the reactor containment building for the purpose of detecting radiation leakage from beamport experiments. Each radiation detector output is fed into one channel of the Area Radiation Monitoring System (ARMS) where the signal is processed and displayed on a meter located in the reactor control room. The meters

are equipped with adjustable set point trips that initiate audible and visual alarms upon detection of a high radiation level.

3. The gamma and neutron radiation levels in a beamport can induce chemical reactions which would normally require extreme temperature and/or pressure conditions in a laboratory. One of the primary sources of a potential explosion due to an induced chemical reaction is the use of nitrogen cryostats in beamport experiments. Liquid nitrogen, in the presence of a high radiation field, may produce ozone from any oxygen impurities in the nitrogen. Ozone, an allotropic form of oxygen, can undergo spontaneous decomposition if exposed to rapid temperature fluctuations or to slight but sudden pressure changes. To ensure a buildup of ozone does not occur, strict operating procedures such as allowing the nitrogen cryostat to warm to ambient temperature at least every five days are established. The LCO of an experiment, as listed in the Technical Specifications, are quite restrictive on the use of, or the generation of, explosive materials in an experiment.

10.3.3 Thermal Column

10.3.3.1 Introduction

The Thermal Column experimental facility is comprised of (up to) a 60-inch (1.5-m) thick graphite stack contained within an aluminum housing and is used for performing neutron radiographs and large sample irradiations. The facility is designed such that dose rates no greater than 2.5 mrem/h are received at points one foot from the thermal column door or from any neighboring shield surface with the reactor operating at 10 MW, the thermal column door in the closed position, and the full graphite stack installed.

10.3.3.2 Description

The Thermal Column is an integral unit which penetrates the biological shield, extends into the reactor pool, and terminates at a removable lead shield positioned between the reflector region (reflector tank outer wall) and the terminal end of the thermal column. The entire assembly is contained within an aluminum housing which reduces in size within the reactor pool (Figure 10.5). The larger section of the aluminum housing is a 4 feet 2-inch (1.3-m) square by 5 feet 8-inch (1.7-m) deep box completely lined with 0.25-inch (6.35-mm) thick BORAL sheeting. The majority of this section is contained within the biological shield. The portion of the aluminum housing extending into the reactor pool is functionally an integral part of the reactor pool liner. This section reduces in size to a 3 foot 1.5-inch (0.95-m) square by 12.25-inch (0.3-m) deep box. The in-pool portion of the aluminum housing is designed and supported to withstand the hydrostatic pressure of the reactor pool water and to support the load of the graphite stack. The removable lead shield, formed to fit the reflector tank, is attached to the front face of the thermal column to attenuate the reactor core gamma radiation. The front face of the thermal column housing is shaped to match the contour of the lead shield. A 0.25-inch (6.35-mm) cooling slot with a +/- 0.125 inch (3.18 mm) tolerance exists between the canned lead shield and the aluminum housing.

The overall length of the graphite stack is 5 feet (1.5 m) as measured along the centerline of the thermal column. The stack is designed such that the central square array of nine graphite stringers are removable from the outer portion of the thermal column. Each stringer may be removed individually and is a single block of graphite with an approximate 4 inches (10.2 cm) square cross section.

Two 4-inch (10.16-cm) vents are installed immediately external to the outer face of the graphite. This allows any radioactive gases produced in the thermal column to be vented to the reactor containment building ventilation exhaust system.

A neutron radiographic variable aperture is positioned through the center of the graphite stack. A bismuth filter used to attenuate gamma rays is affixed between the lead gamma shield and the aperture opening. The aperture is aligned with an exit collimator located in the thermal column door.

The thermal column door is a multi-ton shield piece that requires very deliberate action to be operated, including removing a large radiography beam stop, relocating extremely large shielding pieces, and electrically connecting the drive motor. The thermal column door moves on two (2) level floor tracks and is driven by an electric motor through a gear reducer box. The inner surface of the door is lined with 0.25-inch (0.635-cm) thick BORAL sheeting.

Operation of the door is administratively controlled through reactor operations procedure EX-RO-126, "Thermal Column Door." Included within the "Precautions and Limitations" and "Prerequisites and Initial Conditions" sections of these procedures are specific steps which require the reactor to be shut down and remain shut down while the thermal column door is open. The intent of the Rod Withdrawal Prohibit Circuit is to prevent startup of the reactor unless all five (5) of the conditions listed in Section 7.5.3.1, including the thermal column door, are satisfied. A limit switch (622) is mounted to the side of the door which provides a signal to the Rod Withdrawal Prohibit Circuit, preventing the control rods from being withdrawn with the thermal column door not in the fully closed position.

If, however, the reactor was in operation and the thermal column was inadvertently opened, the associated contact in the Rod Withdrawal Prohibit Circuit would open preventing the withdrawal of the control rods and provide a "Thermal Column Door Open" annunciation in the reactor control room. This would not remove any ability of the reactor operator to insert control rods; however, it would prevent any increase in reactor power and allow the reactor operator to determine what necessary actions need be taken.

The operation of the thermal column door at power would not be a reactor safety issue; it would be a radiation protection concern. The thermal column door would only need to travel a very short distance (a few inches) to cause the west wall Area Radiation Monitor (ARM) to reach the alarm setpoint of 20 mrem/hr. This alarm would provide information to individuals at the door and also in the Reactor Control Room to ensure prompt corrective action. The Control Room Operators would then initiate the immediate and subsequent actions of reactor operations emergency procedure REP-4, "High Radiation Levels." The thermal column door is shown in Figure 10.6. (Reference 10.5)

10.3.4 Pneumatic Tube System

10.3.4.1 Introduction

The Pneumatic Tube (P-Tube) System, shown in Figure 10.6, is designed to quickly transfer individual samples into and out of the graphite reflector region of the reactor core assembly. The samples are placed in small high density polyethylene sample carriers, or "rabbits," and transported at velocities of 30 to 45 feet per second (9.1 to 13.7 m/s) from designated laboratories into a region of relatively high thermal flux (approximately 1×10^{14} n/cm^2-sec) for the purpose of material irradiation.

10.3.4.2 Description

The P-Tube System is a standard 1.5-inch (3.81-cm) diameter, vacuum-operated sample transfer system consisting of two (2) turbo-compressors, a solenoid control valve cabinet, two (2) electric-operated switch assemblies, four (4) in-pool concentric tube terminals, two (2) sending-receiving stations, and associated piping and tubing. The system was originally capable of simultaneously transferring four (4) rabbits either into or out of the graphite reflector region. Only two (2) terminals and sending-receiving stations are currently in use.

A sample prepared for irradiation is placed in a rabbit, which is then inserted into a sending-receiving station. The system is completely automatic once the rabbit is dispatched from a sending-receiving station to the reactor. Each sending-receiving station has a timing circuit which controls the length of the irradiation time and the return of the rabbit. The rabbit travels through the sample carrier tubing from the sending-receiving station into the reactor containment building, past a photocell, and then to the terminus located in a graphite reflector element. It returns to the laboratory along the same path. The photocell starts the timing circuit and illuminates a "rabbit in reactor" light both at the control station and in the reactor control room. Directional air flow, supplied by the two (2) turbo-compressors, moves the rabbit between the sending-receiving station and the terminus. A cabinet located on the basement level of the laboratory building adjacent to the turbo-compressors contains solenoid valves which control direction of the air flow. The air flow is designed to pull the rabbit from place to place rather than push it, minimizing the possibility of fragments from a broken rabbit becoming trapped in the terminus. This arrangement also ensures that airflow direction is into the P-Tube System should a leak develop in the sample carrier tubing.

The length of the sample carrier tubing, which penetrates the biological shield and the reactor pool liner and terminates in the graphite reflector region (in-pool portion), consists of prefabricated concentric tubing rather than a separate sample and air tube. The outer tube (air tube) has a 2.25-inch (5.7-cm) diameter and the inner tube (sample tube) has a 1.5-inch (3.8-cm) diameter. Immediately on the water-side of the reactor pool liner is an aluminum concentric flange which allows for the removal and installation of the in-pool portion of the sample carrier tubing.

Two (2) 7.5-HP (5.6-kw), 190-cfm (5.38-m^3/min) overhung-type turbo-compressors, designated the east and west P-Tube blowers, circulate the required air flow through the P-Tube System. The turbo-

compressors draw air from the laboratory building supply plenum (cold deck), through a HEPA filter, the solenoid cabinet and transfer lines, and yet another HEPA filter prior to discharging into the laboratory building exhaust system. To ensure the reactor operator has overall command of the P-Tube System, electrical power to the turbo-compressors is controlled by a two-position (ON-OFF) switch located on the reactor control room console.

In order to minimize the instantaneous release of argon-41 (Ar-41) which is produced in the terminus, the P-Tube System incorporates two (2) design features. The solenoid-operated control valves are positioned such that a continuous flow path for air exists through the sample carrier tubing even when the P-Tube System is secured. This prevents the buildup of Ar-41 in the terminus. Also, a time delay circuit starts the east p-tube blower approximately 15 seconds after the west p-tube blower, minimizing air surges through the system.

The P-tube system also has isolation valves that can be closed to prevent the continuous flow of air and subsequent Ar-41 production from the system. Isolation of the system can only be performed after all P-tube irradiations are complete, and the system is not required. The isolation valves are administratively controlled and will normally be re-opened after the reactor is shut down and sufficient decay time has elapsed for the removal of Ar-41. (Reference 10.11)

10.3.4.3 General Requirements

- The following are the general requirements of the P-Tube System which are necessary for the system to accomplish its design goal.
- The concentric terminals are designed to adequately remove the heat generated in the inner tube wall when the reactor is operated at 10 MW, the airflow is stopped, and no rabbit is in the terminal.
- The maximum heat production is approximately 4 watts/cm^3 in the aluminum structure at 10-MW operation; the heat transfer coefficient at the outer tube wall is approximately 900 Btu/ft^2-h-°F; and the temperature of the pool water flowing along the outer tube is approximately 105 °F (41 °C).
- The depth of the insertion of the pneumatic terminals into the graphite reflector elements is such that a rabbit is centered at core centerline in the inner facility and at a sufficiently lower elevation in the outer facility to minimize shadowing effects.
- The minimum radius on all tubing bends is 24 inches (61 cm).
- Each sending-receiving station is equipped with an adjustable timer which is initiated by the rabbit. The timer operates the proper solenoid-operated control valves, in the correct sequence, to return the rabbit at the conclusion of the pre-set time.

10.3.5 Graphite Reflector Irradiation Positions

The graphite reflector region of the reactor core assembly consists of removable reflector elements which have been designed to accept aluminum sample holders for the purpose of material irradiation in a region of relatively high thermal flux (approximately 1×10^{14} n/cm^2-sec). These positions are intended for large volume irradiations and for samples of greater size, or which require longer irradiation times than can be provided by the P-Tube System. The graphite reflector region is discussed in greater detail in Chapter 4, "Reactor Description."

Material to be irradiated in this region is first encapsulated in an aluminum sample canister and then inserted into an aluminum sample holder. The sample holder is lowered into a graphite reflector element by a handling line. For samples that require rotation, an extension rod connects the sample holder to a motor drive assembly mounted on the upper bridge above the reactor pool surface. The motor drive assembly, through a reduction gear, rotates the sample holder at a slow rotational speed for uniform irradiation of the sample material. The motor drive assembly is protected by a clutch mechanism in the event the extension rod or sample holder binds or locks in position. Sample materials with very small cross-sections (area) typically do not require rotation.

All sample material which is irradiated in the graphite reflector region is encapsulated in either a seal-welded or crimp-sealed aluminum canister, or a threaded aluminum capsule. The seal-welded aluminum canisters are pressure tested prior to use unless they are to be run as flooded samples. When practical, the sample material will be doubly encapsulated to minimize the possibility of release into the reactor pool. All corrosive material is doubly encapsulated. The primary encapsulation may be an aluminum canister (seal-welded or threaded) or a sealed quartz vial. Sample canisters are weighted, if necessary, to ensure negative buoyancy.

10.3.6 Bulk Pool Irradiation Positions

The bulk pool is the water region above and to the outside of the graphite reflector which provides an area for the placement of sample holders in a region of relatively low thermal flux (less than 5×10^{13} n/cm^2-sec).

Material to be irradiated in this region is first encapsulated in an aluminum sample canister and then inserted into an aluminum sample holder. The sample holder is lowered into a designated bulk pool irradiation position by a handling line. The sample material may or may not require rotation. All sample material which is irradiated in the bulk pool region meets the same encapsulation requirements as stated in Section 10.3.5.

10.4 Experiment Review

10.4.1 Introduction

All experiments conducted at MURR must be approved by the Reactor Manager. The mechanism for obtaining this approval is a Reactor Utilization Request (RUR). This document outlines certain criteria which are considered during the review and approval of any reactor experiment. The criteria may be summarized as follows:

- Criticality and/or Reactivity Considerations;
- Heat Generation Considerations;
- Shielding Considerations; and
- Off-Gassing and/or Chemical Reactions.

After the completion and the submittal of the RUR by the principal experimenter, a review process is initiated to ensure that the experiment does not jeopardize the safe operation of the reactor or constitute a hazard to the safety of the facility staff and general public. This review process includes the following personnel: MURR Staff with technical knowledge of certain criteria, the Reactor Health Physics Manager, the Assistant Reactor Manager-Physics, the Reactor Manager, and if required, the Reactor Safety Subcommittee (RSS) and the RAC.

10.4.2 Reactor Utilization Request

The Reactor Utilization Request (RUR) Summary Sheet is prepared by the principal experimenter with the assistance of the facility staff. The RUR describes the experiment in considerable detail. It presents the activities (and isotopes) which may be produced and details the methods of handling the radioactive waste. The most important section of the RUR and the one which is given paramount consideration in its preparation is the safety analysis. This section analyzes all possible accidents and transients to determine if the experiment involves a question pursuant to 10 CFR 50.59. Each safety analysis includes, but is not limited to, the following major criteria; criticality and/or reactivity considerations (Technical Specifications 3.8.a-e); heat generation considerations; shielding considerations; and off-gassing and/or chemical reactions (Technical Specifications 3.8.f-m). It also includes all credible accident and transient scenarios to ensure that the experiment does not jeopardize the safe operation of the reactor or constitute a hazard to the safety of the facility staff and general public. The complete safety analysis consists of the following seven (7) individual analyses: (Reference 10.8)

1. Thermal Analysis - An estimation of the heat generation and heat transfer rates for an experiment, determining if a cooling system design change is required to prevent the surface temperature of a submerged irradiated sample from exceeding the saturation temperature of the liquid it is submerged in (Technical Specifications 3.8.r and 3.8.t).

2. Sample Decomposition - Pressure Analysis - Describes the form of the sample and component materials of the experiment during irradiation, with reasonable leeway for

normal and abnormal conditions. The analysis should confirm that a potential pressure buildup due to a complete decomposition of the sample material will not exceed the design pressure of the irradiation container (Technical Specification 3.8.s).

3. Experiment Failure Analysis - Used to determine if products or components from the experiment have the potential to violate the limits of 10 CFR 20, Appendix B, Table I, if released to the atmosphere. Exception: fueled experiments (Technical Specification 3.8.n).

4. Loss of Coolant Analysis - Describes how a loss of coolant (e. g., loss of pool coolant flow, loss of experiment cooling, etc.) to the experiment will not result in a release of radioactivity to the atmosphere or affect the safe operation/control of the reactor (Technical Specification 3.8.o).

5. Failure of Other Experiments Analysis - Used to identify the possible effects upon reactor control and other experiments due to operating an experiment under abnormal conditions (failure) (Technical Specification 3.8.p).

6. Corrosion Analysis - If corrosive materials are expected to be generated in an appreciable quantity during normal operation or as a result of the experiment failing, this analysis must show that the encapsulation provides enough corrosion resistance to endure the worst scenario of corrosion for the duration of the experiment (Technical Specifications 3.8.j and 3.8.l).

7. Explosive Analysis - Ensures that if explosive materials are present or are expected to be formed during the irradiation then the total mass of the explosive materials will not exceed the Technical Specification limitation (Technical Specification 3.8.i).

It is important that the experimenter thoroughly research his/her experiment in an attempt to resolve all questions which may arise in the review process.

New and substantive revisions to RURs are reviewed by the Reactor Advisory Committee and approved by the Reactor Manager in accordance with Technical Specification 6.5, Experiment Review and Approval. (Reference 10.12)

10.4.3 Review Process

The review and approval process of experiments conducted at MURR is outlined in the following paragraphs.

The experiment is reviewed by MURR staff with technical expertise, assisting the principal experimenter in preparing the RUR. MURR technical experts should raise safety questions and ensure they are analyzed during preparation of the RUR.

The RUR is reviewed by the Reactor Health Physics Manager to ensure the necessary radiological control measures are implemented during the experiment and that the experimenter possesses the experience and equipment required to deal with the expected radiation levels. The applicability and adequacy of the byproduct license(s) under which the experiment is to be conducted is also determined. If approved, the RUR is signed by the Reactor Health Physics Manager.

The RUR is reviewed by the Assistant Reactor Manager, Reactor Physics, to determine the reactivity effect, heat generation considerations, and the possibility of sample decomposition. If approved, the RUR is signed by the Assistant Reactor Manager, Reactor Physics.

The RUR is reviewed by the Reactor Manager to determine if the experiment represents a new class of experiment or a change to an existing experiment having safety significance, i.e., does the experiment involve a question pursuant to 10 CFR 50.59. If either of these conditions apply, the RUR is submitted to the RSS for review. If the Reactor Manager determines the experiment does not represent a new class of experiment or a safety significant change to an existing experiment, the RUR may be approved and signed by the Reactor Manager.

The RSS reviews experiments for the RAC. This review is primarily to determine if the experiment involves a question pursuant to 10 CFR 50.59. If the RSS determines that the experiment does not involve a potential safety hazard and recommends approval, the RUR is completed. The RSS may refer experiments to the RAC for review due to unusual hazards, special conditions involved, or potential safety hazards. If the RAC determines that the experiment does not involve a potential safety hazard, then the review process is complete.

If the RSS and/or the RAC determine that the proposed experiment does introduce a potential safety hazard pursuant to 10 CFR 50.59, the experiment proposal must be submitted to the NRC for final review.

After all other reviews are completed, the Reactor Manager will indicate on the RUR any additional limitations required beyond those already listed and will then sign the RUR as being approved. Copies of the approved RUR will be distributed to the following personnel:

- Document Control (Original);
- The RSS; and
- The RAC (if the RAC was involved in the review).

The experiment review and approval process is sufficient to protect the operations personnel, experimenters, and general public from radiation and other potential hazards caused by the experiments. Radiation doses will not exceed the limits of 10 CFR 20 and will be consistent with the facility ALARA program.

10.4.4 Reactor Advisory Committee

The Reactor Advisory Committee (RAC) is appointed by the Office of the Chancellor, University of Missouri-Columbia to satisfy the requirements imposed by federal regulation. The RAC has diverse and independent membership as well as acceptable experience and expertise. The University and the NRC expect the RAC to review and make recommendations concerning experimental and operational activities at the facility. The primary responsibility of the RAC with regard to experiment utilization is set forth in the Technical Specifications. This responsibility is to review and make recommendations concerning proposed tests or experiments significantly different from any previously reviewed or which involve a question pursuant to 10 CFR 50.59. The RAC will make a final determination in the safety of any experiment which the Reactor and Reactor Health Physics Managers feel are subject to question. In particular, these individuals possess and continually develop precedence of past experiments which they are confident will not present a hazard to the reactor. In those instances where there is any question as to the safety of an experiment, they will refer the experiment to the RAC for review.

The RAC and its responsibilities are described in greater detail in Chapter 12, “Conduct of Operations.”

10.4.5 Reactor Safety Subcommittee

With regard to experiment utilization, the Reactor Safety Subcommittee (RSS) shall act on behalf of the RAC in performing reviews of proposed tests or experiments significantly different from any previously reviewed or which involve a question pursuant to 10 CFR 50.59. Upon completion of the review, the RSS shall make a recommendation concerning the experiment and report this recommendation to the Chairman of the RAC and to the Reactor Manager. If the review results in a negative recommendation, the RSS shall recommend alternatives for the experiment and report this conclusion to the Chairman of the RAC and the Reactor Manager. In the latter instance, the Reactor Manager shall apprise the Chairman of the RSS of the course of action selected, or he shall submit a new proposal for review.

The RSS and its responsibilities are described in greater detail in Chapter 12, “Conduct of Operations.”

10.5 Experimental Facilities Technical Specifications

The objective of the Limiting Conditions for Operations (LCO) Technical Specifications on experiments is to prevent an accident which would jeopardize the safe operation of the reactor or would constitute a hazard to the safety of the facility staff and general public. Technical Specification 3.8 lists the associated LCOs for experiments.

The objective of the surveillance Technical Specifications on experiments is to prevent the conduct of experiments which may damage the reactor or release excessive amounts of radioactive material as a result of experiment failure. Technical Specification 4.8 lists the associated surveillances for experiments.

Experiments also require specific administrative controls as listed in Technical Specifications 6.5 and 6.7.b(6).

10.6 Hot Cells, Glove Boxes, and Fume Hoods

10.6.1 Control of Hot Cells, Glove Boxes, and Fume Hoods

Use of hot cells, glove boxes, and fume hoods (processing units) at MURR are controlled in several ways with regard to the radiological aspects of their use and with respect to occupational and public dose considerations. Generally, processing units are controlled by their user group at MURR in accordance with an approved reactor license project authorization (RL Project). A 10 CFR 50.59 screen or evaluation is also performed in conjunction with the RL Project. This control process ensures appropriate radiological controls and facilitates for the research or production aspects of MURR's mission to provide high quality radiopharmaceutical products to the research and medical communities.

Prior to MURR research or production groups utilizing a hot cell or glove box, evaluations are conducted by the Reactor Health Physics staff to ensure that the radioactive material used is appropriately shielded by the particular processing unit that will be used. Conversely, if a new isotope is identified for use at MURR, an evaluation occurs to determine if the existing fleet of processing units is sufficient to meet the radiation protection needs of the facility for both occupationally exposed staff and the general public. If no such facility exists, then a review process occurs (either within an existing project or during the creation of a new RL Project) to determine what design characteristics are needed to provide the appropriate level of radiation protection to staff and to the general public, prior to designing or procuring a new processing unit. Similarly, if higher activities are required for the process, existing hot cells, or glove boxes will be evaluated in relationship to the characteristics of the proposed nuclides, including the chemistry and ergonomics of the process, necessary for the safe and effective utilization of the radioisotopes.

RL Project evaluations describe how occupational exposures will be minimized based on the quantity of nuclides expected and chemical form to be utilized in the hot cell, glove box, or fume hood. The RL Project defines and lists the quantities and chemical forms appropriate for the hot cell or glove box, dependent upon the specific processing unit shielding and ventilation capabilities. Historically, isotopes irradiated at MURR are metals or metallic compounds that are not subject to volatilization or aerosolization. Unless specifically called out, any heating during the processing of these isotopes is much less than the melting temperature of the metals or metallic compounds supplied for irradiation. Metallic compounds are usually in the form of nitrates or oxides and are more prone to decomposition rather than volatilization. The heating of compounds during irradiation is considered during the RUR safety evaluation process, prior to placing these compounds in the reactor for irradiation, to ensure that adverse heating conditions do not occur. High temperatures would destroy the compound being irradiated, rendering them useless for processing and further utilization. In RL Projects, when there are no or minimal concerns with airborne contamination due to the inherently safe characteristics of the compound or element being irradiated, no language need exist within the project explicitly stating that this factor is not a concern.

Ultimately, all RL Projects are reviewed by the Isotope Use Subcommittee (IUS) of the RAC for comprehensive scrutiny and approval, after review by Reactor Health Physics staff and review and

approval by the Reactor Health Physics and Reactor Managers. The RL Project process provides a conclusion of any analysis performed during the review process limiting the quantity of radionuclides used within any hot cell, glove box, or fume hood with respect to workers' and the public's safety. (Reference 10.13)

10.6.2 General Use Hot Cells, Glove Boxes, and Fume Hoods

MURR has a variety of hot cells, glove boxes, and fume hoods, from simple stand-alone face ventilators to intricate special purpose units with filtered ventilation for isotope processing. A list of each hot cell, glove box, and fume hood by location, component designation and the typical isotope that is handled within the unit is included in Figure 10.8. Services that are available and could possibly be attached to a hot cell, glove box, or fume hood include exhaust ventilation, electrical power, filtration, domestic cold water, vacuum, and radioactive liquid drains. Each unit is set up with the support systems designated in the RL Project to ensure radiological safety. None of the services are required to mitigate the consequences of any postulated accident in a general use hot cell, glove box, or fume hood. If a hot cell, glove box, or fume hood interfaces with any of the services that are described within this Safety Analysis Report, then a Modification Record and a 10 CFR 50.59 screen or evaluation are performed. Any filtration that is installed is in keeping with the ALARA principles of the MURR Radiation Protection Program to maintain effluent discharges – water and air – as low as is reasonably achievable.

The radiological control boundary for each hot cell, glove box, and fume hood is the area posted with radiological control signs. Typically, that posting is on the room that the hot cell, glove box, or fume hood is located in, and the radiologically controlled area is the room itself. Figures 10.8 and 10.9 provide the locations of all of the hot cells, glove boxes, and fume hoods at MURR.

10.6.3 Main Hot Cell

The main hot cell (HC-01) is an intermediary station used to un-encapsulate large aluminum capsules used during irradiation of the sample and distribute smaller quantities of a given isotopes to designated research or processing facilities. The hot cell is also used to encapsulate previously irradiated samples for additional irradiation. HC-01 is integral to the construction of the facility and is located in the basement of the laboratory building. The hot cell is heavily shielded by four-foot-thick poured concrete walls, floor, and ceiling. Services to the HC-01 include filtered exhaust ventilation, vacuum, electrical power and compressed air. None of these services are required to mitigate the consequences of any postulated accident within HC-01.

TABLE 10-2
HOT CELLS, GLOVE BOXES, AND FUME HOODS

Table Withheld per 10 CFR 2.390

10.7 Iodine-131 Processing Hot Cells

10.7.1 Introduction

A dedicated laboratory in the eastward expansion of the MURR Industrial Building was constructed for the production of radiochemical sodium iodide (I-131). (Reference 10.18) The target material is irradiated at MURR and transferred to the Iodine-131 Processing Hot Cells for; target disassembly, I-131 separation; and Quality Control (QC) testing of the product.

The NRC issued Amendment #37 dated March 11, 2016, authorizing the production of Iodine-131. The Technical Specifications required for I-131 target irradiation and processing are listed in Section 10.7.4. (Reference 10.19)

The letter dated December 30, 2015, "MURR responses to the NRC Requests for Additional Information supporting the application to produce I-131" (which was withheld from public disclosure under 10 CFR 2.390) included details on the following information, summarized here:

- Description of the transfer of irradiated targets from the pool to the processing hot cells with estimates of expected exposures to personnel.
- Estimates of the doses to workers during a typical I-131 process
- Description of the factors considered in the design of the I-131 hot cell exhaust system and the determination of the charcoal filter efficiency requirements to meet TS 3.10.d.

(References 10.14-10.17, Reference 10.19)

10.7.2 Laboratory Design

The design philosophy for the hot cells is to provide adequate personnel shielding and containment of the I-131, precluding or minimizing any potential release from the individual cells to the reactor facility or to any unrestricted area. The design of the hot cell shielding and filtration is based upon a 200-Curie I-131 process; however, the technical specifications limit the process to 150 Curies. This is consistent with technical specification 3.8.h.

The Iodine-131 Processing Hot Cell contains three adjoined and inter-connected hot cells which are located on the building grade level. The area to the rear of the cells is where cell support facilities and equipment are located and where most cell inputs and outputs of targets, equipment, and supplies occur. The area in front of the hot cells is where the cell windows, manipulators, and controls are located.

10.7.2.1 Handling Hot Cell Design

The Handling Hot Cell (HHC or HC-11A) is designed to: receive irradiated targets, house support equipment to remove the target encapsulation, and store process waste for an interim period of time. This cell has a floor access port to allow entry of irradiated targets from a target transfer cask designed

to seal with the cell floor port. Within this cell the target encapsulation is removed, and the irradiated target material is placed in crucibles and transferred via a pass-through to the Processing Hot Cell (PHC).

The cell is designed to have process waste materials from the PHC returned via a pass-through chamber for interim storage. The pass-through has a door at each side and is sized to hold any object that will be moved to and from the PHC. The waste material from the PHC will be placed in containers with press-seal lids.

The volume of the cell is designed to hold the process waste for approximately 10 half-lives of the I-131 (about 80 days) after which time the cans may be moved to longer term shielded storage. The cell has 20-cm-thick vertical lead shielding with a stainless-steel liner and two master-slave-manipulators penetrating the cell from the operator side. The cell also has a sealable glove box for entry of items such as product vials and processing equipment.

10.7.2.2 Processing Hot Cell

The Processing Hot Cell (PHC or HC-11B) is where the processing of the target material is performed. This cell has 20-cm-thick vertical lead shielding within a stainless-steel liner and two master-slave-manipulators which penetrate the cell from the operator side of the cells. The sodium iodide product solution will be moved from the PHC to the Dispensing Hot Cell (DHC) in a sealed vial through a pass-through. The system is also designed where the sealed vial may be passed back into the HHC through the pass-through for transfer from the HHC to the DHC via a shielded cask.

During processing, the PHC provides three (3) barriers to prevent the release of radioactive material to the environment: (1) the processing equipment itself, (2) an additional inner containment for the separation furnace with the PHC with a dedicated filtration system, and (3) the PHC containment provided by an exhaust system that includes high efficiency charcoal filters. The process is designed to contain I-131 within the process equipment such that there should be minimal release of airborne radioactivity to the hot cell.

10.7.2.3 Dispensing Hot Cell

The Dispensing Hot Cell (DHC or HC-11C) is designed to receive the bulk I-131 solution from the PHC via a pass-through. The bulk sodium iodide radiochemical will then be dispensed in quantities to meet customer order requirements. This cell also has a sealable glove box for entry of items such as product vials, lead pigs, and dispensing equipment.

The DHC has 10-cm-thick vertical lead shielding and two master-slave-manipulators penetrating the cell from the operator side.

10.7.2.4 Ventilation System

The ventilation system that supports the Iodine-131 Processing Hot Cells is designed to perform the following radiological control functions: maintain the processing laboratory at a slightly negative

pressure with respect to the surrounding areas to prevent the spread of radioactive contamination; maintain hot cells HC-11A, HC-11B, and HC-11C at a negative pressure to prevent the spread of radioactive contamination; provide the necessary charcoal and HEPA filtration to ensure radioactive airborne being released to the environment are kept at minimum concentrations; and provide continuous monitoring of all radioactive airborne releases discharge through the Iodine-131 Processing Hot Cell ventilation exhaust system.

10.7.3 Hot Cell Ventilation and Filtration Systems

The ventilation supply and exhaust system for hot cells HC-11A, HC-11B and HC-11C consists of a combination of HEPA and charcoal filtration systems (Figure 10.10). The supply and exhaust systems for each cell combine to form a common header, but each hot cell has its own independent filtration system. Air is supplied to each hot cell from a common header through a HEPA and charcoal filter at a rate of approximately 17 cubic feet per minute (cfm) (0.48 m^3/min). Air is exhausted from each hot cell through:

- an internal parallel set of charcoal or combination Charcoal/HEPA (CHEPA) filters (Bank No. 1);
- a (selected) external parallel charcoal filtration bank (Bank No. 2: one in service – the other in standby);
- a second (selected) external parallel charcoal filtration bank (Bank No. 3: one charcoal in service – the other in standby)
- a (selected) external parallel HEPA filter bank (Bank No. 4: one filter on service – one in standby);
- a third (selected) external parallel charcoal filter bank (Bank No. 5: one in service – the other in standby);
- a fourth (selected) external parallel charcoal filter bank (Bank No. 6: one in service – the other in standby);
- a fifth (selected) external parallel charcoal filter bank (Bank No. 7: one in service—the other in standby);
- a sixth (selected) external parallel charcoal filter bank (Bank No. 8: one in service--the other in standby);
- a second external parallel HEPA filter (Bank No.9: one in service)
- into the facility main ventilation exhaust system at a flow rate of approximately 50 cfm to ensure adequate filter residence times are achieved

Air is removed from the hot cells at a rate of approximately 10 air changes per hour (ACPH).

A 4-inch (10.16 cm) line is installed to provide a path to recirculate the effluent from the final charcoal filter bank (Bank No. 6) to the supply line of the HC-11 inlet filters to minimize radioactive release from the facility in the event of an abnormal increase in iodine concentration from HC-11.

There are two glove boxes mounted to the hot cells; one on the north end of HC-11A and one on the south end of HC-11C. These gloves boxes provide the means of introducing or removing items from the

hot cells. Each glove box also has its own independent supply and exhaust system. Air is supplied to each glove box at a rate of approximately 1.7 cfm (0.048 m^3/min) through a combination HEPA/charcoal filter. The air is exhausted from each glove box through a HEPA filter and charcoal filter. The exhaust ducting for each glove box connects to the hot cell common exhaust header downstream of the hot cell filtration systems.

10.7.4 Radiation Monitoring Equipment

There are three (3) separate radiation monitoring systems that can monitor the effluents from the I-131 processing hot cells: (1) the facility Off-Gas Radiation Monitoring System, (2) the I-131 Processing Laboratory Exhaust Duct Monitor, and (3) the ALMO-6 Hot Cell Dose Rate Radiation Monitor.

The air exiting the MURR facility through the ventilation system exhaust stack is continuously monitored for airborne radioactivity by the Off-Gas Radiation Monitoring System. The Off-Gas Radiation Monitoring System is discussed in detail in Chapter 7, "Instrumentation and Control Systems."

The I-131 Processing Laboratory Exhaust Duct Monitor consists of a radiation detection system designed to measure airborne concentrations of radioactive iodine in the exhaust air that is sampled by a probe in the ventilation ducting immediately downstream of all of the I-131 hot cell and room filtration systems. The system is capable of measuring real-time exhaust flow rate as its basis for release concentrations. A pitot tube measurement device and flow transmitter provide input to the system.

The ALMO-6 Hot Cell Dose Rate Radiation Monitor consists of a six (6) detector radiation monitoring system designed to provide radiation dose level information to the process operators. Three (3) of the detectors (G-M) are located at the operator's workstation where the hot cell manipulators are located. These detectors provide real time dose rate information to the operators when they are performing a process. The remaining three (3) detectors (G-M) are located next to the first in a series of charcoal filters (Bank No. 2 and No. 3) located in each of the bays above the three (3) hot cells. These are designed to give the process operators real time information related to the capture and loading of I-131 onto the first charcoal filter in each external bank of the individual cells. This allows the process operators to monitor the condition of the charcoal filters and will alert them of the need to change to a bank of alternately available filters. The ALMO-6 radiation monitor is referred to as the "Iodine-131 Processing Hot Cells Radiation Monitor" in the Technical Specifications.

10.7.5 Evaluation of I-131 Release and Dose Consequences

10.7.5.1 Assumptions

The Processing Hot Cell (PHC) will have the highest possible content of volatile I-131. This peak content builds gradually during the process. During the I-131 processing, the evolved I-131 pass through a quartz separator and is then absorbed in a NaOH solution, producing a much less volatile chemical form of I-131. For the conservative evaluation of a potential release during processing it will be assumed that

150 Curies of I-131 are released within the PHC. This is a very conservative assumption for the following reasons:

1. The I-131 is released from the target over a period of time and is almost immediately absorbed in the NaOH solution. Any indication of leakage from the process equipment into the hot cell will cause the staff to immediately secure the processing equipment resulting in the reduction and then cessation of the I-131 release. This will result in only a fraction of the total target inventory being released; and

2. The processing equipment is designed with a containment (casing) which has dedicated filtration systems on the processing line and exhaust.

10.7.5.2 Dose Consequences in the Restricted Area

The following analysis is specific to determining exposure released from the ventilation exhaust system to a worker, assuming a non-credible, total loss of the I-131 inventory (150 Curies, maximum allowable iodine inventory TS 3.8.h) within the PHC (the same filtration mitigation features exist on all three (3) hot cells; therefore, the analysis applies to any of the three hot cells).

No credit is taken for the plating of iodine within the hot cells and ventilation ducting. Radiation exposure calculations are broken down into two scenarios: (1) exposure rates from the installed charcoal filters, and (2) exposure rates from the ventilation ductwork assume complete filter failure. While calculating exposure rates from the installed filters several adjunct calculations were also performed. The below sequence illustrates the calculation methodology of radiation exposure rates from a loaded charcoal filter.

In order to calculate the activity deposited on a filter and subsequent filters in a series, the following equation was used:

$$\text{(Eq.1)} \qquad A(x) = (A(e)/D^{x-1}) - (A(e)/(DF)^{x})$$

where:

A(x) = Activity deposited on filter “x”;

A(e) = Activity entering the first filter of the series;

DF = Decontamination Factor (must be common between all filters); and

X = The sequential number of the filter bank in a series starting with 1.

Using Equation 1, Table 10-3 presents activities that were determined to be deposited on each of the first four (4) installed charcoal filter banks using a Decontamination Factor (DF) of both 100 (99.0% efficiency) and 1000 (99.9% efficiency) for the maximum processing activity amount and assuming a

failure of one (1) of the installed filters. To model the greatest potential for worker exposures, Filter Bank No. 1 was assumed to fail. The failure of Filter Bank No. 1 represents a greater exposure to workers because this filter is housed within the PHC and shielded with the greatest amount of lead. Filter Bank No. 1 failure would create the largest magnitude of activities external to the PHC and the largest potential of exposure to workers.

TABLE 10-3
I-131 PROCESSING LABORATORY VENTILATION AND FILTRATION FOR 150 CURIES OF I-131

Activity on Filters Post Release (Ci)				
	Bank No. 1	**Bank No. 2**	**Bank No. 3**	**Bank No. 4**
Mitigated (4 of 4 filters - DF 1000)	1.4985E+02	1.4985E-01	1.4985E-04	1.4985E-07
Mitigated (3 of 4 filters - DF 1000)	0.0000E+00	1.4985E+02	1.4985E-01	1.4985E-04
Mitigated (4 of 4 filters - DF 100)	1.4850E+02	1.4850E+00	1.4850E-02	1.4850E-04
Mitigated (3 of 4 filters - DF 100)	0.0000E+00	1.4850E+02	1.4850E+00	1.4850E-02
Unmitigated	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00

Calculation of exposure rates for each of the filters was performed using the computer program MicroShield 8.02 with an Annular Cylinder – External Dose Point geometry for Filter Bank No. 1 through 3 and a Rectangular Volume geometry for Filter Bank No. 4. These geometries most accurately represent the designs of each filter according to the manufacture specification sheets. Also, each filter is designed with surrounding lead shielding. This shielding was included in the geometry model and used in the exposure rate calculations for each filter. Filter shielding thicknesses are described in Table 10-4.

MicroShield 8.02 is a product of Grove Software and is a comprehensive photon/gamma ray shielding and dose assessment program that is widely used in the industry for designing shields.

TABLE 10-4
FILTER SHIELDING THICKNESS

Filter Bank	**Lead Thickness (cm)**
No. 1	10
No. 2	10
No. 3	10
No. 4	0.635

An exposure rate calculation for the case in Tables 10.3 was performed in MicroShield and resulted in the associated exposure rates in Table 10-5. A summary of each MicroShield calculation is attached for review; however, in some cases the calculated exposure rates were scaled from a previous calculation by a ratio of the activities. To be conservative, a one-(1)-foot (30.48-cm) distance from each filter was

selected as the dose point distance. Workers are not expected to come within four (4) feet (1.22 m) of a filter during normal work activities.

TABLE 10-5
150 CURIE TARGET EXPOSURE RATES DUE TO SHINE FROM FILTERS AT 1 FOOT (IN MREM/H)

	Bank No. 1	**Bank No. 2**	**Bank No. 3**	**Bank No. 4**
Mitigated (4 of 4 filters - DF 1000)	5.15E-03	9.72E-06	9.72E-09	1.42E-05
Mitigated (3 of 4 filters - DF 1000)	0.00E+00	9.72E-03	9.72E-06	1.42E-02
Mitigated (4 of 4 filters - DF 100)	5.10E-03	9.64E-05	9.64E-07	1.41E-02
Mitigated (3 of 4 filters - DF 100)	0.00E+00	9.64E-03	9.64E-05	1.41E+00

Tables 10-5 indicates that the highest exposure rate is expected to be from Filter Bank No. 4 at 1.41 mrem/h with three (3) of four (4) filter banks in service during a 150 Curie process and filter DFs of 100. The reason for the highest exposure rates being seen on Filter Bank No. 4 is due to the substantially thinner lead shielding thickness installed around this filter as compared to Filter Bank No. 1 through 3.

A second potential scenario for the ventilation exhaust system to create exposure rates near a worker is that of an unmitigated release resulting in a high activity concentration volume of air moving through the ventilation system. In order to calculate the activity concentration of the air within the facility ventilation exhaust system, the following assumptions were made:

1. 100% of the available iodine inventory is instantaneously released into the hot cell volume free space. In reality it is expected the iodine activity would be released from a target over one (1) hour allowing for greater dilution than is considered by this assumption;

2. 50% of the entire hot cell volume is considered as free space. 50% is a large underestimation of the available free space within a hot cell; however, this results in a conservative determination of airborne activity concentrations and increases the calculated exposure rates; and

3. 100% efficiency of the air to be turned over in the hot cell in the minimum time possible. In reality it is not expected that 100% air turn over efficiency will be achieved; however, for the purposes of calculating exposure rates this assumption will result in a conservative determination of air activity concentrations within the facility ventilation exhaust ducting.

Also, the following parameters were used to calculate the activity concentration of the air within the facility ventilation exhaust system:

1. Hot cell dimensions of 152.5 cm × 152.5 cm × 122 cm (H × W × D); and

2. Hot cell flow rates of: HC-11A = 15 cfm (0.42 m^3/min); HC-11B = 11 cfm (0.31 m^3/min); and HC-11C = 10 cfm (0.28 m^3/min).

In order to calculate the activity concentration of the air entering the facility ventilation exhaust ducting, a determination must first be made of the hot cell air volume turnover time. Equation 2 was used to determine hot cell air volume turn over time.

(Eqn. 2) $T(x) = V(x)/F(x)$

where:

$T(x)$ = Air volume turn over time for hot cell "x" in minutes;

$V(x)$ = Volume of hot cell "x" in cubic feet; and

$F(x)$ = Flow of air exiting hot cell "x" in cfm.

Using Equation 2, the following turnover times listed in Table 10-6 were calculated for the I-131 processing hot cells.

TABLE 10-6
HOT CELL TURNOVER TIME

Hot Cell	Turnover Time (minutes)
HC-11A	3.34
HC-11B	4.55
HC-11C	5.01

Next a calculation was performed to determine the concentration of activity within the hot cell. The below equation was used to calculate the activity concentrations listed in Table 10-7.

(Eqn. 3) $C(x) = A(x)/V(x)$

Where:

$C(x)$ = Concentration of airborne activity in hot cell "x;"

$A(x)$ = Activity instantaneously entering hot cell "x;" and

$V(x)$ = Volume of hot cell "x."

TABLE 10-7
HOT CELL AIR CONCENTRATIONS

Process Activity	Air Concentration (µCi/ml)
150 Ci	105.74
88 Ci	62.03
55 Ci	38.77

From the hot cell, a conservative assumption is made that all of the activity instantaneously introduced into the PHC escapes the hot cell and enters the ventilation exhaust system within approximately five (5) minutes. As stated previously, this assumption will conservatively calculate activity concentrations of the air and result in conservative exposure rate values.

In order to calculate expected activity concentrations throughout the facility ventilation exhaust system a ratio of ventilation flow rates was determined and used to ratio the expected activity concentrations. Table 10-8 lists the flow rates used for the activity determinations and Equation 4 describes the method used to ratio the activity concentrations.

TABLE 10-8
VENTILATION EXHAUST FLOW RATE

Location	Flow Rate (cfm) (m^3/min)
HC-11B	11 (0.31)
6-inch (15.24-cm) Ductwork	47 (1.33)
16-inch (40.64-cm) Ductwork	1000 (28.32)

(Eqn. 4) $A(x) = C[F_o/F(x)]$

where:

$A(x)$ = Activity concentration at location "x";

C = Activity concentration in the hot cell;

F_o = Flow rate from the hot cell; and

$F(x)$ = Flow rate at location "x".

Using Equation 4, the following activity concentrations listed in Table 10-9 were determined at various locations of the ventilation exhaust system. Also, in the table is the calculated exposure rates resulting from the associated activity concentrations. The exposure rate calculations were performed in MicroShield 8.02 using a 100-cm line source geometry with a dose point at a distance of one (1) foot (30.48 cm) from the source.

TABLE 10-9
150 CURIE TARGET EXPOSURE RATES DUE TO SHINE FROM VENTILATION AT 1 FOOT

	Hot Cell Ductwork	**6-inch (15.24-cm) Ductwork**	**16-inch (40.64-cm) Ductwork**
Activity Concentration (µCi/ml)	1.0574E+02	2.4762E+01	1.1632E+00
Unmitigated Exposure Rate (mrem/h)	1.6330E+01	3.8230E+00	1.7960E-01

Assuming the 150 Curie process activity is instantaneously released into the hot cell, the highest exposure rate from an unmitigated release is from the hot cell ductwork at 16.33 mrem/h.

With respect to the expected exposure rates from an unmitigated and mitigated release, the external dose hazards to personnel are minimal and within regulatory compliance for process activities up to 150 Curies. To further enhance ALARA, two (2) additional filter banks (Banks #5 and #6) are installed in the exhaust ventilation flow downstream of filter bank #4. No credit has been taken for the use of these additional filter banks in any safety analysis.

10.7.5.3 Dose Consequences in the Unrestricted Area

Two analyses were performed for potential offsite dose consequences of a failed target and subsequent release of I-131 to the surrounding environment that was produced in the target during irradiation. The first analysis was performed using the COMPLY computer code – a screening tool for evaluating radiation exposure from atmospheric releases of radionuclides and for calculating doses to the public for offsite air effluent radionuclide releases. Updated wind rose data used for this analysis was obtained from the Columbia Regional Airport (CRA) for the time period of 1984 to 1992. The doses calculated are based on the dose to the nearest permanent resident located approximately 760 meters north of MURR in a residential neighborhood and use the frequency distribution of wind directions around MURR. CRA is located approximately 13 miles south-east of MURR and the wind data used was the most readily available and recent wind rose data found published in the available literature.

As previously stated, a 150 Curie activity of I-131 was used in the analysis, which represents the maximum allowable iodine inventory for each non-fueled experiment that is intended to produce I-131 (TS 3.8.h).

The second analysis utilized the Pasquill-Guifford (P-G) dispersion model methodology for atmospheric dispersion of contaminants. This model has been used for years in the nuclear industry and is the basis for subsequent atmospheric modeling methods developed throughout the late 20th century. The “D” stability class and a southern wind direction were chosen for this analysis as they provide the most conservative (highest) concentrations of effluents at the point of interest, with regard to offsite dose calculations and, is the most prominent wind direction in central Missouri. Additionally, the wind direction is assumed to come from the southerly direction the entire two (2) hours of the release event. The above stated three (3) activity amounts were also used in this analysis.

Table 10-10 presents the dose consequences of the I-131 unmitigated (no filtration or plating) release scenario to the environment and expected offsite dose consequences, presented in a matrix of four separate filter status scenarios, for a mitigated release. These calculations are repeated three times to reflect the COMPLY code output and two separate calculations for whole body (WB) and thyroid dose utilizing the P-G methodology.

The initial offsite doses were calculated assuming a total release of the available iodine inventory with no filtration or capture used (unmitigated) and were generated using the COMPLY code. The second and third scenarios were calculated using the P-G model and assumed the dose would be received by a person situated at the Emergency Planning Zone (EPZ) boundary of 150 meters due north of MURR (also highest dose location). The first set of dose calculations are for WB doses due to exposure of I-131 at the EPZ boundary. The second set of dose calculations are doses to the thyroid at the same point. Both sets of calculations show the effects of 1 through 4 banks of charcoal filters, assuming a 99% collection efficiency (DF 100) for I-131, used to mitigate offsite dose. Again, the scenario assumes that no I-131 would plate out in the hot cell or ductwork prior to being released from the facility. The dose calculations assume that a person is located at the EPZ for the entire two (2) hour post-event period prior to being evacuated from that area. This assumes that the air concentration is constant at the calculated values during the release and that no credit is taken for the prevailing wind to disperse the I-131 laden air.

TABLE 10-10
OFFSITE DOSE CONSEQUENCES OF 150 CI I-131 RELEASE (IN MREM)

Unmitigated (no Filtration)			
COMPLY[1] (Whole Body (WB)) – Nearest Residence	Pasquill-Guifford Model[2,3] (WB) – Emergency Planning Zone (EPZ) – 150 meters		Pasquill-Guifford Model[2,3] (Thyroid) – EPZ
124	13.1		7616
MITIGATED (FILTERED) COMPLY[1] Model – WB - Nearest Residence			
1 Filter Bank	2 Filter Banks	3 Filter Banks	4 Filter Banks
1.24	0.01	1.24E-04	1.24E-06
MITIGATED (FILTERED) P-G Model[2,3] – WB – EPZ			
1 Filter Bank	2 Filter Banks	3 Filter Banks	4 Filter Banks
0.13	0.01	1.31E-05	1.31E-07
MITIGATED (FILTERED) P-G Model[2,3] – Thyroid – EPZ			
1 Filter Bank	2 Filter Banks	3 Filter Banks	4 Filter Banks
76.16	0.76	7.62E-03	7.62E-05

1. Dose from COMPLY is to the nearest residence, 760 meters north.

2. Dose conversion factors for the two P-G Models are from NRC Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I"
3. Doses for the two P-G models are assumed to be received at the EPZ, 150 meters north over a two (2) hour period. No correction for dispersion of the cloud is made at the site of the recipient, i.e. the cloud is assumed to be uniform over the entire two (2) hour scenario post release.

For comparison purposes, the dose consequence matrix calculates offsite doses for scenarios where one, two, three and four banks of activated charcoal filters are available to capture released I-131 gases. The filters are assumed to be 99% efficient in capturing elemental iodine, which is a factor of 10x lower than stated values provided by the filter manufacturers. Thus, the offsite doses can be easily compared using a variety of scenarios ranging from having no filtration up to the maximum amount designed into the system of four (4) separated banks of activated charcoal filters.

In conclusion, only one (1) in four (4) filter banks is required to be functional in order to keep offsite dose below the 10 CFR 20 limit for occupationally exposed workers that require monitoring. Two (2) sets of functioning filters keep doses below 10 mrem to the thyroid and below any dose limits applicable to the general public. (Reference 10.15)

10.7.6 Iodine-131 Processing Hot Cells Technical Specifications

The objective of the Limiting Conditions for Operations (LCO) Technical Specifications for the I-131 production experiment is to provide for the safe processing of I-131 and to limit routine exposures to reactor staff or radioactive releases which would jeopardize the safety of the facility staff and general public. The following are Technical Specification LCOs for the I-131 experiment and subsequent processing:

- TS 3.8.h limits the non-fueled experiment total inventory of I-131 to no greater than 150 Ci;
- TS 3.8.v specifies the required processing hot cells and ventilation system components to mitigate accident consequences; and
- TS 3.10, I-131 Processing Hot Cells, specifies the equipment requirements to safely process I-131.

The objective of the surveillance technical specifications on the Iodine-131 Processing Hot Cells is to reasonably assure proper operation of the equipment needed to safely process iodine-131. Technical specification 4.10 lists the associated surveillance technical specification for the Iodine-131 Processing Hot Cells. (Refs. 10.16 and 10.17)

All experiments also require specific administrative controls as listed in TS 6.5 and 6.7.b(6).

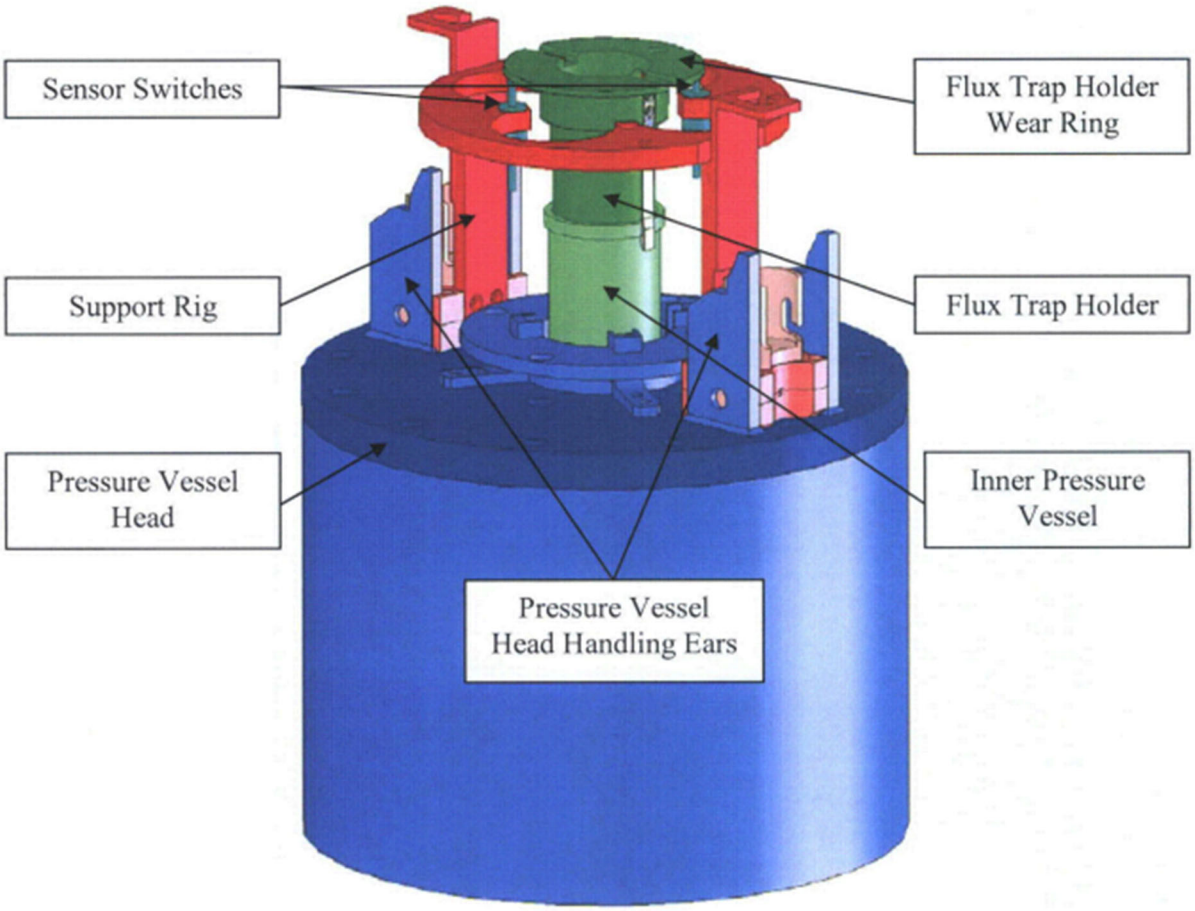
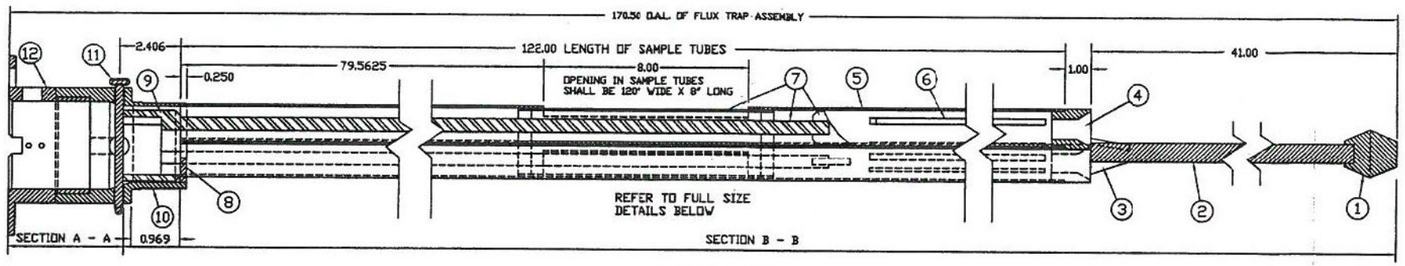
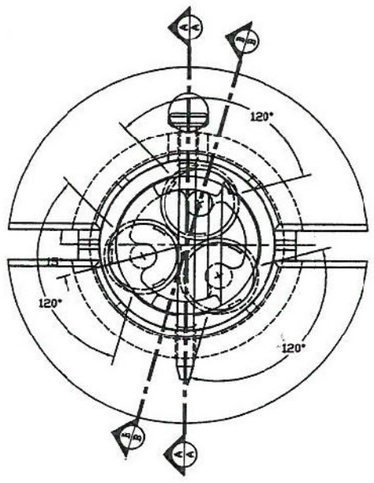


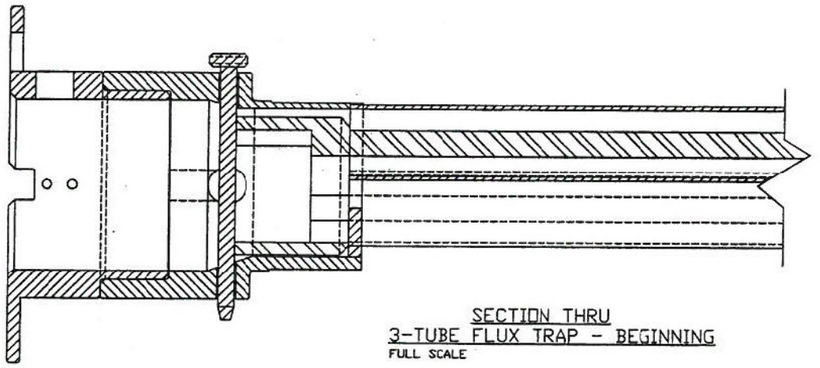
FIGURE 10.1 – FIRST DEVICE



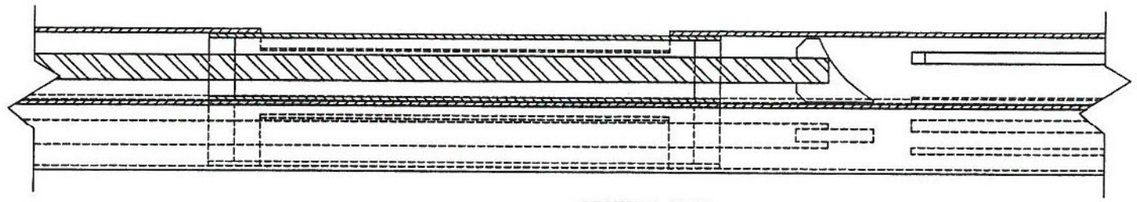
SECTION THRU
3-TUBE FLUX TRAP
1/2 SCALE



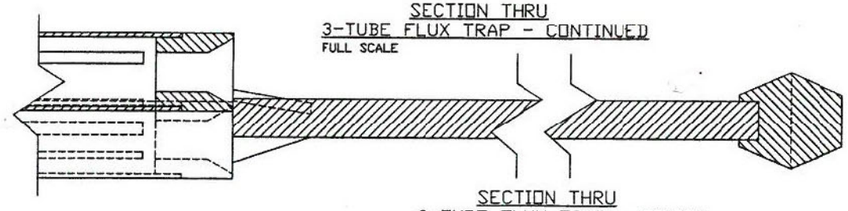
PLAN VIEW
3-TUBE FLUX TRAP
FULL SCALE



SECTION THRU
3-TUBE FLUX TRAP - BEGINNING
FULL SCALE

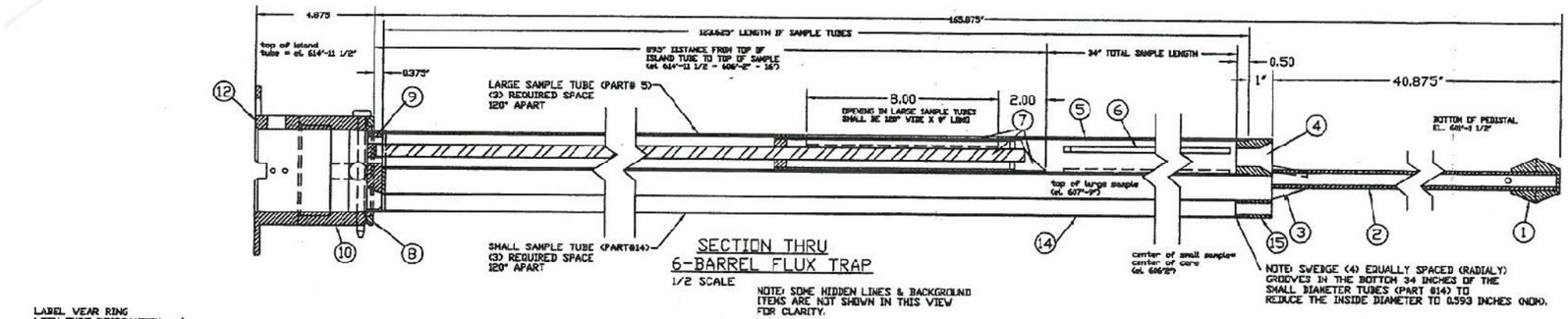


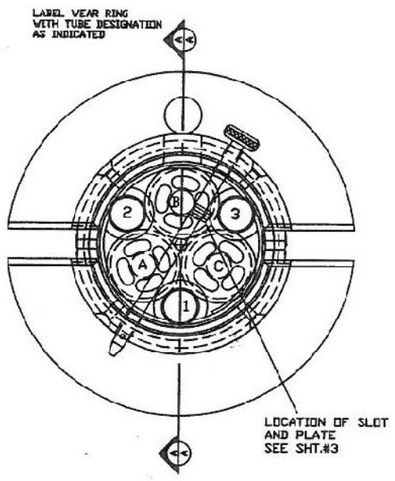
SECTION THRU
3-TUBE FLUX TRAP - CONTINUED
FULL SCALE



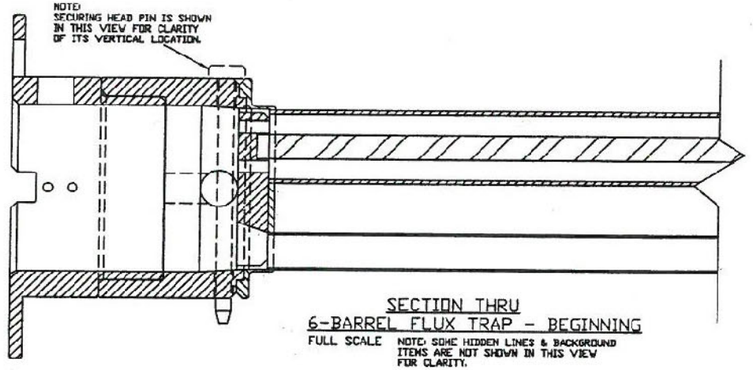
SECTION THRU
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FULL SCALE

FIGURE 10.2 – THREE-TUBE CENTER TEST HOLE CANISTER

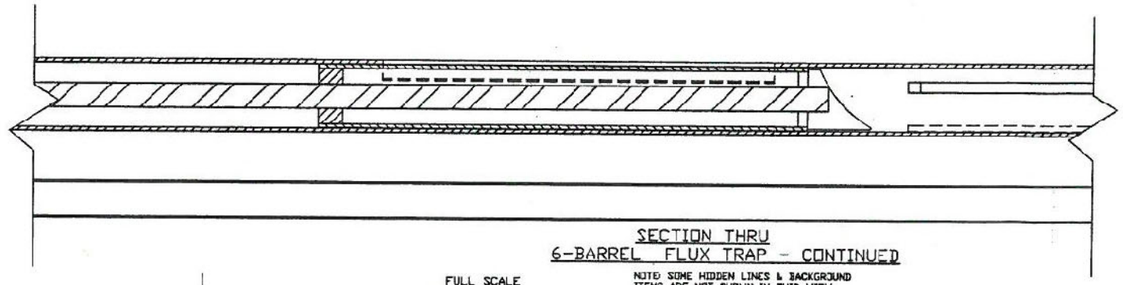




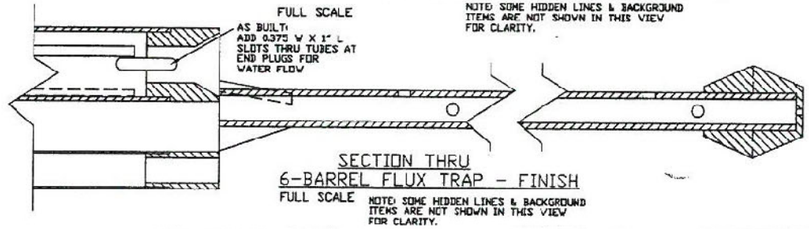
PLAN VIEW
6-BARREL FLUX TRAP
FULL SCALE



SECTION THRU
6-BARREL FLUX TRAP - BEGINNING
FULL SCALE NOTE: SOME HIDDEN LINES & BACKGROUND
ITEMS ARE NOT SHOWN IN THIS VIEW
FOR CLARITY.



SECTION THRU
6-BARREL FLUX TRAP - CONTINUED
NOTE: SOME HIDDEN LINES & BACKGROUND
ITEMS ARE NOT SHOWN IN THIS VIEW
FOR CLARITY.



SECTION THRU
6-BARREL FLUX TRAP - FINISH
FULL SCALE NOTE: SOME HIDDEN LINES & BACKGROUND
ITEMS ARE NOT SHOWN IN THIS VIEW
FOR CLARITY.

FIGURE 10.3 – SIX-TUBE CENTER TEST HOLE CANISTER

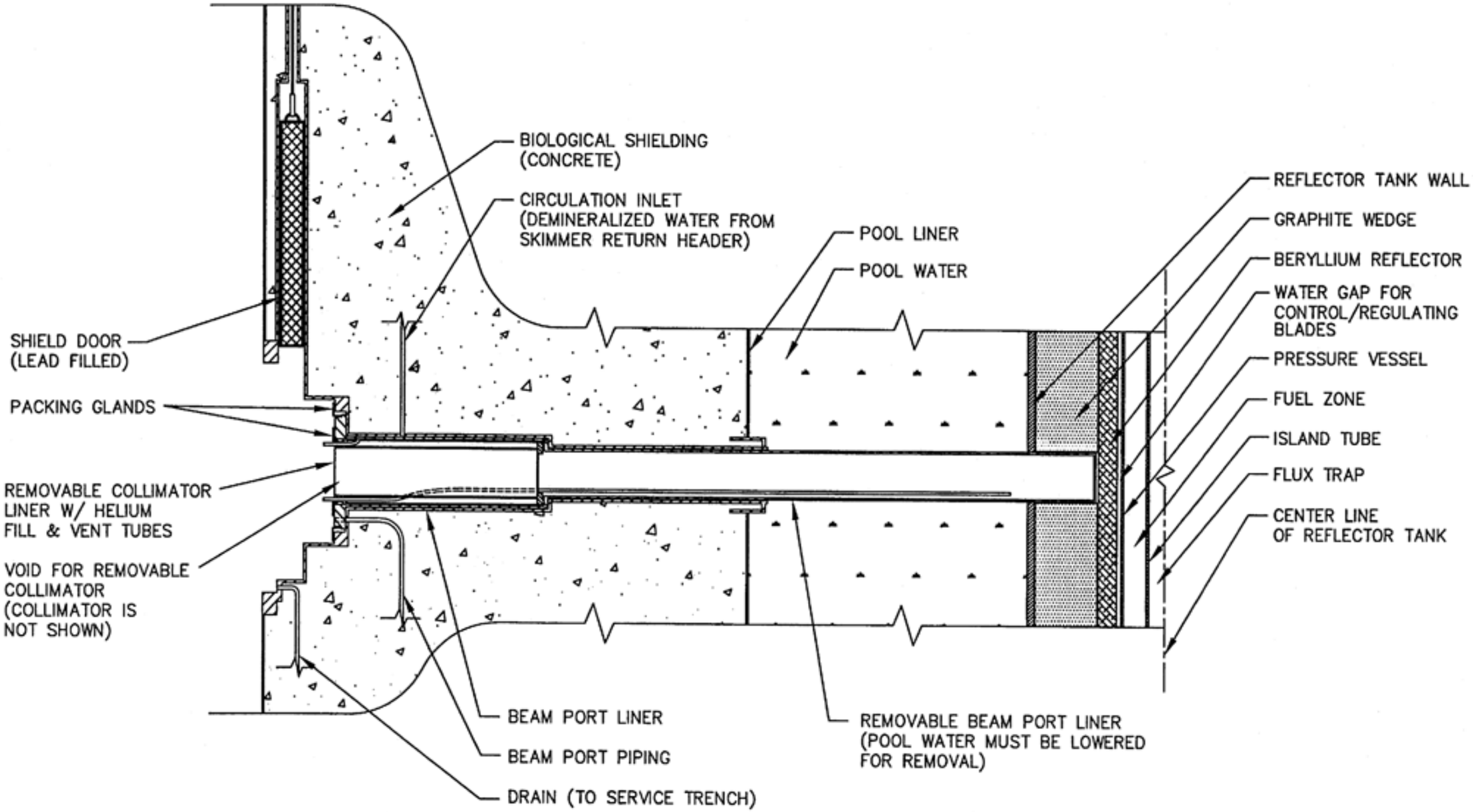


FIGURE 10.4 – CROSS-SECTIONAL VIEW OF A BEAMPORT ASSEMBLY

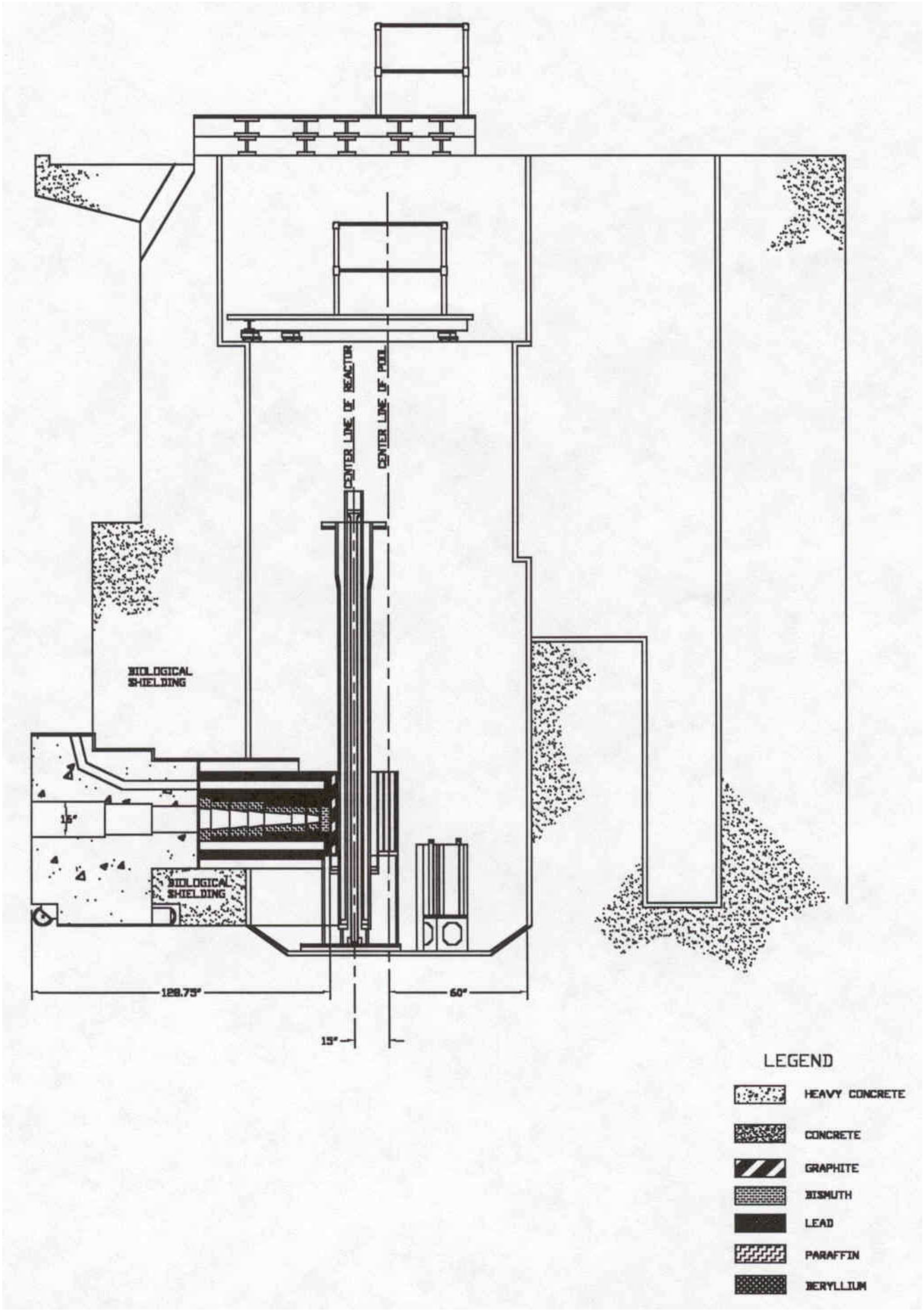


FIGURE 10.5 – THERMAL COLUMN ASSEMBLY

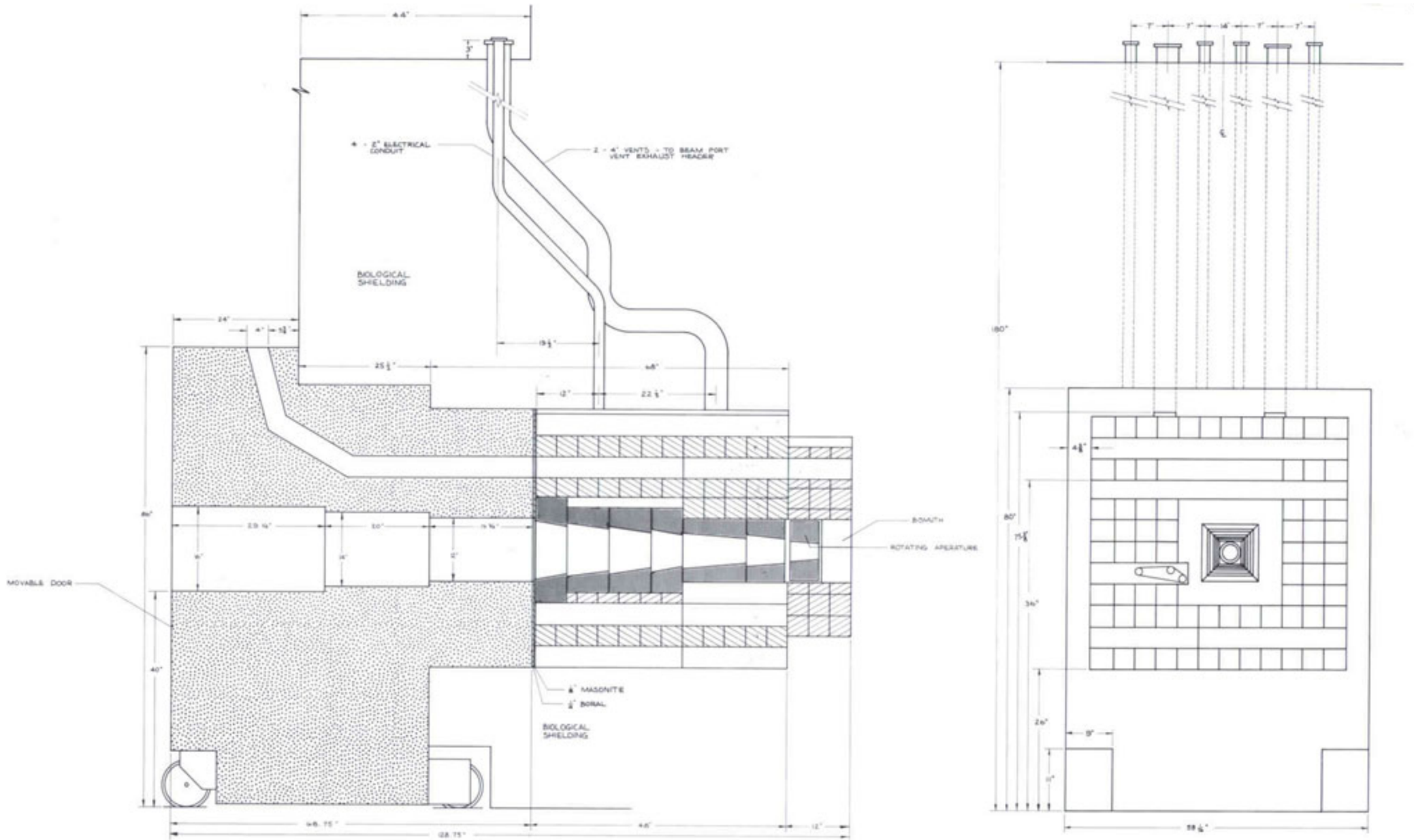


FIGURE 10.6 – THERMAL COLUMN DOOR

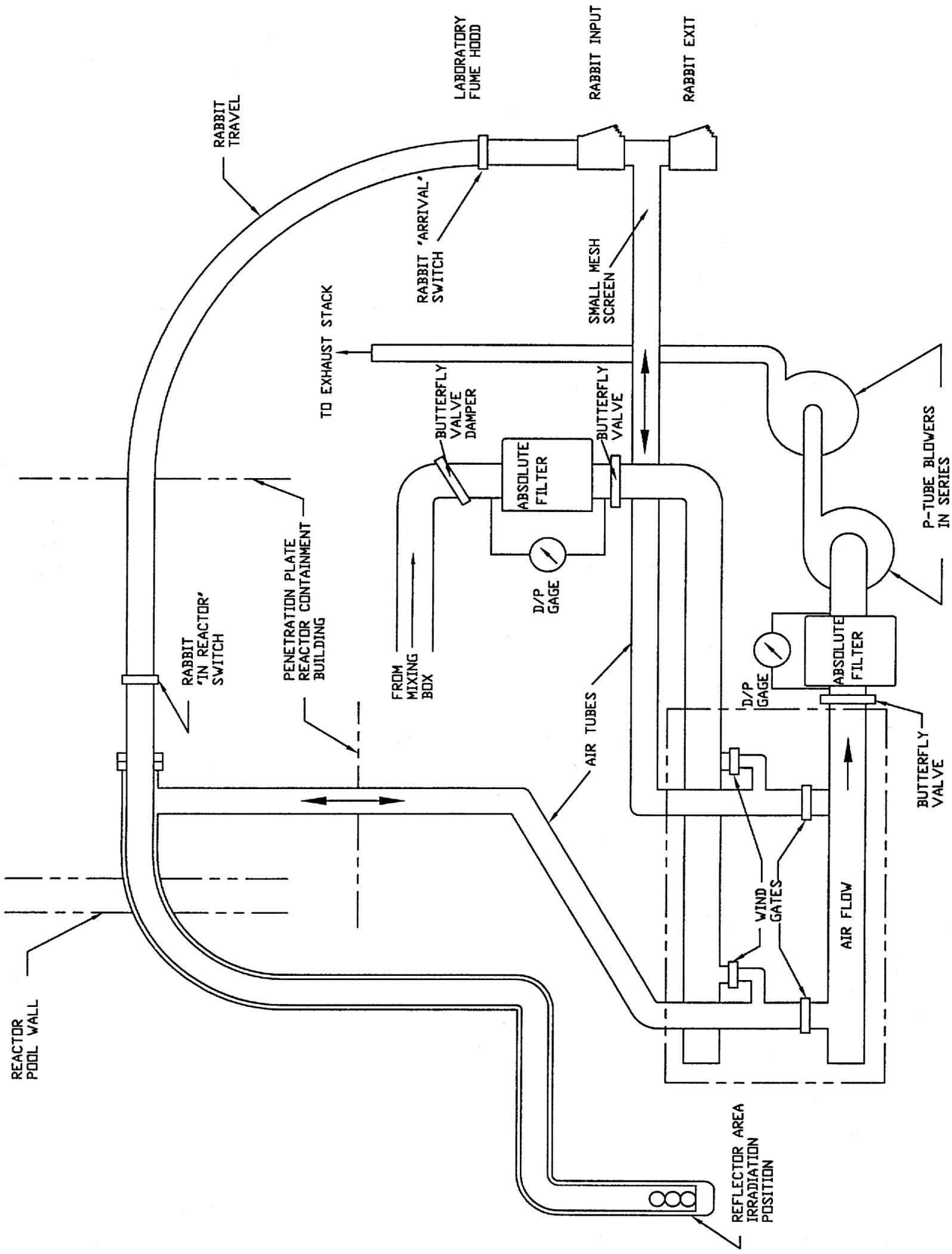


FIGURE 10.7 – PNEUMATIC TUBE SYSTEM

Figure Withheld per 10 CFR 2.390

FIGURE 10.8 – GRADE LEVEL HOT CELLS, GLOVE BOXES, AND FUME HOODS

Figure Withheld per 10 CFR 2.390

FIGURE 10.9 – BELOW GRADE LEVEL HOT CELLS, GLOVE BOXES, AND FUME HOODS

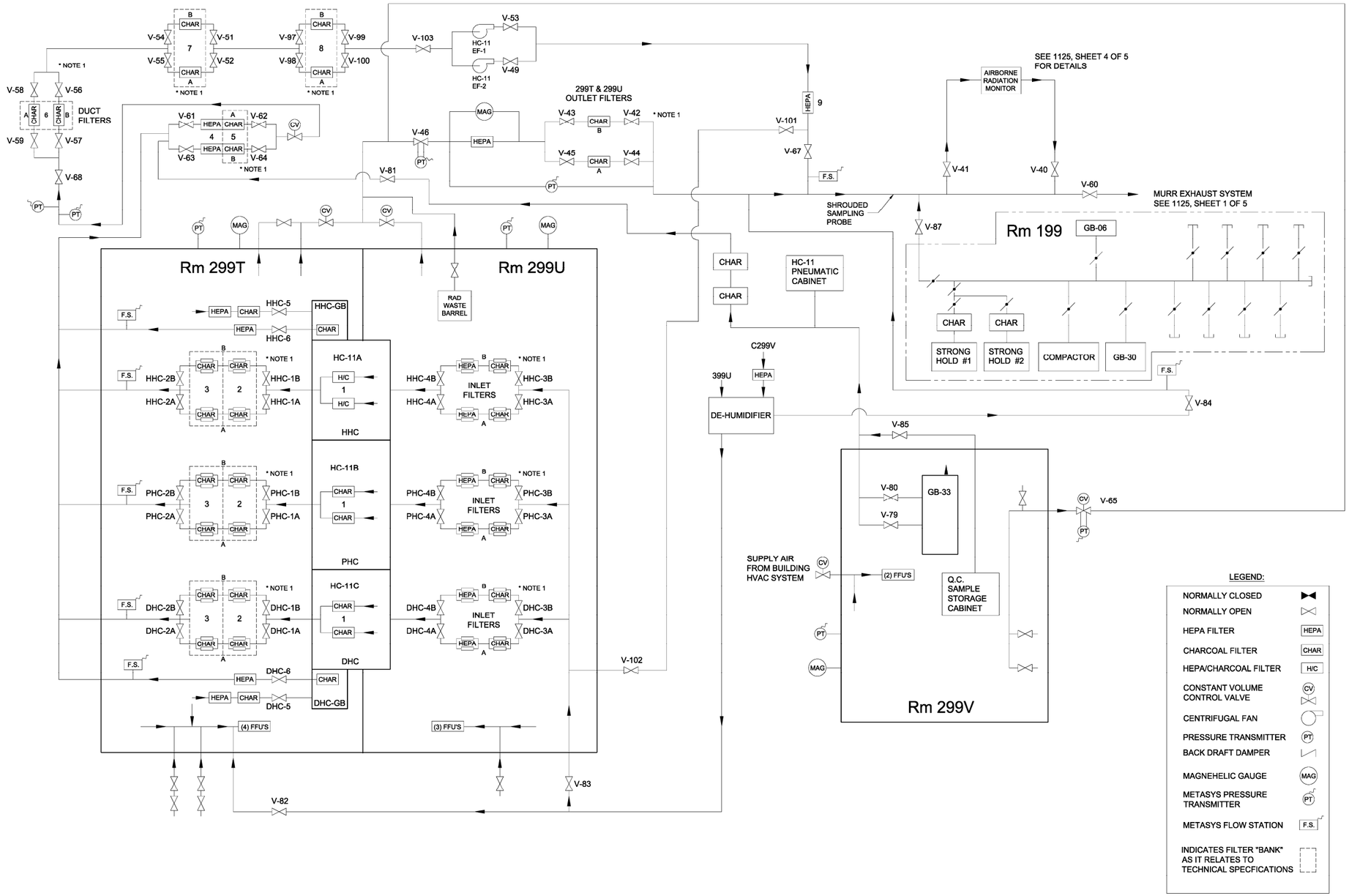


FIGURE 10.10 – I-131 PROCESSING LABORATORY VENTILATION AND FILTRATION

CHAPTER 11

RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

TABLE OF CONTENTS

11.0 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT 4
11.1 Radiation Protection 4
11.1.1 Radiation Sources 5
11.1.1.1 Airborne Radiation Sources 6
11.1.1.1.1 Argon-41 from the Pneumatic Tube System 6
11.1.1.1.2 Argon-41 in the Reactor Containment Building 7
11.1.1.1.3 Argon-41 Release to the Unrestricted Area 7
11.1.1.2 Liquid Radioactive Sources 9
11.1.1.2.1 Radioactivity in the Primary Coolant 10
11.1.1.2.2 Radioactivity in the Pool Coolant 11
11.1.1.2.3 Nitrogen-16 and Oxygen-19 from the Primary and Pool Coolant Systems 11
11.1.1.3 Solid Radioactive Sources 12
11.1.2 Radiation Protection Program (Reference 11.2 and Reference 11.3) 13
11.1.2.1 Reactor Health Physics Staff 14
11.1.2.2 Isotope Use Subcommittee 15
11.1.2.3 Radiation Safety Committee 15
11.1.2.4 Radiation Safety Officer 15
11.1.2.5 Radiation Protection Training 16
11.1.2.6 Reactor Health Physics Procedures 18
11.1.2.7 Reactor Health Physics Audits 19
11.1.2.8 Reactor Health Physics Records 19
11.1.3 ALARA Program 20
11.1.4 Radiation Monitoring and Surveying 23
11.1.4.1 Radiation and Contamination Surveys 23
11.1.4.2 Radiation Monitoring Equipment (Reference 11.1) 25
11.1.4.3 Instrument Calibration 25
11.1.5 Radiation Exposure Control and Dosimetry 28
11.1.5.1 Radiation Shielding 28
11.1.5.1.1 Biological Shield 29
11.1.5.1.2 Spent Fuel Transfer and Storage 30
11.1.5.1.3 Experimental Facilities 32
11.1.5.1.4 Primary and Pool Coolant Systems 35
11.1.5.2 Ventilation System 37
11.1.5.3 Containment 38
11.1.5.4 Entry Control 39
11.1.5.5 Protective Equipment 40
11.1.5.5.1 Respiratory Protection Equipment 40
11.1.5.5.2 Personnel Dosimetry Devices 40
11.1.5.6 Expected Annual Radiation Exposure 42
11.1.5.6.1 Restricted Area 42

11.1.5.6.2 Unrestricted Area 45
11.1.6 Contamination Control 45
11.1.7 Environmental Monitoring 47
11.1.7.1 Environmental Monitoring Technical Specifications 48
11.2 Radioactive Waste Management 50
11.2.1 Radioactive Waste Management Program 50
11.2.2 Radioactive Waste Controls 51
11.2.2.1 Solid Waste 51
11.2.2.2 Liquid Waste 52
11.2.2.3 Gaseous Waste 53
11.2.3 Release of Radioactive Waste 53

LIST OF TABLES

Table 11-1 Ar-41 PRODUCTION AT MURR AT 10 MW 7
Table 11-2 MAXIMUM Ar-41 CONCENTRATIONS AT 150 METERS NORTH FROM THE MURR EXHAUST STACK (REFERENCE 11.5) 8
Table 11-3 MAXIMUM Ar-41 CONCENTRATIONS AT 760 METERS NORTH FROM THE MURR EXHAUST STACK (REFERENCE 11.5) 8
Table 11-4 NORMAL OPERATIONAL Ar-41 CONCENTRATIONS AT 150 METERS NORTH FROM THE MURR EXHAUST STACK 9
Table 11-5 NORMAL OPERATIONAL Ar-41 CONCENTRATIONS AT 760 METERS NORTH FROM THE MURR EXHAUST STACK 9
Table 11-6 TYPICAL RADIONUCLIDES RELEASED INTO CITY OF COLUMBIA SANITARY SEWERAGE 10
Table 11-7 PREDOMINANT RADIONUCLIDES IN MURR PRIMARY COOLANT AND THEIR MEASURED CONCENTRATIONS AT 10 MW 10
Table 11-8 PREDOMINANT RADIONUCLIDES IN MURR POOL COOLANT AND THEIR MEASURED CONCENTRATIONS AT 10 MW 11
Table 11-9 REPRESENTATIVE RADIOACTIVE SOURCES AT MURR 13
Table 11-10 MURR WASTE SHIPMENTS 13
Table 11-11 RADIATION MONITORING AND RELATED EQUIPMENT 27
Table 11-12 THERMAL COLUMN SHIELDING 34
Table 11-13 PROTECTIVE EQUIPMENT FOR RADIATION PROTECTION PROGRAM 40
Table 11-14 PERSONNEL DOSIMETRY DEVICES 41
Table 11-15 DOSE MEASUREMENTS BY FILM BADGE AT SELECTED LOCATIONS WITHIN THE REACTOR CONTAINMENT BUILDING (IN MREM) 43
Table 11-16 REM BY GROUP AT MURR 44
Table 11-17 AVERAGE MONTHLY RADIATION DOSE BY GROUP AT MURR (IN MREM/Y) 44
Table 11-18 ENVIRONMENTAL MONITORING PROGRAM – SAMPLING LOCATIONS 48
Table 11-19 ENVIRONMENTAL DOSIMETERS 49

LIST OF FIGURES

Figure 11.1 – BIOLOGICAL SHIELD MAGNETITE CONCRETE VS. EFFECTIVE POOL RADIUM 54
Figure 11.2 – DIRECT PENETRATION GAMMA-RAY DOSE RATES AS A FUNCTION OF WATER DEPTH ABOVE THE ACTIVE CORE REGION 55
Figure 11.3 – MAGNETITE CONCRETE BIOLOGICAL SHIELDING REQUIREMENTS VS. EFFECTIVE POOL RADIUS TO ACHIEVE ONE-TENTH OF DOSE RATE CRITERIA FROM FAST NEUTRONS 56
Figure 11.4 – SPENT FUEL ELEMENT DOSE RATES THROUGH MAGNETITE SHIELDING FOR A SINGLE ELEMENT AS A FUNCTION OF FISSION PRODUCT DECAY TIME 57
Figure 11.5 – SPENT FUEL ELEMENT DOSE RATES THROUGH MAGNETITE SHIELDING FOR FOUR ELEMENTS AS A FUNCTION OF FISSION PRODUCT DECAY TIME 58
Figure 11.6 – SPENT FUEL ELEMENT DOSE RATES THROUGH MAGNETITE SHIELDING FOR EIGHT ELEMENTS AS A FUNCTION OF FISSION PRODUCT DECAY TIME 59
Figure 11.7 – SPENT FUEL ELEMENT DOSE RATES THROUGH WATER SHIELDING AS A FUNCTION OF FISSION PRODUCT DECAY TIME - HORIZONTAL POSITION 60
Figure 11.8 – SPENT FUEL ELEMENT DOSE RATES THROUGH WATER SHIELDING AS A FUNCTION OF FISSION PRODUCT DECAY TIME - VERTICAL POSITION 61
Figure 11.9 – ENVIRONMENTAL SAMPLING STATIONS 62
Figure 11.10 – SUBSURFACE SOIL SAMPLE LOCATIONS 63

REFERENCES

11.1 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," 2007.

11.2 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.11, "Radiation Protection at Research Reactor Facilities," 2016.

11.3 Title 10, "Energy," Code of Federal Regulations, Part 20, "Standards for Protection Against Radiation," Washington, D.C.

11.4 Written communication to the NRC, dated July 16, 2010, regarding responses to NRC requests for additional information regarding renewal of the Reactor Facility Operating License R-103, [ML 12354A237].

11.5 Written communication to the NRC, dated October 1, 2015, regarding responses to NRC requests for additional information regarding renewal of the Reactor Operating License R-103, [ML15275A314].

11.6 Regulatory Guide 8.10, Operating Philosophy for Maintaining Occupational and Public Radiation Exposures As Low As is Reasonably Achievable, August 2016, Rev. 2.

11.7 Amendment No. 40 to Renewed Facility Operating License R-103, dated June 8, 2021.

11.0 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

This chapter discusses and analyzes the radiological consequences related to normal operation of the reactor. Included are the principal discussions of the facility program to control radiation and expected radiation exposures due to operation, maintenance, and use of the reactor. This chapter outlines the methods for quantitative assessment of radiation doses in the restricted and unrestricted areas; application of these methods to all applicable radiation sources related to the full range of operation; and the program and provisions for protecting the health and safety of all individuals present at MURR, the general public and the environment.

11.1 Radiation Protection

The University of Missouri Research Reactor (MURR) Radiation Protection Program (RPP) (POL-03) has been established to protect the health and safety of all individuals present at MURR, the general public, and the environment. In accordance with Title 10, Chapter I of the Code of Federal Regulations, Part 20.1101 (10 CFR 20.1101), this program has been developed, documented, and implemented to a level commensurate with the scope and extent of licensed activities at MURR, and is sufficient to ensure compliance with the regulations in 10 CFR 20. (Reference 11.3) A primary component of this program is the fundamental principle of maintaining individual radiation exposures and releases of radioactive effluents as low as is reasonably achievable (ALARA). Responsibility for maintaining the MURR ALARA Program extends to all individuals who are granted access to the reactor facility. (Reference 11.6)

All personnel using radioactive materials or radiation sources shall become familiar with the requirements of the MURR RPP and shall conduct their operations in accordance with said program. However, the Reactor Health Physics (HP) Staff have the authority to interdict or terminate the use of radioactive materials or radiation sources if adequate HP support is not available or if significant deviations from established procedures have occurred or are likely to occur.

In addition to the facility operating license, there is currently one other U.S. Nuclear Regulatory Commission (NRC) license granted to The Curators of the University of Missouri governing work at MURR: The Broad Scope Materials License (No. 24-00513-39). The facility operating license is the primary license which covers the authority and responsibilities associated with the reactor and the majority of radioactive materials existing at MURR. Coverage provided by the other license is supplementary and is used to support the research and development mission of MURR which may not currently be covered under the facility operating license.

Additionally, MURR houses a 16 MeV cyclotron for the production of medical diagnostic imaging agents that supply the central Missouri region. The cyclotron is administered under an NRC Materials Licensee not managed by MURR and therefore, the generation of isotopes from the cyclotron do not fall under the auspices of the MURR Radiation Protection Program.

Radioactive materials within the reactor facility, whether licensed under the Broad Scope Materials License or the facility operating license, are subject to the same radiation protection controls. However,

the organizational structure that provides the review and approval process for the use of radiation sources and radioactive materials under the Broad Scope Materials License differs from that of the facility operating license. The Reactor Manager and Reactor HP Manager review and approve the uses of radioactive materials produced by the reactor. A Subcommittee (Isotope Use Subcommittee) of the Reactor Advisory Committee (RAC) has advisory responsibility for the actions of the Reactor Manager and Reactor HP Manager with regard to the use of radioactive materials and radiation sources under the facility operating license. The Radiation Safety Committee (RSC) reviews and approves the uses of radioactive materials and radiation sources that are covered by the Broad Scope Materials License. The records of the review and approval process are maintained by the Radiation Safety Officer (RSO).

11.1.1 Radiation Sources

The radiation sources that are monitored and controlled by the MURR Radiation Protection and Radioactive Waste Management Programs can be categorized as airborne, liquid, or solid. While each of these categories is discussed individually in Sections 11.1.1.1 through 11.1.1.3, the major contributors to each category can be summarized as follows:

Airborne sources consist mainly of argon-41 (Ar-41, half-life 1.8 h), which accounted for about 92.6% of the radioactivity released through the facility ventilation exhaust stack between 2020 through 2024 and 71% of Technical Specification limits during the same period. Ar-41 is produced when the argon-40 in air (~1.0%) is activated by thermal neutrons. The principal production areas within the reactor facility include the pneumatic tube (p-tube) system, the thermal column, and the beamports. Other than Ar-41, no other significant source of airborne radioactivity is produced at the reactor facility as part of its normal operation;

Liquid sources include primary and pool coolant and radioactive liquid waste generated in the laboratories. Liquid waste generated in the laboratories is the most significant source in terms of volume. Since primary and pool coolant is, by design, contained to the maximum extent possible, there are no routine releases of these liquids. However, certain reactor maintenance activities result in small volumes of liquid (containing mainly tritium) being directed to the liquid waste retention system. Limited and strictly controlled quantities of liquid radioactive waste are released to the sanitary sewer in accordance with the requirements of 10 CFR 20.2003; and

Solid sources are more diverse, but are in general, typical of a research reactor facility. Such sources include the reactor fuel in use in the core, irradiated fuel stored in the reactor pool, and new, unirradiated fuel. In addition, other solid sources are present such as the neutron startup source, nuclear instrumentation fission chambers, irradiated material as part of the experiment program, other items irradiated as part of normal reactor use, solid waste, and small instrument check and calibration sources.

11.1.1.1 Airborne Radiation Sources

During normal operation of MURR, Ar-41 is the principal source of airborne radioactivity released through the facility ventilation exhaust stack. The assumptions and calculations used to assess the radiological impact of Ar-41 during normal operation are described in detail in Appendix B, "Radiological Impact of Ar-41 During Normal Operations." This section presents a summary of that information.

Fuel element failure, although not expected, could occur during normal operation of the reactor. Such a failure would usually be caused by a manufacturing defect or pitting corrosion of the fuel cladding. This type of failure would most likely result in a small penetration of the cladding through which fission products would slowly be released into the primary coolant system and quickly detected by the on-line fission product monitor. Realistically it is difficult to postulate even a small fraction of the gaseous activity escaping the primary coolant system due primarily to the properties of iodine, since it is readily absorbed in solution or deposited on materials. This accident, the Fuel Failure during Reactor Operation, is analyzed and discussed in detail in Chapter 13, "Accident Analyses." Another accident, the Maximum Hypothetical Accident (MHA) for MURR, which assumes the complete failure of a fueled experiment, is analyzed and discussed in detail in Chapter 13, "Accident Analyses." This analysis does assume the release of fission products from the experiment to the pool coolant system and into the reactor containment building.

11.1.1.1.1 Argon-41 from the Pneumatic Tube System

As stated above, the principal production areas of Ar-41 within MURR includes the pneumatic tube (p-tube) system, and to a much lesser extent, the thermal column and the beamports. It is estimated that during p-tube operation, approximately 98% of the Ar-41 produced at MURR is from the p-tube terminals located in the graphite reflector region of the reactor core. During operation of the p-tube system, air containing Ar-41 is exhausted from the system through a high efficiency particulate air (HEPA) filter to the facility ventilation exhaust stack. Ar-41 produced in the thermal column and beamports is ducted to the 16-inch (40.64-cm) hot exhaust line which also exhausts to the facility ventilation exhaust stack.

There are no credible failures of the pneumatic tube system that would go unnoticed and uninvestigated at a small fraction of the 10 CFR 20, Appendix B, DAC limits, and thus, no failures that could release significant quantities of argon-41 to the containment or laboratory buildings. The pneumatic tube system is described in detail in Section 10.3.4.

Table 11-1 provides the measured flow rates and typical Ar-41 concentrations for the p-tube system, and the combined thermal column and beamport exhaust ducting.

TABLE 11-1
AR-41 PRODUCTION AT MURR AT 10 MW

Location	Flow Rate(cfm)	Ar-41 Concentration (μCi/ml)
Pneumatic Tube System		
a. With Blowers "On"	173	3.72×10^{-4}
b. With Blowers "Off"	45	1.46×10^{-3}
Thermal Column and Beamports Combined	304	3.66×10^{-5}

11.1.1.1.2 Argon-41 in the Reactor Containment Building

A limited amount of Ar-41 can be found in the reactor containment building during reactor operation. The containment building encloses greater than 225,000 ft^3 (6,371 m^3) of free space and contains the reactor and biological shield, the reactor control room, office spaces, and research instrumentation. Sampling performed within the containment structure indicates the average Ar-41 concentration is well below the NRC regulatory limit stated in 10 CFR 20. When compared to the Ar-41 Derived Air Concentration (DAC) limit of 3×10^{-6} μCi/ml, as listed in 10 CFR 20, Appendix B, facility workers are exposed to less than 1% of the DAC value.

11.1.1.1.3 Argon-41 Release to the Unrestricted Area

Ar-41 produced in the p-tube system, thermal column, and beamports is discharged from MURR through the facility's ventilation exhaust stack, which is nearly 70 feet (21 m) above grade level. Because the effluent exiting the exhaust stack carries momentum and buoyancy, the effective stack height is actually higher than 70 feet (21 m) and is variable depending upon wind conditions. Dilution with other building ventilation air and atmospheric dilution will reduce the concentration of Ar-41 considerably before the exhaust plume returns to ground-level locations which could be occupied by personnel. The detailed calculations relating to the dispersion of Ar-41 released from the exhaust stack are contained in Appendix B, "Radiological Impact of Ar-41 During Normal Operations." It is important to note that only a small amount of dilution is required to reduce the Ar-41 concentration to a level that is well below the 10 CFR 20 limit of 1×10^{-8} μCi/ml for unrestricted areas. This is due in part to the fact that the Ar-41 concentration leaving the ventilation exhaust stack while the reactor is operating is approximately 2.42×10^{-6} μCi/ml at a flow rate of about 30,500 ft^3/min (864 m^3/min).

The analysis of normal operational releases in Table B-4 assumed an annual average Ar-41 release of 1100 Ci/y. Releases of Ar-41 over the period 2020 through 2024 have averaged 1072 Ci/y.

Results of the dispersion calculations for the discharge of Ar-41 through the facility ventilation exhaust stack at a limit of 3.5×10^{-6} μCi/ml (see Section B.4) are shown in Tables 11-2 and 11-3 for various atmospheric conditions at distances of 150 and 760 meters from the exhaust stack.

The maximum annual dose to an individual from exposure to Ar-41 at a limit of 3.5×10^{-6} μCi/ml was calculated at two different distances: 150 meters to the north of the Emergency Planning Zone (EPZ) and

at the nearest residence in relation to the facility (approximately 760 meters north). The maximum annual dose at 150 and 760 meters was approximately 0.7 mrem/y and 4.2 mrem/y, respectively.

TABLE 11-2
MAXIMUM AR-41 CONCENTRATIONS AT 150 METERS NORTH FROM THE MURR EXHAUST STACK
(REFERENCE 11.5)

Atmospheric Stability		**Effective Stack Height**	**Ar-41 Concentration[a]**
Class	Condition	(meters)	(µCi/ml)
A	Extremely Unstable	35	3.14×10^{-9}
B	Moderately Unstable	27	3.04×10^{-9}
C	Slightly Unstable	23	2.28×10^{-9}
D	Neutral	20	5.71×10^{-10}
E	Slightly Stable	23	$2.38 \times 10^{--12}$
F	Moderately Stable	30	2.62×10^{-26}

a. Concentrations are based on an assumed Ar-41 release at concentration limit of 3.5 × 10-6 µCi/ml.

TABLE 11-3
MAXIMUM AR-41 CONCENTRATIONS AT 760 METERS NORTH FROM THE MURR EXHAUST STACK
(REFERENCE 11.5)

Atmospheric Stability		**Effective Stack Height**	**Ar-41 Concentration[a]**
Class	Condition	(meters)	(µCi/ml)
A	Extremely Unstable	16	1.65×10^{-10}
B	Moderately Unstable	8	5.18×10^{-10}
C	Slightly Unstable	4	8.55×10^{-10}
D	Neutral	1	1.99×10^{-9}
E	Slightly Stable	4	5.13×10^{-9}
F	Moderately Stable	11	1.12×10^{-8}

a. Concentrations are based on an assumed Ar-41 release at concentration limit of 3.5×10^{-6} µCi/ml.

Results of the dispersion calculations for the discharge of Ar-41 through the facility ventilation exhaust stack at a normal operational value of 2.42×10^{-6} µCi/ml (see Section B.5) are shown in Tables 11-4 and 11-5 for various atmospheric conditions at distances of 150 and 760 meters from the exhaust stack.

Because calculated dose estimates are proportional to the total amount of Ar-41 released, the dose estimates for normal operating conditions are easily calculated using the ratios of the stack release rates (given Ar-41 production remains constant). The normal operational annual dose at 150 and 760 meters is approximately 0.5 mrem/y and 3.0 mrem/y, respectively.

TABLE 11-4
NORMAL OPERATIONAL AR-41 CONCENTRATIONS AT 150 METERS NORTH FROM THE MURR EXHAUST STACK

Atmospheric Stability		**Effective Stack Height**	**Ar-41 Concentration[a]**
Class	Condition	(meters)	(μCi/ml)
A	Extremely Unstable	35	2.07×10^{-9}
B	Moderately Unstable	27	1.96×10^{-9}
C	Slightly Unstable	23	1.43×10^{-9}
D	Neutral	20	4.00×10^{-10}
E	Slightly Stable	23	1.57×10^{-12}
F	Moderately Stable	30	1.67×10^{-26}

a. Concentrations are based on a normal operational Ar-41 release concentration of 2.42×10^{-6} μCi/ml.

TABLE 11-5
NORMAL OPERATIONAL AR-41 CONCENTRATIONS AT 760 METERS NORTH FROM THE MURR EXHAUST STACK

Atmospheric Stability		**Effective Stack Height**	**Ar-41 Concentration[a]**
Class	Condition	(meters)	(μCi/ml)
A	Extremely Unstable	16	1.00×10^{-10}
B	Moderately Unstable	8	3.74×10^{-10}
C	Slightly Unstable	4	6.53×10^{-10}
D	Neutral	1	1.47×10^{-9}
E	Slightly Stable	4	3.68×10^{-9}
F	Moderately Stable	11	7.82×10^{-9}

a. Concentrations are based on a normal operational Ar-41 release concentration of 2.42×10^{-6} μCi/ml.

11.1.1.2 Liquid Radioactive Sources

All potentially radioactive liquid wastes are directed to a liquid waste retention and disposal system located on the below-grade level of the laboratory building. Liquid waste is then retained or chemically treated until an assay indicates activity levels are less than the limits specified in 10 CFR 20 for disposal by release into sanitary sewerage. Table 11-6 provides a list of the typical radionuclides which are routinely discharged to the sanitary sewer. Tritium accounted for about 88% of the total activity released/year over the period 2020 through 2024.

Radioactive liquid waste generated in the laboratories is the most significant source in terms of volume. However, the amount of radioactivity released to sanitary sewerage from this source is small when compared to the total amount released. Typical laboratory activities which produce radioactive liquid waste include sample preparation, vial and sample washing, and waste cleanup. Because of the diverse nature of the MURR utilization program, radionuclides and their concentrations within the liquid waste vary with time and with the nature of the experiment programs.

Since primary and pool coolant is, by design, contained to the maximum extent possible, there are no routine releases of these liquids. Non-routine liquid radioactive waste can result from maintenance tasks (e. g., resin transfers, filter replacement, etc.) and from decontamination activities. The amount of

this type of liquid is normally small, however, due to the presence of tritium in the primary and pool coolant, this liquid is the major contributor to the total amount of radioactivity released into sanitary sewerage.

11.1.1.2.1 Radioactivity in the Primary Coolant

As mentioned above, primary coolant is one of the three significant liquid radioactive sources at MURR. Radioactivity in this liquid occurs primarily from neutron interactions with oxygen in the water (creating nitrogen-16) and neutron interactions with system components with the subsequent transfer into the primary coolant. Manganese-56 and sodium-24 (Na-24) are common examples of waterborne radioactivity created in this manner. Tritium is also present in the primary coolant due to the activation of D_2O and by other mechanisms.

TABLE 11-6
TYPICAL RADIONUCLIDES RELEASED INTO CITY OF COLUMBIA SANITARY SEWERAGE

Radionuclide	Half-Life	Average Activity (mCi)[a]	Average % of Total Released[b]
Tritium (H-3)	12.3 years	140	87.9%
Sulfur-35 (S-35)	87.2 days	4.0	2.5%
Cobalt-60 (Co-60)	5.27 years	3.6	2.2%
Zinc-65 (Zn-65)	243.8 days	0.6	0.4%
Calcium-45 (Ca-45)	162.7 days	0.2	0.1%
Others	N/A	10.9	6.8%

a. Average amount of activity released into the sanitary sewerage system each year during the reporting period 2020 through 2024.
b. Average percent of total released each year during the reporting period 2020 through 2024.

Radionuclides and their concentrations in the primary coolant vary depending on reactor power, reactor operating time, and time since reactor shutdown, assuming that other variables (e.g., the effectiveness of the water cleanup system) remain constant. Table 11-7 is a list of the predominant radionuclides and their measured concentrations present in the primary coolant at 10 MW.

TABLE 11-7
PREDOMINANT RADIONUCLIDES IN MURR PRIMARY COOLANT AND THEIR MEASURED CONCENTRATIONS AT 10 MW

Radionuclide	Half-Life	Typical Measured Concentration[a]
Tritium (H-3)	12.3 years	7.28×10^{-2} µCi/ml
Magnesium-27 (Mg-27)	9.45 minutes	8.78×10^{-2} µCi/ml
Sodium-24 (Na-24)	14.96 hours	2.74×10^{-2} µCi/ml
Manganese-54 (Mn-54)	312.2 days	3.64×10^{-5} µCi/ml
Manganese-56 (Mn-56)	2.58 hours	1.83×10^{-4} µCi/ml
Xenon-135 (Xe-135)	9.10 hours	3.03×10^{-5} µCi/ml

a. These values are typical of the measured concentrations that exist in the primary coolant at 10 MW two to three days after reactor startup.

Although tritium is the major contributor to the total amount of activity released, it is not released into sanitary sewerage until activity levels are less than the limits specified in 10 CFR 20. Therefore, the primary coolant does not represent a source of exposure to the general public during normal operation. Occupational exposure from primary coolant is limited, as few operations require direct contact with the primary coolant. In cases where there is a potential for contact, such as in certain maintenance activities, the primary coolant is usually allowed to decay in order to significantly reduce radioactivity concentrations. Because of the short half-lives of most of the predominant radionuclides in the primary coolant, many of the radionuclides are essentially gone after 48 hours, with Na-24 reduced by about a factor of 10. In addition, experience at MURR and at other research reactors indicates that tritium is not a source of significant occupational dose.

11.1.1.2.2 Radioactivity in the Pool Coolant

Pool coolant is another significant radioactive liquid source. Radioactivity in this liquid occurs by the same mechanisms as described in Section 11.1.1.2.1: neutron interactions with oxygen in the water and neutron interactions with system and structural components, with the subsequent transfer into the pool coolant. Table 11-8 is a list of the predominant radionuclides and their measured concentrations present in the pool coolant at 10 MW.

TABLE 11-8
PREDOMINANT RADIONUCLIDES IN MURR POOL COOLANT AND THEIR MEASURED CONCENTRATIONS AT 10 MW

Radionuclide	Half-Life	Typical Measured Concentration[a]
Tritium (H-3)	12.3 years	8.29×10^{-2} µCi/ml
Magnesium-27 (Mg-27)	9.45 minutes	1.10×10^{-2} µCi/ml
Sodium-24 (Na-24)	14.96 hours	4.64×10^{-3} µCi/ml
Manganese-56 (Mn-56)	2.58 hours	2.54×10^{-3} µCi/ml
Technetium-101 (Tc-101)	14.2 minutes	4.70×10^{-5} µCi/ml
Technetium-99m (Tc-99m)	6.01 hours	9.73×10^{-6} µCi/ml
Antimony-122 (Sb-122)	2.70 days	1.01×10^{-5} µCi/ml
Xenon-135 (Xe-135)	9.10 hours	1.22×10^{-5} µCi/ml
Silver-110m (Ag-110m)	248.9 days	1.10×10^{-5} µCi/ml

a. These values are typical of the measured concentrations that exist in the pool coolant at 10 MW two to three days after reactor startup.

11.1.1.2.3 Nitrogen-16 and Oxygen-19 from the Primary and Pool Coolant Systems

Nitrogen-16 (N-16, half-life 7.14 sec) is generated by the reaction of fast neutrons with oxygen-16 in the water which passes through or near the reactor core. The amounts of oxygen present in air, either in the path of a beam or entrained in the water near the reactor core, is insignificant when compared to the amount of oxygen in a water molecule in the liquid state. Production of N-16 resulting from neutron interactions with oxygen in air and air entrained in the coolant can therefore be neglected.

Similarly, Oxygen-19 (O-19, half-life 26.9 sec) is generated by the reaction of thermal neutrons with oxygen-18 in the reactor coolant water.

As described in Section 5.7, radiation levels from these activated short-lived radioisotopes are reduced by the use of hold-up tanks in both the primary coolant demineralizer loop and the pool coolant system. These internally baffled tanks hold up, or delay, the primary and pool coolant within a restricted, shielded space (Room 114) for a sufficient amount of time to allow short-lived activity, primarily N-16 and O-19, to decay. During reactor operation, Room 114 is controlled as a high radiation area with access restricted by a gate which is locked and remotely alarmed in the reactor control room. As a result, occupational exposure from these short-lived isotopes is insignificant.

11.1.1.3 Solid Radioactive Sources

The solid radioactive sources associated with normal operation of MURR are summarized in Table 11-9. Because the actual inventory of reactor fuel and other radioactive sources continuously change as part of the normal operation of the reactor and the experimental program, the information presented in Table 11-9 should be considered representative rather than an exact inventory.

Additional information on waste classification, storage, packaging, and shipment is included in Section 11.2. Routinely produced solid waste includes reactor coolant cleanup system demineralizer resin beds, mechanical filters, rags, paper towels, plastic bags, rubber gloves and other materials used for reactor maintenance activities and for the utilization of the experimental facilities. Solid Low-Level Waste (LLW) is packaged in sealed metal containers, typically 55-gallon barrels or B-25 containers. The radioactivity level of each barrel is typically in the millicurie range. Table 11-10 provides a list of the volume of solid waste removed from the site from 2020 through 2024 and its radioactive content, in millicuries.

TABLE 11-9
REPRESENTATIVE RADIOACTIVE SOURCES AT MURR

Source Description	**Radionuclide(s)**	**Nominal Activity (Ci)**	**Physical Characteristics**
Special Nuclear Material	See note		
Small Instrument Calibration & Check Sources	Cs-137, Ba-133, Co-60, Eu-152, Tc-99,& other sources	$< 10^{-4}$ each	Plated Discs, Plastic Buttons, Flux Wires, Planchets, Filter Papers, etc.
Neutron Source	Sb-124	100	In Pool Storage-Sealed Source
Demineralizer Resin	Na-24, Co-60 & other mixed activation products from the primary & pool coolant systems	0.5 to 1.5 each tank	12 ft^3 of mixed bed resin stored in 4 steel tanks Typically: In Primary Service (1); In Pool Service (1); Depleted (2);
Irradiated Items and Materials	Mixed activation products	10^{2} to 10^{-8}	Sealed & unsealed items irradiated in the experimental facilities
Solid Waste	Co-60, Zn-65, Mn-54, and other mixed activation products	See Table 11-10	Typically: 55-gallon barrels and B-25 containers. Total annual waste volumes – see Table 11-10

Note: SNM sources are security plan items and are part of security-related inventory.

TABLE 11-10
MURR WASTE SHIPMENTS

Year	**Volume (Cubic Feet)**	**Activity (mCi)**
2024	3,137	5,480
2023	2,109	8,522
2022	1,742	156,556[a]
2021	1,525	20,249
2020	1,824	297,473[a]

a. Type B reactor component waste shipment occurred this year

11.1.2 Radiation Protection Program (Reference 11.2 and Reference 11.3)

The Reactor HP Staff is located within the organization for the management and operation of MURR as shown in Figure 12.1 and discussed in Chapter 12, “Conduct of Operations.” The Reactor HP Staff includes the Reactor HP Manager and the Reactor HP Staff under their direction.

The Reactor HP Manager reports to the Reactor Facility Director. However, there is a communications/consultation line from the Reactor HP Manager to the Office of the Chancellor. This line of communications/ consultation allows the Reactor HP Manager access to upper University management if Reactor Facility Management does not address radiation protection concerns to the satisfaction of the Reactor HP Manager.

11.1.2.1 Reactor Health Physics Staff

The Reactor HP Staff is the organization which administers the RPP for the reactor facility. The staff consists of the following personnel: the Reactor HP Manager, Reactor Health Physicists, Reactor HP Technician Supervisor, and Reactor HP Technicians.

The qualifications for the positions within the Reactor HP Staff are as follows:

Manager, Reactor Health Physics
The Reactor HP Manager should have a minimum of four years of experience in operational HP and management. The individual should have a recognized master's degree in HP or an equivalent combination of education and experience from which comparable knowledge and abilities can be acquired;

Reactor Health Physicist
The Reactor Health Physicist should have a minimum of two years of experience in applied HP. The individual should have a bachelor's degree in HP or an equivalent combination of education and experience from which comparable knowledge and abilities can be acquired;

Reactor Health Physics Technician Supervisor
The Reactor HP Technician Supervisor should have a minimum of two years of experience in operational HP. The individual should have an associate's degree in HP or an equivalent combination of education and experience from which comparable knowledge and abilities can be acquired; and

Reactor Health Physics Technician
The Reactor HP Technician should have a minimum of a high school diploma in order to become a Reactor HP Technician Trainee. The individual will become internally qualified with a combination of education and experience prior to being entrusted as a Reactor HP Technician.

The positions of authority and responsibility within the Reactor HP Staff are discussed in Chapter 12, "Conduct of Operations."

The working relationship of the Reactor HP Staff relative to the Reactor Operations Staff is shown in Figure 12.1. As shown in this figure, there is a clear separation of responsibilities for the two groups, each with a clear reporting line to the Reactor Facility Director.

11.1.2.2 Isotope Use Subcommittee

The Isotope Use Subcommittee (IUS) may be authorized to act as an advisory group to the RAC regarding matters relating to the custody and use of radiation and radioisotopes within MURR. The RAC and the IUS are described in detail in Chapter 12, "Conduct of Operations."

11.1.2.3 Radiation Safety Committee

The MURR Facility Director is responsible for appointing members and assigning the chairperson to the MURR Radiation Safety Committee (RSC). Committee membership is constituted to satisfy the requirements as outlined in 10 CFR 33. A representative from MURR management and persons knowledgeable in the application and uses of radioactive materials will be included in the committee membership. The MURR Radiation Safety Officer (RSO) will be ex-officio, non-voting member of the MURR RSC.

The RSC's primary duties include:

- Reviewing and approval of the use of radioactive materials covered by the Broad Scope Materials License;
- Ensuring that the use of licensed material is consistent with the ALARA philosophy and program; and
- Reviewing the performance of the RSO and the Reactor HP Staff to assure adequate control of radiation risks with respect to the Broad Scope Materials License.
- Meetings of the RSC are conducted in accordance with a written charter.

11.1.2.4 Radiation Safety Officer

The RSO fulfills the radiation protection duties required under the Broad Scope Materials License. The Reactor HP Manager is responsible for the radiation protection duties under the Reactor License. These can be the same individual. (Reference 11.4)

The RSO is responsible for the implementation of the policies established by the RSC. The RSO is appointed by the Facility Director, as delegated by the Office of the Chancellor. The RSO reports through to the Facility Director. The RSO's primary duties, with the assistance of the Reactor HP Staff, include:

- Assisting the RSC in the performance of its duties, e.g., coordinating the review of safety evaluations of all proposed uses of radioactive materials, providing staff assistance, and implementing the policies established by the RSC;
- Issuing all authorizations for use of radioactive material covered by the Broad Scope Materials License on behalf of the RSC;
- Maintaining a list of the current authorizations and approvals to provide an accountability of radioactive materials used under the Broad Scope Materials License;

- Determining compliance with Federal regulations, with consistency and compatibility for all appropriate licenses and their conditions, and with the conditions of project approvals as specified by the RSC; and
- Regularly evaluating and reviewing (including routine surveys and inspecting) projects authorized for use of radioactive materials.

11.1.2.5 Radiation Protection Training

The MURR Radiation Protection Training Program ensures that initial and refresher training is provided to all individuals who will use, and/or may come in contact with radioactive materials. The program is structured at different levels in order to meet the diverse needs of the reactor facility staff, researchers, and students using the facility. In accordance with 10 CFR 19.12, such training will be commensurate with the level of radiological safety established for the type of work/research being performed.

All personnel and visitors entering the reactor facility shall receive training in radiation protection sufficient for their work/visit or shall be under the constant escort of an individual who has received such training. Visitors entering beyond the front lobby are required to sign in and out on the MURR visitors' log or a tour sheet. A statement is provided in the visitors' log which alerts the visitor that radioactive materials are used at the facility and that their exposure to this material will be maintained below the Federal limits.

The general levels of radiation protection training at the reactor facility are listed below.

Class III

Individuals granted Class III access are granted unescorted access to the reactor facility. Individuals are required to complete an initial training program which normally includes the viewing of a visual presentation instructing the individual in the general security, emergency, and radiation safety procedures established for MURR. The initial training program also includes the completion of documentation for the level of unescorted access, assignment of personnel dosimetry when required, and a tour of the reactor facility. Radiation safety training topics shall cover the following areas in sufficient depth for the work being done:

- Biological effects of ionizing radiation;
- Principles and practices of radiation safety (ALARA);
- Radioactivity measurements and monitoring techniques;
- Applicable regulations and license requirements;
- Areas where radioactive materials are used and stored;
- Appropriate radiation protection procedures and practices;
- The individual's responsibility to report unsafe conditions (potential regulatory and license violations) to the Reactor HP Staff and/or applicable authorities;
- Appropriate response to emergencies or unsafe conditions;
- Worker's right to be informed of occupation radiation exposure and bioassay results;
- Employee protection as described in 10 CFR 50.7; and

- Locations where copies of pertinent regulations/notices are posted or made available.

Individuals receiving this level of training normally include University Police, campus and other maintenance/construction staff, custodians, clerical staff, etc. Individuals who have unescorted access to the facility are considered to be radiation workers even if they do not actively work with radioactive materials.

Personnel that maintain their unescorted access must be periodically re-trained. This consists of the individual updating any changes in their personal information and reviewing a training presentation. Periodic re-training programs may also include additional update and review discussions and/or facility tours. In addition, training programs and drills specific to the emergency procedures required under the Emergency Plan are conducted.

Personnel having unescorted access are allowed to handle radioactive materials without direct supervision only at levels less than the exempt quantities as defined in 10 CFR 30.71, Schedule B. Individuals having Class III access are allowed to work with radioactive materials greater than exempt quantities only under the direct supervision of a radiation worker with Class I or II status, or under the supervision of a member of the Reactor HP Staff.

Class II

In addition to expanding certain subject areas listed for the Class III level of training (unescorted access), Class II may include radiation safety training related to the following areas:

- Radioactive decay, and radiation units and quantities;
- Specific radiation protection techniques, including external protection;
- Radiation instrumentation, including the use of air monitoring and special personnel dosimetry;
- Contamination checks and surveys;
- Basic calculations for the use and measurement of radioactivity;
- Decontamination techniques;
- Radioactive materials transfer/release requirements;
- Bioassay considerations; and
- Radioactive waste disposal.

Additional training time ranges from a minimum of twenty hours of combined formal and on-the-job training depending on the scope of work, the quantity and type of radioisotope, and the level of hazard involved. In addition, individuals seeking Class II status must have greater than 1 month of Radiation Worker experience. In general, an individual who has a Class II level of training is approved to work with radioactive materials without direct supervision and may be described as "self-supervised." Prior documented training and experience may be used to fulfill part of the training requirements.

Class I

This level of training is required for individuals requesting permission to direct or supervise the work of others in utilizing radioactive materials under Reactor License or Materials License Projects. All training at MURR applies to and is sufficient to allow work with byproduct material for either the Broad Scope

Material or Reactor license (Reference 11.4). In addition to the requirements for Class II level, training for Class I will focus on subjects specific to administrative controls, such as:

- Requirements for the application of project authorization;
- Safety evaluation criteria;
- Regulatory requirements and license conditions;
- Personnel training and approval; and
- Documentation requirements.

This training shall include a minimum of an additional twenty hours of combined formal and on-the-job training beyond Class II status. In addition, individuals seeking Class I status must have greater than 6 months of Radiation Worker experience. Prior documented training and experience may be used to fulfill part of the training requirements.

Records of formal training are maintained by MURR. These records will normally include the following information: course/lecture/visual outline and/or material used; date(s) of training; names of trainees; means of indicating participation and understanding; duration and name of instructor/lecturer, if applicable. Formal training will be provided by the Reactor HP Staff, authorized supervisors, or other individuals who, by their training and experience, are approved by the Reactor HP Manager or RSO to provide given elements of formal radiation protection training.

11.1.2.6 Reactor Health Physics Procedures

The Reactor HP Manager shall approve the radiological control procedures and the procedures for the preparation for shipping and the shipping of byproduct material (Reference 11.7). The Reactor HP Operating Procedures provide the methods and guidelines for the implementation and the maintenance of the MURR RPP. Changes to these procedures are made by the Reactor HP Manager, or his authorized delegate, and are subsequently reviewed by the Isotope Use Procedures Review Subcommittee (IUPRS), when applicable.

While not intended to be all inclusive, the following list provides an indication of the typical evolutions or programs that require reviewed written radiation protection procedures for the Reactor Health Physics Staff:

- Reactor facility radiation monitoring program including surveys, personnel monitoring, radioactive waste management, and sampling and analysis of solid, liquid, and gaseous wastes released from the facility;
- Calibration of area radiation monitors, facility air monitors, laboratory radiation detection systems, personal radiation monitoring devices, and portable radiation monitoring instruments;
- Administrative guidelines for the facility personnel indoctrination training program;
- Receiving and opening packages of radioactive materials and any subsequent transfer within the facility;
- Monitoring of radioactivity in the environment surrounding the facility;

- Leak testing of sealed sources containing radioactive materials;
- Shipment of radioactive materials;
- Radioactive analysis of the primary and pool coolant;
- Radiation Work Permit procedures; (Radiation Work Permits (RWPs) provide instructions to workers prior to performing radiological operations. A RWP is used for work not controlled by an operating procedure which has the potential to release unusual radioactive airborne or surface contamination. RWPs also document exposures received during the radiological operations and provide documentation of specific comments germane to future radiological work.)
- Controlled special exposures (ALARA considerations);
- Unplanned personnel radiation exposure investigation procedures;
- Secondary coolant and facility sump water analyses; and
- Monitoring a beamport area during a reactor startup.

11.1.2.7 Reactor Health Physics Audits

Periodic audits are performed by MURR management or their authorized delegates in order to verify the adequacy and the implementation of the programs and operating procedures designed to ensure that radiation safety and compliance with applicable regulations are maintained. The audits will normally be conducted annually and shall include a selective (but comprehensive) examination of logs, operating records, data sheets, and other documents. Discussions with personnel and observation of operations are also performed as appropriate.

The audits shall include, but are not limited to, the following (source document that either requires or recommends that the audit be performed included):

- Radiation Protection Program - 10 CFR 20.1101;
- As Low As is Reasonably Achievable (ALARA) Program - Regulatory Guide 8.10.C.1 and C.2;
- Reactor HP Operating Procedures - MURR Technical Specification No. 6.4.b;
- Type B Radioactive Materials Shipping Program - 10 CFR 71.137; and
- Broad Scope Materials License – Materials License Renewal Application.

11.1.2.8 Reactor Health Physics Records

Records that document compliance with regulations regarding radiation protection are maintained by the Reactor HP Staff. These records may be in the form of logs, data sheets, or other suitable forms or documents. The required information may be contained in single or multiple records, or a combination thereof. Normally, records for the current and previous year are retained on-site by the Reactor HP Office. Other records are retained in long-term storage (remote record retention facilities). Radiation protection records are reviewed by the Reactor HP Manager or their authorized delegate prior to filing.

In addition to the records listed in Section 12.5, records of the following activities shall be maintained and retained for the periods specified below.

- Lifetime Records - The following records are to be retained for the lifetime of the reactor facility:
    - Provisions of the RPP;
    - Surveys conducted for dose assessments;
    - Measurements and calculations used to determine intakes of radioactive material;
    - Results of air samples, surveys, and bioassays;
    - NRC Form 4 records for each individual;
    - Records of planned special exposures;
    - Records of doses received during work, accidents, emergency conditions and doses received by an embryo/fetus of a declared pregnant woman;
    - Declaration of pregnancy;
    - Records to demonstrate compliance with the dose limit for individual members of the public;
    - Records of the disposal of licensed materials; and
    - Records of restricted areas, areas documented for contamination outside restricted areas, buried waste outside restricted areas, and areas where radioactive materials would be required to be moved if the license were terminated.
- Five Year Records - The following records are to be retained for a period of at least five years:
    - Reactor facility radiation and contamination surveys required by applicable regulations;
    - Audits of the RPP.
- Three Year Records - The following records are to be retained for a period of at least three years:
    - Surveys conducted for the receipt of radioactive material;
    - Records of instrument calibrations;
    - Records used to prepare NRC Form 4; and
    - Records of receipt and transfer of byproduct material; receipt records for as long as the material is possessed, three years following transfer.

Radiation protection records are used for developing trend analysis, for keeping management informed regarding radiation protection matters, and for reporting to regulatory agencies. In addition, they are used for planning radiation-protection-related work or research, e.g., radiological surveys as required by RWPs or to evaluate the effectiveness of decontamination efforts or temporary shielding placement.

11.1.3 ALARA Program

The ALARA Program for MURR is dedicated to the fundamental principle of maintaining individual exposures and radioactive effluents ALARA. Responsibilities for maintaining the ALARA Program extend to all individuals who are granted access to the reactor facility. The Reactor Facility Director has the ultimate responsibility for the ALARA program but has delegated this responsibility to the Reactor HP Manager.

The MURR ALARA program, which is a component of the RPP, has been established in accordance with 10 CFR 20.1101 and is audited annually as specified in Regulatory Guide 8.10, sections C.1 and C.2. (Reference 11.6).

Personnel radiation doses at the reactor facility are minimized by considering the use of the following ALARA principles when planning or performing work with radiation or radioactive materials:

- Allowing radiation source(s) to decay;
- Installing portable or temporary shielding;
- Installing portable or temporary ventilation systems, or temporary enclosures and covering, or both;
- Conducting "dry runs" on mockup equipment to identify problems which might be encountered in the actual job and to select and qualify special tools and procedures;
- Conducting pre-operational briefings with those assigned to perform tasks in high radiation areas;
- Ensuring that unnecessary personnel are kept out of areas where radiation exposure may occur;
- Ensuring that sufficient radiation monitoring instruments are available to permit accurate measurements and rapid evaluations of the radiation and contamination levels encountered; and
- Relocating component(s) to be worked on to a lower radiation area.

The MURR ALARA program (POL-03, RPP) also contains the following elements which enhance the effectiveness and reliability of the overall program.

Commitment

MURR management is committed to the ALARA program for keeping individual and collective radiation exposures and radioactive effluents ALARA. In addition, the sum of the doses received by all exposed individuals is also maintained at the lowest practicable level.

Review

All new uses of radiation sources and radioactive materials licensed under the facility operating license must be approved by the Reactor Manager and Reactor HP Manager. These approvals are reviewed by the RAC. Work with radioactive materials licensed under the Broad Scope Materials License must be approved by the RSC prior to starting such work.

Responsibilities

Operational responsibilities of the following groups or individuals for adherence to the ALARA program are:

- Reactor HP Staff:
    - Inform personnel of ALARA program efforts;
    - Ensure that personnel who work with radioactive materials are instructed in the ALARA philosophy;

- Establish programs for routine radiation surveys in areas where radioactive materials and/or sealed sources of radiation are used, handled or stored;
- Procure, distribute, calibrate, and maintain the necessary radiation safety instrumentation;
- Quantify all radioactive effluents from the reactor facility;
- Trend selected effluents to ensure the adequacy of engineered controls in minimizing radioactive effluents;
- Provide advice and assistance regarding disposal of liquid effluents and potential releases of gaseous effluents;
- Provide advice and assistance for decontamination and for special surveys or inspections of the facility;
- Monitor requisition, receipt, and delivery of all incoming shipments of radioactive materials;
- Monitor proper radioactive material transfer, waste handling, and disposal in accordance with ALARA principles;
- Perform safety (ALARA) evaluations of facilities, equipment, and procedures employed in areas where radioactive materials are used, handled, or stored; and
- Evaluate the personnel dose and effluent release investigation levels on an annual basis to ensure that they are still appropriate.

- Project Leaders and Authorized Supervisors:
  - To all individuals under their supervision, explain the ALARA concept and their commitment to it;
  - Ensure that personnel under their supervision adhere to established protocols and good HP practices during the performance of occupational duties;
  - Consult with the Reactor HP Staff and obtain the approval of the Reactor Manager and Reactor HP Manager for work under the facility operating license, or approval of the RSC before using radioactive materials under the Broad Scope Materials License; and
  - Evaluate all applicable practices and procedures before using radioactive materials to ensure that exposures and releases adhere to the ALARA principle.

- Radiation Workers and Approved Workers:
  - Assume responsibility for their own participation in the ALARA program, for adhering to project procedures/protocols, and for employing good HP practices in daily work;
  - Know what recourse is available if ALARA is not being promoted on the job;
  - Take the opportunity to participate in the formulation or modification of the procedures that they are required to follow; and
  - Make any suggestions, as expected, to their supervisors and/or the Reactor HP Staff about improving HP practices for working with radioactive materials or radiation sources.

<u>Investigation Levels</u>

Investigation levels for occupational radiation exposures and effluent concentrations are established

which, when exceeded, initiate a review or investigation by the Reactor HP Staff. This investigation or review is focused on determining the cause of the exposure so that appropriate ALARA actions, if any, can be applied.

Actions for Investigational levels are described in MURR ALARA Program, of MURR Policy Manual POL-3, "Radiation Protection Program," and are excerpted here:

- Personnel dose less than or equal to Investigational Level I

  No action will be taken in those cases where an individual's dose is less than values for Investigational Level I.

- Personnel dose greater than Investigational Level I but less than or equal to Investigational Level II

  The HP Staff will notify the appropriate group's ALARA reviewer of each individual whose dose exceeds Investigational Level I and will report the results of the reviews to MURR Management and actions will be taken if they are deemed appropriate.

- Personnel dose greater than Investigational Level II

  HP Staff will investigate with the appropriate group's ALARA reviewer the cause(s) of all personnel exposures exceeding Investigational Level II and will, if warranted, take corrective action. A report of the investigation will be presented to MURR Management and to the Isotope Use Subcommittee of the Reactor Advisory Committee for information and review of actions. (Reference 11.4)

11.1.4 Radiation Monitoring and Surveying

The radiation monitoring program for MURR has been established to meet the requirements for the utilization of radioactive materials and radiation sources licensed under the facility operating license. This program is structured such that all three categories of radiation sources (air, liquids, and solids) are detected, measured, and assessed in a timely manner. To achieve this, the monitoring program consists of two major types of surveys: routine surveys and monitoring of specific areas and activities within the facility, and special surveys or monitoring to support non-routine operations.

11.1.4.1 Radiation and Contamination Surveys

Periodic surveys are performed throughout the reactor facility in order to monitor radiation and contamination levels. The frequency of these routine surveys is determined by the basis of degree of utilization and levels of radioactivity handled in the various work areas. Special surveys are performed as required.

While not intended to be all inclusive, the following list provides examples of the typical evolutions or activities performed at MURR which would require a radiation survey:

- Changes to a beamport or other reactor experiment which could lead to significant alterations in area radiation levels;
- Entry into the mechanical equipment room (Room 114) following a reactor shutdown;
- Lowering water level in the reactor pool below the normal operating level;
- Removal of a Nuclear Instrumentation (NI) detector assembly from a drywell;
- Opening the thermal column door;
- Receipt and opening of a package containing radioactive material(s) from another facility or organization;
- Removal of an activated or contaminated component or irradiated sample from the reactor pool;
- As determined by a RWP;
- Periodically in accessible radiation areas and high radiation areas, and in all other occupied areas in the facility;
- Upon initial opening of the primary or pool coolant system for inspection, maintenance, or repair;
- Prior to inspecting an irradiated fuel element;
- Receipt of new, unirradiated reactor fuel; and
- Shipment of irradiated reactor fuel.

While not intended to be all inclusive, the following list provides examples of the typical evolutions or activities performed at MURR which would require a contamination survey:

- Receipt and opening of a package containing radioactive material(s) from another facility or organization;
- Release of equipment, supplies, or materials from the reactor facility for unrestricted use;
- Periodically in accessible contaminated areas and occupied areas surrounding contaminated areas;
- Periodically in occupied non-contaminated areas of the facility;
- Upon initial entry into potentially contaminated ventilation ducting for its inspection, repair, or filter replacement;
- Decontamination of equipment;
- Receipt of new, unirradiated reactor fuel;
- Shipment of irradiated reactor fuel;
- As determined by a RWP; and
- Whenever operations are performed that are known to result in, or are expected to result in, the spread of contamination.

In addition to the surveys used to assess radiation and contamination levels, air in the reactor containment building is routinely sampled for tritium and Ar-41 to ensure that levels remain below the regulatory limits.

11.1.4.2 Radiation Monitoring Equipment (Reference 11.1)

Radiation monitoring equipment used at MURR is summarized in Table 11-11. The vast scope of work performed under the facility operating license requires a wide variety of radiation detection and monitoring equipment to provide adequate radiation protection. If a safety evaluation identifies the need for additional and/or different instrumentation to support a particular research project, then the instrumentation will be acquired for that project. The instrumentation listed in Table 11-11 is available to both the Reactor HP Staff and to MURR radiation workers. Because periodic updating and replacement of instrumentation are performed to take advantage of technological improvements, Table 11-11 is not intended to be all inclusive and should be considered representative rather than an exact listing. However, the required function that the radiation monitoring equipment performs remains the same.

The permanently installed radiation monitoring equipment at the reactor facility, including the Area Radiation Monitoring System (ARMS), the Fuel Element Failure Monitoring System, the Secondary Coolant Monitoring System, and the Off-Gas Radiation Monitoring System, is discussed in detail in Chapter 7, "Instrumentation and Control Systems." Personnel monitoring is discussed in Section 11.1.5.5.2.

11.1.4.3 Instrument Calibration

Radiation protection instrumentation used at MURR is periodically calibrated according to written procedures. Portable radiation survey instruments used for personnel radiation monitoring are calibrated in accordance with the requirements of the American National Standards Institute (ANSI) Standard N323A-1997, "American National Standard Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments." Other radiation protection equipment is calibrated in accordance with the manufacturers' suggested procedures as applicable to the use of the instrument at MURR. It is the policy of the facility to use radiation sources traceable to the National Institute of Standards and Technology (NIST) for instrument calibrations whenever possible.

The following instrumentation is normally calibrated at MURR by individuals trained to perform primary instrument calibration:

- Portable Contamination Meters ("friskers");
- Gamma Spectroscopy Counting Systems;
- Dose Calibrators;
- Alpha/Beta Planchet Counters;
- Portal Monitors;
- Portable Air Samplers;
- Continuous Air Monitors;
- Off-Gas Radiation Monitor;
- Area Radiation Monitors; and
- Portable G-M Survey Meters.

The following instrumentation is normally calibrated at a vendor's calibration facility:

- Portable Ionization Chamber Survey Meters;
- Portable Alpha Survey Meters;
- Portable Neutron Survey Meters; and
- Air Flow Measurement Instruments;

The frequencies at which instrument calibrations are required to be performed are monitored by a computer-based tracking system. Instrument calibration records are maintained by the Reactor HP Staff for a period of at least three years. These include the model and serial number of each source used and the identity of the radionuclide contained in the calibration source, as well as the instrument model and serial number and the individual or vendor performing the calibration. Calibration stickers showing pertinent calibration information (e.g., date of most recent calibration, initials of calibrator, and the date the next calibration is due) are attached to all radiation survey meters.

TABLE 11-11
RADIATION MONITORING AND RELATED EQUIPMENT

Equipment	**Function**	**Typical Location(s)**
Portable Contamination Meters	Measure alpha/beta/gamma contamination levels	Laboratories HP Spaces
Portable Alpha Survey Meters	Measure alpha contamination levels only	Laboratories HP Spaces
Portable G-M Survey Meters	Measure beta/gamma radiation dose rates	Laboratories HP Spaces Control Room
Portable Ionization Chamber Survey Meters	Measure beta/gamma radiation dose rates	Laboratories HP Spaces
Portable Neutron Survey Meters	Measure neutron radiation dose rates	Beamport Area HP Spaces
Gamma Spectroscopy Systems	Perform gamma-ray spectroscopy	Laboratories HP Spaces
Dose Calibrators	Measure beta/gamma activities	Laboratories
Alpha/Beta Planchet Counters	Measure alpha/beta/gamma contamination on swipes	HP Spaces
Liquid Scintillation Counters	Measure beta contamination in liquid	HP Spaces
Optically Stimulated Luminescence Dosimeters (OSLs)	Measure environmental gamma radiation doses	Various On-Site and Off-Site Locations
Electronic Personal Dosimeters	Measure personnel gamma dose	Laboratory Building Kiosks
Hand and Foot Monitors	Measure potential contamination on hands and feet	Exit from Potentially Contaminated Areas
Portal Monitors Personnel Contamination Monitors	Measure potential contamination on the whole body prior to exiting the facility	Exits from Radiological Control Boundaries
Air Monitors	Measure radioactivity in stack effluent; Measure area airborne radioactivity; Collect grab air samples	West Tower HP Spaces Various Other Locations
Area Radiation Monitors (ARMs)	Measure gamma radiation fields at various locations in the facility	Various Locations
Fuel Element Failure Monitoring System	Monitor the primary coolant system for fission product activity buildup	Mechanical Equipment Room (Room114)
Secondary Coolant Monitoring System	Monitor the secondary coolant system for the presence of radioactive isotopes	Cooling Tower Tunnel
Air Flow Measurement Instruments	Measure facility ventilation flow rates	HP Spaces

11.1.5 Radiation Exposure Control and Dosimetry

Radiation exposure control at MURR depends on many different factors including reactor facility design features, operating procedures, training, adherence to the ALARA program, etc. Radiation protection training and operating procedures and the ALARA program are covered in Section 11.1.2 and Section 11.1.3, respectively. This section covers design features such as radiation shielding, ventilation, containment, and entry control devices for high radiation areas as well as personnel protective equipment and dosimetry devices.

11.1.5.1 Radiation Shielding

Shielding is the paramount design feature in controlling radiation exposure during operation of the reactor. Shielding has been installed to keep radiation levels in areas occupied by personnel ALARA. All shielding thicknesses are based on an operating power level of 10 MW. Fuel storage and handling requirements are based on 40-day continuous operation at 10 MW prior to shutting down and removing fuel.

The following dose rate schedule was used in the initial analysis and design of the primary reactor shielding:

- 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 mrem/h;
- 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 mrem/h;
- At three feet (0.9 m) from any experimental facility opening in the biological shield and on the centerline of the opening, the dose rate shall not exceed 2.5 mrem/h;
- At one foot (0.3 m) from the mechanical equipment room's (Room 114) walls and ceiling, the radiation level shall not exceed 2.5 mrem/h;
- At one foot (0.3 m) from the demineralizer cell's walls and ceiling, the radiation level shall not exceed 2.5 mrem/h;
- At one foot (0.3 m) above the beamport floor, but ten feet (3 m) from the primary reactor shield, the radiation level shall not exceed 2.0 mrem/h; and
- At any location at the top surface of the reactor pool, the radiation level shall not exceed 20 mrem/h.

Several additional dose rate criteria regarding fuel handling were selected for design consideration. The design conditions are 40-day continuous operation at 10 MW, followed by a 1.16-day fission product decay time prior to fuel handling and storage. Shield thicknesses for fuel storage are based on these conditions and the same dose rate criteria as for the bulk shield. The required water level in the reactor pool during fuel handling is based on a 100 mrem/h maximum dose rate from a fuel element at these reference conditions during the fuel transfer.

The shield thicknesses for the primary reactor shield are discussed below. These include the required magnetite concrete thicknesses for the bulk shield structure, water shielding requirements above an operating reactor core, and the shielding requirements during spent fuel element transfer and storage as a function of the fission product decay.

11.1.5.1.1 Biological Shield

The initial analyses used to determine the required pool water depth and thickness of the MURR biological shield are discussed below. Gamma-ray and neutron attenuation are considered individually. The physical construction of the biological shield is described in Section 4.4.

Gamma-Ray Attenuation

The attenuation of the gamma-rays originating in the reactor core from prompt fission, fission products, and radiative neutron capture gamma-rays generated throughout the reactor core assembly was considered in the analysis of the biological shield requirements. An analytical model of the reactor core and the surrounding regions was developed. External to this analytical model of the reactor system, the thickness of the water and magnetite concrete cylindrical annuluses were varied to account for the eccentricity of location of the core in the reactor pool.

All calculations of required shield thicknesses were based on regional gamma-ray source spectrums generated by the Internuclear Company. The calculations of the gamma-ray dose external to the biological shield were performed using a computer program which included gamma-ray buildup factors. In an effort to assure the validity of using the selected buildup factors, a series of identical applicable problems were calculated using iron dose buildup factors and water dose buildup factors. The iron and water calculations bracketed the value obtained for the concrete shielding and served to substantiate the use of the constants selected.

The concrete thicknesses required to achieve the applicable dose rate schedule, as defined earlier (Section 11.1.5.1), are shown in Figure 11.1 as a function of reactor pool water radius between the core and the biological shield.

Attenuation of the gamma-rays in an axial direction was calculated to determine the required reactor pool water depth for an operating power level of 10 MW. Calculations were based on the gamma-ray sources as determined in the analytical model. The dose rate as a function of pool water depth over the reactor core is shown in Figure 11.2. It shall be noted that to reach a tolerable dose rate from the direct penetration reactor core gamma-ray contribution in comparison to the pool water activity dose rate, a minimum water depth of 23.6 feet (7.2 m) is required over the fuel region at an operating power level of 10 MW.

Neutron Attenuation

Neutron attenuation in the biological shield was calculated to insure adequate neutron removal. A calculation was performed by the Internuclear Company using a computer to integrate the point water attenuation kernel over the reactor core volume. Correcting this kernel by the exponential attenuation

of the non-hydrogenous materials in the system, the dose rate from fast neutrons is then calculated assuming a dose rate conversion of 0.15 mrem/h per unit neutron flux.

The magnetite concrete thickness required, as a function of reactor pool radius, to achieve a neutron dose rate of 10% of the dose criteria is shown in Figure 11.3. It can be seen that the neutron dose rate is of negligible importance in comparison to the bulk shielding requirements to attenuate the gamma-ray radiation.

11.1.5.1.2 Spent Fuel Transfer and Storage

The transfer and storage of spent fuel elements in the reactor pool were studied in order to limit dose rates during these operations to a reasonable level or, if applicable, to the dose rate criteria as defined in Section 11.1.5.1. During fuel element transfer, the philosophy used in design was to limit the dose rate during the transfer operation to less than 100 mrem/h at some reasonable time after shutdown. The shielding requirements for the storage of spent fuel elements in the reactor pool (main pool or spent fuel storage tank) were calculated to meet the dose rate criteria of the bulk shielding.

The calculations were performed by the Internuclear Company using a computer program. The cases of spent fuel elements, in arrays of one, four, and eight elements adjacent to the biological shield wall and of a single element in a horizontal or vertical position shielded by water were calculated as a function of fission product decay time. All calculations were based on an average fuel element with uniform axial burnup and 400 megawatt days (MWD) of reactor operation at 10 MW.

The dose rates through magnetite concrete shielding from one, four, and eight spent fuel element arrays adjacent to the shielding wall are shown in Figures 11.4, 11.5, and 11.6. In these figures, the results are presented for fission product decay times of 10^3, 10^5, and 10^6 seconds. Figures 11.7 and 11.8 show the dose rate through water shielding from a spent fuel element in a horizontal and vertical position for decay times of 10^3, 10^5, and 10^6 seconds.

Application of these results to the design shield thicknesses assumes that the following operational procedures would be followed:

- Lowering of the water level in the reactor pool to the elevation of the lower bridge prior to the removal of fuel from the reactor pressure vessel; (Water level in the reactor pool is no longer required to be lowered to the elevation of the lower bridge during fuel transfers. The air-operated fuel handling tool was modified and placed in service in 1979, allowing the handling of fuel with the pool water at the normal operating level. This provides greater shielding during fuel handling. The fuel handling tools are described in greater detail in Section 9.2.2.)
- Transferring of the fuel from the core to the storage baskets located at the bottom of the main (deep) pool. However, it is possible to transfer the fuel directly from the core or from these storage baskets to the storage baskets located in the east end of the pool (semi-cylindrical spent fuel storage tank) while the water level is at the refuel level; and

- No transferring of the fuel over the weir into a shipping (transfer) cask at the reduced pool level and no storing of spent fuel on the ledge behind the weir.

The functional shielding requirements can then be defined as follows:

- The minimum concrete thicknesses around stored fuel;
- The minimum elevation of the lowered pool water level to facilitate fuel transfer from the reactor pressure vessel and into storage baskets located in the deep pool;
- The minimum required submergence of the weir and ledge to facilitate fuel transfer to the semi-cylindrical spent fuel storage tank or into a transfer cask; and
- The minimum required depth of water above spent fuel stored in the spent fuel storage tank in the event the deep pool is drained. This minimum depth is maintained by the weir separating the deep pool from the spent fuel storage tank.

For a fission product decay time of 10^5 seconds (1.16 days), the shielding requirements for the described conditions are as follows:

- Storage of Fuel Elements Adjacent to the Primary Reactor Shield:
    - Storage of four spent fuel elements adjacent to the primary reactor shield or in the spent fuel storage tank with a dose rate criterion of "25 mrem/h at one foot (0.3 m) from the shield surface" requires 52 inches (1.3 m) of magnetite concrete; and
    - Storage of eight spent fuel elements in the spent fuel storage tank adjacent to the dividing wall between the deep pool and the storage tank with a dose rate criterion of "50 mrem/h for a worker in the deep pool" requires that the dividing wall have a magnetite concrete thickness of 47 inches (1.2 m);
- Transfer Operations at a Lowered Water Level in the Reactor Pool:
    - The minimum water shielding depth over the reactor pressure vessel top flange during the transfer of a spent fuel element in a vertical position with a dose rate criterion of "100 mrem/h at the water surface" is 11 feet (3.4 m) [this includes a 12-inch (0.3 m) clearance of the fuel element over the pressure vessel top flange]; and
    - The minimum water shielding depth over the fuel storage baskets in the spent fuel storage tank with seven elements in the storage baskets and one element in transfer, or eight elements in the storage baskets with a dose rate criterion of "10 mrem/h from all elements during a single element transfer, and the dose rate with 8 elements in the storage baskets not to exceed 1.0 mrem/h" requires 12.5 feet (3.8 m) of water [this includes a 12-inch (0.3-m) clearance of the fuel element being transferred into the storage baskets];
- Transfer Operations at Normal Operating Level in the Reactor Pool:
    - The minimum water shielding depth over the pool weir, or over the shipping cask, during transfer of a spent fuel element from the fuel storage baskets in the deep pool to the shipping cask, or to the spent fuel storage tank with a dose rate

criterion of “1.0 mrem/h from a single fuel element” requires 14 feet (4.3 m) of water over the weir or over the shipping cask [this includes a 12-inch (0.3-m) clearance of the fuel element being transferred over the weir]; and

- Reactor Pool Water Level Lowered to the Pool Weir:
    - The minimum water shielding depth over the spent fuel storage tank with 8 fuel elements in the fuel storage baskets with a dose rate criterion of “50 mrem/h at the water surface” is 10 feet (3 m) over the baskets.

11.1.5.1.3 Experimental Facilities

This section describes the initial analyses performed to determine the shielding criteria for the beamports, thermal column, and the spent fuel gamma irradiation facility. The beamport analysis included a determination of supplementary shielding requirements for the beamport plug and vestibule as necessitated by the loss of concrete created by the vestibule, by the radiation streaming around the beamport plug, and the induced activity of the beamport coolant. All calculations were based on an operating power level of 10 MW.

The shielding requirements for the experimental facilities were based on the radiation dose rate schedule provided earlier (Section 11.1.5.1). The radiation level criterion of “2.5 mrem/h at one foot (0.3 m) from the surface of the primary reactor shield” was used as the design basis in determining shielding for the beamports and the thermal column. In the shielding analysis for the beamports, the basis was a typical 6-inch beamport. Since the primary reactor shield is designed to compensate for the eccentricity of location of the core in the reactor pool, the analysis of beamport shielding was required on only one beamport. The shielding analysis for the thermal column is based on a 60-inch (1.5-m) thick graphite stack arrangement along the axis of the column followed by the lead shield nose piece.

Beamport Shielding

Supplementary shielding for the beamports is based on material replacement necessary to maintain the primary reactor shield integrity. In the calculations used to establish the supplementary shielding requirements, the following areas were studied:

- Annuli (water or air filled, normally, the volume in front of the shield plug is filled with either helium or demineralized water) between the primary reactor shield casing (fixed beamport liner), the beam tube (removable beamport liner), and the beamport shield plug; (The original design of a beamport assembly utilized a shield plug to provide the pressure boundary in order to fill a beam tube. A removable collimator liner is presently used (Section 10.3.2.2).)
- Annuli of materials (primary reactor shield casing, beam tube, collimator liner) which have densities less than the primary reactor shield design density of 3.5 grams/cm;
- The beamport vestibule at the primary reactor shield surface; and
- The coolant activation in the beam tube and/or experiment can and the subsequent passage to the outer portion of the reactor shield. (Experiment cans are no longer utilized in the beamport assemblies.)

The desired radiation level criteria, as defined in Section 11.1.5.1, are satisfied in the beamport design by stepping the aluminum and water or air annuli of the beamport system and including lead shielding.

Calculations were performed on the following annular geometries:

- An air or water annulus of varying thickness followed by lead and concrete shielding or vice versa;
- A homogenized annulus of aluminum and air or water of varying thicknesses followed by lead and concrete shielding or vice versa;
- An air or water annulus of varying thickness traversing the entire primary reactor shield; and
- A homogenized annulus of aluminum and water or air traversing the entire primary reactor shield.

From the analysis of the various contributions from the combination of annuli which exists in the design, the amount of lead required at the step was established and the required step was sized.

It should be noted that, since a collimated beam of gamma-rays is studied, no gamma-ray buildup was included in the calculations to account for scattering.

For the beamport assembly design as shown in Figure 10.3, the thickness of the aluminum and water or air did not exceed 7/8 of an inch (22.2 mm) and any single annular thickness of water or air did not exceed 1/16 inch (1.6 mm), thereby providing the lead shielding and step requirements as follows:

- At the step, a minimum of 3 inches (7.6 cm) of lead in the “line of sight” is required for each annular channel; and
- A minimum step of 2 inches (5.1 cm) is required with the inclusion of the above lead shielding. (The step is defined as the distance from the inner radius of the smaller annulus of the step to the outer radius of the larger annulus or the radial increment from the inner magnetite concrete radius to the outer concrete radius at the step.)

The additional considerations of the vestibule and the coolant activation were solved using standard techniques. The vestibule represents a deficiency of concrete which must be supplemented by additional lead. A minimum thickness of 3 inches (7.6 cm) of lead has been provided based on this lack of concrete and the relative material densities. This thickness of lead backs up the entire vestibule and overlaps the edges of the vestibule by a minimum of 1 inch (2.5 cm). The beamport plug requires a similar minimum thickness of lead.

The activation of the experiment coolant was based on a light water cooled experiment with a total flowrate of 2 gpm (7.6 lpm) to the experiment can or the flooded volume. Consideration was given to the transient times in the irradiation region, the mixing of the activity in the beam tube, and to shielding of the drain lines from the beamport. For those lines carrying the activated coolant, a minimum of 2 feet (0.6 m) of magnetite concrete has been provided between the lines and the surface of the shield.

Thermal Column Shielding

The thermal column penetrates the biological shield, extends into the reactor pool, and terminates at a removable lead shield positioned between the reflector region (reflector tank outer wall) and the terminal end of the thermal column. The lead shield, formed to fit the reflector tank, is attached to the front face of the thermal column to attenuate the core gamma radiation. The front face of the thermal column housing is shaped to match the contour of the lead shield. The minimum thickness of the lead shield is 4 inches (10.2 cm) while the edges are 6 inches (15.2 cm). The whole is encased in aluminum.

Minimum shielding thickness occurs at the centerline of the thermal column. Table 11-12 describes the materials and thicknesses of the thermal column as traversed from the inner face of the gamma shield to the external face of the thermal column door.

The thermal column case is constructed of two square boxes. The in-pool portion immediately behind the lead shield is a 3 foot 1.5-inch (95.3-cm) square by 12.25-inch (31.1- cm) deep box where it is stepped to the next box. The second box is 4 feet 2 inches (127 cm) square by 5 feet 8 inches (172.7 cm) deep, completely lined with 0.25-inch (0.635-cm) thick BORAL sheeting. The analysis of the shielding requirements was made by the Internuclear Company using the computer codes GH-4 and GRACE-I. Calculations performed with the GH-4 code utilized an approximation of the thermal column geometry of successive cylindrical annular segments to describe the lead, graphite, and shield door of the column. Calculations using GRACE-I, which is a multi-region, multi-group gamma-ray attenuation program utilizing slab geometry with the option of truncated cone geometry, were performed by describing the regions as truncated cones.

The off-axis shielding of the column and the streaming of radiation down the gaps around the door were also studied. The off-axis and streaming shielding is provided by a door overlap of 5 inches (12.7 cm) on all four sides of the 4-foot (122-cm) square thermal column face.

TABLE 11-12
THERMAL COLUMN SHIELDING

Description	**Material**	**Thickness in Inches**
Gamma Shield Face Plate	Aluminum	0.25 (0.635 cm)
Gamma Shield	Lead	4.00 (10.16 cm)
Gamma Shield Back Plate	Aluminum	0.25 (0.635 cm)
Thermal Column Face Plate	Aluminum	0.75 (1.905 cm)
Thermal Column Stack (full)	Graphite	60.00 (152.4 cm)
Thermal Column Door Face Plate	BORAL®	0.25 (0.635 cm)
Thermal Column Door[a]	Magnetite Concrete	(174.6 cm)

a. The original thermal column door was constructed of steel plating [25-inch (63.5 cm) total thickness]. A design modification was performed in 1977 which increased the size of the door, providing increased shielding and experiment flexibility.

Spent Fuel Element Irradiation Facility Shielding

The capability of using an array of spent fuel elements as a gamma radiation source has been installed

but is not currently in use. A section of shielding in the wall of the semi-cylindrical spent fuel storage tank is removable and may be replaced with an irradiation unit fabricated of lead.

The spent fuel element gamma irradiation facility consists of a cavity in the biological shield wall of the spent fuel storage tank. The cavity is comprised of three boxes which are fabricated of aluminum. The innermost box is 2 feet (61 cm) square by 20 inches (51 cm) deep. This box steps to the second box which is 2 feet 4 inches (71 cm) square and 10.5 inches (27 cm) deep. The third (outer) box is 2 feet 8 inches (81 cm) square and 2 feet (61 cm) deep.

Presently, this cavity is filled with cast blocks of magnetite concrete. The blocks are of two sizes: 4 × 4 × 10 inches (10.2 × 10.2 × 25.4 cm) and 4 × 4 × 10.5 inches (10.2 × 10.2 × 26.7 cm). The concrete blocks are positioned in the cavity in staggered rows to minimize gap lengths. When the cavity is filled with blocks, the effective shielding is 54.5 inches (138.4 cm) of magnetite concrete. The calculated dose rate external to the biological shield with six spent fuel elements in the storage rack is 1.4 mrem/h for elements subjected to 400-MWD reactor operation and 10^5-second fission product decay time.

11.1.5.1.4 Primary and Pool Coolant Systems

Piping and the Mechanical Equipment Room

The shielding requirements for the primary and pool coolant systems' piping and the mechanical equipment room (Room 114) are based entirely on the N-16 activity in the two coolant systems. N-16, 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. Required thicknesses of the primary reactor shield were calculated on the basis of concrete densities of 2.2 grams/cm^3 for ordinary concrete and the radiation level criteria as defined in Section 11.1.5.1. The calculated equilibrium specific activities of N-16 are 6.6×10^6 and 1.54×10^7 MeV/cm^2 at the exit of the reactor core activation areas of the primary and pool coolant systems, respectively.

The initial shielding calculations were based on the locations of the invert loop in the reactor pool and the piping in the Room 114 tunnel. Decay time, transient times, and source geometries were based on the 12-inch (30.48-cm) primary coolant piping, the 6-inch (15.24-cm) pool coolant piping, and the design system flow rates.

The calculations for Room 114 assume that one inlet pipe is located at any point along the wall or ceiling. The shielding requirements for a heat exchanger room, such as Room 114, are generally determined by such a critical pipe location, since the radiation dose rate from this geometry exceeds that from the distributed components in the room. This assumption provides the most conservative estimate of shielding requirements. The resulting shielding requirements are as follows:

- Invert Loop:
    - The invert loop is located in the reactor pool adjacent to the primary reactor shield;
    - The radiation level criterion is "2.5 mrem/h at one foot (0.3 m) from the shield surface;" and

- The shielding requirement is “3.5 feet (1.1 m) of magnetite concrete or its equivalent in all directions from the pipe section.”

- Mechanical Equipment Room (Room 114) Pipe Tunnel:
  - The pipe tunnel referred to is located beneath the reactor containment building beamport floor (below grade level) and runs from the reactor pool to Room 114;
  - The radiation level criterion is “2.0 mrem/h at one foot (0.3 m) from the shield surface;” and
  - The shielding requirement is “5 feet (1.5 m) of ordinary concrete or its equivalent.”

- Mechanical Equipment Room:
  - Room 114 is located below grade level and adjacent to the reactor containment building. The areas of concern are the walls and ceiling. An occupied area is located immediately above the room;
  - The radiation level criterion is “2.5 mrem/h at one foot (0.3 m) from the shield surface;” and
  - The shielding requirement is “5 feet (1.5 m) of ordinary concrete or its equivalent.”

Demineralizer System

The shielding requirements for the demineralizer system are reduced by the use of hold-up tanks which hold up, or delay, the primary and pool coolant within Room 114 for a sufficient amount of time to allow short-lived activity to decay, avoiding the need to shield N-16 gamma radiation (Section 5.7). The shielding requirements are, therefore, determined by the specific activities deposited on the demineralizer beds. The coolant lines supplying the demineralizer beds, however, still have appreciable oxygen-19 (O-19) activity which influences the shielding requirements for these lines.

Separate demineralizer tanks are used for the primary and pool coolant systems. However, these tanks are interchangeable and the shielding requirements are identical for all three tanks (including the spare unit). Similarly, the regeneration station for the depleted resin beds has identical requirements because it can receive the activity from any one tank.

The equilibrium activities present in the demineralizer beds during reactor operation have been calculated. Based on the specific activities calculated and on a minimum ordinary concrete density of 2.2 grams/cm^3, the shielding requirements for the demineralizer system are as follows:

- Demineralizer System Piping:
  - The piping extends from the primary and pool coolant systems in Room 114 to the demineralizer tanks;
  - The radiation level criterion is “2.5 mrem/h at one foot (0.3 m) from the shield surface;” and
  - The shielding requirement is “one foot (0.3 m) of ordinary concrete or its equivalent.”

- Demineralizer Tanks:

- The demineralizer tanks are located in cells adjacent to Room 114. The calculation of shielding requirements was based on occupied areas above and readily accessible areas to the front of the demineralizer cells;
- The radiation level criterion is "2.5 mrem/h at one foot (0.3 m) from the shield surface;" and
- The shielding requirement is "3 feet (0.9 m) of ordinary concrete or its equivalent in those directions where access is provided during operation of the units." The assumption is made that a sufficient delay time has been built into the system to permit N-16 and O-19 activities to decay.

- Demineralizer Piping:
  - The location of the resin handling and dump station piping is in the valve tunnel adjacent to and in from of the demineralizer cells. These lines are used to sluice the resin beds from the demineralizer tanks to the handling and dump station.
  - The radiation level criterion is "2.5 mrem/h at one foot (0.3 m) from the shield surface;" and
  - The shielding requirement is "2 feet (0.6 m) of ordinary concrete or its equivalent."
- Radioactive Liquid Waste Tanks:
  - Location of the radioactive liquid waste tanks is immediately to the west of the demineralizer cells. The area of concern is located above the waste tanks where there are occupied areas;
  - The radiation level criterion is "2.5 mrem/h at one foot (0.3 m) from the shield surface;" and
  - The shielding requirement above and to the sides of the waste tanks is "2 feet (0.6 m) of ordinary concrete or its equivalent."

11.1.5.2 Ventilation System

The ventilation system for the reactor facility is described in detail in Section 9.1. Here follows a discussion of the design features that are incorporated into the ventilation system for the purpose of radiation protection.

The ventilation system provides the necessary air exchanges within the laboratory and reactor containment buildings, ensuring that concentrations of airborne radionuclides are maintained at levels below the 10 CFR 20 limits for occupational exposure.

The ventilation system maintains the reactor containment and laboratory buildings at a slightly negative pressure with respect to the surrounding environment to prevent the spread of radioactive contamination.

The ventilation system ensures that maximum dilution of potentially contaminated air is attained, resulting in minimum concentrations of radioactive gases being released to the environment.

Exhaust air from areas in the reactor containment building that produce radioactive gases or airborne contamination is ducted directly to the exhaust plenum via a 16-inch (40.64-cm) line. These areas include the reactor pool sweep system, the beamport storage ports (hot storage ports), the thermal column, and the beamport vestibules.

Air flow through all laboratory fume hoods and registers exhausts through stainless steel filter housings containing banks of pre-filters and HEPA filters.

The mechanical equipment room's (Room 114) exhaust system contains a pre-filter, two HEPA filters, and two activated charcoal filters.

The p-tube system exhausts through a HEPA filter. The system is designed to minimize the instantaneous release of Ar-41 that is produced in the terminus of the sample carrier tubing by ensuring that a continuous flow path for air exists through the carrier tubing even when the p-tube system is secured. A time delay circuit starts the second p-tube blower approximately 15 seconds after the first p-tube blower, minimizing air surges through the system.

The ventilation system also prevents the uncontrolled release of radioactive materials to the environment in the event of an accident. This design feature is discussed in detail in Chapter 6, "Engineered Safety Features."

11.1.5.3 Containment

The containment of radioactivity within the reactor facility is primarily a concern with respect to the sample materials being irradiated in the various experimental facilities and with the reactor fuel. Containment of radionuclides generated during the use of the experimental facilities is achieved through strict encapsulation procedures for samples and strict limits on what materials can be irradiated, as described in Chapter 10, "Experimental Facilities and Utilization" and in Operating Procedures. Containment of fission products in the fuel elements is achieved by maintaining the integrity of the fuel's aluminum cladding, which is accomplished by operating the reactor within the Safety Limits developed for MURR. Operation within the Safety Limits will prevent fuel plate meltdown or cladding damage resulting from the departure from nucleate boiling (DNB).

To further improve containment and minimize the potential release of radioactivity from samples irradiated in the experimental facilities, sample cans or capsules are only opened in laboratory fume hoods or hot cells. The fume hoods and hot cells, which exhaust through HEPA filters, have an inflow of air to prevent the release of radioactivity to the surrounding area.

In addition, the Containment System, an engineered safety feature, 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. The Containment System is discussed in detail in Chapter 6, "Engineered Safety Features."

11.1.5.4 Entry Control

There are three main areas within the reactor facility which require entry control during operation of the reactor in order to meet the requirements of 10 CFR 20, Subpart G for limiting access into high or very high radiation areas. Access to the mechanical equipment room (Room 114), the demineralizer cell area (Rooms 115, 120, and 121), and the beamport area (Room 101) is controlled according to the following descriptions.

Mechanical Equipment Room

Entry control to the mechanical equipment room is because it is a high radiation area during operation of the reactor. This is due primarily to the N-16 activity in the primary and pool coolant systems. A locked gate is located at the point of entry into the area which energizes an audible and visual alarm locally and in the reactor control room when opened. This barrier satisfies requirements of 10 CFR 20.1601(a)(2) and 10 CFR 20.1601(a)(3) for controlling access to a high radiation area.

Demineralizer Cell Area

Entry control to Rooms 115, 120, and 121 is because these rooms are high radiation areas during operation of the reactor. This is due primarily to the O-19 activity in the primary and pool coolant lines supplying the demineralizer tanks and the buildup of activation products (mainly Na-24 and Co-60) in the resin beds. This location may also be used as a storage area for radioactive material, creating the potential for a high radiation area even when the reactor is in a shutdown condition. Two locked gates are located at the entry points into the area which cause an audible and visual alarm locally and in the reactor control room when opened. These barriers satisfy requirements of 10 CFR 20.1601(a)(2) and 10 CFR 20.1601(a)(3) for controlling access to a high radiation area.

Beamport Area

Entry control to the beamport area is because, when a neutron beam is accessed from any one of the six beamports, the open beam is considered a high radiation area. All individuals granted access to the beamport area are made aware that these beams are sources of potentially dangerous radiation. It is not possible to completely enclose the neutron beams due to research considerations, and, therefore, it is the responsibility of the individuals having access to the area to use proper established procedures for work with neutron beam instruments and around the beam paths. The path of an open neutron or gamma beam is clearly marked by a set of yellow plastic streamers. The streamer is a highly visible indication of the presence of the beam. The only high radiation areas that are permitted under normal operation in the beamport area are directly in the beams. All other areas accessible to personnel are limited to being radiation areas as defined by 10 CFR 20. A locked gate is located at the entrance to the beamport area. This barrier satisfies the requirement of 10 CFR 20.1601(a)(3) for controlling access to a high radiation area.

In addition to the three areas mentioned above, other locations within the facility may also require access control as specified in 10 CFR 20, irrespective of the status of the reactor. These locations typically include, but are not limited to, radioactive material storage (waste tank room, isotope closet, etc.) and sample processing areas (hot cells) where radiation levels can exceed those levels defined for a

high or very high radiation area. Entryways into these areas will be locked, except during periods when access to the area is required, with positive control over each entry.

11.1.5.5 Protective Equipment

Typical protective equipment and related materials used in the MURR RPP are summarized in Table 11-13. While not intended to be all inclusive, this table provides an indication of the types of protective equipment and materials used during routine and emergency operations at the reactor facility.

MURR maintains two Emergency Lockers which contain emergency equipment and supplies required by the Site Emergency Procedures. The primary Emergency Locker is located at a readily accessible location within the reactor facility while the backup locker is located at the University Police Headquarters. In addition to some of the protective equipment listed in Table 11-13, the Emergency Lockers normally contain flashlights, a first-aid kit, swipes, absorbent paper, assorted plastic bags, yellow and magenta rope, etc. These lockers are periodically audited to verify that the contents meet, at a minimum, the required levels stated on the inventory checklists.

TABLE 11-13
PROTECTIVE EQUIPMENT FOR RADIATION PROTECTION PROGRAM

Equipment	**Use[a]**
Coveralls	R & E
Lab Coats	R & E
Rubber Gloves	R & E
Chemical Resistant Gloves	R & E
Safety Glasses	R & E
Hoods/Caps	R & E
Plastic Shoe Covers	R & E
Rubber Overshoes	R & E
Decontamination Sink	R & E
Decontamination Showers	E

a. Use: R = Routine; E = Emergency

11.1.5.5.1 Respiratory Protection Equipment

Other than Ar-41, no airborne radioactivity is expected to occur at the reactor facility as part of its normal operation. Consequently, respiratory equipment is not part of the protective equipment typically used at MURR. However, in special instances, after reasonable engineering and process controls have been exhausted and the atmosphere does not constitute as immediately dangerous to life and health, MURR has provided for the use of respirators. Policy POL-25, “MURR Respiratory Protection Program,” is designed to protect personnel from radiological and non-radiological hazards.

11.1.5.5.2 Personnel Dosimetry Devices

Personnel dosimetry devices have been selected to provide monitoring of all radiation types likely to be encountered. Table 11-14 provides a summary of the dosimetry devices typically used at MURR.

TABLE 11-14
PERSONNEL DOSIMETRY DEVICES

Device	Radiation Measured	Dose
Optically Stimulated Luminescence (OSL) Dosimeter (with track-etch option)	X-Ray, Beta, Gamma, and Neutron (when required)	Deep-Dose Equivalent Eye Dose Equivalent Shallow-Dose Equivalent
Ring Badge (TLD)	X-Ray, Beta, and Gamma	Shallow-Dose Equivalent, Extremity
Electronic Personal Dosimeters (EPD)	X-Ray and Gamma	Deep-Dose Equivalent

In accordance with the requirements of 10 CFR 20, personnel who may receive significant external exposures shall wear a dosimetry device. Individuals who are assigned personnel dosimetry devices are instructed as to when, where, and how to properly wear their assigned dosimetry. EPDs are recalibrated annually by a service provider and other dosimetry devices are exchanged at regular time intervals. The dosimetry devices are processed by a contractor and the results are returned to the Reactor HP Staff for review and compilation. Any contractor employed by the University shall hold a current personnel dosimetry accreditation from the National Voluntary Laboratory Accreditation Program (NVLAP).

Occupational exposure records are maintained by the Reactor HP Staff and are retained for the life of the facility. In addition, RWPs which document radiation exposures received during a radiological operation are also maintained by the Reactor HP Staff and are retained for a period of at least three years. It is the policy of the facility that individuals who enter the restricted area beyond the front lobby, offices, or the loading dock areas of MURR will be monitored for external radiation exposure.

Administrative investigation levels for occupational radiation exposures have been established which, when exceeded, initiate a review or investigation by the Reactor HP Staff. The investigation or review is focused on determining the cause of the exposure so that appropriate ALARA actions, if any, can be applied. This is part of the MURR ALARA program described in Section 11.1.3.

Since there are no normal routine operations at the reactor facility that would result in the potential for the internal deposition of radionuclides, internal dosimetry is not a routine personnel dosimetry consideration. Nonetheless, monitoring by urine sampling for tritium uptake is periodically performed on individuals frequenting the reactor containment building. The tritium bioassay is performed to verify that individual monitoring as described in 10 CFR 20.1502 is not required.

The one exception for the need for routine internal dosimetry is for the individuals involved with the processing of I-131. The MURR Iodine Bioassay Program is designed to measure and monitor occupational exposure from radioactive iodine form licensed radiation sources. This program ensures compliance with 10 CFR 20.1204, as well as other subparts within Part 20 and the MURR Reactor License.

Based upon the ALARA dose trend analysis charts maintained by the Reactor HP Staff, there has been a relatively downward trend in total person-rem over the five-year period, 2020-2024 (84.7% of the total

person-rem for the previous five-year period, 2015-2019). The average annual occupational whole-body exposure (deep dose equivalent) during the period 2020 through 2024 for a Reactor Operations and a Reactor Health Physics Staff member is approximately 567 mrem/y (assuming an average Reactor Operations staff of 25 persons) and 178 mrem/y (assuming an average HP staff of 25 persons), respectively.

11.1.5.6 Expected Annual Radiation Exposure

The guidelines for radiation doses and for airborne concentrations of radionuclides during normal operation of the reactor facility are contained in 10 CFR 20. These guidelines establish levels for both "restricted" and "unrestricted" areas. The MURR "restricted" area is considered to be all locations within the operations boundary (as defined in Section 2.1.2). The "unrestricted" area includes all locations and personnel outside the operations boundary. The following sections contain an estimate of annual radiation exposure in these two areas.

11.1.5.6.1 Restricted Area

Although MURR operates 24 hours a day, 6.5 days per week, 52 weeks per year, it is assumed that an individual working at MURR will be in the facility only one shift per day (40 hours per week). It is further assumed that an occupationally exposed individual will spend only a limited amount of time in areas where there is potential for significant radiation levels (within the demineralizer cells, the waste tank room, or in the mechanical equipment room). Predicted occupational doses are based on an estimate of the actual time an individual will spend in areas where there are measurable radiation levels. Also, radiation surveys of the reactor facility within the "restricted" area are repeatedly performed and there is a great deal of actual personnel dosimetry data (e.g., ALARA dose trend analysis charts) available to use as a basis for future dose estimates. Where radiation dose rate measurements and actual personnel doses are available, they are included in the following discussions.

With the reactor operating, radiation levels on the below-grade level of the reactor containment building (the beamport floor) vary significantly from location to location depending upon the status of the beamports and/or the type of experiments being conducted. Due to the potential radiation hazards associated with a beamport, changes to the physical arrangement of instruments, beam stops, and shielding, which can potentially alter area radiation levels, are not allowed without prior approval of the Reactor HP Staff and Reactor Operations Staff. The dosimetry currently utilized for area radiation monitoring within MURR are Optically Stimulated Luminescence (OSL) Dosimeters provided by an NVLAP accredited contractor.

Table 11-15 shows the OSL dosimeter recorded annual dose measurements as well as the monthly average at certain locations in the containment building for the period 2020 through 2024. Using the location which has the highest monthly average as a conservative estimate, the calculated average radiation level on the beam port floor with the reactor operating at 10 MW is approximately 0.4 mrem/h (280 mrem/month divided by 30.5 days/month divided by 24 hours/day). Past exposure history

on research personnel who utilize the beamport experimental facilities shows little recorded dose each month.

TABLE 11-15
DOSE MEASUREMENTS BY FILM BADGE AT SELECTED LOCATIONS WITHIN THE REACTOR CONTAINMENT BUILDING (IN MREM)

Location	Total for Year					Monthly Average
	2020	2021	2022	2023	2024	
North Beamport Floor	5,678	4,843	2,279	1,993	2,003	280
South Beamport Floor	5,198	4,615	2,125	891	1,208	234
East Beamport Floor	3,932	2,618	875	549	628	143
West Beamport Floor	6,516	4,753	1,878	1,861	1,656	278
Upper Bridge Area	9,654	8,329	6,862	7,812	7,529	670
Reactor Control Room	1,309	1,094	579	558	552	68

Reactor operators spend a large portion of their time in the reactor control room and on the upper bridge where OSL dosimeters placed at these locations routinely record average radiation doses on the order of 68 mrem/month and 670 mrem/month, respectively. In addition, operators perform maintenance tasks in the mechanical equipment room (Room 114) and on the lower bridge when the reactor is in a shutdown condition. General area radiation levels at these two locations are typically 20 mrem/h and 10 mrem/h, respectively. Entry into Room 114 is normally not allowed while the reactor is operating, and consequently, personnel exposures from activity in this area should be insignificant.

With the exception of loading a new resin bed or dumping a depleted resin bed, maintenance of equipment located in the demineralizer cell area (Rooms 115, 120, and 121) is typically performed while the reactor is in a shutdown condition. This location is also used as a potential storage area for radioactive material. However, personnel doses from these storage areas are expected to be low because they are isolated and posted and seldom entered. Radiation levels in other areas of these rooms are considerably lower (10 to 50 mrem/h). In addition, personnel doses from work performed in the demineralizer cell area are typically documented on a RWP.

Handling and inspection of irradiated fuel is performed in the reactor pool. Pool water is maintained at a level which provides maximum shielding for the operators. Removing or replacing fuel elements, either in the reactor core or in the spent fuel storage baskets, requires that the element be raised in the vertical direction high enough to clear the weir divider or the top of the pressure vessel. The highest dose rate during fuel handling is approximately 25 mrem/h and occurs during the short interval of time when an element is passed into the weir area, or when an element is raised to clear the top of the pressure vessel. Using the air-operated fuel handling tool, the fuel element at its highest point is still covered by approximately 14 feet (4.7) of water, including a 12-inch (0.3-m) clearance when a fuel element is being transferred over the weir divider.

The removal of irradiated fuel elements from the reactor pool is accomplished using an NRC-approved transfer cask. The transfer cask is lowered into the reactor pool and placed on the shelf behind the weir

divider. An irradiated fuel element is then removed from a storage location and placed into the transfer cask. For this operation, there will be about 11 feet (3.7 m) of water between the operator and the fuel element. Although the radiation level could be as high as 15 mrem/h, the radiation dose to an operator will be insignificant since the time required for the operation is normally less than one minute per element.

General area radiation levels in the laboratories are typically less than 0.5 mrem/h.

While the above predictions provide some indication of what the annual Total Effective Dose Equivalent (TEDE) might be for occupationally exposed individuals at MURR, the facility has measured staff personnel radiation doses for many years. Using this data to predict annual doses provides even more assurance that the personnel doses will remain low. Table 11-16 shows the total rem by group from 2020 through 2024. Table 11-17 shows the average monthly dose for each group during the same reporting period.

TABLE 11-16
REM BY GROUP AT MURR

Group Name	**Year**				
	2020	**2021**	**2022**	**2023**	**2024**
HP / Shipping	3.81	3.92	4.42	4.85	5.34
Irradiations/Hot Cell	1.67	2.04	1.92	1.85	2.09
Production	1.90	2.59	2.05	4.17	4.29
Reactor Operations	12.07	12.09	16.80	12.78	17.13

TABLE 11-17
AVERAGE MONTHLY RADIATION DOSE BY GROUP AT MURR (IN MREM/Y)

Group Name	**Year**				
	2020	**2021**	**2022**	**2023**	**2024**
HP/Shipping	317	326	368	404	445
Irradiations/Hot Cell	140	170	160	154	174
Production	158	216	171	348	358
Reactor Operations	1,006	1,008	1,400	1,065	1,428

Notes:

1. The groups represented in Tables 11-16 and 11-17 are the top four groups/combined groups, in terms of total dose, of individuals badged at MURR and collectively represent over 86% of the total facility dose received at MURR over the period.
2. Reactor Operations and HP 2022 and 2024 data includes doses incurred during deep pool maintenance
3. 2023 and 2024 data reflect combined doses for Production and QA

The dose limit for an embryo/fetus of a declared pregnant worker over the entire course of the pregnancy is stated in 10 CFR 20. If a worker at MURR declares their pregnancy, in most cases they may continue their job with no changes and still meet the NRC's limit for exposure to the embryo/fetus. In

some instances, it may be necessary to change some or all of the job responsibilities during the pregnancy in order to meet regulatory limits.

Other categories of individuals who might receive exposure at MURR include research and service personnel, students, and visitors. Past exposure history on these groups shows little or no recorded dose and there does not appear to be any reason to expect this situation to change.

An individual may be authorized to receive a Controlled Special Exposure (non-emergency) if the anticipated radiation exposure from a work project exceeds 10% of any limit stated in 10 CFR 20. A Controlled Special Exposure Authorization Form is prepared by the Reactor HP Staff which documents authorization of the exposure by MURR management and the approved dose limit. A RWP is also prepared in conjunction with the authorization form in order to establish adequate control of any potential radiation hazards.

Emergency Exposures may be authorized for individuals who voluntarily expect to receive radiation exposures in excess of the 10 CFR 20 limits during a site emergency. The exposure limits for saving a life or for any action which prevents exposure to members of the general public in excess of the Protective Action Guides (PAGs) of 1 rem dose equivalent-whole body and 5 rem dose equivalent-thyroid are stated in the MURR Emergency Plan Implementing Procedures. An Emergency Exposure Authorization and Record Form documents authorization by the Emergency Director along with the approved limits.

11.1.5.6.2 Unrestricted Area

A detailed discussion of the estimated annual Total Effective Dose Equivalent (TEDE) in the unrestricted area from Ar-41 production during normal operation of MURR is contained in Section 11.1.1.1.3 and in Appendix B, "Radiological Impact of Ar-41 During Normal Operations." Using the nearest occupied facilities and most probable wind direction, the annual dose values for the unrestricted area indicate a maximum TEDE, primarily from Ar-41, at two different distances: 150 meters north [at the Emergency Planning Zone (EPZ) boundary] and at the nearest residence in relation to the facility (approximately 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 EPZ is discussed in Section 2.1.2. These values are not expected to increase.

11.1.6 Contamination Control

Radioactive contamination is controlled at MURR by trained personnel using written procedures for the proper handling of radioactive material and by maintaining a monitoring program designed to detect and identify loose and fixed surface contamination in a timely manner. The monitoring program is described in Section 11.1.4. In addition to the monitoring program, the following items are also a part of the overall approach taken for controlling contamination at the facility.

For work being performed in areas of the reactor facility that are known or are considered likely to be contaminated, a detailed written procedure or a RWP shall provide the necessary instructions for

contamination control. Each individual performing work under a RWP shall be informed of the required contamination controls, shall review the RWP, and shall acknowledge such by initialing the form.

After working in a contaminated area, personnel are required to perform surveys to ensure that no contamination is present on exposed skin, clothing, shoes, etc. before leaving the work location. Additional contamination monitoring shall be performed when deemed necessary by Reactor HP personnel.

Approval of a work area to ensure that proper facilities, equipment, procedures, and controls needed to conduct the work are established before work begins is required. Adequate control of contamination is dependent upon proper work area design. This includes the proper location of fume hoods and glove boxes, the proper layout of work and counting areas, and the use of appropriate construction materials.

Anti-contamination (Anti-C) clothing designed to protect personnel against removable contamination is used as appropriate. Normally, Anti-C clothing is specified in a written procedure or in a RWP. Anti-C clothing is typically laundered after each use or disposed of as solid radioactive waste.

All contaminated equipment or areas of surface contamination shall be properly labeled. Additionally, items containing radioactive materials and/or their containers are labeled as containing radioactive material. As appropriate, such labels will normally identify the major isotope(s), form(s), quantity(ies), dose rate, and date.

After handling material in the reactor pool, personnel are required to scan, or “frisk,” their hands after exiting the upper (operating) bridge. An additional frisking station is located in the containment building lobby and used frequently even during periods of routine evolutions, e.g., sample handling, removal of samples from the reactor pool, etc.

Written procedures address the proper handling of personnel who have become contaminated. Contamination events are documented on a Report of Personnel Contamination form. These reports provide a permanent record of all such events and can be used in determining a corrective action to avoid future incidents of unplanned personnel contamination. Reports of Personnel Contamination are maintained by the Reactor HP Staff and are retained for the life of the facility.

Encapsulation requirements for samples likely to cause contamination during or after irradiation are described in Chapter 10, “Experimental Facilities and Utilization,” and in Operating Procedures.

All personnel and visitors entering the reactor facility are trained in radiation protection to a level sufficient for their work/visit or shall be under the constant escort of an individual who has received such training. This training includes the risks of contamination and techniques for avoiding, limiting, and controlling contamination.

11.1.7 Environmental Monitoring

The MURR Environmental Monitoring Program has continued from initial operation in October 1966 to the present day in order to determine if operation of the reactor facility is contributing to any increase in environmental radioactivity. No particulate, gaseous or liquid effluents from the reactor or its ancillary facilities will be released to the environment with an average radioactive content greater than the limits specified by 10 CFR 20, Appendix B, unless specifically documented in the Technical Specifications. Effluents are monitored before release to ensure that levels of radioactivity are less than these limits. Thus, any normal operating mode of the reactor facility will not produce an increase of radioactivity in the environment.

A series of pre-operational environmental samples were collected during the period from September 1965 to July 1966 in order to measure and establish baseline values for the level of ambient radioactivity in the vicinity of the reactor facility. The locations of the sampling stations are shown in Figure 11.9 and Figure 11.10. The radioactive content of the environment was determined by analysis of alpha, beta, and gamma-ray emissions from samples of grass, soil, water, and air. In addition, the radiation dose rate due to background gamma radiation was measured. Slight changes have been made over the course of time in the locations of several sampling stations in order to accommodate changes in the terrain resulting from weathering and construction. In addition to the routine sampling, occasional nonroutine samples are also collected and analyzed as specified by the Reactor HP Manager.

While many different types of samples have been collected and analyzed to date, there has been no statistically valid indication that the environmental samples differ significantly from the pre-operational data acquired for this program. Averages and range of values per sample type are comparable. Gamma-ray spectroscopy of water and air samples has not shown any radioactivity that might have been released in the effluents discharged from the facility as a result of its operation. This result is consistent with the expectations for a facility of this type.

The current environmental monitoring program consists of the following basic components:

- Soil samples obtained semi-annually at eight (8) locations (sample stations 1-7, and 10) with typical sensitivity based on average minimum detectable activity from 2020 through 2024 for gamma emitters ~0.70 pCi/gm;
- Vegetation samples obtained semi-annually at eight (8) locations (sample stations 1-7, and 10) with typical sensitivity based on average minimum detectable activity from 2020 through 2024 for gamma emitters ~1.33 pCi/gm;
- Water samples obtained semi-annually at three (3) locations (sample stations 4, 6, and 10) with typical sensitivity based on average minimum detectable activity from 2020 through 2024 for gamma emitters ~192.4 pCi/liter; and
- Integrated gamma dose measurements using Optically stimulated luminescence (OSL) dosimeters which are exchanged quarterly at up to forty-six (46) locations.

- Subsurface soil samples are taken at six (6) locations indicated by an "x" in Figure 11.20, Subsurface Soil Sample Locations. The typical sensitivity based on average minimum detectable activity from 2020 through 2024 for gamma emitters~0.57 pCi/g.

Water and vegetation samples undergo gamma-ray spectroscopy and are analyzed for gross alpha, gross beta, and tritium. Soil samples undergo gamma-ray spectroscopy and are analyzed for gross alpha and gross beta. OSL dosimeters are processed by a contractor and the results are returned to the Reactor HP Staff for review and compilation. The OSLs are designed to withstand the variety of environmental challenges associated with Missouri weather conditions and still be sensitive and accurate for measuring very low levels of radiation exposure.

MURR is focused on ensuring a comprehensive environmental monitoring program which incorporates an adequate number of sample types (collected at the correct locations and at the appropriate frequencies) which are then analyzed with sufficient sensitivity and reported in the Reactor Operations Annual Report. Document control measures for these procedures are described in Section 11.1.2.8.

Environmental monitoring locations and the types of measurements made or the samples collected are summarized in Tables 11-18 and 11-19. A wastewater treatment plant operates in the southwest section of the city, approximately 3.7 miles (6 km) west of the reactor facility. Sample Station 10 is located at this sewage treatment facility.

TABLE 11-18
ENVIRONMENTAL MONITORING PROGRAM – SAMPLING LOCATIONS

Location Number	Description	Vegetation (>50 grams)	Soil (>120 ml)	Water (1 liter)
10	Columbia Waste Water Treatment Plant	X	X	X
1	University Hall	X	X	
2	University Golf Course, Hole 3	X	X	
3	Research Park Development Building Grounds	X	X	
4	Hinkson Creek at Research Park	X	X	X
5	President's Drive behind Physical Plant	X	X	
6	Hinkson Nature Preserve – back corner of rugby field where sewage line passes under the creek	X	X	X
7	Lower Golf Course	X	X	

11.1.7.1 Environmental Monitoring Technical Specifications

The objective of the limiting conditions for operations (LCO) Technical Specification on the Environmental Monitoring Program is to ensure that sufficient radiation monitoring information is available to verify that continued operation of the facility presents no significant risk to the health and safety of the general public. Technical Specification 3.7.c provides this assurance for the Environmental Monitoring Program.

The objective of the surveillance Technical Specification on the Environmental Monitoring Program is to reasonably assure that radiation limits are not being exceeded. Technical Specification 4.7.c provides this reasonable assurance for the Environmental Monitoring Program.

Technical Specification 6.6.e (7) requires the description of any environmental surveys performed outside the reactor facility be included in the Annual Report to the NRC. Technical Specification 6.7.a (2) requires offsite environmental monitoring surveys be retained for the lifetime of the facility. This Technical Specification is consistent with Section 12.5.1 regarding retention of records for the lifetime of the facility.

TABLE 11-19
ENVIRONMENTAL DOSIMETERS

Badge Number	Direction from MURR	Meters from MURR Stack
000	Control	N/A*
9	W	30
10	SW	59
11	ENE	110
12	NNE	84
13	ENE	55
14	SW	32
15	SSE	27
16	NE	139
17	N	135
18	NE	284
19	NNE	305
20	S	168
21	SSE	74
22	SE	113
23	E	299
24	NE	453
25	NE	673
26	NE	893
27	SSE	239
28	SE	158
29	NW	89
30	SSW	308
31	SSW	435
32	SSW	365
33	SW	170
34	NW	229
35	NW	260
36	N	335
37	NNE	677
38	NW	760
39	ESE	578

TABLE 11-19
ENVIRONMENTAL DOSIMETERS (CONT.)

Badge Number	Direction from MURR	Meters from MURR Stack
40	ENE	596
41	SSE	477
42	SE	446
43	NE	732
44	NW	487
45	W	528
46	N	503
47	NE	161
48	In Building	N/A
49	In Building	N/A
50	SW	102
51	SE	94
52	SE	105

*The control monitors are approximately 10 miles (16.1 km) NW of MURR

11.2 Radioactive Waste Management

MURR has a comprehensive Radioactive Waste Management Program that supports operation of the reactor, its ancillary facilities, and their utilization programs. All radioactive waste materials released from the facility through the Ventilation and Air Treatment System, the Radioactive Liquid Waste Retention and Disposal System, and the Solid Radioactive Waste Program are identified, assessed, and released or disposed of in conformance with all applicable regulations and in a manner that protects the health and safety of the general public and the environment. It is the policy of the facility to keep the volume of waste materials being generated to the absolute minimum by the efficient use of experiment materials, by the use of proper techniques, and by any other means available.

The MURR facility contracts LLW acquisition and disposal services to a licensed waste broker. Through the licensed waste broker, MURR disposes of LLW waste . Most waste is shipped through the waste broker to a licensed waste processing facility prior to final disposal (Reference 11.4).

11.2.1 Radioactive Waste Management Program

The objective of the Radioactive Waste Management Program is to ensure that radioactive waste materials generated at the facility are minimized, and that they are properly handled, stored, and disposed of. It is the responsibility of every individual who uses the facilities of MURR to ensure contaminated and non-contaminated material is properly segregated, contained, and labeled.

The Reactor HP Manager, with the assistance of the HP Staff, is responsible for the safe disposal of radioactive waste generated from materials under the Reactor License and administers the Radioactive Waste Management Program. (Reference 11.4) The organizational structure, the authorities and responsibilities, and the position qualifications for the Reactor HP Staff are discussed in Section 11.1.2.1 and Chapter 12, “Conduct of Operations.” The RSO, with the assistance of the Reactor HP Staff, is

responsible for the safe disposal of radioactive waste from materials licensed under the MURR Broad Scope Materials License. The responsibilities and primary duties of the RSO are discussed in Section 11.1.2.4.

Radioactive waste management training is part of the initial and refresher radiation protection training provided to individuals who use and/or may come in contact with radioactive materials. The radiation protection training program, including the topics covered, is described in Section 11.1.2.5.

The Radioactive Waste Management Program is periodically audited as part of the RPP and other radiation safety programs (e.g., ALARA program). The audit is performed by MURR management or its authorized delegates in order to verify the adequacy of the program and its compliance with applicable regulations.

Radioactive waste management records, including radioactive material shipment and transfer records, are maintained by the Reactor HP Staff. All records are retained for the life of the facility.

11.2.2 Radioactive Waste Controls

Radioactive waste is generally considered to be any item or substance which is no longer of use to the facility and which contains, or is suspected of containing, radioactivity above the established natural background radioactivity. Because of the operational history of the facility and its experimental programs and the fact that the waste items are generally repetitive and easily identifiable, there is usually little question about what is, or what is not, radioactive waste. Reactor equipment or components are categorized as radioactive waste by the Reactor Operations Staff with assistance from the Reactor HP Staff, while standard consumable supplies such as plastic bags, gloves, absorbent material, contamination wipes, etc., become waste if detectable radioactivity above background is found to be present.

Written procedures provide guidance in the segregation and preparation of radioactive waste prior to its transfer to the Reactor HP Staff for disposal. Safe disposal may be accomplished by any one of the following methods: storage for decay and ultimate disposal as ordinary trash; release of limited and strictly controlled quantities into the sanitary sewage system; transfer to another of the University of Missouri licenses for storage, incineration, or processing for other disposal; or packaging for subsequent shipment to a commercial waste facility.

11.2.2.1 Solid Waste

Solid radioactive waste, for the most part, is generated from reactor maintenance activities and from the utilization of experimental facilities. Section 11.1.1.3 summarizes the sources of solid waste at MURR.

Receptacles lined with plastic radioactive materials bags or in other ways made acceptable for the disposal of radioactive waste are located in laboratories and other work areas that create solid radioactive waste. The solid waste is placed in these receptacles and then collected on a routine basis

and stored in a designated waste processing location until a sufficient volume has accumulated and then it is packaged in sealed containers (typically metal drums). Appropriate radiation monitoring equipment is used in identifying and segregating solid radioactive waste. All items and materials initially categorized as radioactive waste are monitored a second time before packaging for disposal in order to confirm data needed for waste shipment records, and in order to provide a final opportunity for decontamination/reclamation of an item. This helps reduce the volume of waste by eliminating the disposal of items that can still be used. The containers are then processed and prepared for shipment by the Reactor HP Staff according to Department of Transportation (DOT) specifications. The containers are shipped directly to a waste disposal site for final disposal or transferred to an authorized radioactive waste broker or brokerage service for further processing. No solid waste is intended to be retained or permanently stored on the MURR site.

Highly radioactive materials such as spent fuel elements require special handling and shipment containers. Such shipments are planned in accordance with NRC and DOT regulations.

While it is the policy of the facility to keep the volume of waste materials generated to the absolute minimum, there are no numerical volume goals set at the facility for this purpose. The Reactor and Reactor HP Managers periodically assess operations for the purpose of identifying opportunities or technological improvements that will reduce or eliminate the generation of radioactive waste.

11.2.2.2 Liquid Waste

Non-Resource Conservation Recovery Act (RCRA) regulated radioactive liquid wastes may be solidified and further processed as solid radioactive waste. All other potentially radioactive liquid wastes are directed to a liquid waste retention and disposal system located on the below-grade level of the laboratory building. The liquid waste retention and disposal system consists of four (4) tanks, three (3) with a capacity of approximately 5,000 gallons (18,927 l) and a fourth (1) with a capacity of 550 gallons (2,082 l), three (3) transfer pumps, three (3) filter banks, and associated piping and valves.

Liquid waste is retained or chemically treated until an assay indicates that activity levels are less than the limits specified in 10 CFR 20 for disposal by release into sanitary sewerage. Chemical treatment consists of introducing a carrier solution to precipitate-out the radionuclides and/or neutralization of acidic or basic solutions in order to meet the pH requirements of the local sewerage waste treatment facility. The precipitate can then be removed by filtration or pumped to a different waste retention tank. In addition to having the activity levels measured, all liquid waste is circulated through a filter bank until no suspended solids of a visible size remain prior to release to the sanitary sewer. It is the policy of the reactor facility to hold all liquid waste as long as practical to minimize the total activity released to the environment.

Section 11.1.1.2 describes the radioactive liquid sources associated with the operation of the reactor and its utilization programs. It indicates that radioactive liquid waste generated in the laboratories is the most significant source in terms of volume. Since primary and pool coolants are, by design, contained to the maximum extent possible, there are no routine releases of these liquids. However,

certain maintenance operations, such as the transfer of resin from a demineralizer tank, result in small amounts of radioactivity (mainly tritium) being directed to the liquid waste retention system.

11.2.2.3 Gaseous Waste

Although Ar-41 and other radioactive gases are released from the facility through the ventilation system exhaust stack, this release is not considered to be waste in the same sense as the solid and liquid wastes. Releases through the MURR exhaust stack are usually classified as effluent, which is a routine part of the normal operation of the reactor. In the MURR facility, as in many other non-power reactors, there are no radioactive waste off-gas collection systems. Exhaust air from both the laboratory and reactor containment buildings is combined in the facility ventilation exhaust plenum prior to being discharged to the atmosphere. Since air from both buildings is not mixed until this point, potentially contaminated air is diluted by mixing with uncontaminated air, resulting in reduced concentrations of radioactive gases being released to the environment.

A complete description of Ar-41 production and its subsequent discharge into the unrestricted environment is contained in Section 11.1.1.1 and Appendix B, "Radiological Impact of Ar-41 During Normal Operations." The design features which are incorporated into the facility ventilation system to minimize the release of airborne radioactivity are described in Section 11.1.5.2.

11.2.3 Release of Radioactive Waste

The release of gaseous (mainly Ar-41) and particulate activity through the facility ventilation exhaust is described in Section 11.1.1.1 and Appendix B, "Radiological Impact of Ar-41 During Normal Operations." The maximum rate of discharge shall not exceed 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.

Liquid radioactive 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 for disposal by release into sanitary sewerage. In addition to the limit on each isotope, 10 CFR 20 also limits the total activity that can be annually released from the MURR to the sanitary sewerage. Investigational levels are described in MURR ALARA Program, of MURR Policy Manual POL-3, "Radiation Protection Program".

Normally, the transfer of solid radioactive waste is to an authorized solid waste broker or brokerage service. However, the facility may opt to ship solid radioactive waste directly to a waste disposal site without the use of a broker. The individual responsible for making the waste shipment must have documented training which meets the requirements of 49 CFR 172, Subpart H.

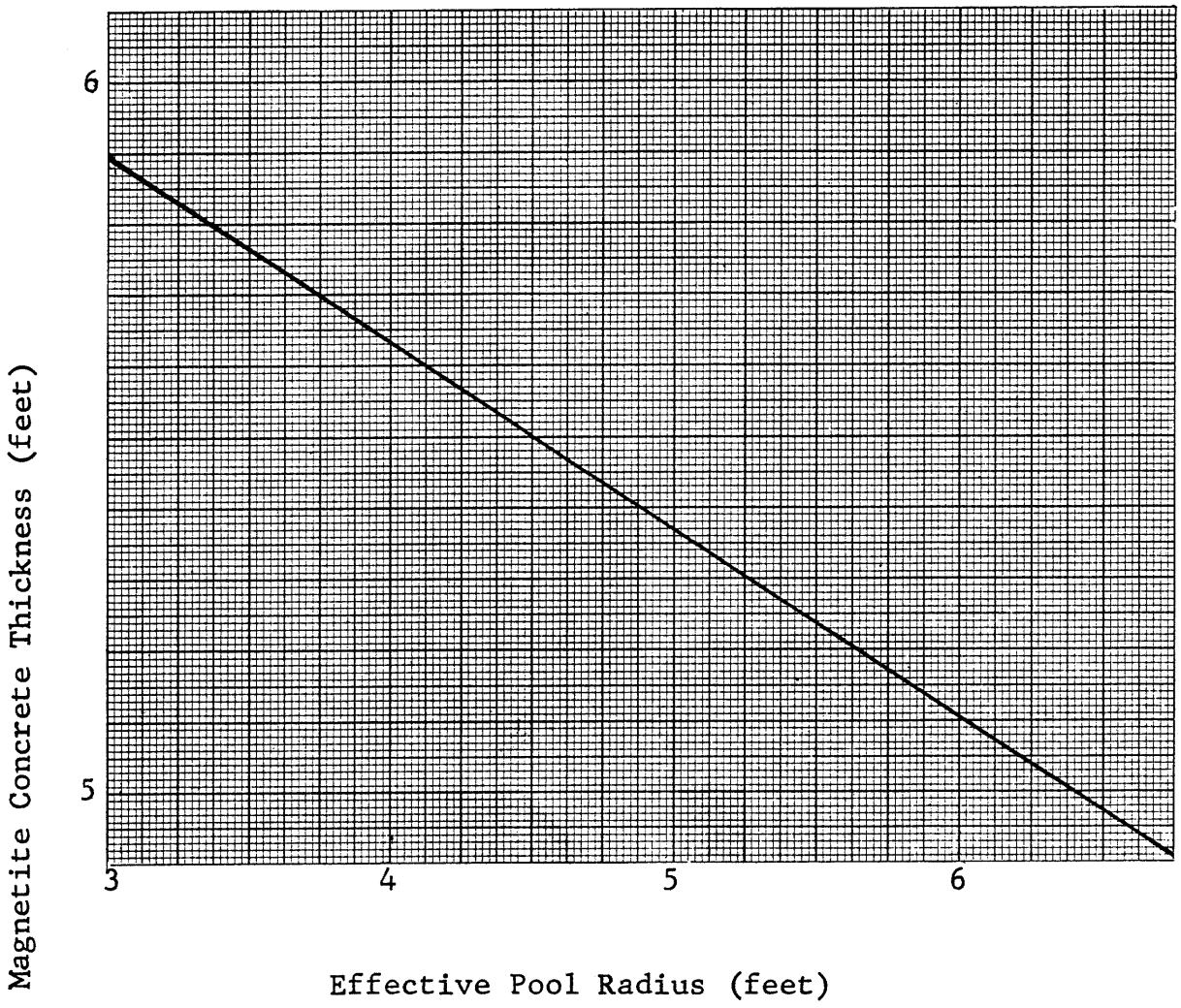


FIGURE 11.1 – BIOLOGICAL SHIELD MAGNETITE CONCRETE VS. EFFECTIVE POOL RADIUM

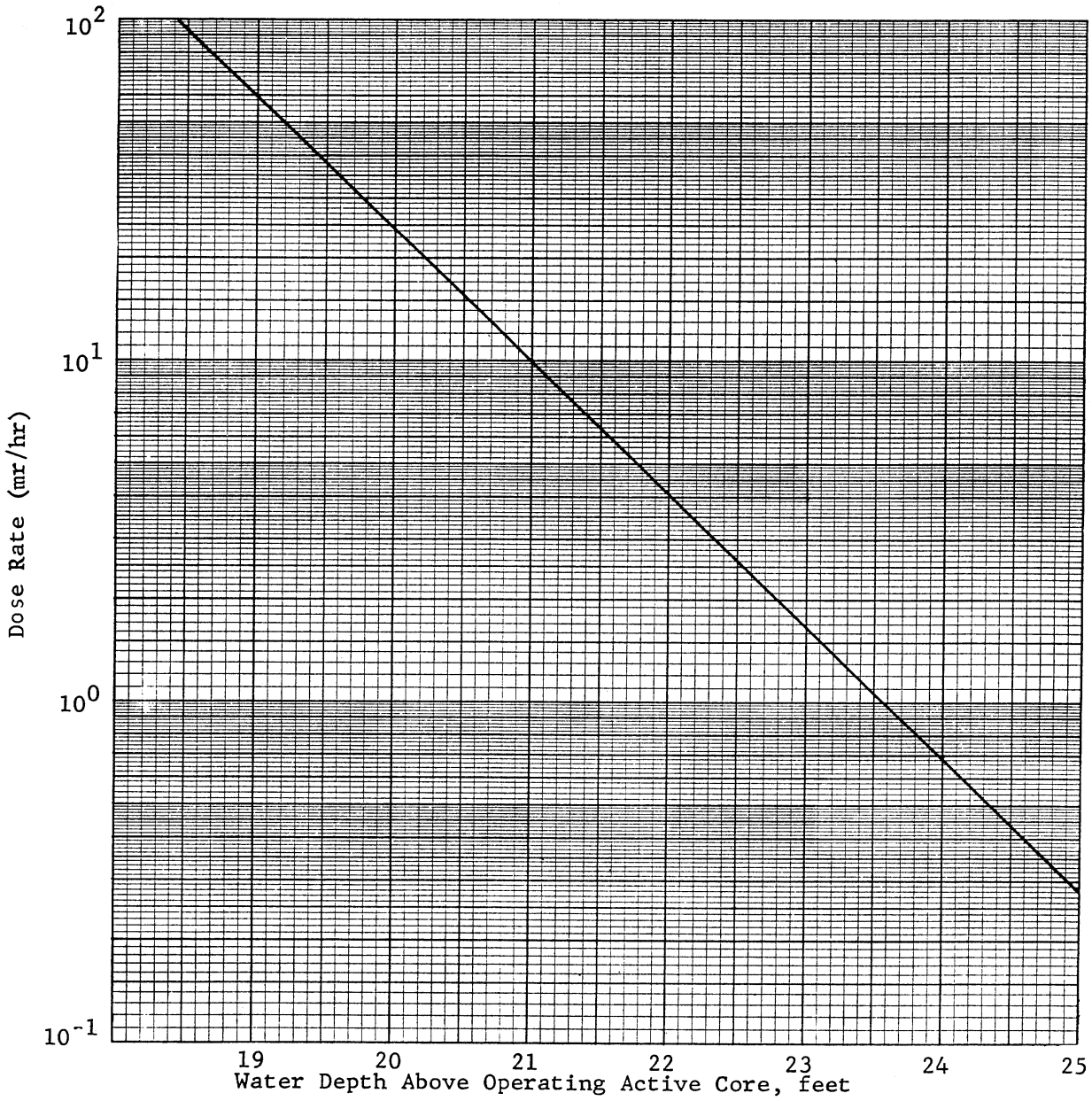


FIGURE 11.2 – DIRECT PENETRATION GAMMA-RAY DOSE RATES AS A FUNCTION OF WATER DEPTH ABOVE THE ACTIVE CORE REGION

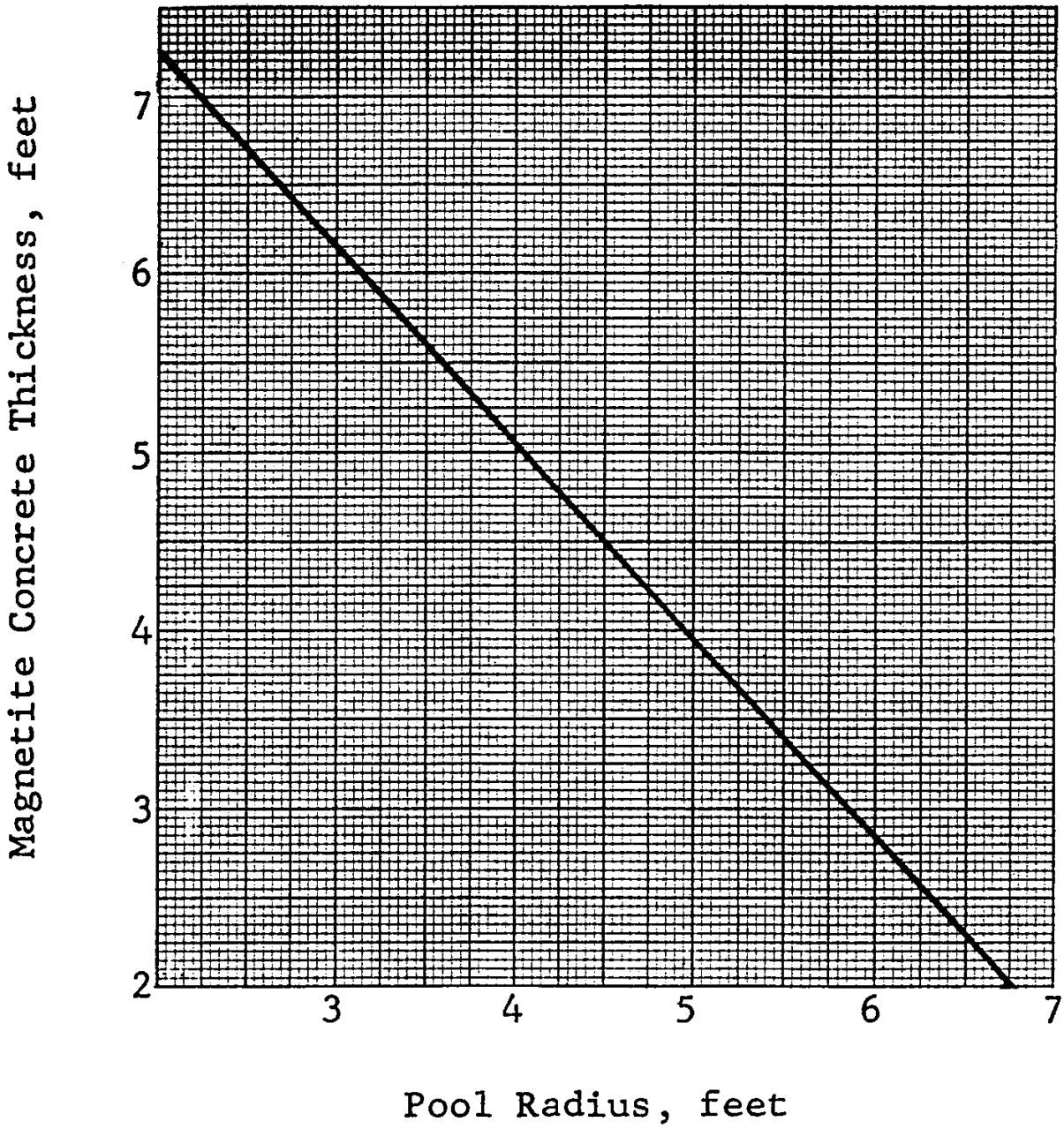


FIGURE 11.3 – MAGNETITE CONCRETE BIOLOGICAL SHIELDING REQUIREMENTS VS. EFFECTIVE POOL RADIUS TO ACHIEVE ONE-TENTH OF DOSE RATE CRITERIA FROM FAST NEUTRONS

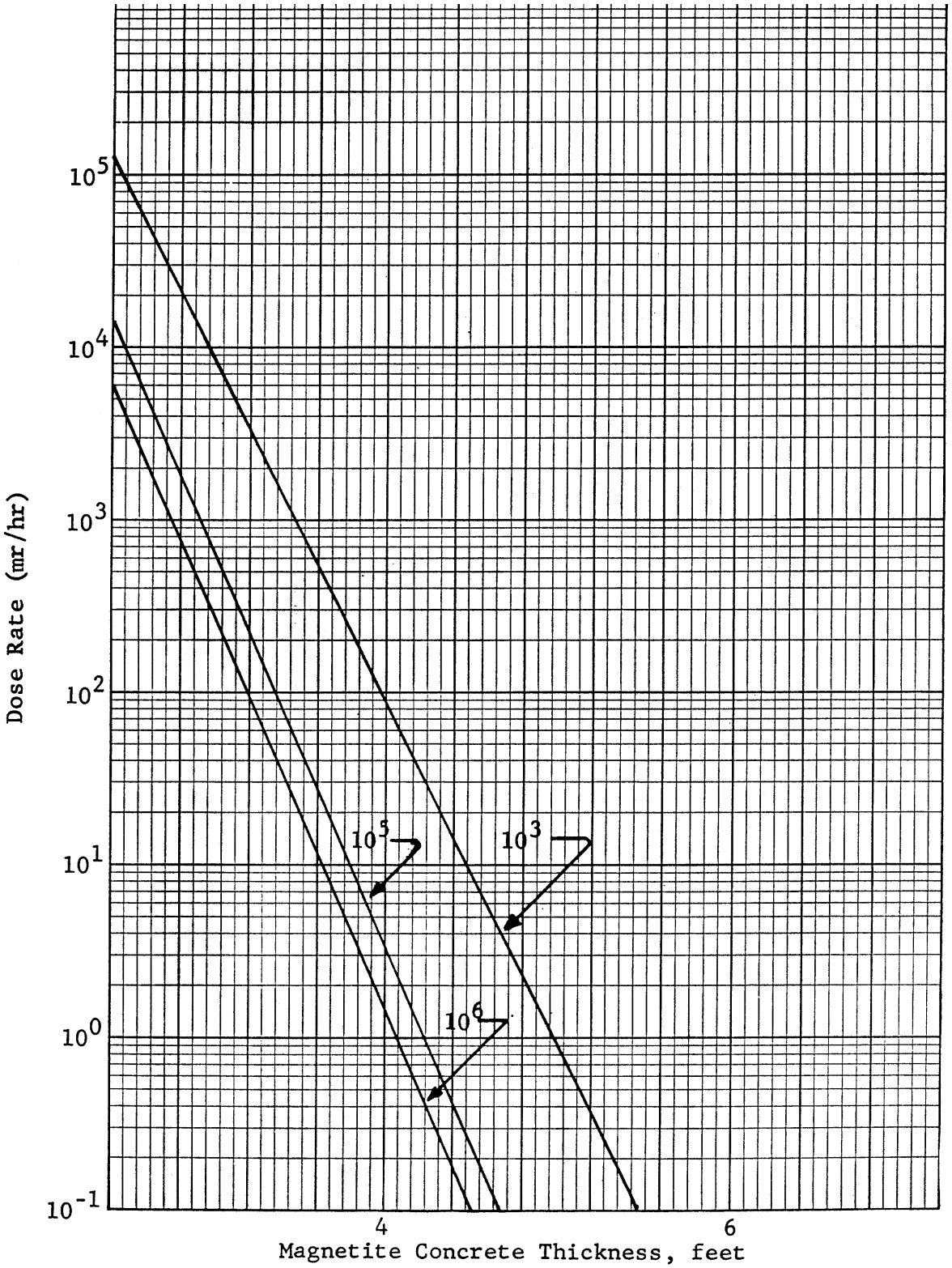


FIGURE 11.4 – SPENT FUEL ELEMENT DOSE RATES THROUGH MAGNETITE SHIELDING FOR A SINGLE ELEMENT AS A FUNCTION OF FISSION PRODUCT DECAY TIME

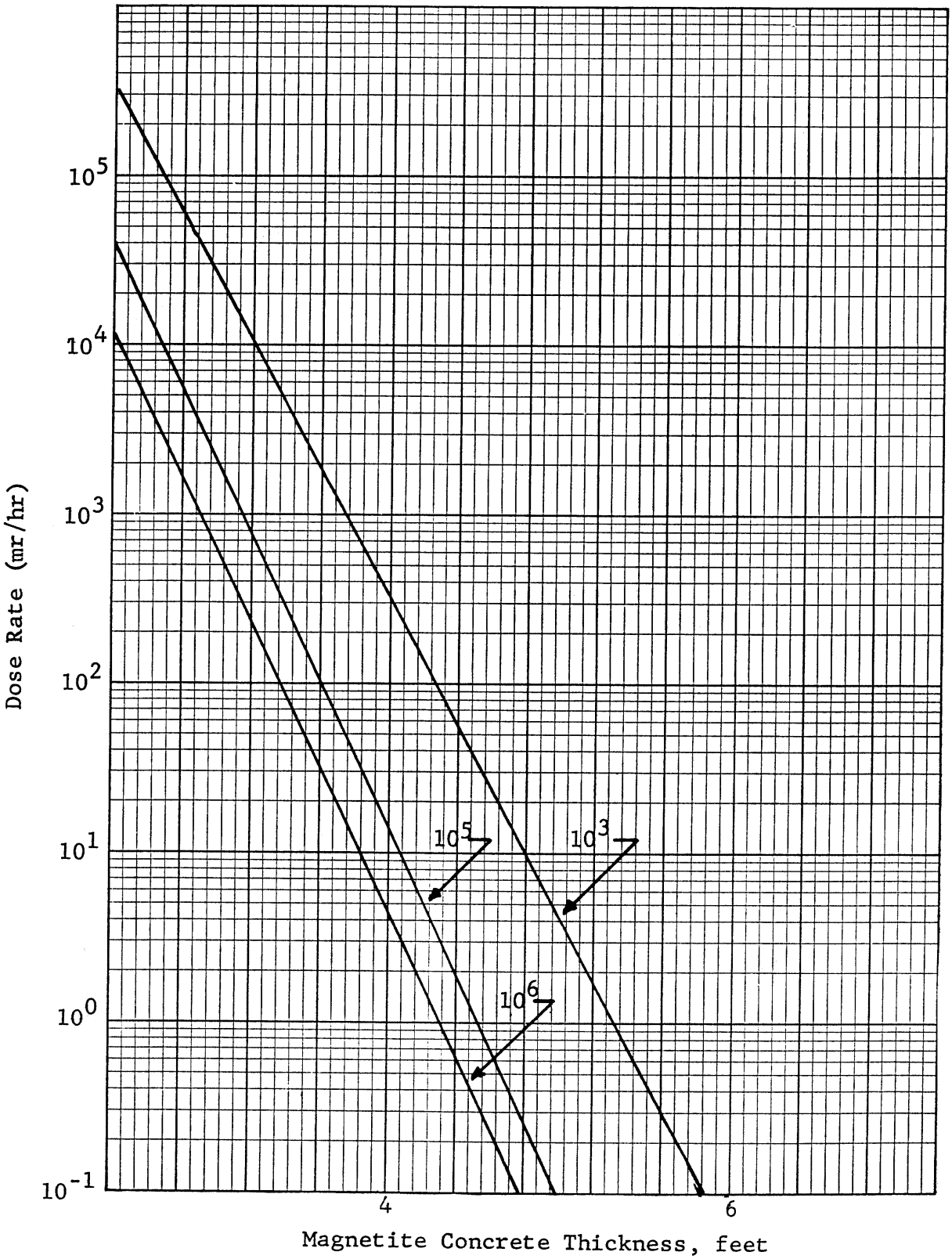


FIGURE 11.5 – SPENT FUEL ELEMENT DOSE RATES THROUGH MAGNETITE SHIELDING FOR FOUR ELEMENTS AS A FUNCTION OF FISSION PRODUCT DECAY TIME

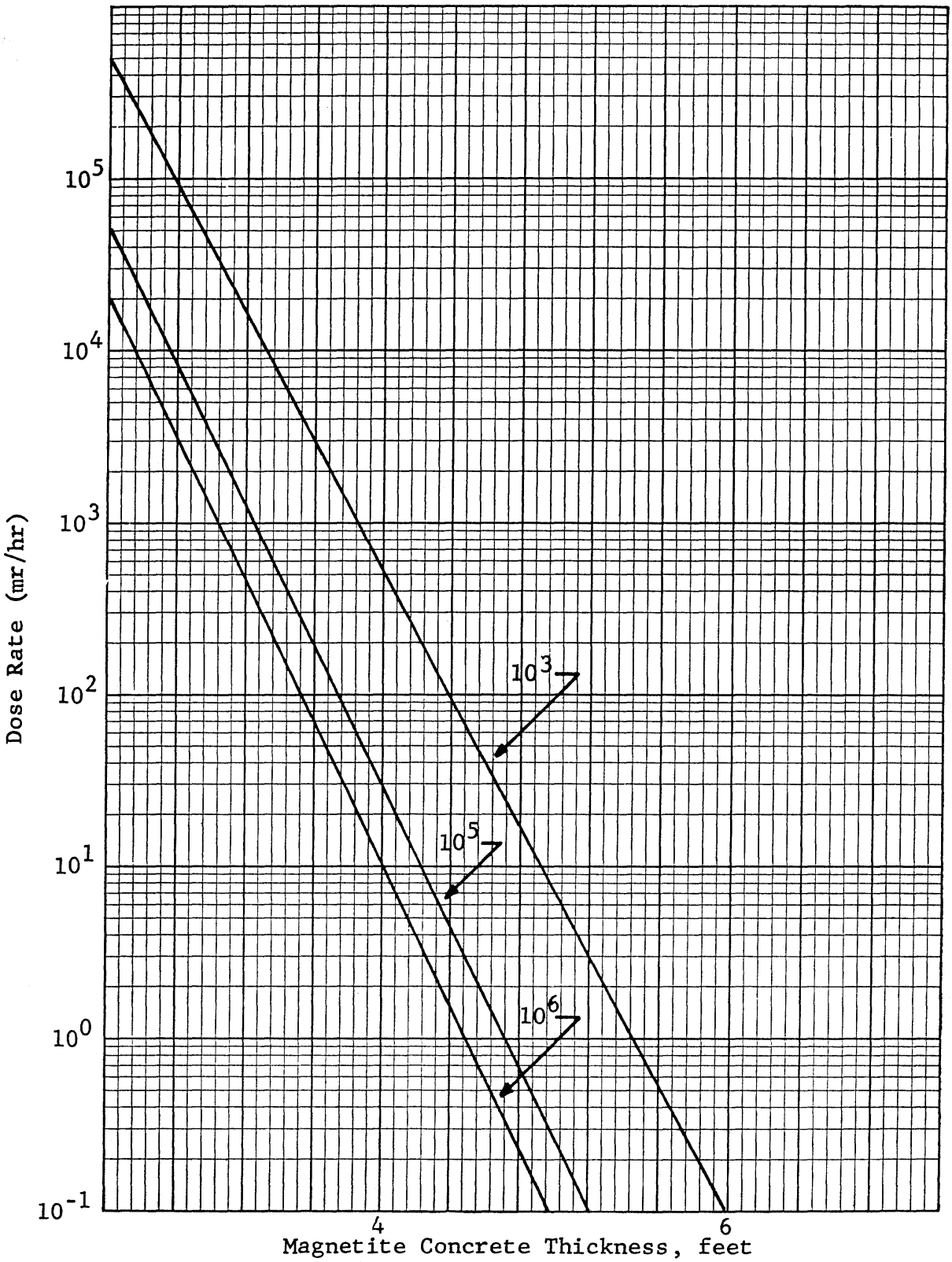


FIGURE 11.6 – SPENT FUEL ELEMENT DOSE RATES THROUGH MAGNETITE SHIELDING FOR EIGHT ELEMENTS AS A FUNCTION OF FISSION PRODUCT DECAY TIME

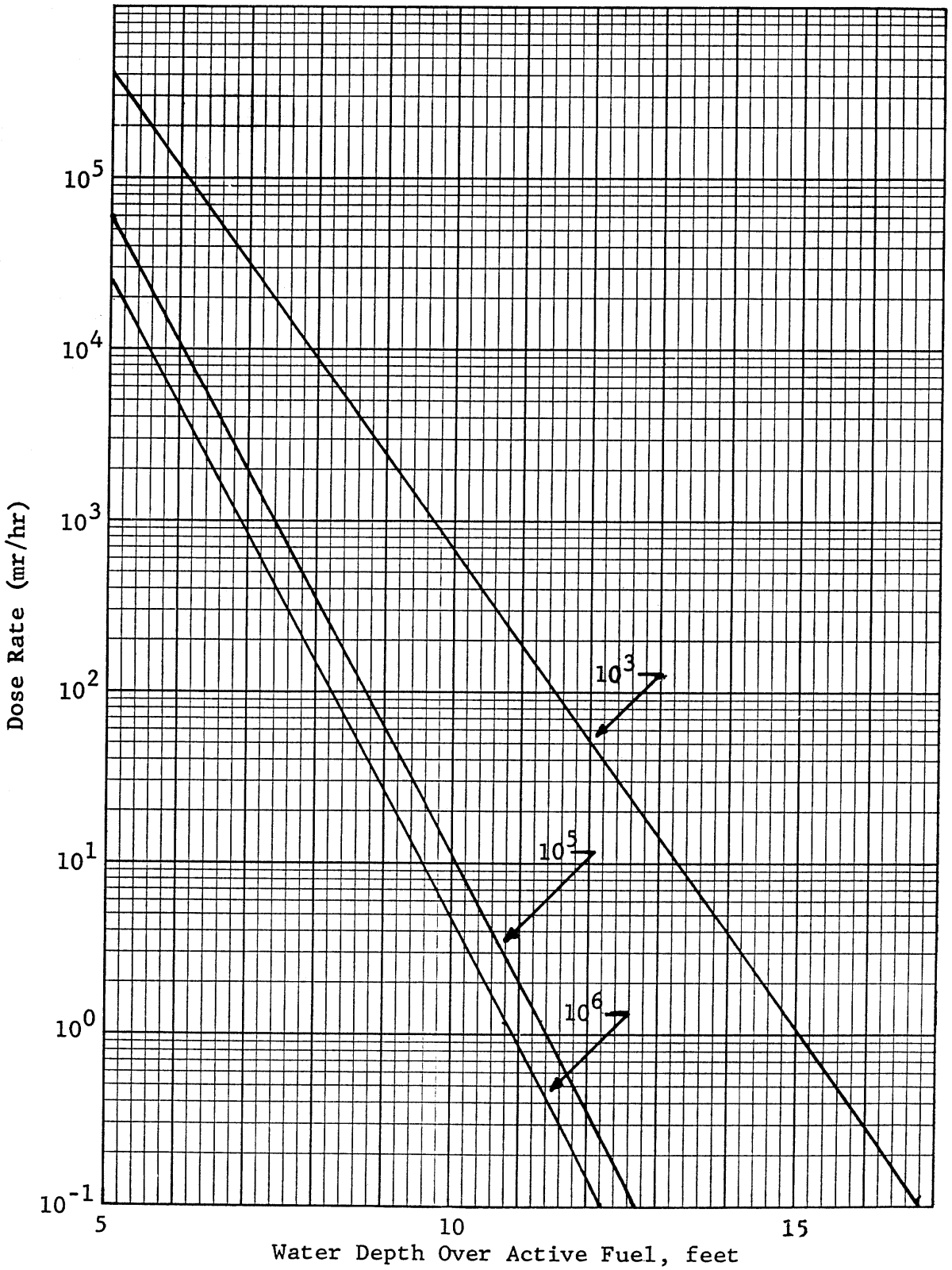


FIGURE 11.7 – SPENT FUEL ELEMENT DOSE RATES THROUGH WATER SHIELDING AS A FUNCTION OF FISSION PRODUCT DECAY TIME - HORIZONTAL POSITION

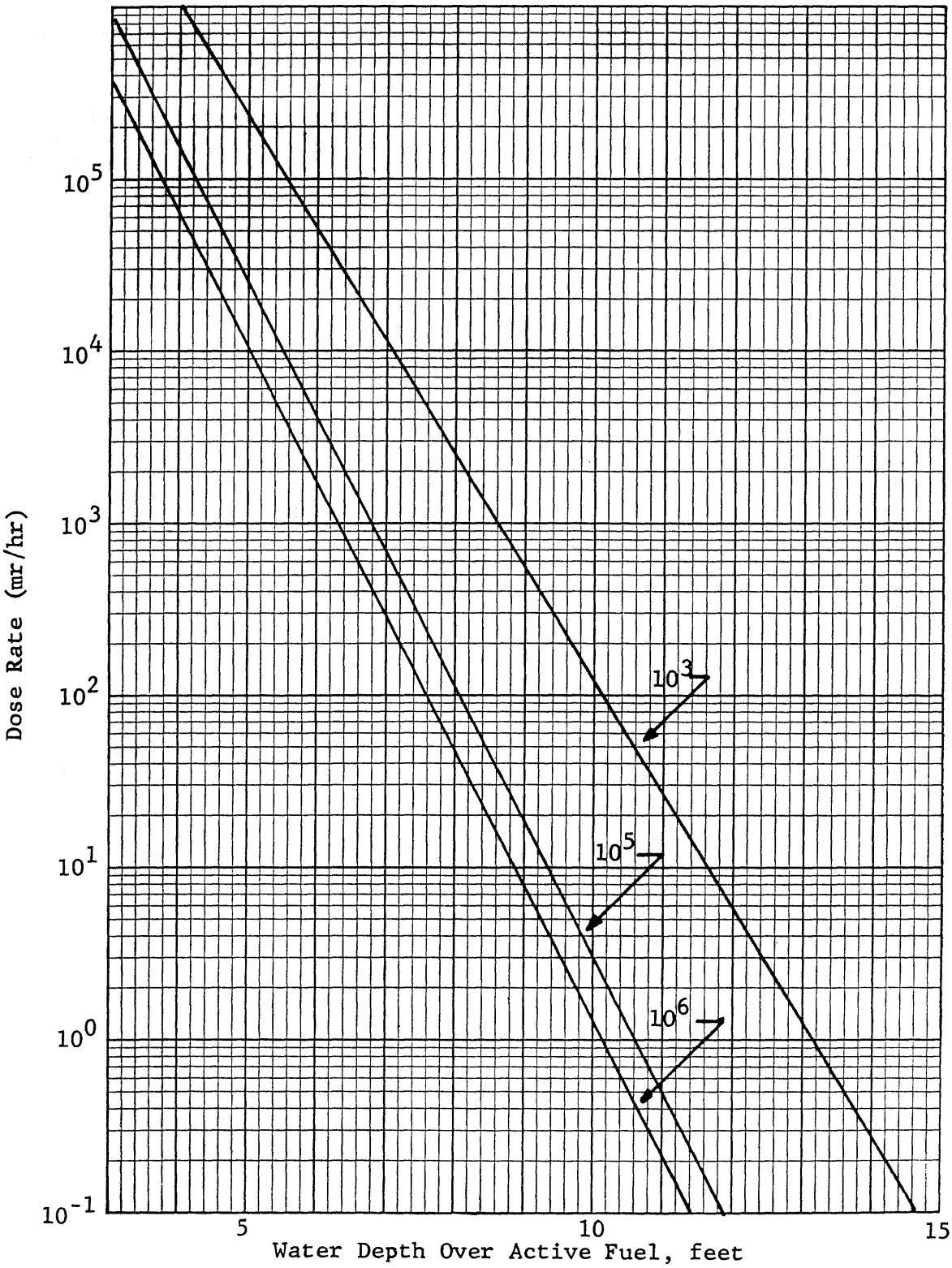


FIGURE 11.8 – SPENT FUEL ELEMENT DOSE RATES THROUGH WATER SHIELDING AS A FUNCTION OF FISSION PRODUCT DECAY TIME - VERTICAL POSITION

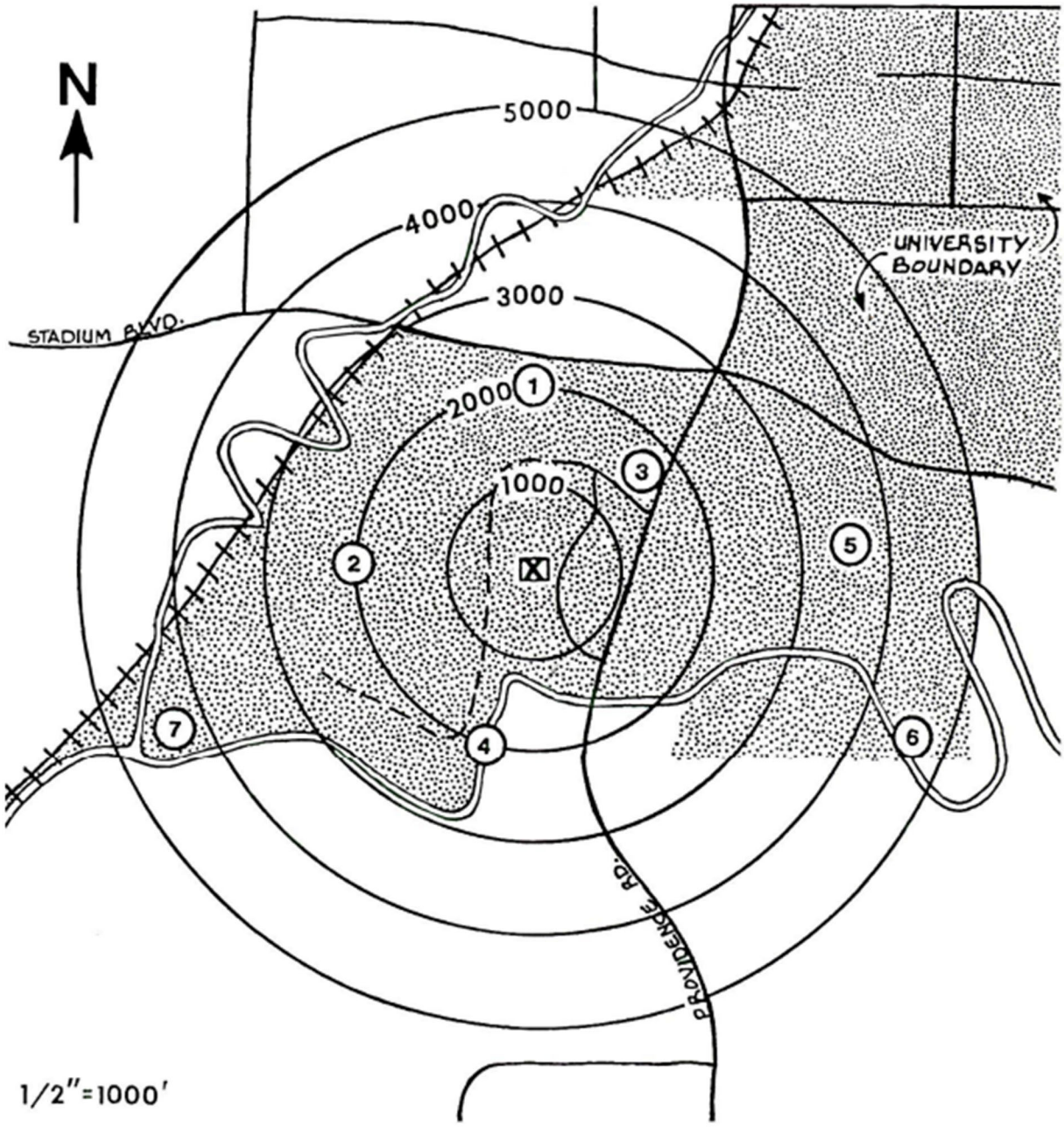


FIGURE 11.9 – ENVIRONMENTAL SAMPLING STATIONS

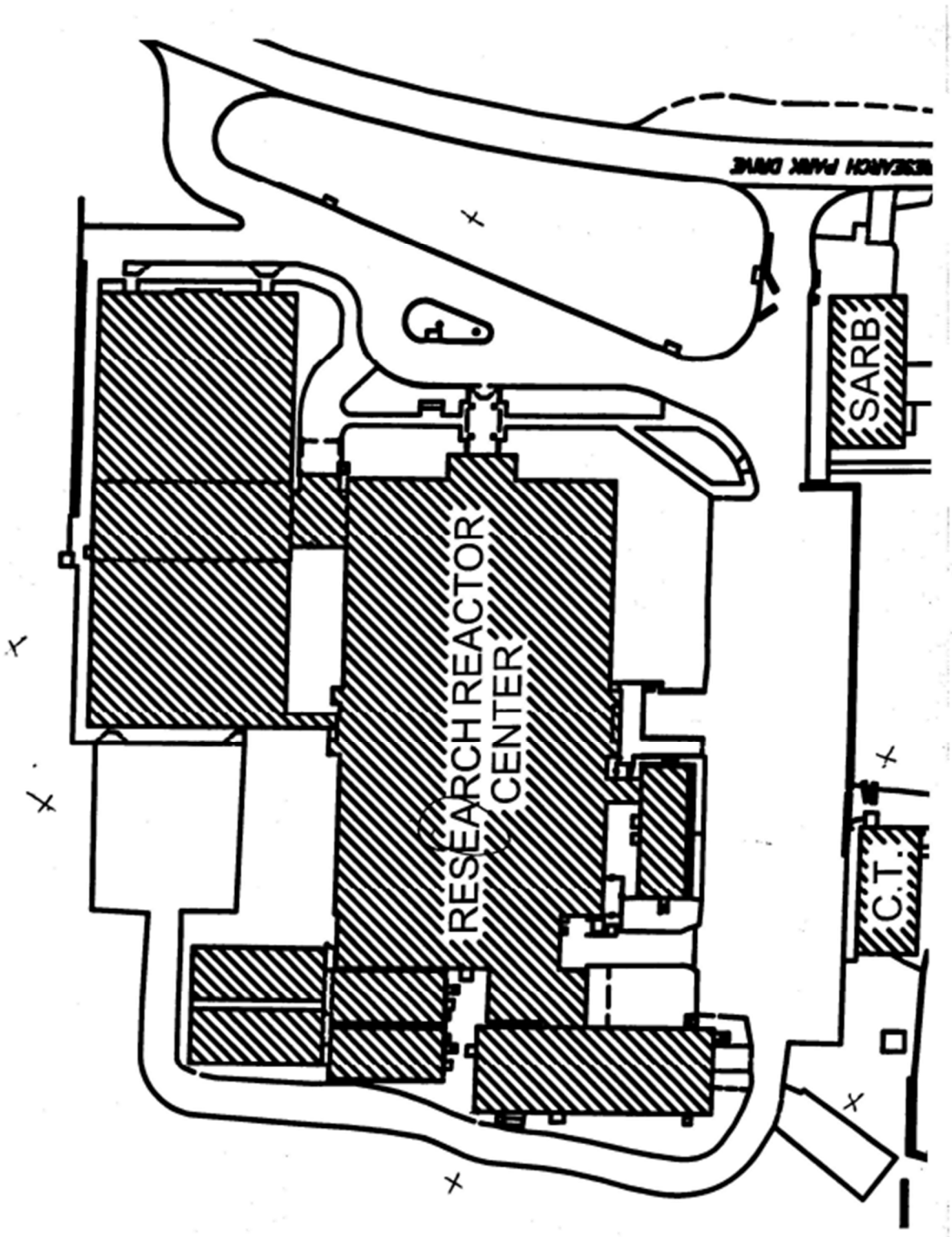


FIGURE 11.10 – SUBSURFACE SOIL SAMPLE LOCATIONS

CHAPTER 12

CONDUCT OF OPERATIONS

TABLE OF CONTENTS

12.0 CONDUCT OF OPERATIONS 3
12.1 Organization 3
12.1.1 Structure 3
12.1.2 Responsibility 4
12.1.2.1 Office of the Chancellor 4
12.1.2.2 Reactor Facility Director 4
12.1.2.3 Reactor Manager 4
12.1.2.4 Reactor Health Physics Manager 5
12.1.2.5 Reactor Operations Staff 5
12.1.2.6 Reactor Health Physics Staff 5
12.1.3 Staffing 6
12.1.4 Selection and Training of Personnel 7
12.1.4.1 Initial Training and Licensure 7
12.1.4.2 Requalification 7
12.1.5 Radiation Safety 7
12.2 Review and Audit Activities 8
12.2.1 Composition and Qualification 8
12.2.2 Charter and Rules 8
12.2.3 Review Function 9
12.2.4 Audit Function 10
12.2.5 Subcommittees 10
12.2.5.1 Reactor Safety Subcommittee 10
12.2.5.2 Reactor Action Subcommittee 12
12.2.5.3 Reactor Safety Procedure Review Subcommittee 12
12.2.5.4 Isotope Use Subcommittee 14
12.2.5.5 Isotope Use Procedure Review Subcommittee 15
12.3 Procedures 16
12.3.1 Reactor Operations 16
12.3.2 Reactor Health Physics 17
12.4 Reportable Events and Required Actions 17
12.4.1 Safety Limit Violation 17
12.4.2 Release of Radioactivity 18
12.4.3 Other Reportable Occurrences 18
12.4.4 Other Reports 18
12.4.5 Annual Report 18
12.5 Records 19
12.5.1 Lifetime Records 19
12.5.2 Five Year Records 20
12.5.3 Operator Licensing Records 20
12.6 Emergency Planning 20

12.8 Quality Assurance 21
12.9 Operator Training and Requalification 22
12.10 Environmental Reports 23

LIST OF FIGURES

Figure 12.1 – UNIVERSITY OF MISSOURI RESEARCH REACTOR (MURR) ORGANIZATION 24

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12.1 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," 2007.

12.2 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.4, "Selection and Training of Personnel for Research Reactors," 2007.

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12.7 Written communication dated September 30, 2010, regarding responses to the "University of Missouri at Columbia – Request for Additional Information Re; License Renewal, Safety Analysis Report, [ML12355A019].

12.8 Written communication dated March 11, 2011, regarding responses to the "University of Missouri at Columbia – Request for Additional Information Re; License Renewal, Safety Analysis Report, [ML110740249].

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12.0 CONDUCT OF OPERATIONS

This chapter describes and discusses the conduct of operations at the reactor facility. The conduct of operations involves the administrative aspects of facility operation, the facility emergency plan, the security plan, the quality assurance plan, the reactor operator requalification plan, and environmental reports. The administrative aspects of facility operations are the facility organization, review and audit activities, organizational aspects of radiation safety, facility procedures, required actions in case of license or Technical Specifications violations, reporting requirements, and record keeping. This chapter forms the basis for Section 6.0 of the Technical Specifications (Reference 12.1) and (Reference 12.9).

12.1 Organization

University of Missouri Research Reactor (MURR) falls within the organizational structure of the University of Missouri System and is administratively controlled and operated as shown in Figure 12.1. The operating and service groups of the University can be divided into three (3) main categories: academic departments, each of which is supervised by a dean; academic research and service organizations, which include budget, construction, auditing, etc.; and non-academic enterprise operations, which include the MU hospital, athletic organizations, and MURR.

The reactor and laboratory facilities of MURR are available to faculty members or graduate and undergraduate students interested in pursuing research involving radiation, radioisotopes, or the reactor. The research programs are coordinated, supervised, and monitored by the permanent staff employed by the facility. Some of the research staff, composed of faculty members and graduate students, is semi-transient, with no permanent assignment of space or facilities. Administration of MURR is separately maintained and removed from the administration of the research programs to eliminate the possibility of a compromise in safety for the sake of experimental expediency.

12.1.1 Structure

The University of Missouri System is governed by a nine-voting member Board of Curators appointed by the Governor and confirmed by the State Senate to serve a six-year term. One non-voting student representative serves a two-year term. As the facility licensee, The Board of Curators is responsible for ensuring adherence to all the requirements of the facility operating license and the Technical Specifications, thus reasonably ensuring that the health and safety of the general public will not be endangered as a result of operating the reactor. The Board of Curators delegates this responsibility through the President and the Office of the Chancellor to the Reactor Facility Director. The Reactor Facility Director has overall responsibility for the direction and operation of MURR. He delegates the internal direction of the Reactor Operations Branch to the Reactor Manager and the Reactor Health Physics (HP) Branch to the Reactor HP Manager. The Reactor Facility Director reports to the Office of the Chancellor.

The Reactor Operations Branch includes the Reactor Manager and the Reactor Operations Staff. The Reactor Operations Staff consists of Assistant Reactor Managers, Senior Reactor Operators (SROs), and Reactor Operators (ROs).

The Reactor HP Branch includes the Reactor HP Manager and the Reactor HP Staff. The Reactor HP Staff consists Reactor Health Physicists and Reactor HP Technicians.

The Reactor Advisory Committee (RAC) provides independent oversight in matters pertaining to the safe operation of the reactor and with regard to planned research activities and uses of the facility building and equipment. The RAC is appointed by, and reports to, the Office of the Chancellor.

There is a communications/consultation line from the Reactor HP Manager to the Office of the Chancellor. This line of communications/consultation allows the Reactor HP Manager access to upper University of Missouri (MU) management if Reactor Facility Management does not address radiation protection concerns to the satisfaction of the Reactor HP Manager.

12.1.2 Responsibility

12.1.2.1 Office of the Chancellor

The Office of the Chancellor (Level 1) is responsible for directing MU's research mission, the quality and effectiveness of all programs and dedicating MU resources necessary to ensure that all research, education, and service are conducted in accordance with applicable federal, state, and local regulations and accreditation requirements.

12.1.2.2 Reactor Facility Director

The Reactor Facility Director (Level 2) is responsible for establishing the policies that minimize radiation exposure to the public and to radiation workers and that ensure the requirements of the license and Technical Specifications are met.

The Reactor Facility Director is assisted with providing supervision and guidance to the facility staff by a select group of Directors. However, all decisions regarding the reactor license are made by the Reactor Facility Director, Reactor Manager, and Reactor HP Manager.

12.1.2.3 Reactor Manager

The Reactor Manager (Level 3) is responsible for operating the reactor in a safe, efficient, and reliable manner. To safeguard the public and facility personnel from undue radiation exposure, the Reactor Manager is responsible for (1) compliance with Technical Specifications and license requirements regarding reactor operation, maintenance and surveillance; and (2) oversight of the experiment review process.

The Reactor Manager is assisted in the performance of these duties by the Reactor Operations Staff. The Reactor Manager shall be certified at the SRO level for the MURR facility pursuant to Title 10, Chapter I of the Code of Federal Regulations, Part 55 (10 CFR 55).

12.1.2.4 Reactor Health Physics Manager

The Reactor HP Manager (Level 3) is responsible for all aspects of radiation safety at MURR. To safeguard the public and facility personnel from undue radiation exposure, the Reactor HP Manager is responsible for (1) compliance with Technical Specifications and license requirements regarding radiation safety, byproduct material handling, the shipment of byproduct material; and (2) implementation of the Radiation Protection Program.

The Reactor HP Manager is assisted in the performance of these duties by the Reactor HP Staff.

12.1.2.5 Reactor Operations Staff

The Reactor Operations Staff (Level 4) assist the Reactor Manager in the performance of their duties. The Reactor Operations Staff includes Assistant Reactor Managers, SROs, ROs.

The Assistant Reactor Managers perform various administrative tasks associated with operation of the reactor and MURR facility. The Assistant Reactor Managers may be licensed as an SRO or RO pursuant to 10 CFR 55 or possess other qualifications such as substantial experience or technical degrees, which qualify them for the position.

SROs are responsible for directing the licensed and non-licensed activities of the ROs. SROs report to the Lead SRO (LSRO) of their assigned shift. (The importance of one supervisor to coordinate the licensed and non-licensed activities is recognized for safety and effective control of reactor operation. The LSRO is responsible for the supervision of the staff which supports the operation of the reactor during his assigned shift.) SROs also assume the duties and responsibilities of a RO when required. The SROs are licensed pursuant to 10 CFR 55.

ROs are responsible for the safe operation of the reactor through the assiduous manipulation of the reactor controls, the monitoring of reactor plant instrumentation, and the operating and maintaining of reactor-related equipment. ROs report to the LSRO of their assigned shift. The ROs are licensed pursuant to 10 CFR 55.

12.1.2.6 Reactor Health Physics Staff

The Reactor HP Staff (Level 4) assist the Reactor HP Manager in the performance of their duties. The Reactor HP Staff includes Reactor Health Physicists and the Reactor HP Technicians. The Reactor HP Staff are responsible for directing research, training, and monitoring programs in order to protect personnel from radiation hazards and to assure compliance with federal, state, and MU regulations.

12.1.3 Staffing

A list of reactor facility personnel by name and telephone number is available in the reactor control room for use by the reactor operators whenever a facility emergency exists. The emergency call list shall include:

- *Director's Office personnel;
- * Reactor Operations personnel;
- *Reactor HP (Radiation Safety) personnel;
    - Additional personnel (based on their expertise); and
    - Emergency Support Organizations (e.g., MU Police, MU Hospital and Clinics, City of Columbia Fire Department, etc.).

* Denotes call list items also required by Technical Specification 6.1.d

The following staffing requirements shall be satisfied:

- At a minimum during reactor operation, there shall be two facility staff personnel at the facility. One of these individuals shall be an RO or an SRO licensed pursuant to 10 CFR 55. The other individual shall be knowledgeable of the facility (A Reactor Operator Trainee who has successfully completed a 50% board, as defined in AP-RO-110). The knowledgeable person must be able to assist the on-site operator in performing actions to place the reactor in a safe condition and supporting actions to facilitate a reactor isolation or facility evacuation; (Reference 12.6)
- At a minimum, there will be one SRO or RO licensed pursuant to 10 CFR 55 in the reactor control room whenever the reactor is not considered secured as defined by the Technical Specifications. While the reactor is shutdown, but not secured, the reactor control room boundary may be expanded to include the operating (upper) bridge during refueling or maintenance activities;
- An SRO licensed pursuant to 10 CFR 55 shall be present at the facility or readily available on call at all times during operations. Readily available on call means an individual who:
    - Has been specifically designated and the designation know to the operator on duty;
    - Can be rapidly contacted by phone, by the operator on duty; and
    - Is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 minutes or within a 15-mile radius);
- An SRO must be present at the facility during:
    - Initial startup and approach to power;
    - All fuel or control rod relocations within the reactor core region;
    - Relocation of any experiment with a reactivity worth greater than 0.0074 Δk/k; and
    - Recovery from an unplanned or unscheduled shutdown or a power reduction of 2 MWs or greater.

- A Reactor HP Technician, Electronic Technician, and an Engineering (Mechanical Support) Technician will normally be readily available for emergencies (i.e., typically capable of arriving at the reactor facility within one hour of notification).

12.1.4 Selection and Training of Personnel

In order to develop and maintain an organization qualified for operation and maintenance of MURR, personnel will be selected and trained as operators using the guidelines described in Reference 12.2. Personnel who have been selected and trained to operate a research reactor shall have that combination of academic training, job-related experience, health, and skills commensurate with their level of responsibility in order to provide reasonable assurance that decisions and actions during all normal and abnormal conditions will be such that the reactor is operated in a safe manner. To ensure that the above qualifications are satisfied, all personnel selected to be licensed at the RO level and SRO level pursuant to 10 CFR 55 will participate in an initial training program and then a subsequent requalification program after their license is received from the NRC.

12.1.4.1 Initial Training and Licensure

Initial training of personnel (trainees) to be licensed as ROs and SROs will consist of documented stages of self-study and on-the-job training. The content of the training shall cover the physical facility, applicable theory and design, procedures, and applicable rules and regulations. The anticipated result of this training is a confident, well-versed, decisive individual capable of performing the duties of a licensed operator during normal and abnormal situations. Licensure of a candidate is achieved after extensive training followed by the successful completion of an examination administered by the NRC. (Reference 12.5)

12.1.4.2 Requalification

The objectives of the requalification program are to review/retrain in areas of infrequent operation, to review facility and procedural changes, to address subject matter not reinforced by direct use, and to improve in areas of performance weakness. The program is designed to evaluate an operator's knowledge level and proficiency and to retrain where necessary. Emphasis is on subjects necessary for continued proficiency. Licensed individuals who have successfully completed the requalification program may be recertified by the NRC when required. The MURR Operator Requalification Program is described in greater detail in Section 12.9.

12.1.5 Radiation Safety

The Radiation Protection Program has been established to protect the health and safety of MURR staff, research associates, students, and the general public. A primary component of this program is dedicated to the fundamental principle of maintaining individual exposures and radioactive effluents to the ALARA principle. Responsibilities for maintaining the MURR ALARA Program extend to all individuals who are granted unescorted access to the reactor facility.

All personnel using radioactive materials or radiation sources shall become familiar with the requirements of the Radiation Protection Program and conduct their operations in accordance with them. However, the Reactor HP Staff has the authority to interdict or terminate the use of radioactive materials or radiation sources if adequate HP support is not available or if significant deviations from established procedures have occurred or are likely to occur.

The Radiation Protection Program uses Reference 12.3 as a guide and is described in detail in Chapter 11, "Radiation Protection Program and Waste Management."

12.2 Review and Audit Activities

The Reactor Advisory Committee (RAC) is the Committee of the University of Missouri appointed by the University of Missouri (MU) Office of the Chancellor to satisfy requirements imposed by the U.S. Nuclear Regulatory Commission (NRC). The University and the NRC expect the Committee to act in an advisory capacity to the Reactor Facility Director and to provide independent oversight in matters pertaining to the safe operation of the reactor and with regard to planned research activities and use of the facility building and equipment. (Reference 12.9)

12.2.1 Composition and Qualification

The RAC shall be composed of at least five (5) members. Members of the RAC shall be appointed to the RAC with respect to their expert knowledge of experimental activities, reactor operations, MU business policy, or related subjects. Members of the RAC shall be appointed by, and report to, the Office of the Chancellor. Members may be reappointed indefinitely.

The RAC, through its Chairperson, may appoint subcommittees (See Section 12.2.5) consisting of knowledgeable members of the public, students, faculty, and MU staff when it is deemed necessary to delegate a part of its responsibilities. Membership on the subcommittees need not be limited to appointed members of the RAC. When subcommittees are appointed, these subcommittees shall consist of no less than three (3) members with no more than one (1) student appointed to each subcommittee. Subcommittees may be authorized to act on behalf of the RAC.

12.2.2 Charter and Rules

The RAC shall meet at least once in each four (4) month period. The RAC and its subcommittees shall maintain minutes of meetings in which the items considered and the committees' recommendations are recorded. Dissemination of the minutes to the Office of the Chancellor, the RAC and its subcommittees shall be done within four (4) months after the meetings. Subcommittee recommendations or independent actions made or taken in the name of the RAC are to be reviewed and approved by the RAC at its next regular meeting.

A quorum of the RAC or the subcommittees, consisting of at least 50 percent of the appointed members, shall be present at any meeting to conduct the business of the RAC, or a subcommittee. *Ex officio* members shall be without vote. A quorum must be present at any meeting to conduct the business of

the RAC or subcommittee. The Chairperson will establish that a quorum is present for the conduct of business before the committee, and this determination will be documented by recording the names of members present. Additionally, reactor facility staff shall not constitute greater than fifty percent of the quorum for a meeting of the RAC. Reactor facility staff shall not constitute a majority of the RAC. A meeting of a subcommittee shall not be deemed to satisfy the requirement for the RAC to meet at least once in each four (4) month period.

If a member resigns during the year in which the member was appointed to serve, the position shall be considered vacant and not count in the voting membership until filled by a new appointment made by the Office of the Chancellor acting on the recommendation of the Reactor Facility Director.

12.2.3 Review Function

Responsibilities of the RAC shall include the following reviews required by TS 6.2.a. These reviews may be delegated to the Reactor Safety Subcommittee (RSS) as detailed in 12.2.5.1.

- Determine that proposed changes in the facility, and procedures, and conduct of tests or experiments allowed without prior authorization by the NRC, pursuant to 10 CFR 50.59;
- Reviewing all new procedures and major revisions thereto having safety significance, proposed changes to reactor facility equipment, or systems having safety significance. Changes to procedures that do not change their original intent may be approved without RAC review if approved by the TS-designated manager, either the Reactor HP Manager or Reactor Manager, or a designated alternate who is a member of Reactor HP or Senior Operator, respectively. All such changes to the procedures shall be documented, reviewed pursuant to 10 CFR 50.59, and subsequently reviewed by the RAC;
- Reviewing proposed experiments which are significantly different from any previously reviewed or which involve a question pursuant to 10 CFR 50.59;
- Reviewing proposed changes in the Technical Specifications or the license;
- Reviewing the circumstances of all reportable occurrences and violations of the Technical Specifications or license and the measures taken to prevent a recurrence;
- Reviewing violations of internal procedures or operating abnormalities having safety significance; and
- Reviewing reports from audits required by T.S. 6.2.e. These audits shall include:
    - Facility Operations, for conformance to the Technical Specifications and license conditions, at least annually;
    - Operator Requalification Program, for compliance with the approved program, at least every two (2) years;
    - Corrective Action items associated with reactor safety at least annually, and
    - Emergency Plan and implementing procedures, at least every two (2) years.

The RAC may independently explore policies and procedures as they relate to interaction with other administrative elements of MU and with clients of the reactor facility that are not part of the University.

It will respond to matters brought before it by the Reactor Facility Director, researchers, or other University administrative officials.

12.2.4 Audit Function

The TS required audits listed above shall be conducted by an individual or group without immediate responsibility in the area to be audited. Audit findings which affect reactor safety shall be immediately reported to the Reactor Facility Director. A written report of the findings shall be submitted to the Reactor Facility Director, the RAC and its subcommittees within three (3) months after the audit has been completed.

The charter for the RAC states the RAC shall review the TS required audits listed above (the RAC may authorize the RSS to perform the review of these TS required audits). The RAC reviews other licensing documents performed by the Reactor and Reactor HP Managers or their authorized delegates to verify the adequacy and the implementation of operating procedures and programs designed to ensure safe operation of the reactor facility and to ensure the protection of the health and safety of the public. Additional audits shall include, but are not limited to the following:

- Shipping Quality Assurance Program for Type B Shipping Casks;
- Physical Security Plan and Security Procedures;
- Operating Procedures (Reactor Operations and Reactor HP);
- Radiation Protection Program; and
- As Low As is Reasonably Achievable (ALARA) Program.

12.2.5 Subcommittees

The following five (5) subcommittees have been appointed by the Reactor Advisory Committee: Reactor Safety, Reactor Action, Reactor Safety Procedure Review, Isotope Use, and Isotope Use Procedure Review.

12.2.5.1 Reactor Safety Subcommittee

The Reactor Safety Subcommittee (RSS) may be authorized to act in behalf of the RAC in performing reviews of the following:

- Determinations that proposed changes in the facility and the conduct of tests or experiments are allowed without prior authorization by the NRC, pursuant to 10 CFR 50.59;
- Proposed reactor experiments significantly different from any previously reviewed or which involve a question pursuant to 10 CFR 50.59;
- Proposed changes in the Technical Specifications or the license;
- Circumstances of reportable occurrences and violations of the Technical Specifications or the license and the measures taken to prevent a recurrence;
- Violations of internal procedures or operating abnormalities having safety significance; and
- Reports from audits required by Technical Specification 6.2.e. These audits shall include:

- Facility Operations, for conformance to the Technical Specifications and license conditions, at least annually;
- Operator Requalification Program, for compliance with the approved program, at least every two (2) years;
- Corrective Action items associated with reactor safety at least annually, and
- Emergency Plan and implementing procedures, at least every two (2) years.

Upon completion of the review, the RSS shall make a recommendation concerning the proposed change, experiment, or remedial measure and report this recommendation to the Chairperson of the RAC, the Reactor Facility Director, and the Reactor Manager. When the review results in a negative recommendation, the RSS shall recommend alternatives and report this conclusion to the Chairperson of the RAC, the Reactor Facility Director, and the Reactor Manager. In the latter instance, the Reactor Manager shall apprise the Chairperson of the RSS of the course of action selected or submit a new proposal for review.

A subset of the RSS may act in behalf of the RAC as the Reactor Action Subcommittee (See Section 12.2.5.2).

The RSS shall consist of knowledgeable members from the public, student body, faculty, or the staff of MU. The RSS shall consist of no less than three (3) members with no more than one (1) student appointee.

The Chairperson of the RSS shall be appointed by the Chairperson of the RAC with the concurrence of the Committee upon recommendation from the Reactor Facility Director. The term of appointment shall be coincident with the term of appointment of the RAC. The Chairperson of the RSS shall be a member of the RAC. Members of the RSS shall be appointed by the Chairperson of the RAC upon recommendation from the Reactor Facility Director.

A meeting of the RSS shall not be deemed to satisfy the requirement of the RAC to meet at least once in each four (4) month period. A quorum of the RSS consisting of at least fifty percent of the appointed members shall be present at any meeting to conduct the business of the subcommittee. Independent actions of the RSS shall be reviewed by the RAC at the next regular meeting. Meetings will also be called as necessary by the RSS Chairperson, the Reactor Manager, the Reactor HP Manager or the Chairperson of the RAC. A majority of the voting membership is required to approve any business brought before the subcommittee. Meetings shall be conducted in accordance with *Robert's Rules of Order.* Dissemination of the minutes to the Office of the Chancellor, the RAC and RSS members shall be done within four (4) months after the meeting. The Chairperson of the RSS, or an individual designated by the Reactor Facility Director, will report actions of the RSS at the regular meetings of the RAC.

Requests submitted to the RSS for review may be acted upon by email or mail ballot. A request will be acted upon by ballot only if at least two-thirds of the ballots indicate approval without question. If the subcommittee believes that a request introduces a question pursuant to 10 CFR 50.59, it will refer the request to the RAC with a recommended disposition.

12.2.5.2 Reactor Action Subcommittee

The Reactor Action Subcommittee (RAS) may be authorized to act in behalf of the RAC in an advisory capacity to the Reactor Facility Director in matters that pertain to the safe operation of the reactor and that may require immediate consideration.

The RAS may be authorized to act in behalf of the RAC in performing reviews of:

- The circumstances of reportable occurrences and violations of the Technical Specifications or license and the measures taken to prevent a recurrence; and
- Violations of internal procedures or operating abnormalities having safety significance.

Upon completion of the review, the RAS shall make recommendations concerning the items above and report these recommendations to the Chairperson of the RAC, the Reactor Facility Director, and the Reactor Manager. When the review results in a negative recommendation, the RAS shall recommend alternatives and report this conclusion to the Chairperson of the RAC, the Reactor Facility Director, and the Reactor Manager. In the later instance, the Reactor Manager shall apprise the Chairperson of the RAS of the course of actions selected, or submit a new proposal for review.

The RAS is a subset of the Reactor Safety Subcommittee (RSS) and shall consist of no less than three (3) members of the RSS with no more than one (1) student appointed to each subcommittee. The Chairperson of the RSS shall serve as the Chairperson of the RAS. The term of appointment shall be coincident with the term of appointment of the RAC. The Chairperson of the RAS shall be a member of the RAC.

A meeting of the RAS shall not be deemed to satisfy the requirement of the RAC to meet at least once in each four (4) month period. A quorum of the RAS consisting of at least fifty percent of the appointed members shall be present at any meeting to conduct the business of the subcommittee. Independent actions of the RAS shall be reviewed by the RAC at the next regular meeting.

Meetings will also be called as necessary by the Reactor Manager, the Reactor HP Manager, or the Chairperson of the RAC. Meetings shall be conducted in accordance with *Robert's Rules of Order.* Dissemination of the minutes to the Office of the Chancellor, the RAC and the RAS members shall be done within four (4) months after the meeting. The Chairperson of the RAS, or an individual designated by the Reactor Facility Director, will report actions of the RAS at the regular meetings of the RAC.

12.2.5.3 Reactor Safety Procedure Review Subcommittee

The Reactor Safety Procedure Review Subcommittee (RSPRS) may be authorized to act in behalf of the RAC in performing reviews of the following:

- Determinations that proposed changes to procedures are allowed without prior authorization by the NRC, pursuant to 10 CFR 50.59;

- All new procedures and major revisions thereto having safety significance. Changes to procedures that do not change their original intent may be made without prior RAC review if approved by the Technical Specification-designated manager, either the Reactor HP Manager or Reactor Manager, or a designated alternate who is a member of Reactor HP or a SRO, respectively. All such changes to the procedures shall be documented, reviewed pursuant to 10 CFR 50.59, and subsequently reviewed by the RAC.

Specifically, the RSPRS may be required to review the following procedures, as described within TS 6.4.a:

- Startup, operation, and shutdown of the reactor;
- Fuel loading, unloading and movement within the reactor;
- Maintenance of major components of systems that could have an effect on reactor safety;
- Surveillance checks, calibrations and inspections that may affect reactor safety;
- Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity; and
- Implementation of the Emergency and Physical Security Plans.

The RSPRS consists of knowledgeable members from the public, student body, faculty, or the staff of MU. The RSS shall consist of no less than three (3) members with no more than one (1) student appointed to each subcommittee. The Reactor Manager and the Reactor HP Manager, or designated delegate, must be included in the membership. The Chairperson of the RSPRS shall be appointed by the Chairperson of the RAC with concurrence of the Committee upon recommendation from the Reactor Facility Director. The term of appointment shall be coincident with the term of appointment of the RAC. The Chairperson of the RSPRS shall be a member of the RAC. Members of the RSPRS shall be appointed by the Chairperson of the RAC upon recommendation from the Reactor Facility Director.

A meeting of the RSPRS shall not be deemed to satisfy the requirement of the RAC to meet at least once in each four (4) month period. A quorum of the RSPRS consisting of at least fifty percent of the appointed members shall be present at any meeting to conduct the business of the subcommittee. Independent actions of the RSPRS shall be reviewed by the RAC at the next regular meeting. Meetings will also be called as necessary by the RSPRS Chairperson, the Reactor Manager, the Reactor HP Manager or the Chairperson of the RAC. Meetings shall be conducted in accordance with *Robert's Rules of Order*. Dissemination of the minutes to the Office of the Chancellor, the RAC and RSPRS members shall be done within four (4) months after the meeting. The Chairperson of the RSPRS, or an individual designated by the Reactor Facility Director, will report the actions of the RSPRS at the regular meetings of the RAC.

Requests submitted to the RSPRS for review may be acted upon by email or mail ballot. A request will be acted upon by ballot only if at least two-thirds of the ballots indicate approval without question. If the subcommittee believes that a request introduces a question pursuant to 10 CFR 50.59, it will refer the request to the RAC with a recommended disposition.

12.2.5.4 Isotope Use Subcommittee

The Isotope Use Subcommittee (IUS) may be authorized to act in behalf of the RAC in performing reviews of the following:

- Matters relating to custody and use of radiation and radioisotope experiments within MURR;
- Determinations that proposed changes in the radiation and radioisotope experiments are allowed without prior authorization by the NRC, pursuant to 10 CFR 50.59;
- Reviewing radiation safety for project applications for use of radiation sources at MURR covered by Renewed Facility Operating License No. R-103 and not covered under the MURR Broad Scope Materials License;
- Proposed changes in the Technical Specifications or the license, if applicable;
- Circumstances of reportable occurrences and violations of the Technical Specifications or the license and the measures taken to prevent a recurrence, if applicable;
- Violations of internal procedures or operating abnormalities having radiological safety significance; and
- Reports from audits required by 10 CFR 20.1101(c).

Upon completion of the review, the IUS shall make a recommendation concerning the items above and report this recommendation to the Chairperson of the RAC, the Reactor Facility Director, and the Reactor HP Manager. When the review results in a negative recommendation, the IUS shall recommend alternatives and report this conclusion to the Chairperson of the RAC, the Reactor Facility Director, and the Reactor HP Manager. In the latter instance, the Reactor HP Manager shall apprise the Chairperson of the IUS of the course of action selected or submit a new proposal for review.

The IUS shall consist of knowledgeable members from the public, student body, faculty, or the staff of MU. The RSS shall consist of no less than three (3) members with no more than one (1) student appointed to each subcommittee. The Chairperson of the IUS shall be appointed by the Chairperson of the RAC with concurrence of the Committee upon recommendation from the Reactor Facility Director. The term of appointment shall be coincident with the term of appointment of the RAC. The Chairperson of the IUS shall be a member of the RAC. Members of the IUS shall be appointed by the Chairperson of the RAC upon recommendation from the Reactor Facility Director.

A meeting of the IUS shall not be deemed to satisfy the requirement of the RAC to meet at least once in each four (4) month period. A quorum of the IUS consisting of at least fifty percent of the appointed members shall be present at any meeting to conduct the business of the subcommittee. Independent actions of the IUS shall be reviewed by the RAC at the next regular meeting.

Meetings will also be called as necessary by the IUS Chairperson, the Reactor Manager, the Reactor HP Manager or the Chairperson of the RAC. Meetings shall be conducted in accordance with *Robert's Rules of Order.* Dissemination of the minutes to the Office of the Chancellor, the RAC and IUS members shall be done within four (4) months after the meeting. The Chairperson of the IUS, or an individual

designated by the Reactor Facility Director, will report actions of the IUS at the regular meetings of the RAC.

Requests submitted to the IUS for review may be acted upon by email or mail ballot. A request will be acted upon by ballot only if at least two-thirds of the ballots indicate approval without question. If the subcommittee believes that a request introduces a question pursuant to 10 CFR 50.59, it will refer the request to the RAC with a recommended disposition.

12.2.5.5 Isotope Use Procedure Review Subcommittee

The Isotope Use Procedure Review Subcommittee (IUPRS) may be authorized to act in behalf of the RAC in performing reviews of the following:

- Determinations that proposed changes to procedures are allowed without prior authorization by the NRC, pursuant to 10 CFR 50.59;
- All new procedures and major revisions thereto having safety significance. Changes to procedures that do not change their original intent may be made without prior RAC review if approved by the Technical Specification-designated manager, either the Reactor HP Manager or Reactor Manager, or a designated alternate who is a member of Reactor HP or a SRO, respectively. All such changes to the procedures shall be documented, reviewed pursuant to 10 CFR 50.59, and subsequently reviewed by the RAC.

Specifically, the IUPRS may be required to review the following procedures, as described within TS 6.4.b:

- Radiological control;
- Preparation for shipping byproduct material; and
- Shipping of byproduct material produced under the facility operating license.

The IUPRS shall consist of knowledgeable members from the public, student body, faculty, or the staff of MU. The IUPRS shall consist of no less than three (3) members with no more than one (1) student appointed to each subcommittee. The Reactor Manager and the Reactor HP Manager, or designated delegate, must be included in the membership. The Chairperson of the IUPRS shall be appointed by the Chairperson of the RAC with concurrence of the Committee upon recommendation from the Reactor Facility Director. The term of appointment shall be coincident with the term of appointment of the RAC. The Chairperson of the IUPRS shall be a member of the RAC. Members of the IUPRS shall be appointed by the Chairperson of the RAC upon recommendation from the Reactor Facility Director.

The Chairperson of the IUPRS may request the Reactor Safety Procedure Review Committee (RSPRS) perform a courtesy review of any procedure brought before the IUPRS if deemed necessary to ensure an adequate review. Conflicts in selection of the more appropriate subcommittee for review of a procedure will be decided by mutual agreement of the Chairpersons of the IUPRS and RSPRS.

A meeting of the IUPRS shall not be deemed to satisfy the requirement of the RAC to meet at least once in a four (4) month period. A quorum of the IUPRS consisting of at least fifty percent of the appointed

members shall be present at any meeting to conduct the business of the subcommittee. Independent actions of the IUPRS shall be reviewed by the RAC at the next regular meeting.

Meetings will also be called as necessary by the IUPRS Chairperson, the Reactor Manager, the Reactor HP Manager or the Chairperson of the RAC. Meetings shall be conducted in accordance with *Robert's Rules of Order*. Dissemination of the minutes to the Office of the Chancellor, the RAC and IUPRS members shall be done within four (4) months after the meeting. The Chairperson of the IUPRS, or an individual designated by the Reactor Facility Director, will report actions of the IUPRS at the regular meetings of the RAC.

Requests submitted to the IUPRS for review may be acted upon by email or mail ballot. A request will be acted upon by ballot only if at least two-thirds of the ballots indicate approval without question. If the subcommittee believes that a request introduces a question pursuant to 10 CFR 50.59, it will refer the request to the RAC with a recommended disposition.

12.3 Procedures

Written procedures shall be in effect for the Reactor Operations and Reactor HP Branches. These procedures provide detailed guidance in the operation and utilization of the reactor and the laboratory facilities and shall be adequate to assure the safe operation of the reactor, the protection of the health and safety of the general public and the staff at the facility, and the protection of the environment (Reference 12.9).

12.3.1 Reactor Operations

Reactor operating procedures provide methods and guidelines for operation of the reactor and associated systems to ensure safety and performance within the limits of the Technical Specifications. Changes to these procedures, and to any other special operating or maintenance procedures which have safety significance, must be reviewed by the RSPRS prior to the approval by the Reactor Manager. Changes which are editorial or have no safety significance may be made by the Reactor Manager, or his authorized delegate, but must be documented and subsequently reviewed by the RSPRS. Written procedures shall be in effect for operation of the reactor, including the following:

- Start-up, operation, and shutdown of the reactor;
- Fuel loading, unloading, and movement in the reactor;
- Maintenance of major components of systems that could have an effect on reactor safety;
- Surveillance checks, calibrations and inspections that may affect reactor safety;
- Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity; and
- Implementation of the Emergency and Physical Security Plans.

The Reactor Manager shall approve the procedures for normal operations of the reactor and the Emergency Plan implementing procedures. Deviations from reactor operations procedures may be

enacted by an SRO. Such deviations shall be documented, reviewed pursuant to 10 CFR 50.59, and reported within 24 hours or the next working day to the Reactor Manager or designated alternate.

12.3.2 Reactor Health Physics

The Reactor HP Manager shall be responsible for the implementation of the Radiation Protection Program. The requirements of the Radiation Protection Program are established in 10 CFR 20. The program should use the guidelines of American National Standard "Radiation Protection at Research Reactor Facilities," ANSI/ANS-15.11-1993 (R2004) (Reference 12.3).

Reactor HP operating procedures provide methods and guidelines for the implementation and maintenance of the Radiation Protection Program. This program has been established to protect the health and safety of MURR staff, research associates, students, and the general public. Changes to these procedures are made by the Reactor HP Manager, or his authorized delegate, and are subsequently reviewed by the IUPRS. Written procedures shall be in effect for radiological control, and the preparation for shipping and the shipping of byproduct material produced under the facility operating license.

The Reactor HP Manager shall approve the radiological control procedures and the procedures for preparation for shipping and the shipping of byproduct material. Deviations from these procedures may be enacted by a member of Reactor HP. Such deviations shall be documented, reviewed pursuant to 10 CFR 50.59, and reported within 24 hours or the next working day to the Reactor HP Manager or designated alternate.

12.4 Reportable Events and Required Actions

The following incidents and conditions relating to the operation of the reactor require that the NRC be informed (TS 6.6). Abnormal Occurrences which are considered reportable events also require certain actions prior to returning the reactor to its normal condition. These actions are outlined below. (References 12.7 and 12.8)

12.4.1 Safety Limit Violation

If a safety limit, as defined by the Technical Specifications, is violated, cessation of reactor operations is required until resumption is authorized by the NRC pursuant to 10 CFR 50.36(c)(1). The safety limit violation shall be promptly reported to the Reactor Manager and Reactor Facility Director, or designated alternates.

The safety limit violation shall be promptly reported to the NRC. Prompt reporting of the violation shall be made by MU, by telephone and subsequently confirmed in writing or email, to the NRC Operations Center no later than the following working day. A detailed follow-up report shall be submitted within fourteen (14) days to the NRC Document Control Desk. The report shall include: the circumstances leading to the violation including, when known, the causes and contributing factors; date and approximate time of the occurrence; effect of the violation upon the reactor and associated systems;

effect of the violation on the health and safety of the facility staff and general public; and the corrective actions to prevent recurrence.

12.4.2 Release of Radioactivity

Should a release of radioactivity of greater than allowable limits occur from the reactor facility boundary, reactor conditions shall be returned to normal operation or the reactor shall be shut down. The release of radioactivity shall be promptly reported to the Reactor Manager and Reactor Facility Director, or designated alternates. The release of radioactivity shall be promptly reported to the NRC. Prompt reporting of the violation shall be made by MU, by telephone and subsequently confirmed in writing or email, to the NRC Operations Center no later than the following working day. If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed until authorized by the Reactor Facility Director, or designated alternate. A detailed follow-up report shall be prepared. The follow-up report shall be submitted within fourteen (14) days to the NRC Document Control Desk,

12.4.3 Other Reportable Occurrences

In the event an Abnormal Occurrence, as defined by Technical Specification 1.1, the following actions shall be taken. The Abnormal Occurrence shall be promptly reported to the Reactor Manager and the Reactor Facility Director, or designated alternates. The reactor shall be shut down or placed in a safe condition. Return to normal operation shall not be allowed until authorized by the Reactor Facility Director, or designated alternate. A prompt report of the Abnormal Occurrence to the NRC with a subsequent detailed follow-up report is required. Prompt reporting of the Abnormal Occurrence shall be made to the NRC Operations Center by telephone no later than the following working day. The follow-up report shall be submitted to the NRC Document Control Desk within fourteen (14) days.

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

12.4.4 Other Reports

A written report shall be submitted to the NRC Document Control Desk within thirty (30) days of:

- Any significant change(s) in the transient or accident analyses as described in the SAR; and
- Permanent changes in the facility organization involving the Office of the Chancellor or the Reactor Facility Director.

12.4.5 Annual Report

Annual reports detailing the activities of the reactor facility in connection with the operation of the reactor will be submitted to the NRC Document Control Desk within sixty (60) days following each calendar year. Each annual report shall include the following information:

- A brief narrative summary including:
    - Operating experience (including operations designed to measure reactor characteristics);
    - Changes in the reactor facility design, performance characteristics, and operating procedures related to reactor safety occurring during the reporting period; and
    - Results of surveillance tests and inspections;
- A tabulation showing the energy generated by the reactor (in megawatt-days);
- The number of emergency shutdowns and inadvertent scrams (unscheduled shutdowns), including the reason therefore and corrective actions, if any, taken;
- Discussion of the major maintenance operations performed during the period, including the effects, if any, on the safe operation of the reactor;
- A summary of changes to the facility and procedures, and conduct of tests or experiments carried out under the conditions of 10 CFR 50.59;
- A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge;
- A description of any environmental surveys performed outside the reactor facility; and
- A summary of radiation exposures received by facility staff, experimenters, and visitors, including the dates and times of significant exposure, and a brief summary of the results of radiation and contamination surveys performed within the facility.

12.5 Records

Records of the following activities shall be maintained and retained for the periods specified below in accordance with TS 6.7. The records may be in the form of logs, data sheets, or other suitable forms or documents. The required information may be contained in single or multiple records, or a combination thereof. (Reference 12.9)

12.5.1 Lifetime Records

The following records are to be retained for the lifetime of the reactor facility: (Note: Applicable annual reports, if they contain all of the required information, may be used as records in this section.)

- Gaseous and liquid radioactive effluents released to the environs;
- Off-site environmental-monitoring surveys required by the Technical Specifications;
- Radiation exposure for all monitored personnel;
- Updated drawings of the reactor facility; and
- Reviews and reports pertaining to a violation of a safety limit, limiting safety system setting, or limiting conditions for operations.

12.5.2 Five Year Records

The following records are to be maintained for a period of at least five (5) years or for the life of the component involved, whichever is shorter:

- Normal reactor facility operation (but not including supporting documents such as checklists, log sheets, etc. which shall be maintained for a period of at least one year);
- Principal maintenance operations;
- Reportable occurrences;
- Surveillance activities required by the Technical Specifications;
- Reactor facility radiation and contamination surveys required by applicable regulations;
- Experiments performed with the reactor;
- Fuel inventories, receipts, and shipments;
- Approved changes to operating procedures; and
- Records of meetings and audit reports of the review and audit group.

12.5.3 Operator Licensing Records

Records of retraining and requalification of ROs and SROs who are licensed pursuant to 10 CFR 55 shall be retained at all times the individual is employed or until the license is renewed.

12.6 Emergency Planning

The MURR Emergency Plan contains a detailed description of the elements of advanced planning to contend with emergency situations connected with the operation of the reactor facility. The Emergency Plan focuses primarily on situations that may cause or may threaten to cause radiological hazards affecting the health and safety of the MU staff or the general public. It outlines the objectives that are met by the Emergency Plan Implementing Procedures and defines the authority and the responsibilities in order to achieve these objectives. (Reference 12.4)

Emergency preparedness for the reactor facility is the responsibility of the Reactor Manager and is maintained by the following programs:

Training

MURR staff and users annually receive training in radiation safety and emergency procedures. Emergency Support Organizations (e. g., MU Police, the MU Hospital and Clinics, City of Columbia Fire Department, etc.) shall biennially be invited to train for their role in maintaining emergency preparedness;

Drills

An annual on-site emergency drill shall be conducted as an action drill, with each required emergency measure being executed as realistically as is reasonably possible, including the use of appropriate emergency equipment. At least every two years, the drill shall contain provisions for coordination with

emergency support personnel and should test, as a minimum, the communication links and notification procedures with the Emergency Support Organizations;

Equipment Maintenance

The operational readiness of emergency equipment and supplies required by the Emergency Plan Implementing Procedures shall be maintained, calibrated, tested, and periodically inventoried as detailed in the Equipment Maintenance Procedure. The Equipment Maintenance Procedure shall cover detailed requirements such as the required inventory of emergency supplies to be maintained at designated readily accessible locations; and

Emergency Plan Review and Update

The Emergency Plan and the Emergency Plan Implementing Procedures shall be annually reviewed and revised as necessary. The revisions will be reviewed and approved in accordance with the Technical Specifications. The Reactor Manager shall provide for any necessary retraining needed due to a revision in the Emergency Plan or its Implementing Procedures.

The MURR Emergency Plan is written using the guidelines of Regulatory Guide 2.6, Rev.2 (2017). This regulatory guide (RG) provides licensees and applicants with a method that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in complying with the regulations on the content of emergency plans for research and test reactors and other non-power production and utilization facilities licensed under Title 10 of the Code of Federal Regulations (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities" (10 CFR Part 50).

12.7 Security Planning

The MURR Physical Security Plan describes the physical protection system and the security organization which will detect the attempted theft or theft of Special Nuclear Material (SNM) at MURR. It outlines the objectives that are met by the Security Procedures, a separate document which describes the security requirements and security measures for the reactor facility.

The Reactor Facility Director or his designated representative has overall responsibility for the initiation and implementation of the Physical Security Plan.

12.8 Quality Assurance

The MURR Shipping Quality Assurance (QA) Program for Type B Shipping Casks (POL-14) describes the use, testing, maintenance, and repair of shipping containers identified by Title 10, Chapter I of the Code of Federal Regulations, Part 71 (10 CFR 71). Any activity which could significantly affect the ability of such a structure, system, or component to perform safely and as specified falls within the scope of the QA Program. Shipping casks covered under 10 CFR 71 will be released for shipping only after they have satisfactorily met the requirements of the QA Program.

The Type B Shipping QA Program Manager is responsible for the Shipping QA Program for Type B Shipping Casks. The program shall be annually reviewed and revised as necessary.

12.9 Operator Training and Requalification

The MURR Operator Requalification Program is designed to provide assurance that all operators certified at the RO and SRO levels, pursuant to 10 CFR 55, maintain competence and proficiency in all aspects of licensed activities. The objectives of the program are to review/retrain in areas of infrequent operation, to review facility and procedural changes, to address subject matter not reinforced by direct use, and to improve in areas of performance by direct use and to improve in areas of performance weakness.

The MURR Operator Requalification Program uses Reference 12.2 as a guide and is divided into the following main components:

- Written Examinations;
- On-The-Job Training;
- Operating Tests; and
- Documented Review of Changes.

A biennial written examination is given to each licensed operator to verify the individual's knowledge level in the categories mentioned below. The examinations will be of a scope and complexity equivalent to the licensing examinations administered by the NRC. The results of the examination shall provide the basis for a determination of those areas in which an operator needs retraining. Preplanned lectures shall be used to retrain those operators who demonstrate deficiencies in any part of the examination. The examination shall contain questions from each of the following categories as described in References 12.2 and 12.5:

- Reactor Theory, Thermodynamics, and Facility Operating Characteristics;
- Normal and Emergency Procedures, and Radiological Controls; and
- Facility and Reactor Plant, and Radiation Monitoring Systems.

The minimum acceptance score in any one category and on the entire examination shall be established. Failure in one category will require retraining the operator until a satisfactory passing grade is attained in that category. Failure of the entire test will place the operator in an accelerated training program until retraining results in a satisfactory passing of the re-examination. Furthermore, the individual will be removed from licensed activities until the written re-examination is passed.

On-the-job training consists of performing evolutions which are typically accomplished only by licensed operators. These evolutions include plant control manipulations and plant evolutions (e.g., start-ups, shutdowns, significant reactivity changes, etc.) required by 10 CFR 55.59 (c) (3). On-the-job training provides assurance that (1) the operator maintains their competence in manipulating the plant controls and in operating all apparatuses and mechanisms required by their license, and (2) that they have a thorough understanding of all emergency procedures.

An annual operating test is given to each RO or SRO in order to demonstrate an understanding of, and ability to perform, the actions necessary to accomplish a broad sample of applicable items specified in 10 CFR 55.45 (a) (2) through (13).

Documented reviews ensure that all licensed individuals are cognizant of all design, procedural, Technical Specifications, and facility operating license changes. The operators sign an attached review sheet indicating that the documents describing these changes have been read and understood.

All licensed operators shall participate in the Requalification Program. The Requalification Program is conducted over a period not to exceed two (2) years and is to be followed by successive two-year programs. To maintain active status as defined by 10 CFR 55.53 (e), each licensed operator shall actively perform the functions of an RO or SRO for a minimum of four (4) hours per calendar quarter. For ROs, these functions include refueling, reactor start-ups, or time spent as a console operator. For SROs, these functions include directing refueling, start-ups, and shift activities. If a licensed operator has not been actively performing the functions of an RO or SRO, the Reactor Manager shall verify that the operator's license is current and valid. The operator shall complete a minimum of six (6) hours of operation, under the supervision of an operator or senior operator, as appropriate, covering the functions described above.

12.10 Environmental Reports

NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Nuclear Reactors," identifies that license renewal of research reactors is an action that requires an Environmental Assessment (EA). An Environmental Report (ER) was prepared by MURR staff to aid the NRC in preparing the EA to meet the requirements of the National Environmental Protection Act of 1969, as amended.

Regulatory guidance for the preparation of ERs for research reactors is minimal. However, the guidance originating from the renewal of operating licenses for nuclear power plants is significant and comprehensive. To provide a thorough and comprehensive ER, MURR staff used the guidance provided by the following documents:

- "Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses," Regulatory Guide 4.2, Supplement 1, September 2000;
- "Standard Review Plans for Environmental Reviews for Nuclear Power Plants," NUREG-1555, Supplement 1, September 2000; and
- "Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants," NUREG-1437.

The ER supported the application for a twenty-year renewal of the Class 104c Amended Facility License No. R-103 (NRC Docket No. 50-186). The NRC Renewed Facility Operating License R-103 by letter dated January 4, 2017. NRC rulemaking, effective January 29, 2025, eliminated term limits for Class 104(c) research reactors, such as MURR.

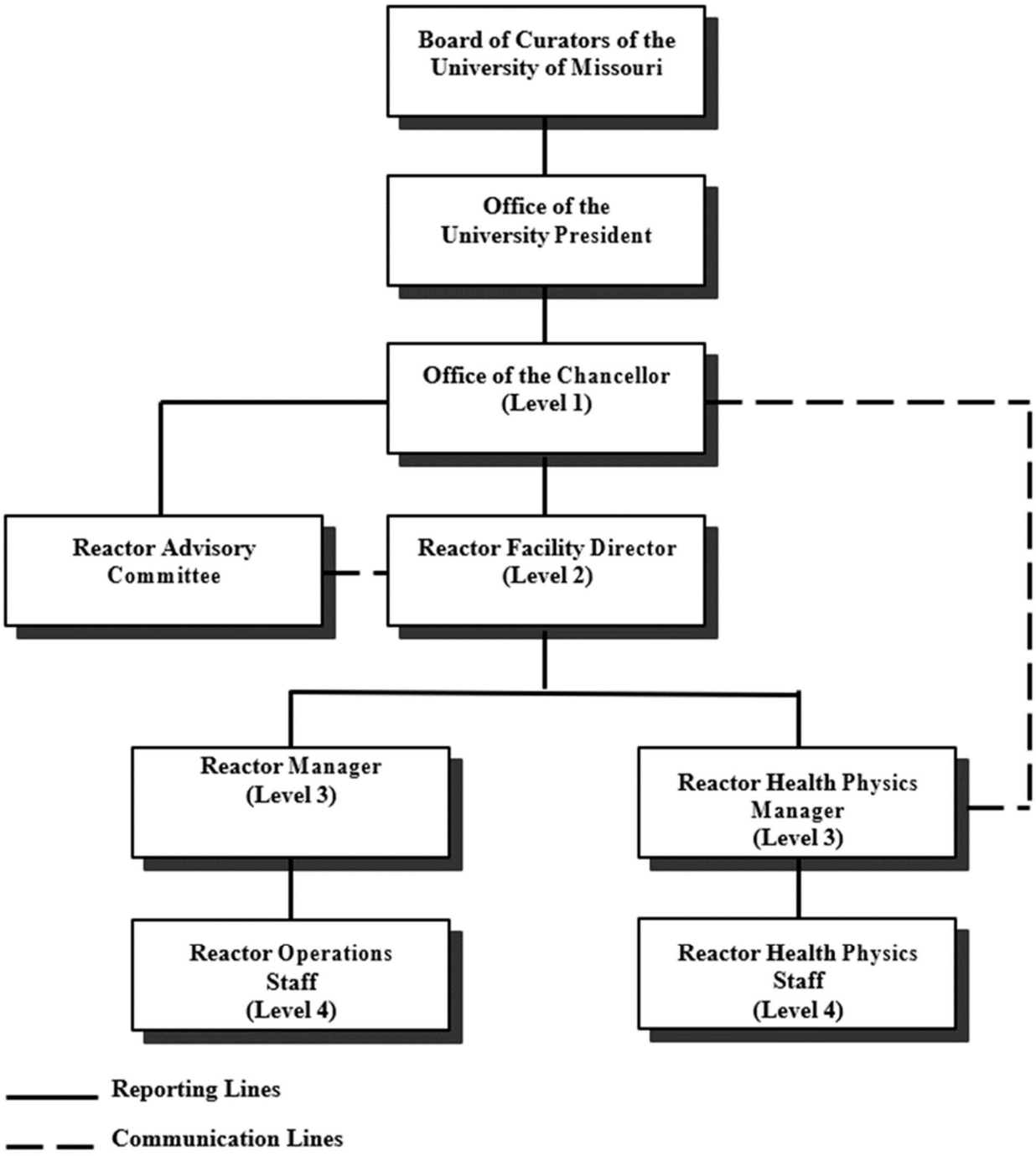


FIGURE 12.1 – UNIVERSITY OF MISSOURI RESEARCH REACTOR (MURR) ORGANIZATION

CHAPTER 13

ACCIDENT ANALYSES

TABLE OF CONTENTS

13.0 ACCIDENT ANALYSES 11
13.1 Introduction 11
13.2 Accident-Initiating Events and Scenarios, Accident Analyses, and the Determination of Consequences 12
13.2.1 Maximum Hypothetical Accident 12
13.2.1.1 Accident-Initiating Events and Scenarios 12
13.2.1.2 Accident Analysis and Consequences 12
13.2.1.3 Conclusions 25
13.2.2 Insertion of Excess Reactivity 25
13.2.2.1 Accident-Initiating Events and Scenarios 25
13.2.2.1.1 Rapid Step Insertions of Positive Reactivity 25
13.2.2.1.2 Continuous Control Blade Withdrawal 30
13.2.2.2 Conclusions 31
13.2.3 Loss of Primary Coolant 32
13.2.3.1 Accident-Initiating Events and Scenarios 32
13.2.3.2 Accident Analysis and Consequences 34
13.2.3.2.1 Rupture of Piping Between Valve V507B and the Reactor Pool 36
13.2.3.2.2 Rupture of Piping Between Valve V507A and the Reactor Pool 41
13.2.3.3 Conclusions 42
13.2.4 Loss of Primary Coolant Flow 42
13.2.4.1 Accident-Initiating Events and Scenarios 42
13.2.4.2 Accident Analysis and Consequences 43
13.2.4.3 Conclusions 46
13.2.5 Mishandling or Malfunction of Fuel 46
13.2.5.1 Fuel Failure during Reactor Operation 46
13.2.5.1.1 Accident-Initiating Events and Scenarios 46
13.2.5.1.2 Accident Analysis and Consequences 47
13.2.5.2 Fuel Handling Accident (FHA) 60
13.2.5.2.1 Accident-Initiating Events and Scenarios 60
13.2.5.2.2 Accident Analysis and Consequences 60
13.2.5.3 Fuel Element Malfunction Due to Cladding Deformation or Corrosion 71
13.2.5.4 Conclusions 73
13.2.6 Experiment Malfunction 74
13.2.6.1 Accident-Initiating Events and Scenarios 74
13.2.6.2 Accident Analysis and Consequences 74
13.2.6.3 Conclusions 76
13.2.7 Loss of Electrical Power 76
13.2.7.1 Accident-Initiating Events and Scenarios 76
13.2.7.2 Accident Analysis and Consequences 77
13.2.7.2.1 Power Loss with the Reactor Operating 77

13.2.7.2.1.1 Reactor Control System 77
13.2.7.2.1.2 Instrumentation and Control (I&C) Systems 77
13.2.7.2.1.3 Reactor Process Systems 77
13.2.7.2.1.4 Containment Building Ventilation Isolation Doors 504 and 505 78
13.2.7.2.1.5 Emergency Air Compressor 78
13.2.7.2.1.6 Truck Entry Door 101 78
13.2.7.2.1.7 Personnel Air Lock Doors 276 and 277 78
13.2.7.2.1.8 Facility Ventilation Exhaust Fans (EF-13 and EF-14) 78
13.2.7.2.1.9 Reactor and Laboratory Corridor and Exit Lights 78
13.2.7.2.1.10 Fan Failure Alarm System 79
13.2.7.2.1.11 Intercommunication System 79
13.2.7.2.1.12 Evacuation/Isolation Alarm System 79
13.2.7.2.1.13 Diesel Generator Room Distribution Panel 79
13.2.7.2.1.14 Nitrogen Station 79
13.2.7.2.1.15 Fire Protection System 79
13.2.7.2.2 Power Loss with the Reactor Shutdown 80
13.2.7.2.2.1 Reactor Control System 80
13.2.7.2.2.2 Personnel within the Reactor Containment Building 80
13.2.7.2.2.3 Reactor Containment Building 80
13.2.7.3 Conclusions 80
13.2.8 External Events 80
13.2.8.1 Accident-Initiating Events and Scenarios 80
13.2.8.2 Accident Analysis and Consequences 81
13.2.8.3 Conclusions 81
13.2.9 Mishandling or Malfunction of Equipment 82
13.2.9.1 Leak in the Pool Coolant System 82
13.2.9.1.1 Accident-Initiating Events and Scenarios 82
13.2.9.1.2 Accident Analysis and Consequences 83
13.2.9.1.3 Conclusions 83
13.2.9.2 Shearing of a Beamport 84
13.2.9.2.1 Accident-Initiating Events and Scenarios 84
13.2.9.2.2 Accident Analysis and Consequences 85
13.2.9.2.3 Conclusions 85
13.2.9.3 Failure of In-Pool Heat Exchanger Isolation Valves to Open 86
13.2.9.3.1 Accident-Initiating Events and Scenarios 86
13.2.9.3.2 Conclusions 86
13.2.9.4 High-Pressure Transient 86
13.2.9.4.1 Accident-Initiating Events and Scenarios 87
13.2.9.4.2 Accident Analysis and Consequences 87
13.2.9.4.3 Conclusions 88
13.2.9.5 Failure of the Neutron Startup Source 88
13.2.9.5.1 Accident-Initiating Events and Scenarios 89

13.2.9.5.2 Accident Analysis and Consequences 89
13.2.9.5.3 Conclusions 90

LIST OF TABLES

Table 13-1 RADIOIODINE AND NOBLE GAS ACTIVITIES IN A 5-GRAM LEU TARGET 13
Table 13-2 RADIOIODINE CONCENTRATIONS IN THE POOL WATER FROM A 5-GRAM LEU TARGET 14
Table 13-3 AVERAGE RADIOIODINE CONCENTRATIONS IN THE CONTAINMENT BUILDING AIR DURING THE 5 MINUTES FROM A 5-GRAM LEU TARGET 14
Table 13-4 AVERAGE NOBLE GAS CONCENTRATIONS IN THE CONTAINMENT BUILDING AIR DURING THE 5 MINUTES FROM A 5-GRAM LEU TARGET 15
Table 13-5 DERIVED AIR CONCENTRATION VALUES AND 5-MINUTE EXPOSURES – RADIOIODINE – FROM A 5-GRAM LEU TARGET 16
Table 13-6 DERIVED AIR CONCENTRATION VALUES AND 5-MINUTE EXPOSURES – NOBLE GASES – FROM A 5-GRAM LEU TARGET 17
Table 13-7 5-MINUTE DOSE FROM RADIOIODINES AND NOBLE GASES IN THE CONTAINMENT BUILDING FROM A 5-GRAM LEU TARGET 18
Table 13-8 RADIATION SHINE THROUGH THE CONTAINMENT BUILDING FROM A 5-GRAM LEU TARGET 18
Table 13-9 RADIOIODINE CONCENTRATIONS IN CONTAINMENT AIR DURING THE FIRST HOUR FROM A 5-GRAM LEU TARGET 20
Table 13-10 AVERAGE NOBLE GAS CONCENTRATIONS IN THE CONTAINMENT AIR DURING THE FIRST HOUR FROM A 5-GRAM LEU TARGET 21
Table 13-11 AVERAGE RADIOIODINE CONCENTRATIONS IN AIR LEAKING FROM CONTAINMENT AND EXITING THE EXHAUST STACK OVER THE 16.5 HOUR PERIOD FROM A 5-GRAM LEU TARGET 22
Table 13-12 AVERAGE NOBLE GAS CONCENTRATIONS IN AIR LEAKING FROM CONTAINMENT AND EXITING THE EXHAUST STACK OVER THE 16.5 HOUR PERIOD FROM A 5-GRAM LEU TARGET 22
Table 13-13 EFFLUENT CONCENTRATION LIMITS, CONCENTRATIONS AT POINT OF MAXIMUM CONCENTRATION AND RADIATION DOSES IN THE UNRESTRICTED AREA – RADIOIODINE – FROM A 5-GRAM LEU TARGET 24
Table 13-14 EFFLUENT CONCENTRATION LIMITS, CONCENTRATIONS AT POINT OF MAXIMUM CONCENTRATION, AND RADIATION DOSES IN THE UNRESTRICTED AREA – NOBLE GASES – FROM A 5-GRAM LEU TARGET 24
Table 13-15 DOSE FROM RADIOIODINES AND NOBLE GASES IN THE UNRESTRICTED AREA FROM A 5-GRAM LEU TARGET 25
Table 13-16 PEAKING FACTORS IN THE HOT AND AVERAGE CHANNELS 27
Table 13-17 NORMAL REACTOR OPERATING CONDITIONS AND CONSERVATIVE ASSUMPTIONS 32
Table 13-18 MINIMUM LIQUID FRACTION IN ALL FOUR VOLUMES OF COOLANT CHANNELS 2, 3, 4 & 5 AT THE 0.5-SECOND POINT 36
Table 13-19 HEAT TRANSFER MODES IN COOLANT CHANNELS 3 & 4 AT THE 0.5-SECOND POINT 37

Table 13-20 MINIMUM LIQUID FRACTION IN ALL FOUR VOLUMES OF COOLANT CHANNELS 2, 3, 4 & 5 AT OR BEFORE THE 4.0-SECOND POINT 38
Table 13-21 HEAT TRANSFER MODES IN COOLANT CHANNELS 3, 4 & 5 AT THE 4.0-SECOND POINT 39
Table 13-22 MINIMUM LIQUID FRACTION IN ALL FOUR VOLUMES OF COOLANT CHANNELS 11, 12, 13 & 23 AT THE 13.0-SECOND POINT 39
Table 13-23 RADIOIODINE AND NOBLE GAS ACTIVITIES IN THE CORE FROM FUEL FAILURE 48
Table 13-24 RADIOIODINE AND NOBLE GAS ACTIVITIES RELEASED INTO THE POOL AFTER 10 MINUTES FOLLOWING FUEL FAILURE 49
Table 13-25 RADIOIODINE CONCENTRATIONS IN THE POOL WATER AT 10 MINUTES FOLLOWING FUEL FAILURE 50
Table 13-26 AVERAGE RADIOIODINE CONCENTRATIONS IN THE CONTAINMENT BUILDING AIR DURING THE 10 MINUTES FOLLOWING FUEL FAILURE 51
Table 13-27 AVERAGE NOBLE GAS CONCENTRATIONS IN THE CONTAINMENT BUILDING AIR DURING THE 10 MINUTES FOLLOWING FUEL FAILURE 51
Table 13-28 DERIVED AIR CONCENTRATION VALUES AND 10-MINUTE EXPOSURES – RADIOIODINE – FOLLOWING FUEL FAILURE 52
Table 13-29 DERIVED AIR CONCENTRATION VALUES AND 10-MINUTE EXPOSURES – NOBLE GASES – FOLLOWING FUEL FAILURE 54
Table 13-30 10-MINUTE DOSE FROM RADIOIODINES AND NOBLE GASES IN THE CONTAINMENT BUILDING FOLLOWING FUEL FAILURE 54
Table 13-31 RADIATION SHINE THROUGH THE CONTAINMENT BUILDING FOLLOWING FUEL FAILURE 55
Table 13-32 RADIOIODINE CONCENTRATIONS IN AIR LEAKING FROM CONTAINMENT AND EXITING THE EXHAUST STACK DURING THE FURST HOUR FOLLOWING FUEL FAILURE 55
Table 13-33 NOBLE GAS CONCENTRATIONS IN AIR LEAKING FROM CONTAINMENT AND EXITING THE EXHAUST STACK IN THE FIRST HOUR FOLLOWING FUEL FAILURE 56
Table 13-34 AVERAGE RADIOIODINE CONCENTRATIONS IN AIR LEAKING FRON CONTAINMENT AND EXITING THE EXHAUST STACK FOLLOWING FUEL FAILURE 56
Table 13-35 NOBLE GAS CONCENTRATIONS IN AIR LEAKING FROM CONTAINMENT AND EXITING THE EXHAUST STACK FOLLOWING FUEL FAILURE 56
Table 13-36 EFFLUENT CONCENTRATION LIMITS, CONCENTRATIONS AT POINT OF MAXIMUM CONCENTRATION AND RADIATION DOSES IN THE UNRESTRICTED AREA – RADIOIODINE – FOLLOWING FUEL FAILURE 58
Table 13-37 EFFLUENT CONCENTRATION LIMITS, CONCENTRATIONS AND POINT OF MAXIMUM CONCENTRATION, AND RADIATION DOSES IN THE UNRESTRICTED AREA – NOBLE GASES – FOLLOWING FUEL FAILURE 59
Table 13-38 DOSE FROM RADIOIODINES AND NOBLE GASES IN THE UNRESTRICTED AREA FOLLOWING FUEL FAILURE 59
Table 13-39 RADIOIODINE AND NOBLE GAS ACTIVITIES IN THE CORE AFTER 30-MINUTE DECAY FROM FHA 61
Table 13-40 RADIOIODINE AND NOBLE GAS ACTIVITIES RELEASED INTO THE POOL FROM FHA 62
Table 13-41 RADIOIODINE CONCENTRATIONS IN THE POOL WATER FROM FHA 63

Table 13-42 AVERAGE RADIOIODINE CONCENTRATIONS IN THE CONTAINMENT BUILDING AIR DURING THE 5-MINUTE PERIOD FROM FHA 63
Table 13-43 AVERAGE NOBLE GAS CONCENTRATIONS IN THE CONTAINMENT BUILDING AIR DURING THE 5-MINUTE PERIOD FROM FHA 64
Table 13-44 DERIVED AIR CONCENTRATION VALUES AND 5-MINUTE EXPOSURES – RADIOIODINES – FROM FHA 65
Table 13-45 DERIVED AIR CONCENTRATION VALUES AND 5-MINUTE EXPOSURES – NOBLE GASES – FROM FHA 66
Table 13-46 5-MINUTE DOSE FROM RADIOIODINES AND NOBLE GASES IN THE CONTAINMENT BUILDING FROM FHA 67
Table 13-47 RADIATION SHINE THROUGH THE CONTAINMENT BUILDING FROM FHA 67
Table 13-48 RADIOIODINE CONCENTRATIONS IN CONTAINMENT AIR DURING THE FIRST HOUR FROM FHA 68
Table 13-49 AVERAGE NOBLE GAS CONCENTRATIONS IN THE CONTAINMENT AIR DURING THE FIRST HOUR FROM FHA 68
Table 13-50 AVERAGE RADIOIODINE CONCENTRATIONS IN AIR LEAKING FROM CONTAINMENT AND EXITING THE EXHAUST STACK DURING THE FIRST HOUR FROM FHA 69
Table 13-51 AVERAGE NOBLE GAS CONCENTRATIONS IN AIR LEAKING FROM CONTAINMENT AND EXITING THE EXHAUST STACK DURING THE FIRST HOUR FROM FHA 69
Table 13-52 EFFLUENT CONCENTRATION LIMITS, CONCENTRATIONS AT POINT OF MAXIMUM CONCENTRATION, AND RADIATION DOSES IN THE UNRESTRICTED AREA – RADIOIODINE – FROM FHA. 70
Table 13-53 EFFLUENT CONCENTRATION LIMITS, CONCENTRATIONS AT POINT OF MAXIMUM CONCENTRATION, AND RADIATION DOSES IN THE UNRESTRICTED AREA – NOBLE GASES – FROM fha ... 71
Table 13-54 DOSE FROM RADIOIODINES AND NOBLE GASES IN THE UNRESTRICTED AREA FROM FHA ... 71

LIST OF FIGURES

Figure 13.1 – REACTOR POWER, FUEL, AND CLADDING TEMPERATURES VS. TIME FOR A POSITIVE STEP INSERTION OF 0.006 ΔK/K 91
Figure 13.2 – REACTOR POWER VERSUS TIME FOR VARIOUS SMALLER POSITIVE REACTIVITY STEP INSERTIONS 92
Figure 13.3 – MURR OPERATING LIMIT CURVES (PRESSURIZER PRESSURE AT 75 PSIA) 93
Figure 13.4 – REACTOR POWER VERSUS TIME FOR CONTINUOUS CONTROL BLADE WITHDRAWAL 94
Figure 13.5 – SCHEMATIC OF IN-POOL PORTION OF PRIMARY COOLANT SYSTEM 95
Figure 13.6 – JUNCTION FLOW RATES DURING THE FIRST 20 SECONDS OF THE COLD-LEG LOCA 96
Figure 13.7 – COOLANT FLOW THROUGH THE 25 INDIVIDUAL CHANNELS DURING THE FIRST 20 SECONDS OF THE COLD-LEG LOCA 97
Figure 13.8 – CENTERLINE TEMPERATURE OF THE 24 FUEL PLATES (SECTION 4) DURING THE FIRST 20 SECONDS OF THE COLD-LEG LOCA 98
Figure 13.9 – LIQUID FRACTION OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 1) DURING THE FIRST 600 SECONDS OF THE COLD-LEG LOCA 99

Figure 13.10 – LIQUID FRACTION OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 2) DURING THE FIRST 600 SECONDS OF THE COLD-LEG LOCA 100
Figure 13.11 – LIQUID FRACTION OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 3) DURING THE FIRST 600 SECONDS OF THE COLD-LEG LOCA 101
Figure 13.12 – LIQUID FRACTION OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 4) DURING THE FIRST 600 SECONDS OF THE COLD-LEG LOCA 102
Figure 13.13 – CENTERLINE TEMPERATURE OF THE 24 FUEL PLATES (SECTION 1) DURING THE FIRST 20 SECONDS OF THE COLD-LEG LOCA 103
Figure 13.14 – CENTERLINE TEMPERATURE OF THE 24 FUEL PLATES (SECTION 2) DURING THE FIRST 20 SECONDS OF THE COLD-LEG LOCA 104
Figure 13.15 – CENTERLINE TEMPERATURE OF THE 24 FUEL PLATES (SECTION 3) DURING THE FIRST 20 SECONDS OF THE COLD-LEG LOCA 105
Figure 13.16 – TEMPERATURE OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 1) DURING THE FIRST 20 SECONDS OF THE COLD-LEG LOCA 106
Figure 13.17 – TEMPERATURE OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 4) DURING THE FIRST 20 SECONDS OF THE COLD-LEG LOCA 107
Figure 13.18 – VOLUME PRESSURES DURING THE FIRST 20 SECONDS OF THE COLD-LEG LOCA 108
Figure 13.19 – CENTERLINE TEMPERATURE OF THE 24 FUEL PLATES (SECTION 3)DURING THE FIRST 40 SECONDS OF THE HOT-LEG LOCA 109
Figure 13.20 – TEMPERATURE OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 1) DURING THE FIRST 200 SECONDS OF THE HOT-LEG LOCA 110
Figure 13.21 – SIMPLIFIED SCHEMATIC OF THE PRIMARY COOLANT SYSTEM 111
Figure 13.22 – DOWNWARD, UPWARD, AND NET COOLANT FLOW THROUGH THE CORE DURING THE FIRST 12 SECONDS OF THE LOF ACCIDENT 112
Figure 13.23 – DOWNWARD, UPWARD, AND NET COOLANT FLOW THROUGH THE CORE DURING THE FIRST 100 SECONDS OF THE LOF ACCIDENT 113
Figure 13.24 – COOLANT FLOW THROUGH THE 25 INDIVIDUAL COOLANT CHANNELS DURING THE FIRST 60 SECONDS OF THE LOF ACCIDENT 114
Figure 13.25 – CENTERLINE TEMPERATURE OF THE 24 FUEL PLATES DURING THE FIRST 60 SECONDS OF THE LOF ACCIDENT 115
Figure 13.26 – TEMPERATURE OF THE 25 INDIVIDUAL COOLANT CHANNELS DURING THE FIRST 60 SECONDS OF THE LOF ACCIDENT 116

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13.0 ACCIDENT ANALYSES

This chapter demonstrates that various facility design features, the safety limit, limiting safety system settings (LSSSs), and limiting conditions for operation have been selected to ensure that no credible accident can lead to unacceptable radiological consequences to people or the environment.

13.1 Introduction

Operational records have shown that light-water moderated, open pool-type reactors are safe in design and construction. This safety is predicated upon the demonstrated ability of these water-moderated reactors to absorb reactivity additions by slight changes in their moderator. Negative void and temperature coefficients are intrinsic features of this type of reactor. Numerous experiments performed by the BORAX and SPERT programs have repeatedly demonstrated the inherent safety characteristics of water-moderated/cooled reactors.

The Missouri University Research Reactor (MURR) described in this document, though more complex than the ordinary open pool-type research reactor, still possesses these same inherent safety characteristics as demonstrated by the history of safe operation. Although the inherent safety of MURR can be adequately shown, this inherent safety serves solely as a fail-safe or back-up mechanism in case of the failure of other control provisions. An extensive system of sensing devices, electronic circuits, signal conditioning equipment, and electro-mechanical devices are provided to protect the reactor against all credible postulated accident scenarios. The MURR Reactor Protective System has been designed to (1) initiate automatic actions to ensure that the reactor safety limit is not exceeded as a result of anticipated operational occurrences, and (2) sense accident conditions and to initiate the operation of systems and components important to safety.

The nine postulated accident events or categories for research reactors are identified in NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors” (Reference 13.1). They are as follows:

- Maximum Hypothetical Accident;
- Insertion of Excess Reactivity;
- Loss of Primary Coolant;
- Loss of Primary Coolant Flow;
- Mishandling or Malfunction of Fuel;
- Experiment Malfunction;
- Loss of Electrical Power;
- External Events; and
- Mishandling or Malfunction of Equipment.

This chapter contains analyses of the nine postulated accident events or categories as they apply to MURR. Some categories contain more than one accident even though one is usually most limiting in terms of impact. The limiting event has potential consequences that exceed all others in that category

and is used for the detailed quantitative analysis. Any accident having potential significant radiological consequences was included. No credible accident scenario has been identified that can lead to the release of fission products to the primary coolant system. However, one accident scenario, termed the Maximum Hypothetical Accident (MHA), assumes the complete failure of a fueled experiment and the subsequent release of fission products to the pool coolant system. The MHA is intended to postulate conditions which lead to consequences worse than those resulting from any other credible accident. Because the MHA is not expected to occur, the initiating event and the scenario details are immaterial to the analysis and are not analyzed.

13.2 Accident-Initiating Events and Scenarios, Accident Analyses, and the Determination of Consequences

13.2.1 Maximum Hypothetical Accident

13.2.1.1 Accident-Initiating Events and Scenarios

Many types of accidents have been considered in conjunction with the operation of MURR. In all cases, safety systems have been designed such that the likelihood of an accident involving the release of a significant amount of fission products has essentially been eliminated. The safety systems take the form of automatic reactor shutdown circuits and process systems designed to ensure, through redundancy, that the reactor will shut down upon a significant deviation from normal operating conditions. Additionally, the reactor is housed within a containment building which provides further protection against a significant release of radioactive material to the environment.

The Maximum Hypothetical Accident (MHA) postulates conditions leading to consequences worse than those from any credible accident. In the MHA for MURR, it is assumed that an accident condition has caused the complete failure of a 5-gram, low-enriched uranium (LEU) fueled experiment and the subsequent release of fission products to the pool coolant system. Because this postulated accident is considered worse than any credible accident, the conditions that lead to this event are immaterial to the analysis. (Reference 13.49)

13.2.1.2 Accident Analysis and Consequences

The release of the radioisotopes of krypton, xenon and radioiodine from a 5-gram LEU target is the major source of radiation exposure to an individual and serves as the basis for the source term for the dose calculations of a Fueled Experiment Failure. A 5-gram LEU target irradiated for 150 hours (normal weekly operating cycle) at a thermal neutron flux of 1.5×10^{13} n/cm^2-sec will produce the following radioiodine, krypton and xenon activities (additionally, approximately 1.40×10^4 μCi of Sr-90 will be produced):

TABLE 13-1
RADIOIODINE AND NOBLE GAS ACTIVITIES IN A 5-GRAM LEU TARGET

I-131 – 6.755 Ci	Kr-85 – 0.0020 Ci	Xe-133 – 18.925 Ci
I-132 – 18.635 Ci	Kr-85m – 7.580 Ci	Xe-135 – 13.630 Ci
I-133 – 39.875 Ci	Kr-87 – 15.405 Ci	Xe-135m – 6.760 Ci
I-134 – 45.405 Ci	Kr-88 – 21.660 Ci	Xe-137 – 35.800 Ci
I-135 – 37.695 Ci	Kr-89 – 27.740 Ci	Xe-138 – 37.380 Ci
	Kr-90 – 27.410 Ci	Xe-139 – 30.675 Ci
Total Iodine – 148.365 Ci	**Total Krypton – 99.797 Ci**	**Total Xenon – 143.170 Ci**

The source term for the Fueled Experiment Failure accident analysis was determined using the ORIGEN program. The fueled experiment target was irradiated for 150 hours at the peak flux level it is exposed to during full power reactor operation and the resulting activities were calculated using ORIGEN 2.2.

A complete failure of the fueled target is unrealistic for many reasons. The worst that can be expected is partial melting; however, to present a worst-case dose assessment for an individual that remains in the containment building following target failure, 100% of the total activity of the target is assumed to be released into the reactor pool.

Fission products released into the reactor pool will be detected by the pool surface and ventilation system exhaust plenum radiation monitors. However, for the purposes of this analysis, it is assumed that a reactor scram and actuation of the containment building isolation system occurs by action of the pool surface radiation monitor. Actuation of the isolation system will prompt Operations personnel to ensure that a total evacuation of the containment building is accomplished promptly, usually within two (2) to two-and-a-half (2.5) minutes. A conservative 5-minute evacuation time is used as the basis for the stay time in the dose calculations for personnel that are in containment during target failure.

The radioiodine released into the reactor pool over a 5-minute interval is conservatively assumed to be instantly and uniformly mixed into the 20,000 gallons (75,708 l) of bulk pool water, which results in pool water concentrations for the radioiodine isotopes as shown in Table 13-2. The water solubility of the krypton and xenon noble gases released into the pool over this same time period is conservatively ignored. The gas bubble rise time in the reactor pool from where the target is situated in the graphite reflector region to the pool surface has been measured at 17 seconds, so it is assumed the earliest any of the radioisotopes from the fueled experiment enter into the containment building air volume is after 17 seconds. It is also assumed when the radioactivity enters the containment air volume it instantaneously forms a uniform concentration in the isolated containment structure.

TABLE 13-2
RADIOIODINE CONCENTRATIONS IN THE POOL WATER FROM A 5-GRAM LEU TARGET

I-131 – 3.38×10^2 µCi/gal	I-133 – 1.99×10^3 µCi/gal	I-135 – 1.88×10^3 µCi/gal
I-132 – 9.32×10^2 µCi/gal	I-134 – 2.27×10^3 µCi/gal	

When the reactor is at 10 MW and the containment building ventilation system is in operation, the evaporation rate from the reactor pool surface is approximately 80 gallons (302.8 L) of water per day. For the purposes of this calculation, it is assumed that a total of 20 gallons (75.7 L) of pool water containing the previously listed radioiodine concentrations evaporates into the containment building over the 5-minute period. Containment air with a temperature of 75 °F (23.9 °C) and 100% relative humidity contains H_2O vapor equal to 40 gallons (151.4 L) of water. Since the air in containment is normally at about 50% relative humidity, thus containing 20 gallons (75.7 L) of water vapor, the assumed addition of 20 gallons (75.7 L) of water vapor over five (5) minutes will not cause the containment air to be supersaturated. It is also conservatively assumed that all of the radioiodine activity in the 20 gallons (75.7 L) of evaporated pool water instantaneously forms a uniform concentration in the containment building air. When distributed into the containment building, this would result in radioiodine concentrations in the 225,000 ft^3 (6,371.3 m^3) air volume as shown in Table 13-3, using the following method:

Example calculation of the average I-131 concentration released into containment air during the first minute:

= I-131 concentration in pool water/gal x 20 gal/5 min x 0.5 min x e^[(-0.693 x (17 + 30 sec)) / (8.02 day x 8.64×10^4 sec/day)] /(225,000 ft^3 x 28317 ml/ft^3)

= (3.38×10^2 µCi/gal) x 2 gal x 0.99995 / (6.371×10^9 ml)

= 1.06×10^{-7} µCi/ml

The same calculation is used for the isotopes listed in Table 13-3 and was performed for each one minute interval. The average radioiodine concentration over the 5-minute interval is the average of the five (5) 1-minute intervals.

TABLE 13-3
AVERAGE RADIOIODINE CONCENTRATIONS IN THE CONTAINMENT BUILDING AIR DURING THE 5 MINUTES FROM A 5-GRAM LEU TARGET

I-131 – 5.30×10^{-7} µCi/ml	I-133 – 3.12×10^{-6} µCi/ml	I-135 – 2.96×10^{-6} µCi/ml
I-132 – 1.44×10^{-6} µCi/ml	I-134 – 3.40×10^{-6} µCi/ml	

As noted previously, the krypton and xenon noble gases released into the reactor pool from the 5-gram LEU target during the 5-minute interval following failure are assumed to have no absorption in the pool water, rise through the pool in 17 seconds (thus slightly decaying) and enter the containment building

air volume where they are assumed to instantaneously form a uniform concentration in the isolated structure. Based on the 225,000-ft^3 (6,371.3 m^3) volume of containment building air, and the previously listed Curie quantities of these gases released into the reactor pool, the maximum noble gas concentrations in the containment structure at the end of five (5) minutes would be as follows:

Example calculation of the average Kr-85 released into containment air during the first minute after the gas enters the containment air:

= Kr-85 activity x e^[(-0.693 x (17 + 300 sec)) / (10.76 yrs x 3.156 x 10^7 sec/yr)] / (225,000 ft^3 x 28317 ml / ft^3)

= 2.00 x 10^3 µCi x 1.000 / (6.371 x 10^9 ml)

= 3.14 x 10^{-7} µCi/ml

The same calculation is used for the isotopes listed in Table 13-4. The average noble gas concentrations are the average of the five (5) 1-minute decay corrected concentrations.

TABLE 13-4
AVERAGE NOBLE GAS CONCENTRATIONS IN THE CONTAINMENT BUILDING AIR DURING THE 5 MINUTES FROM A 5-GRAM LEU TARGET

Kr-85 – 3.14 x 10^{-7} µCi/ml	Xe-133 – 2.97 x 10^{-3} µCi/ml
Kr-85m – 1.18 x 10^{-3} µCi/ml	Xe-135 – 2.13 x 10^{-3} µCi/ml
Kr-87 – 2.36 x 10^{-3} µCi/ml	Xe-135m – 9.37 x 10^{-4} µCi/ml
Kr-88 – 3.36 x 10^{-3} µCi/ml	Xe-137 – 3.50 x 10^{-3} µCi/ml
Kr-89 – 2.48 x 10^{-3} µCi/ml	Xe-138 – 5.13 x 10^{-3} µCi/ml
Kr-90 – 4.33 x 10^{-4} µCi/ml	Xe-139 – 6.50 x 10^{-4} µCi/ml

The objective of this calculation is to present a worst-case dose assessment for an individual who remains in the containment building for five (5) minutes following target failure. Based on the source term data provided, it is possible to determine the radiation dose to the thyroid from radioiodine and the dose to the whole body resulting from submersion in the airborne noble gases and radioiodine inside the containment building.

Because the airborne radioiodine source is composed of five (5) different iodine isotopes, it will be necessary to determine the dose contribution from each individual isotope and to then sum the results before calculating an effective dose equivalent for the radioiodines. These results were then added to the submersion doses calculated for the noble gases. Dose multiplication factors were established using the Derived Air Concentrations (DACs) for the "listed" isotopes in Appendix B of 10 CFR 20 and were calculated using the methodology presented in Federal Guidance Report (FGR) No. 12 for those isotopes that are "unlisted" submersion isotopes, and the radionuclide concentrations in the containment building (Reference 13.49, Attachment 4).

Example calculation of thyroid dose due to I-131:

The DAC can also be defined as 50,000 mrem (thyroid target organ limit)/2,000 hrs, or 25 mrem/DAC-hr. Additionally, five (5) minutes of one DAC-hr is 8.33×10^{-2} DAC-hr.

I-131 concentration in containment = 5.30×10^{-7} µCi/ml

I-131 DAC (10 CFR 20) = 2.00×10^{-8} µCi/ml

Dose Multiplication Factor = (I-131 concentration) / (I-131 DAC)

= (5.30×10^{-7} µCi/ml) / (2.00×10^{-8} µCi/ml)

= 26.5

A 5-minute thyroid exposure from I-131 is:

= Dose Multiplication Factor x DAC Dose Rate x 5 minutes of a DAC-hr

= 26.5 x (25 mrem/DAC-hr) x (8.33×10^{-2} DAC-hr)

= 55.2 mrem

The same calculation is used for the radioiodines listed in Table 13-5.

TABLE 13-5
DERIVED AIR CONCENTRATION VALUES AND 5-MINUTE EXPOSURES – RADIOIODINE – FROM A 5-GRAM LEU TARGET

Radionuclide	Derived Air Concentration	5-Minute Exposure
I-131	2.00×10^{-8} µCi/ml	55.2 mrem
I-132	3.00×10^{-6} µCi/ml	0.997 mrem
I-133	1.00×10^{-7} µCi/ml	65.0 mrem
I-134	2.00×10^{-5} µCi/ml	0.354 mrem
I-135	7.00×10^{-7} µCi/ml	8.80 mrem
Total = 130.4 mrem		

Doses from the kryptons and xenons present in the containment building are assessed in much the same manner as the radioiodines, and the dose contribution from each individual radionuclide must be calculated and then added together to arrive at the final noble gas dose. Because the dose from the noble gases is only an external dose due to submersion, and because the DACs for these radionuclides are based on this type of exposure, the individual noble gas doses for five (5) minutes in containment were based on their average concentration in the containment air and the corresponding DAC.

Example calculation of whole-body dose (CEDE) due to Kr-85:

The DAC can also be defined as 5,000 mrem/2,000 hrs, or 2.5 mrem/DAC-hr. Additionally, five (5) minutes of one DAC-hr is 8.33×10^{-2} DAC-hr.

Kr-85 concentration in containment = 3.14×10^{-7} µCi/ml

Kr-85 DAC (10 CFR 20) = 1.00×10^{-4} µCi/ml

Dose Multiplication Factor = (Kr-85 concentration) / (Kr-85 DAC)

= (3.14×10^{-7} µCi/ml) / (1.00×10^{-4} µCi/ml)

= 0.00314

A 5-minute whole body exposure from Kr-85 is:

= Dose Multiplication Factor x DAC Dose Rate x 5 minutes of a DAC-hr

= 0.00314 x (2.5 mrem/DAC-hr) x (8.33×10^{-2} DAC-hr)

= 6.54×10^{-4} mrem

The same calculation is used for the noble gases listed in Table 13-6.

TABLE 13-6
DERIVED AIR CONCENTRATION VALUES AND 5-MINUTE EXPOSURES – NOBLE GASES – FROM A 5-GRAM LEU TARGET

Radionuclide	**Derived Air Concentration**	**5-Minute Exposure**
Kr-85	1.00×10^{-4} µCi/ml	6.54×10^{-4} mrem
Kr-85m	2.00×10^{-5} µCi/ml	1.23×10^{1} mrem
Kr-87	5.00×10^{-6} µCi/ml	9.82×10^{1} mrem
Kr-88	2.00×10^{-6} µCi/ml	3.50×10^{2} mrem
Kr-89	1.90×10^{-6} µCi/ml	2.71×10^{2} mrem
Kr-90	2.80×10^{-6} µCi/ml	3.22×10^{1} mrem
Xe-133	1.00×10^{-4} µCi/ml	6.18×10^{0} mrem
Xe-135	1.00×10^{-5} µCi/ml	4.44×10^{1} mrem
Xe-135m	9.00×10^{-6} µCi/ml	2.17×10^{1} mrem
Xe-137	2.00×10^{-5} µCi/ml	3.65×10^{1} mrem
Xe-138	4.00×10^{-6} µCi/ml	2.67×10^{2} mrem
Xe-139	3.70×10^{-6} µCi/ml	3.66×10^{1} mrem
Total = 1176.42 mrem		

To finalize the occupational dose in terms of Total Effective Dose Equivalent (TEDE) for a 5-minute exposure in the containment building after target failure, the doses from the radioiodines and noble gases must be added together, and result in the following values:

TABLE 13-7
5-MINUTE DOSE FROM RADIOIODINES AND NOBLE GASES IN THE CONTAINMENT BUILDING FROM A 5-GRAM LEU TARGET

Total Iodine – Committed Dose Equivalent (CDE)	**130.36 mrem**
Iodine – Committed Effective Dose Equivalent (CEDE) (CEDE = CDE x 0.03)	**3.91 mrem**
Noble Gas – Committed Effective Dose Equivalent (CEDE)	**1176.42 mrem**
Total Dose – Total Effective Dose Equivalent (TEDE)	**1180.33 mrem**

Note: The addition of Sr-90 will increase the above stated TEDE (whole body) by <1%. However, due to the chemical form of the strontium, which would not be prone to evaporation from the pool into the containment atmosphere, it would remain as a residue or concentrate in the unevaporated pool water and its contribution to dose would essentially be zero.

By comparison of the maximum TEDE and CDE for those occupationally exposed during target failure to applicable NRC dose limits in 10 CFR 20, the final values are shown to be well within the published regulatory limits and, in fact, lower than 25% of any occupational limit.

Radiation shine through the containment structure was also evaluated when considering accident conditions and dose consequences to the public and MURR staff. Calculation of exposure rate from the target failure was performed using the computer program MicroShield 8.02 with a Rectangular Volume – External Dose Point geometry for the representation of the containment structure (Reference 13.42). MicroShield 8.02 is a product of Grove Software and is a comprehensive photon/gamma ray shielding and dose assessment program that is widely used by industry for designing radiation shields.

The exposure rate values provided in Table 13-8 represent the radiation fields at 1 foot (30.5 cm) from a 12-inch (30.5 cm) thick ordinary concrete containment wall and at the Emergency Planning Zone (EPZ) boundary of 150 meters (492.1 ft). The airborne concentration source terms used to develop the exposure rate values are identical to those used for determining the dose to a worker within containment from noble gases. For radioiodine, the total iodine activity of the target was used for the dose calculations, not the amount that evaporated in five (5) minutes. The source term also assumes a homogenous mixture of nuclides within the containment free volume.

TABLE 13-8
RADIATION SHINE THROUGH THE CONTAINMENT BUILDING FROM A 5-GRAM LEU TARGET

Exposure Rate at 1-Foot from Containment Building Wall:	**74.69 mrem/hr**
Exposure Rate at Emergency Planning Zone Boundary (150 meters):	**0.514 mrem/hr**

As noted earlier in this analysis, the containment building ventilation system will shut down and the building itself will be isolated from the surrounding areas. Target failure will not cause an increase in

pressure inside the reactor containment structure; therefore, any air leakage from the building will occur as a result of normal changes in atmospheric pressure and pressure equilibrium between the inside of the containment structure and the outside atmosphere. It is highly probable that there will be no pressure differential between the inside of the containment building and the outside atmosphere, and consequently there will be no air leakage from the building and no radiation dose to members of the public in the unrestricted area. However, to develop what would clearly be a worst-case scenario, this analysis assumes that a barometric pressure drop had occurred in conjunction with target failure. A worst-case assumption would be a pressure change on the order of 0.7 inches of Hg (25.4 mm of Hg at 60 °C), which would then create a pressure differential of about 0.33 psig (2.28 kPa above atmosphere) between the inside of the isolated containment building and the inside of the adjacent laboratory building, which surrounds most of the containment building.

The conservative assumption is made that the containment building will leak at a rate slightly greater than the Technical Specification (TS) leakage rate limit. The TS leakage rate limit shall not exceed either 16.3 ft^3/min (STP) with an overpressure of one pound per square inch gauge or 10% of the contained volume over a 24-hour period from an initial overpressure of two pounds per square inch gauge. Additionally, the minimum TS free volume of the containment building is 225,000-ft^3 (6,371.3 m^3) at standard pressure and temperature.

The following equation represents the air leakage rate from the containment building in standard cubic feet per minute (scfm) as a function of containment pressure which at 1 psi over pressure (6.89 kpa above atmosphere) would correspond to 17.68 ft^3/minute (0.5 m^3/min). This would correspond to a leakage rate 8.4% greater than the TS limit of 16.3 ft^3/minute (0.46 m^3/min) at 1.0 psig (6.89 kpa above atmosphere).

$$LR = 17.68 \times (CP-14.7)^{0.5};$$

where:

LR = leakage rate from containment (scfm); and

CP = containment pressure (psia).

Using this equation for the assumed initial overpressure condition of 0.33 psig (2.28 kPa above atmosphere), it would take approximately 16.5 hours for the leakage rate to decrease to zero from an initial leakage rate of approximately 10.25 scfm (0.29 m^3/min), which would occur at the start of the event. The average leakage rate over the 16.5-hour period would be approximately 5.2 scfm (0.147 m^3/min). This conservatively overcalculates the actual amount of activity that would leak out of the containment building and potentially expose someone due to this assumed accident.

Several factors exist that will mitigate the radiological impact of any air leakage from the containment building following target failure. First of all, most leakage pathways from containment discharge into the laboratory building, which surrounds the containment structure. Since the laboratory building

ventilation system continues to operate during target failure, leakage air captured by the ventilation exhaust system is mixed with other building air and then discharged from the facility through the exhaust stack at a rate of approximately 30,500 scfm (863.66 m^3/min). Mixing of the containment air leakage with the laboratory building ventilation flow, followed by discharge out the exhaust stack and subsequent atmospheric dispersion, results in low radionuclide concentrations and small radiation doses in the unrestricted area. A tabulation of these concentrations and doses is given below.

A second factor which reduces the potential radiation dose in the unrestricted area relates to the behavior of radioiodine, which has been studied extensively in the containment mockup facility at Oak Ridge National Laboratory (ORNL) (Reference 13.4). From these experiments, it was shown that up to 75% of the iodine released will be deposited in the containment vessel. For the purposes of this analysis, MURR used the NRC-accepted value of 50% reduction of radioiodines from plate-out and deposition (NUREG 0800, Section 15.6.5, Appendix B "Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Engineered Safety Feature Components Outside Containment"). Due to this 50% iodine deposition in the containment building, each cubic foot of air released from containment has a radioiodine concentration that is 50% of each cubic foot within containment building air. The radioiodine concentrations leaking from the containment structure into the laboratory building, in microcuries per standard cubic foot (scf), during the first hour will be (average I-131 concentration in containment):

= (I-131 activity in pool water x 20 gal) x e^[(-0.693 x 0.5 hr) / I-131 $T_{1/2}$]) / (scf in containment)

= (3.38 x 10^2 µCi/gal x 20 gal) x 0.9982 / (229,800 scf)

= 2.94 x 10^{-2} µCi/scf

The same calculation is used for the radioiodines listed in Table 13-9.

TABLE 13-9
RADIOIODINE CONCENTRATIONS IN CONTAINMENT AIR DURING THE FIRST HOUR
FROM A 5-GRAM LEU TARGET

I-131 – 2.94 x 10^{-2} µCi/scf	I-133 – 1.70 x 10^{-1} µCi/scf	I-135 – 1.55 x 10^{-1} µCi/scf
I-132 – 6.97 x 10^{-2} µCi/scf	I-134 – 1.33 x 10^{-1} µCi/scf	

Example calculation of the Kr-85 concentration released into containment during the First Hour:

= Kr85 activity x e^[(-0.693 x 0.5 hr) / (10.76 yr x 365.25 days/yr x 24 hr/day)] / 229,800 scf

= 2.00 x 10^3 µCi x 0.999996 / 229,800 scf

= 8.70 x 10^{-3} µCi/scf

The same calculation is used for the noble gases listed in Table 13-10.

TABLE 13-10
AVERAGE NOBLE GAS CONCENTRATIONS IN THE CONTAINMENT AIR DURING THE FIRST HOUR FROM A 5-GRAM LEU TARGET

Kr-85 – 8.70×10^{-3} µCi/scf	Xe-133 – 8.21×10^{1} µCi/scf
Kr-85m – 3.05×10^{1} µCi/scf	Xe-135 – 5.71×10^{1} µCi/scf
Kr-87 – 5.10×10^{1} µCi/scf	Xe-135m – 7.56×10^{0} µCi/scf
Kr-88 – 8.34×10^{1} µCi/scf	Xe-137 – 6.74×10^{-1} µCi/scf
Kr-89 – 1.64×10^{-1} µCi/scf	Xe-138 – 3.72×10^{1} µCi/scf
Kr-90 – 2.02×10^{-15} µCi/scf	Xe-139 – 3.02×10^{-12} µCi/scf

The average containment building leakage rate was calculated for each of the first five (5) hours and for the following three (3) 4-hour intervals:

Hours:	0 - 1	1 - 2	2 - 3	3 - 4	4 - 5	5 - 9	9 - 12	12 - 16.5
scf/hr:	595.6	558.7	521.8	485.0	448.1	355.8	207.9	67.9

The average concentration of the radioactive iodine's and noble gasses in the facility's exhaust stack was calculated based on the average isotope concentration in containment during each of the time periods and the average leak rate for the time interval. The iodine activities are reduced to 50% due to the previous stated 50% deposition in the containment structure in their leakage path. This leakage of activity out of containment is drawn into the facility exhaust ventilation system. The facility exhaust ventilation system has a flow rate of 30,500 scfm (863.66 m^3/min) and results in the following average exhaust ventilation system activities:

Example calculation of the average concentration of I-131 released through the exhaust stack during the first hour:

= (I-131 activity in containment in µCi/scf x scf leakage rate x 0.5) / (30,500 ft^3/min x 60 min/hr x 28,317 ml/ft^3)

= (2.934×10^{-2} µCi/scf x 595.6 scf/hr x 1 hr x 0.5) / 5.182×10^{10} ml/hr

= 1.686×10^{-10} µCi/ml

A similar, but longer, calculation is used to determine the average concentration in air exiting the exhaust stack over the full 16.5 hours for the radioiodines listed in Table 13-11.

TABLE 13-11
AVERAGE RADIOIODINE CONCENTRATIONS IN AIR LEAKING FROM CONTAINMENT AND EXITING THE EXHAUST STACK OVER THE 16.5 HOUR PERIOD FROM A 5-GRAM LEU TARGET

I-131 – 8.60×10^{-11} µCi/ml	I-133 – 4.32×10^{-10} µCi/ml	I-135 – 2.92×10^{-10} µCi/ml
I-132 – 7.56×10^{-11} µCi/ml	I-134 – 8.01×10^{-11} µCi/ml	

Example calculation of Kr-85 released through the exhaust stack during the first hour:

= (Kr-85 activity/scf x 595.6 scf/hr x 1 hr) / (30,500 ft^3/min x 60 min/hr x 1 hr x 28,317 ml/ft^3)

= (8.70×10^{-3} µCi/scf x 595.6 scf) / 5.182×10^{10} ml

= 1.00×10^{-10} µCi/ml

The same calculation is used for the noble gases listed in Table 13-12.

TABLE 13-12
AVERAGE NOBLE GAS CONCENTRATIONS IN AIR LEAKING FROM CONTAINMENT AND EXITING THE EXHAUST STACK OVER THE 16.5 HOUR PERIOD FROM A 5-GRAM LEU TARGET

Kr-85 – 5.19×10^{-11} µCi/ml	Kr-87 – 7.66×10^{-8} µCi/ml	Kr-89 – 1.14×10^{-10} µCi/ml
Kr-85m – 9.69×10^{-8} µCi/ml	Kr-88 – 2.07×10^{-7} µCi/ml	Kr-90 – 1.40×10^{-24} µCi/ml
Xe-133 – 4.76×10^{-7} µCi/ml	Xe-135m – 5.61×10^{-9} µCi/ml	Xe-138 – 2.73×10^{-8} µCi/ml
Xe-135 – 2.40×10^{-7} µCi/ml	Xe-137 – 4.70×10^{-10} µCi/ml	Xe-139 – 2.10×10^{-21} µCi/ml

Assuming, as stated earlier, that (1) the average leakage rate from the containment building is 5.2 scfm (0.147 m^3/min), (2) the leak continues for about 16.5 hours in order to equalize the containment building pressure with atmospheric pressure, (3) the flow rate through the facility's ventilation exhaust stack is 30,500 scfm (863.66 m^3/min), (4) the reduction in concentration from the point of discharge at the exhaust stack to the point of maximum concentration in the unrestricted area is a factor of 292, and (5) there is no decay of any radioiodines or noble gases following release from the stack, then the concentrations of radioiodines and noble gases with their corresponding radiation doses will occur in the unrestricted area as shown in Tables 13-13 and 13-14. The values listed are for the point of maximum concentration in the unrestricted area assuming a uniform, semi-spherical cloud geometry for noble gas submersion and assuming that the most conservative (worst-case) meteorological conditions exist for the entire 16.5-hour period of containment leakage following target failure. Radiation doses are calculated based on a 16.5-hour exposure period. Dose values for the unrestricted area were obtained using the same methodology that was used to determine doses inside the containment building, and it was assumed that an individual was present at the point of maximum concentration for the full 16.5 hours that the containment building was leaking.

A worst-case scenario dilution factor of 292 for effluent dilution using the Pasquill-Guifford Model for atmospheric dilution is used in this analysis. All offsite (public) dose is assumed to occur under these atmospheric conditions at the site of interest, i.e. 760 meters North of MURR. At 760 meters, this occurs only during Stability Class 'F' conditions, which normally only occurs 11.4% of the time when the wind blows from the south. Thus, this calculation is conservative.

10 CFR 20 Appendix B Effluent Concentration Limits are used for the "listed" isotopes. Effluent Concentration Limits were calculated for each of the four (4) "unlisted" noble gases (Kr-89, Kr-90, Xe-137 and Xe-139) using the data and methodology contained in FGR No. 12, External Exposure to Radionuclides in Air, Water, and Soil for submersion isotopes (Reference 13.37). The DAC value was first calculated and then a factor of 219 was applied using 10 CFR 20, Appendix B, as a reference point for DAC values from submersion isotopes. Exposure at 1 DAC equates to 5000 mrem per year whereas at the Effluent Concentration Limit it is 50 mrem per year (for noble gases, the Effluent Concentration Limit is determined by a submersion dose of 100 mrem per year). Thus, there is a factor of 100 times lower allowable dose for the Effluent Concentration Limit as compared to the DAC. Exposure at the Effluent Concentration Limit assumes an individual is within that effluent concentration for 8760 hours per year. Therefore, the time assumed to be exposed to the Effluent Concentration Limit is a factor of 4.38 times longer than the 2000 hours per year that defines a DAC. No credit is taken for transit time from the exhaust stack to the receptor point nor is credit taken for decay inside the containment building until release. In the case of Kr-89 and Xe-137, the transit time alone would be approximately one (1) half-life while the transit time for Kr-90 and Xe-139 would be at least four (4) half-lives.

Example calculation of whole-body dose in the unrestricted area due to I-131:

Conversion Factor: (Public dose limit of 50 mrem/yr) x (1 yr/8760 hrs) = 5.71×10^{-3} mrem/hr

I-131 concentration = 8.65×10^{-11} µCi/ml

I-131 effluent concentration limit = 2.00×10^{-10} µCi/ml

I-131 Conversion Factor = 5.71×10^{-3} mrem/hr

A 16.5-hour whole body exposure from I-131 is:

= I-131 concentration / (I-131 effluent concentration limit x Conversion Factor x 16.5 hrs)

= (8.60×10^{-11} µCi/ml / 2.00×10^{-10} µCi/ml) x 5.71×10^{-3} mrem/hr x 16.5 hrs/292

= 1.39×10^{-4} mrem

The same calculation is used for the isotopes (radioiodines and noble gases) listed in Tables 13-13 and 13-14.

TABLE 13-13
EFFLUENT CONCENTRATION LIMITS, CONCENTRATIONS AT POINT OF MAXIMUM CONCENTRATION AND RADIATION DOSES IN THE UNRESTRICTED AREA – RADIOIODINE – FROM A 5-GRAM LEU TARGET

Radionuclide	Effluent Limit	Maximum Concentration[1]	Radiation Dose
I-131	2.00×10^{-10} µCi/ml	2.95×10^{-13} µCi/ml	1.39×10^{-4} mrem
I-132	2.00×10^{-8} µCi/ml	2.59×10^{-13} µCi/ml	1.22×10^{-6} mrem
I-133	1.00×10^{-9} µCi/ml	1.48×10^{-12} µCi/ml	1.39×10^{-4} mrem
I-134	6.00×10^{-8} µCi/ml	2.74×10^{-13} µCi/ml	4.31×10^{-7} mrem
I-135	6.00×10^{-9} µCi/ml	9.99×10^{-13} µCi/ml	1.57×10^{-5} mrem
Total = 2.95E-4 mrem			

Note 1: Maximum Concentrations are radioiodine and noble gas concentrations leaking from the containment building and exiting the exhaust stack reduced by a dilution factor of 292.

TABLE 13-14
EFFLUENT CONCENTRATION LIMITS, CONCENTRATIONS AT POINT OF MAXIMUM CONCENTRATION, AND RADIATION DOSES IN THE UNRESTRICTED AREA – NOBLE GASES – FROM A 5-GRAM LEU TARGET

Radionuclide	Effluent Limit	Maximum Concentration[1]	Radiation Dose[2]
Kr-85	7.00×10^{-7} µCi/ml	1.78×10^{-13} µCi/ml	4.78×10^{-8} mrem
Kr-85m	1.00×10^{-7} µCi/ml	3.32×10^{-10} µCi/ml	6.26×10^{-4} mrem
Kr-87	2.00×10^{-8} µCi/ml	2.62×10^{-10} µCi/ml	2.48×10^{-3} mrem
Kr-88	9.00×10^{-9} µCi/ml	7.08×10^{-10} µCi/ml	1.48×10^{-2} mrem
Kr-89	8.60×10^{-9} µCi/ml	3.92×10^{-13} µCi/ml	8.58×10^{-6} mrem
Kr-90	1.20×10^{-8} µCi/ml	4.81×10^{-27} µCi/ml	7.56×10^{-20} mrem
Xe-133	5.00×10^{-7} µCi/ml	1.63×10^{-9} µCi/ml	6.14×10^{-4} mrem
Xe-135	7.00×10^{-8} µCi/ml	8.23×10^{-10} µCi/ml	2.22×10^{-3} mrem
Xe-135m	4.00×10^{-8} µCi/ml	1.92×10^{-11} µCi/ml	9.06×10^{-5} mrem
Xe-137	9.10×10^{-8} µCi/ml	1.61×10^{-12} µCi/ml	3.34×10^{-6} mrem
Xe-138	2.00×10^{-8} µCi/ml	9.34×10^{-11} µCi/ml	8.80×10^{-4} mrem
Xe-139	1.60×10^{-8} µCi/ml	7.20×10^{-24} µCi/ml	8.48×10^{-17} mrem
Total = 2.18E-2 mrem			

Note 1: Maximum Concentrations are radioiodine and noble gas concentrations leaking from the containment building and exiting the exhaust stack reduced by a dilution factor of 292.

To finalize the unrestricted dose in terms of Total Effective Dose Equivalent (TEDE), the doses from the radioiodines and noble gases must be added together and result in the values displayed in Table 13-15.

TABLE 13-15
DOSE FROM RADIOIODINES AND NOBLE GASES IN THE UNRESTRICTED AREA
FROM A 5-GRAM LEU TARGET

Total Iodine – Committed Effective Dose Equivalent (CEDE)	**2.95E-4 mrem**
Noble Gas – Committed Effective Dose Equivalent (CEDE)	**2.18E-2 mrem**
Total Dose – Total Effective Dose Equivalent (TEDE)	**2.21E-2 mrem**

Summing the doses from the noble gases and the radioiodines simply substantiates earlier statements regarding the low levels in the unrestricted area should a failure of a fueled experiment occur and should the containment building leak following such an event. Due to low concentrations present at the modeled site of interest, dose to the public is low with the noble gases and iodines each contributing about equally. The overall TEDE is much less than 1 mrem and is well below the applicable 10 CFR 20 regulatory limit for dose to the public in an unrestricted area.

13.2.1.3 Conclusions

The selection of the complete failure of a 5-gram, low-enriched uranium fueled experiment as the MHA represents a condition worse than any credible postulated accident. However, the analyses show that the release of fission products from the fueled target show no occupational radiation doses or doses in the unrestricted area that exceed any regulatory limits. The NRC documented their review of this analysis and performed their own confirmatory analyses to ensure the MHA remained within applicable regulatory limits, in the Safety Evaluation Report documented under Reference 13.50.

13.2.2 Insertion of Excess Reactivity

13.2.2.1 Accident-Initiating Events and Scenarios

Two different accident scenarios for an insertion of excess positive reactivity are evaluated in this section. First, a step insertion of positive reactivity based upon the maximum step insertion that MURR can withstand with no core damage, and second, a continuous ramp insertion of positive reactivity based on the continuous withdrawal of MURR's four shim control blades. The exact mechanisms or events that could cause these reactivity insertions can vary but could include an inadvertent rapid insertion or removal of an experiment from the center test hole or reactor operator error.

13.2.2.1.1 Rapid Step Insertions of Positive Reactivity

Historical studies (Reference 13.12, Reference 13.13, and Reference 13.15) have extensively evaluated the expected results of a sudden step insertion of positive reactivity in the MURR core with an initial power of 10 MW and concluded that MURR could withstand a positive reactivity step insertion of 0.008 Δk/k without fuel damage. However, this step insertion of reactivity analysis for the updated Safety Analysis Report for license renewal was conservatively based on an initial power level of 11.5 MW, greater than the license power (10 MW) but less than the limiting safety system setting (LSSS) of 12.5

MW. This initial power level is equal to the power that would cause an automatic rod run-in; which is conservative and is equivalent to the assumption of a single failure. The following other initial conditions were chosen based on worst case conditions established by the LSSS; a reactor coolant flow rate of 3200 gallons per minute (gpm), (12,113 lpm), a pressurizer pressure of 75 pounds per square inch, absolute (psia) (517 kPa), and a reactor core coolant inlet temperature of 155 °F (68 °C).

Core voiding and coolant temperature increase are the two major negative reactivity feedback mechanisms which halt the rapid power escalation following a positive reactivity step insertion. During the low power testing program for MURR's uranium-aluminide core, the temperature and void coefficients were remeasured and found to be close to the original calculated values. The values observed were -7.0×10^{-5} Δk/k/°F and -2.51×10^{-3} Δk/k/% void, respectively (Reference 13.14). In this analysis a conservative fuel temperature coefficient of -6.0×10^{-5} Δk/k/°F and void coefficient of -2.0×10^{-3} Δk/k/% void were assumed based on the limits of Technical Specification (TS) 5.3.

The control blade insertion times were based on the TS 3.2.c requirement of insertion to the 20% withdrawn position in less than 0.7 seconds. The insertion rate was calculated based on the shim control blades travelling from 26 inches (66.04 cm) (fully withdrawn) to 5.2 inches (13.21 cm) (20% withdrawn or 80% inserted) in 0.7 seconds. Even though the shim control blades, after they reach their 80% inserted position, will continue traveling to their 100% fully inserted position at a reduced rate of insertion because of a dashpot assembly. To be conservative, for the accident analysis it was assumed that the maximum negative reactivity inserted following a scram corresponds to only the 80% of the fully inserted value. This negative reactivity corresponds to over 91% of the total negative worth of the shim control blades

The Reactivity Transient Analysis program PARET (V7.5), maintained and distributed by the Nuclear Engineering Division of Argonne National Laboratory (ANL) was used to model two channels; a hot channel representing worst-case conditions inside the core and an average channel representing the rest of the core experiencing "average" conditions. The axial power profiles used for this 2-channel PARET reactivity transient analysis are given in Table 13-16. The power profiles are calculated based on detailed MCNP calculations of a mixed burnup MURR core under non-xenon conditions and with a set of all-fresh shim control blades installed (worst-case peaking conditions). Additionally, the step reactivity insertion accident was run with 91% of the total shim control blade worth inserted.

TABLE 13-16
PEAKING FACTORS IN THE HOT AND AVERAGE CHANNELS

Hot Channel	Average Channel
(Top of Fuel) 2.046	(Top of Fuel) 1.058
1.971	0.920
2.145	1.018
2.335	1.132
2.497	1.219
2.672	1.307
2.835	1.360
2.986	1.411
3.105	1.430
3.164	1.437
3.169	1.420
3.098	1.383
2.953	1.326
2.775	1.243
2.542	1.140
2.290	0.989
2.069	0.828
1.888	0.701
1.703	0.615
1.499	0.530
1.277	0.460
1.080	0.386
0.904	0.329
(Bottom of Fuel) 0.880	(Bottom of Fuel) 0.358

As indicated earlier, the transient was started from an initial power level of 11.5 MW with core coolant flow rate as well as core coolant inlet temperatures set at their LSSS values of 3,200 gpm (12,113 lpm) and 155 °F (68 °C), respectively and pressurizer pressure at 75 psia (517 kPa). Since the Insertion of Excess Reactivity transient was analyzed from a starting power of 11.5 MW, the rod run-in that would be initiated by the rod run-in system at 11.5 MW was bypassed and only the high-power scram set point of 12.5 MW was modeled. A delay of 150 milliseconds was incorporated into the control blade scram model so that the control blades would only start to insert 0.15 seconds after the power level has exceeded the scram set point of 12.5 MW. The 150 millisecond delay accounts for the total delay required for electronic system responses.

The results of a step reactivity insertion of 600 pcm (+0.006 Δk/k) are shown in Figure 13.1. The peak power during the transient momentarily reaches approximately 37.4 MW.

In a reactivity transient, the product of the power level and time must yield enough total energy to reach the melting temperature of the fuel plate. Such reactor behavior has been conclusively shown by hundreds of power excursion tests performed at the Special Power Excursion Reactor Test (SPERT)

facility at the National Reactor Testing Station in Idaho (Reference 13.42). The fuel plate surface temperature during such fast transients lags the power trace due to fuel plate heat capacity and the time required for heat transfer from the fuel meat to the plate surface. SPERT tests have shown that the maximum fuel plate surface temperature remains below the saturation temperature as long as the power burst lasts only a few tenths of a second and is reduced either by the inherent core shut down mechanisms or, as a backup, by the reactor safety system once the short period or high-power limit is exceeded. The peak fuel temperature during the Insertion of Excess Reactivity accident in the worst (Hot) channel is only 441 °F (227.4 °C) – well below the Safety Limit of 986 °F (530 °C) for the aluminide fuel.

Additional modelling was performed to determine the effects of oxide layer buildup of fuel plates over time. During reactor operation, a thin film of aluminum oxide ($Al_2O_3.H_2O$ or boehmite) forms on the fuel plate surface. Based on fuel element oxide thickness measurements taken in July 1987 and using the Griess Correlation, the average hot spot oxide thickness was calculated to be 0.631 mils (0.016 mm) on a fully burned up MURR fuel element. The "worst case" hot spot oxide thickness was calculated to be 1.27 mils (0.032 mm) (Reference 13.43). Previous studies of the oxide formation on aluminum 6061 have shown that the spallation of the oxide layer does not occur until a thickness in the range of 2 mils (0.0508 mm) has developed (Reference 13.44).

To investigate the effect of this oxide layer on reactivity transients, a 2.0 mil (0.0508 mm) thick oxide layer was added to the MURR PARET reactivity transient analyses model. Typically, the oxide layer gradually builds up on fuel plate surface during reactor operation. By the time the layer reaches a thickness of 2.0 mils (0.0508 mm), the fuel element is near its end-of-life and will have a lower power load. However, to be conservative in modeling, the maximum power load was applied to the fuel as if it were a fresh fuel element. Additionally, conservative values were selected for the oxide thermal properties in the PARET model.

With these conservative assumptions incorporated, and with an initial reactor power of 10 MW, the MURR limiting step reactivity insertion transient of 0.006 Δk/k was reanalyzed.

The initial peak power burst reached during the transient went from 34.4 MW to 36.8 MW. This change is to be expected since the insulating oxide layer will delay the heat transfer from the fuel to the coolant and delay the inherent feedback mechanisms that help limit the transient. This effect was also evident since the peak power is reached in 0.057 seconds from the start of the transient compared to 0.047 seconds for the no oxide layer case.

The peak centerline temperature of the fuel increased from 418.5 °F (214.7 °C) for the no-oxide layer case to 581.7 °F (305.4 °C) with the 2.0 mil (0.0508 mm) oxide layer. Again, this is consistent since the heat transfer from the fuel to the coolant is inhibited by the oxide layer thus raising the peak centerline temperature attained during the transient. The maximum temperature reached is still well below the melting point or the blister temperature of the fuel.

The 600 pcm (+0.006 Δk/k) value analysis is used to verify the established Technical Specification limit of 0.006 Δk/k for the reactivity worth of each secured removable experiment, for the absolute total reactivity worth of all experiments in the center test hole, and for the total reactivity worth of all unsecured experiments in the reactor.

Rapid removal of an unsecured or movable experiment can also potentially cause a rapid step insertion of positive reactivity. Regulatory Guide 2.2 (Reference 13.35) states that the magnitude of the potential reactivity worth of each unsecured experiment should be less than the reactivity value which would cause a violation of a safety limit. In order to establish this reactivity limit, the PARET-ANL code was also used to analyze the reactor transient behavior following step insertions of various smaller amounts of positive reactivity.

Results of this study are illustrated in Figure 13.2. Employing these curves and the MURR operating limit curves (Reference 13.45) for a pressurizer pressure of 75 psia (517 kPa) (see Figure 13.3), the following conclusions can be drawn. First, for a step insertion of 0.003 Δk/k, the peak power reached is approximately 16.0 MW, which is less than the operating limit of 17.0 MW with a reactor coolant inlet temperature of 130 °F (54.4 °C) and a total primary coolant flow rate of 3,600 gpm (13,627 lpm). The 150-millisecond scram response time also ensures that the transient will be terminated before any safety limit is exceeded. Thus, the chosen limit of 0.0025 Δk/k placed on each unsecured experiment ensures that a safety limit will not be violated.

In addition to unsecured experiments, Regulatory Guide 2.2 (Reference 13.35) states that the rate of reactivity change from any movable experiment be such that, when the experiment is intentionally set in motion, the capacity of the control system to provide compensation is not exceeded. For the purposes of this analysis, this requirement was interpreted to mean (1) in manual control, the operator and/or rod run-in circuit would have sufficient time to shim the control blades and control the transient before a high power scram is initiated, and (2) in automatic control, the capacity of the regulating blade will be sufficient to compensate for the reactivity inserted.

Based on the PARET-ANL analysis, a positive reactivity step insertion of 0.001 Δk/k will result in a prompt power jump of approximately 15.0% (see Figure 13.2), followed by a steady power rise on a positive period ranging from 50 to 140 seconds. From an initial power level of 10 MW, and assuming no regulating blade insertion from automatic control, the prompt jump would increase power level to approximately 11.5 MW. However, a rod run-in would initiate at or before 11.5 MW thus ensuring that no safety limit is exceeded. If reactor control was in the manual mode, and a rod run-in was not considered, reactor power would increase on an average positive period of 100 seconds, taking approximately 8.6 seconds to reach the high-power scram set point of 12.5 MW. This is sufficient time for the operator to take control of the transient because, while in manual control, the operator would be continuously monitoring reactor power level.

In automatic control with the regulating blade near the top (15.6 inches-withdrawn (39.62 cm)) of its normal operating range, it would take approximately 12 seconds to insert 0.001 Δk/k of negative reactivity and eliminate the positive reactivity step insertion. A positive reactivity insertion of 0.001

Δk/k due to the handling of a movable experiment will not be instantaneous and if it occurs over a few seconds, the negative reactivity caused by the insertion of the regulating blade would prevent a rod run-in from occurring. If the regulating blade is near the bottom (10.0 inches-withdrawn (25.4 cm)) of its normal operating range, the positive step insertion of 0.001 Δk/k would cause the regulating blade to insert and reach the 5.2 inches-withdrawn (13.208 cm) position in approximately 7.2 seconds. The automatic shim control circuit would then activate and insert the shim rods to neutralize the positive reactivity insertion effect. Therefore, no safety limit would be exceeded if reactor control was in either the manual or automatic control modes.

A negative reactivity insertion of 0.001 Δk/k would require operator action in both the automatic and manual control modes. In automatic control, the regulating blade would be driven out to its fully withdrawn position (26 inches (66.04 cm)), but the reactor would still be subcritical. The "Regulating Blade 60% Withdrawn" alarm would annunciate and alert the operator to the transient. If the operator did not assume manual control, the nuclear instrumentation Wide Range Monitor interlock would cause the control system to shift to manual control when power level decreased below 75%.

In conclusion, the control system has sufficient capacity to compensate for a step reactivity insertion of 0.001 Δk/k and this becomes the maximum reactivity worth of each movable experiment.

13.2.2.1.2 Continuous Control Blade Withdrawal

To assess core behavior during a reactor startup accident, a PARET-ANL analysis was performed using the following worst-case reactor conditions.

A positive reactivity ramp insertion rate of 0.0003 Δk/k/sec (the Technical Specification limit on the maximum rate of reactivity insertion for all four shim control blades operating simultaneously) was introduced to the reactor starting at an initial subcritical power level of 1.0 watt and a shutdown reactivity value of negative 0.042 Δk/k. Even though the reactivity addition resulting from the simultaneous withdrawal of all four shim control blades follows the typical differential rod worth curve behavior (with reduced worths at the beginning and at the end of rod withdrawal), the maximum value allowed by the Technical Specification was imposed during the entire transient.

Behavior of the reactor due to this ramp insertion of positive reactivity was observed during a total transient time of 150.0 seconds (2.5 minutes). Since power level remains low during the entire transient, as shown in Figure 13.4, no major reactivity feedback mechanisms exist to effect reactor behavior. The entire reactivity effect is due to the withdrawal of the control blades.

Approximately 140.0 seconds into the transient, the reactor passes through the critical state. At this time, the reactor power level is approximately 17.0 watts with a reactor period of approximately 11.0 seconds. Once the reactor passes through critical, the increase in power is rapid. At 150.0 seconds after initiation of the transient (ten seconds after passing through the critical state), reactor power level exceeds 64 watts with a reactor period of only 4.0 seconds. At this point, a short period reactor scram will have terminated the transient at the reactor safety system set point of 8.0 seconds.

The strong neutron emission rate from the beryllium reflector (γ, n) reaction eliminates the possibility of a reactor startup with a neutron level below the minimum sensitivity of the installed instruments. This means that the transient can be detected earlier by the nuclear instrumentation and the accident would be terminated by the reactor safety system. With the continuous indication provided by the nuclear instrumentation, the probability that the operator would recognize the accident and take corrective actions is considerably greater.

A continuous control blade withdrawal accident from source power without a reactor scram would cause significant damage to the reactor core. However, a failure to scram on short period is not considered credible because the reactor safety system has been designed to conform with the criteria of IEEE-279 (Reference 13.31). Upon reaching the short period scram set point the reactor will shut down. No credit was taken for the short period rod run-in which would be activated before the short period scram set point is reached. The rod run-in circuit has also been designed to comply with IEEE-279 (Reference 13.31) criteria and its complete failure is not considered credible.

Additionally, a reactor startup is performed in a controlled manner by the reactor operator. Control blade withdrawals are paused at 5.0-inch (12.7 cm) intervals to monitor and record various reactor parameters as well as plot 1/M criticality data. After the control blades have been withdrawn to 2.0 inches (5.08 cm) below the estimated critical position (ECP), only individual blade withdrawals are permitted. All of these administrative controls mitigate the effects of a continuous control blade withdrawal accident during a reactor startup.

A continuous control blade withdrawal accident starting from an initial power level of 10 MW was also evaluated using PARET-ANL. The same Technical Specification limit on the maximum reactivity insertion rate of 0.0003 Δk/k/sec was introduced to the reactor operating at full power. In this case, the transient will be terminated at 4.53 seconds by a high-power scram when reactor power exceeds 12.5 MW. Failure of the high-power scram and/or high-power rod run-in circuits is not considered credible because of the safety system's conformance to IEEE-279 (Reference 13.31).

MURR also analyzed a continuous control blade withdrawal accident involving the simultaneous withdrawal of the regulating blade with the four shim blades, even though this scenario is highly unlikely because the inadvertent simultaneous manipulation of both the regulating blade and the four shim blades involves two separate control switches. The conclusion stated that even though the simultaneous withdrawal of the regulating blade and the four shim blades would result in a higher rate of reactivity insertion, the reactor protective system would initiate a rod run-in or scram earlier, resulting in termination of the transient before any fuel damage can occur. (Reference 13.51)

13.2.2.2 Conclusions

The insertion of excess reactivity at MURR was analyzed to assess the impact of a rapid step insertion of positive reactivity and the impact of a continuous control blade withdrawal accident. The results of the first analysis showed that MURR can, under the worst possible conditions, withstand a positive reactivity step insertion of 0.006 Δk/k without core damage. This value was then used to establish Technical

Specification limit of 0.006 Δk/k for the reactivity worth for each secured removable experiment, for the total reactivity worth of all experiments in the center test hole, and for the total reactivity worth of all unsecured experiments in the reactor. For each unsecured experiment, a limit of 0.0025 Δk/k was established as a safe value, while each movable experiment shall be limited to 0.001 Δk/k.

The consequences of a continuous blade withdrawal accident at the MURR will be a rod run-in or a reactor scram on either short period or high power and there will be no resulting fuel damage.

13.2.3 Loss of Primary Coolant

13.2.3.1 Accident-Initiating Events and Scenarios

Historically, the most serious accident considered in the safety analyses of most reactors is the postulated loss of coolant accident (LOCA) for the primary coolant system, frequently initiated, in theory, by the double-ended rupture in a section of main coolant piping. The use of Engineered Safety Features (ESFs) greatly helps to mitigate the effects of this type of accident; however, the consequences of such an accident should still be considered. The benchmarking performed on the RELAP model for the LOCA and LOFA analyses is described in the following reference. (Reference 13.47 and Reference 13.55)

This section analyzes the sequence of events and expected results of a double-ended rupture of the largest diameter primary coolant piping. The consequences of rupturing this pipe were analyzed using the MURR RELAP5 model (see Appendix C). Table 13-17 provides a comparison of the normal reactor operating parameters and the conservative assumptions that were used in the LOCA analysis when the transient starts.

TABLE 13-17
NORMAL REACTOR OPERATING CONDITIONS AND CONSERVATIVE ASSUMPTIONS
WHEN THE LOCA INITIATES

Parameter	**Conservative Assumption**	**Normal Condition**
Reactor Power	11 MW	10 MW
Coolant Inlet Temperature	155 °F (68 °C)	120 °F (49 °C)
Core Inlet Flow Rate	3,800 gpm (14,385 lpm)	3,800 gpm (14,385 lpm)
Pool Temperature	120 °F (49 °C)	100 °F (38 °C)
Pressurizer Pressure	60 psig (414 kPa)[1]	60 - 66 psig (414 - 455 kPa)[1,2]
Anti-Siphon Pressure	26 psig (179 kPa)[1]	36 psig (248 kPa)[1]

Note 1: Pressure above atmosphere.
Note 2: Primary coolant system pressure as indicated on PT-943.

The following explains why the above stated conservative assumptions were selected for each of the reactor operating parameters used in the LOCA analysis at the start of the transient. (Reference 13.52)

Reactor Power

Class 104c Amended Facility License R-103, issued by the U.S. Nuclear Regulatory Commission, authorizes MURR to operate a maximum steady-state power level of 10 MW, so a conservative steady-

state power level of 11 MW is assumed at the start of the transient, which produces 10% more decay heat in the LOCA analysis. The high reactor power rod run-in and scram are currently set at 114% and 119%, respectively.

Coolant Inlet Temperature

MURR Technical Specification 2.2, "Limiting Safety System Settings," and Technical Specification 3.2.a, "Reactor Control and Reactor Safety Systems," requires instrumentation necessary for two (2) reactor inlet water temperature scrams to be set at a maximum value of 155 °F (68 °C). This value of 155 °F (68 °C) is the most conservative assumption for coolant inlet temperature used in the LOCA analysis. The reactor inlet water temperature scram is currently set at 148 °F (64 °C) while the normal operating temperature is 120 °F (49 °C).

Core Inlet Flow Rate

MURR Technical Specification 2.2, "Limiting Safety System Settings," and Technical Specification 3.2.a, "Reactor Control and Reactor Safety Systems," requires instrumentation necessary for four (4) primary coolant flow scrams, two instruments per loop, to be set at a minimum value of 1,625 gpm (6,151 lpm).

Since 50 gpm (189 lpm) of the primary coolant flow is diverted to the cleanup (demineralizer) system, the actual total flow rate through the core equates to 3,200 gpm (12,113 lpm). Additionally, a differential pressure across the core scram, which corresponds to a flow value of 3200 gpm (12,113 lpm), provides a backup to the primary coolant low flow scrams. However, a core inlet flow rate of 3,800 gpm (14,385 lpm) is assumed in the LOCA analysis. A higher flow rate is more conservative since it will result in less coolant remaining in the U-shaped primary coolant piping exiting the core. The primary coolant flow scrams are currently set at 1,725 gpm (6,530 lpm) (each loop) whereas the differential pressure across the core scram is currently set at 3,400 gpm (12,870 lpm). As stated above, the normal core inlet flow rate is approximately 3800 gpm (14385 lpm).

Pool Temperature

MURR Technical Specification 3.5.b, "Reactor Instrumentation," requires sufficient instrumentation to assure that reactor pool water temperature is limited to a maximum of 120 °F (49 °C). Since the pool coolant system serves as the heat sink during the LOCA, this assumed value is the most conservative temperature for the pool coolant system in the LOCA analysis. Nominal operating temperature, as stated above, is 100 °F (38 °C).

Pressurizer Pressure

MURR Technical Specification 2.2, "Limiting Safety System Settings," and Technical Specification 3.2.a, "Reactor Control and Reactor Safety Systems," requires instrumentation necessary for four (4) primary coolant low pressure scrams to be set at a minimum value of 75 psia (517 kpa) (which corresponds to the pressurizer pressure Limiting Safety System Setting with normal primary coolant flow). The RELAP model uses 60 psig (414 kpa above atmosphere) as the steady-state pressurizer pressure, which is the most conservative value since the worst-case LOCA is a shear of the primary coolant cold-leg inlet piping between the primary coolant isolation valve and the reactor pool. Primary coolant system pressure will then become 0 psig (0 kpa above atmosphere) on the core side of the cold-leg inlet valve immediately

after the transient starts. The lower pressurizer pressure reduces the primary coolant system hot-leg (core outlet) pressure and with the primary system break occurring before the core in the cold-leg, this decreases the pressure differential that helps slow down the primary coolant flowing through the core.

Anti-Siphon Pressure

MURR Technical Specification 3.5, "Reactor Instrumentation," requires sufficient instrumentation to assure anti-siphon system pressure is maintained at a value greater than 27 psig (186 kpa above atmosphere). A pressure of 26 psig (179 kpa above atmosphere) is assumed in the LOCA analysis. Since a lower anti-siphon pressure means less air will be injected into the inverted primary coolant loop, this assumption is conservative. (Reference 13.52)

13.2.3.2 Accident Analysis and Consequences

Figure 13.5 is a schematic of the in-pool portion of the primary coolant system. A rupture in a section of the 12-inch (30.48-cm) diameter primary coolant piping would cause a loss of pressure as sensed by the following four pressure transmitters and sensors: PT 944A, PT 944B, PT 943 and PS 938. Each pressure transmitter or sensor will initiate a reactor scram when system pressure decreases to approximately 95% of the normal operating value.

In addition to initiating a reactor scram, PT 944A and PT 944B will de-energize relays 2K13 and 2K28; either of which will cause the following actions to occur:

1. Primary coolant circulation pumps P501A and P501B will stop;
2. Primary coolant isolation valves V507A and V507B will close; and
3. Anti-siphon system isolation valves V543A and V543B will open.

As primary coolant isolation valves V507A and V507B leave their fully open position, a limit switch on each valve actuator will cause relays 2K10, 2K11, and 2K17 to de-energize, and, in addition to causing a reactor scram (which has already been initiated), cause the following actions to occur:

- Actions 1 and 3 listed above;
- In-pool heat exchanger isolation valves V546A and V546B will open; and
- Pressurizer surge line isolation valve V527C will close.

A reduction in primary coolant flow caused by pumps P501A and P501B stopping will also cause isolation valves V546A and V546B to open when flow decreases to approximately 90% of the normal operating value as sensed by differential pressure sensor DPS 929 and flow transmitters FT 912A and 912H. The MURR RELAP5 model of the LOCA has valves V546A and V546B opening due to the reduction in flow as sensed by the above listed differential pressure and flow sensors, which is before isolation valves V507A and V507B leave their fully open position.

If the rupture is at a considerable distance upstream or downstream from the core, the closure of the primary coolant isolation valves adjacent to the reactor pool penetrations would prevent the core from being uncovered. The core would be protected as described in the loss of flow accident analysis (Section

13.2.4). A rupture in a section of in-pool primary coolant piping would cause the reactor to scram, and pool water to be admitted into the ruptured system until flow would be stopped by the isolation valves. The core would remain covered and no significant reactor safety issues would exist. In either accident, the decay heat would be transferred to the pool water through the inner and outer reactor pressure vessels and in-pool heat exchanger. The accident of greatest consequence is a rupture in the short section of primary coolant piping between the reactor pool and either isolation valve. However, the automatic protective actions mentioned above would all still occur. Additionally, the primary coolant isolation valves are located as close as practical to the biological shield in order to minimize this piping length.

The MURR RELAP5 analysis shows that the core would remain covered even if a double-ended rupture occurred in the section of primary coolant piping between the reactor pool and either primary coolant isolation valve. As stated above, upon a rupture and loss of pressure, the following actions would occur: a reactor scram, primary coolant circulation pumps stop, primary coolant isolation valves close, anti-siphon system isolation valves open, in-pool heat exchanger isolation valves open, and the pressurizer surge line isolation valve closes. When anti-siphon system isolation valves V543A and V543B open, the anti-siphon system pressurized air is applied to the top of the in-pool primary coolant system inverted loop. How actuation of the anti-siphon system ensures that the core remains covered differs depending on the location of the rupture: between isolation valve V507B and the reactor pool (cold-leg break) or between isolation valve V507A and the reactor pool (hot-leg break).

In the RELAP5 model, the reactor is operating at the assumed conservative operating parameters listed in Table 13-17, which results in a peak steady-state temperature of 272.1 °F (133.4 °C) at the center line of fuel plate number-1 [third section (heat structure 651)]. After the piping rupture occurs, the peak fuel plate centerline temperatures take place within the first second of the transient. Due to the rapid decrease in primary coolant pressure, a reactor scram signal is automatically initiated, which in turn causes the control rods to start dropping in 166.6 milliseconds. (The 166.6 millisecond delay accounts for electrical system delay of 150 milliseconds and mechanical response time of 16.6 milliseconds.) During the cold-leg break, the highest peak center line temperature of 311.7 °F (155.4 °C) occurs in fuel plate number-3, 0.5 seconds after the rupture occurs. During the hot-leg break, the highest peak center line temperature of 281.2 °F (138.4 °C) occurs in fuel plate number-1 at 0.2 seconds. Because peak fuel plate temperatures remain more than 500 °F (260 °C) below the “no fuel plate blister verification temperature” of 900 °F (482 °C), the temperature at which every MURR fuel plate is tested to during fabrication, no fuel damage is caused by either LOCA scenario (Reference 13.41). A detailed description of the Quality Assurance step regarding the blister test performed during fuel plate fabrication is provided in the following reference. (Reference 13.53).

A more severe cold-leg break analysis is described below in detail.

References to volumes and heat structures in this section are explained in Appendix C of this report.

13.2.3.2.1 Rupture of Piping Between Valve V507B and the Reactor Pool

During a rupture of the primary coolant piping between isolation valve V507B and the reactor pool (cold-leg break), primary coolant circulation pump discharge pressure immediately decreases to zero downstream of valve V507B. Since the coolant piping rises approximately 15 feet (4.6 m) in the vertical direction between valve V507B and core inlet check valve V502, primary coolant flows back down the pipe and out through the open rupture. This, combined with the air pressure that is admitted to the top of the inverted loop by the anti-siphon system, causes coolant flow rate through check valve V502 to stop within the first second of the transient (see Figure 13.6).

The net downward coolant flow rate at core centerline is a minimum of 70.3 gpm (4.4 lps) at 1.1 seconds into the transient. However, the total flow rate through the fuel element coolant channels is greater because of the flow reversal that occurs in channels 1 through 7 during this first second. As shown in Figure 13.7, when the minimum net flow rate of 70.3 gpm (4.4 lps) occurs, the total downward flow rate of 244.9 gpm (15.5 lps) is offset by a peak upward (reversed) flow rate of 174.6 gpm (11 lps). The highest peak fuel plate centerline temperature of 311.7 °F (155.4 °C) occurs in fuel plate number-3 at 0.5 seconds (see Figure 13.8). This coincides with the flow rate through channels 2, 3, 4 and 5 passing through zero as the flow reverses. This results in a peak coolant temperature of 261.2 °F (127.3 °C) in the third volume of channel 5, which also occurs during this transient zero flow rate condition at 0.5 seconds. This is 70.1 °F (21.2 °C) above the pre-accident peak steady-state temperature of 191.1 °F (88.4 °C) that occurs in coolant channel 2.

At the 0.5-second point, with several coolant channel flow rates passing through zero and yet still high energy generation rates in the fuel plates, significant voiding occurs in coolant channels 2, 3, 4 and 5. All four volumes of channels 3 and 4 momentarily approach a 0.0 liquid fraction (see Figures 13.9 through 13.12). Table 13-18 provides a list of the minimum liquid fractions in all four volumes of coolant channels 2 through 5. Volume 4 of channels 6 through 25 vary in liquid fractions between 0.761 and 0.880. A liquid fraction of 0.982 occurs in volume 4 of coolant channel 1 and 1.0 in the other three volumes. At the 1.0-second point, the liquid fraction in all four volumes of all twenty-five channels is 1.0.

TABLE 13-18
MINIMUM LIQUID FRACTION IN ALL FOUR VOLUMES
OF COOLANT CHANNELS 2, 3, 4 & 5 AT THE 0.5-SECOND POINT

Coolant Channel	Volume 1	Volume 2	Volume 3	Volume 4
2	0.859	0.584	0.216	0.600
3	0.001	0.003	0.004	0.001
4	0.010	0.004	0.004	0.001
5	0.334	0.091	0.050	0.006

Various heat transfer modes occur during the analyzed 2,500 seconds of the cold-leg break LOCA. The following list provides the RELAP5 heat transfer modes that occur and their descriptions. Modes indicate which regime is being used to transfer the heat between heat structure surfaces and the circulating fluid contained in the primary coolant system.

Mode	Description
2	Single-phase liquid convection at subcritical pressure, subcooled wall and low void fraction
3	Subcooled nucleate boiling
4	Saturated nucleate boiling
5	Subcooled transition boiling
6	Saturated transition boiling
7	Subcooled film boiling
8	Saturated film boiling
9	Single-phase vapor/gas or supercritical convection
10	Condensation when void fraction is less than one

If the non-condensable quality (based on vapor/gas mass) is greater than 1×10^{-9}, then 20 is added to the mode. Thus Mode 23 would be subcooled nucleate boiling with non-condensable quality greater than 1×10^{-9}. For comparison, with the reactor operating at a steady-state power level of 10 MW, the only heat transfer mode that occurs in the core is Mode 2.

The most challenging heat transfer modes occur for a fraction of a second within the first second of the transient in channels 3 and 4 when steam almost totally fills the coolant channels as flow rates pass through zero. The steam formed at that instant helps to quickly promote flow reversal, which in turn refills the coolant channels with liquid, thus limiting the duration of the more extreme heat transfer modes. Table 13-19 provides a list of the heat transfer modes in coolant channels 3 and 4 that occur at the 0.5-second point. After the 0.5-second point, Modes 22 and 23 occurs in the following number of coolant channels and volumes for about 1 to 2 seconds: 18 channels – volume 1, 16 channels – volume 2, 15 channels – volume 3, and 16 channels – volume 4. At the 1.0-second point, the heat transfer mode in all four volumes of channels 3 and 4 is Mode 2.

TABLE 13-19
HEAT TRANSFER MODES IN COOLANT CHANNELS 3 & 4 AT THE 0.5-SECOND POINT

Coolant Channel	Volume 1	Volume 2	Volume 3	Volume 4
3	Mode 9	Mode 6	Mode 4	Mode 9
4	Mode 4	Mode 6	Mode 6	Mode 9

As shown in Figures 13.8, and 13.13 through 13.17, substantial variation occurs in the fuel plate and coolant channel temperatures within the first 20 seconds of the LOCA. The movement of coolant in the pressure vessel and outlet piping region (U-loop) causes flow rate variations in the coolant channels, which in turn results in variations in coolant channel and fuel plate temperatures. Figure 13.7 shows the

overall downward flow rate through the core increasing from a minimum at 1.1 seconds to a maximum at around 4 seconds. The downward flow is caused by the coolant draining from the in-pool heat exchanger through its 6-inch (15.24-cm) inlet piping (indicated as reverse flow through junction 13901-40101). This action increases the amount of coolant that drains from the horizontal primary coolant inlet piping (volume 13901), through junction 13901-50101, to the pressure vessel (volume 50101), and down into the core. Coolant flow in the downward (normal) direction can still occur because coolant isolation valve V507A does not fully close until 9.5 seconds after the LOCA begins.

As net downward flow peaks between 3 to 4 seconds, reversed flow through coolant channels 2 through 5 decreases to about zero, which results in additional temperature peaks in the corresponding coolant channels and fuel plates. The higher temperatures result in the formation of steam that momentarily fills a major portion of the lower half of coolant channels 3, 4, and 5. This results in a significant jump in flow reversal in these channels at 4 seconds. A similar situation occurs between 13 to 14 seconds in channels 13 and 23. As downward flow approaches zero, steam also forms in these coolant channels, causing a quick reversal in flow to the upward direction.

As described above, voiding occurs in channels 2 through 5 between 3 to 4 seconds as flow decreases to zero. Table 13-20 provides a list of the minimum liquid fractions in all four volumes of these coolant channels. By the 4.2-second point, the liquid fraction in all four volumes of all twenty-five channels is back to 1.0.

TABLE 13-20
MINIMUM LIQUID FRACTION IN ALL FOUR VOLUMES OF
COOLANT CHANNELS 2, 3, 4 & 5 AT OR BEFORE THE 4.0-SECOND POINT

Coolant Channel	Volume 1	Volume 2	Volume 3	Volume 4
2	0.928	0.659	0.979	1.000
3	0.122	0.114	0.116	0.904
4	0.059	0.068	0.156	0.759
5	0.406	0.193	0.268	0.897

After the first second of the transient, the reduced flow in channels 3, 4 and 5 at the 4.0-second point causes the next most challenging heat transfer modes. Again, steam formed in the coolant channels helps to quickly re-promote flow reversal, as a result refilling the channels with liquid and preventing the higher heat transfer modes.

Table 13-21 lists the heat transfer modes in coolant channels 3, 4, and 5 that occur at the 4.0-seconds. Modes 22 and 23 occur again in several channels for about 1 to 2 seconds: 4 channels – volume 1, 3 channels – volume 2, 3 channels – volume 3, and 4 channels – volume 4. By 6.0 seconds, all four volumes of all twenty-five coolant channels are in Mode 2, with the exception of the two surfaces of fuel plate number-1 in volume 3, which are in Mode 3.

TABLE 13-21
HEAT TRANSFER MODES IN COOLANT CHANNELS 3, 4 & 5 AT THE 4.0-SECOND POINT

Coolant Channel	Volume 1	Volume 2	Volume 3	Volume 4
3	Mode 24	Mode 24	Mode 24	Mode 23
4	Mode 24	Mode 24	Mode 24	Mode 23
5	Mode 23	Mode 24	Mode 23	Mode 23

Between 13 and 14 seconds, as flow rates through channels 11, 12, 13, and 23 approach zero, steam formed in these channels causes a quick reversal to upward flow. Table 13-22 provides a list of the minimum liquid fractions that momentarily occur in all four volumes of these coolant channels. By the 14.0-second point, the liquid fraction in all four volumes of all twenty-five channels is 1.0.

TABLE 13-22
MINIMUM LIQUID FRACTION IN ALL FOUR VOLUMES
OF COOLANT CHANNELS 11, 12, 13 & 23 AT THE 13.0-SECOND POINT

Coolant Channel	Volume 1	Volume 2	Volume 3	Volume 4
11	0.985	0.916	1.000	1.000
12	0.908	0.513	0.947	0.999
13	0.933	0.324	0.339	0.927
23	0.780	0.081	0.201	0.173

The third flow transient, between 13 to 14 seconds, has only a minor effect on the heat transfer modes. When flow reversal occurs in channel 23, Mode 4 occurs for a fraction of a second in volumes 2 and 4.

After the initial three flow transients, Figure 13.7 illustrates a momentary increase in flow rate through all twenty-five coolant channels in the normal downward direction, 16 seconds after the LOCA begins. This increase in coolant flow is caused by the upward flow of air through core inlet check valve V502, which rapidly displaces the coolant in the piping above the check valve (volume 13701) into the horizontal inlet piping (volume 13901) and down into the pressure vessel (volume 50101). The cause of this air flow through check valve V502 is explained in detail in the next paragraph. Additionally, pressures within the volumes of the RELAP5 model, which create the differential pressures that cause this flow of air, are shown in Figure 13.18.

After the break occurs, coolant draining out of the cold-leg rupture creates a pressure of less than 10 psia (69 kPa) in the primary coolant piping below check valve V502 (inlet side). After 10 seconds, enough coolant has drained to allow the return of air, which then causes pressure in the piping to increase towards atmospheric pressure. While coolant is draining out of the cold-leg piping, air pressure in the voided volume of the in-pool hot-leg piping is also drawn into a vacuum as coolant flows down past isolation valve V507A. At 9.5 seconds after the LOCA started, pressure decreases to around 10 psia

(69 kPa) as isolation valve V507A fully closes. Between 12 to 14 seconds, pressure on the inlet side of check valve V502 (volume 13507) increases to a few psi greater than the pressure on the outlet side (volume 13701). This differential pressure causes the check valve to chatter open, allowing air to flow upward while some of the water leaks downward past the valve during these momentary openings. Just prior to 16 seconds, the check valve opens far enough to cause a rapid upward flow of air, which forces some of the coolant up into the horizontal primary piping (volume 13901), which in turn then drains into the upper portion of the pressure vessel. This slug of coolant flow through the core causes coolant channel and fuel plate temperatures to drop 20 to 30 °F (11 to 17 °C) in less than a second. This is the final addition of coolant into the U-loop as determined by the MURR RELAP5 model.

From 20 to 80 seconds, the overall coolant channel and fuel plate centerline temperatures increase as coolant channels 14 through 22 independently enter flow reversal conditions. By 72 seconds, all channels have upward flow with the exception of channels 1, 24, and 25, which still maintain normal downward flow. At this point, the transfer of heat from the primary coolant (from coolant channels 1 and 25, and the coolant above and below the core) is to the reactor pool through the inner and outer reactor pressure vessels.

Coolant channel temperatures rise and peak as flow rates go through zero during the flow reversal periods. A peak fuel plate centerline temperature of 229.6 °F (109.8 °C) occurs at 74.2 seconds in plate number-2 (second section down – the five inches (12.7 cm) above core horizontal centerline). The fourth time that liquid fractions decrease below 0.8 occur around the 70-second point when coolant channel 17 enters flow reversal. The other smaller momentary voiding events that occur between 30 and 80 seconds are due to flow reversals in the other channels.

From 80 to 2,500 seconds, flows through coolant channels 1 and 25 remain in the normal downward direction. Flow through channel 24 starts in the downward direction but goes through a series of reversals between 250 to 500 seconds, with the most significant flow reversal occurring at 325 seconds. At this point, it reverses again to normal downward flow for about a 30-second period. It then cycles between upward and downward flow a few times over the next 120 seconds. These reversals disturb the flow in the other coolant channels and cause momentary up and down spikes in channel and fuel plate temperatures. Other than the initial temperature oscillations and those caused by channel 24, the peak coolant channel and fuel plate temperatures are fairly constant over the first 600 seconds of the LOCA.

After the first 80 seconds, the peak coolant channel temperature is approximately 215 °F (102 °C) for the 600 second period. This results in fairly constant peak centerline temperatures of 225 to 230 °F (107 to 110 °C) in all of the fuel plates except for plate number-1 and number-24, which are lower because of the cooler inner and outer coolant channels. A peak centerline temperature of 229.9 °F (109.9 °C) occurs in the third section (the 5 inches (12.7 cm) below core centerline) of plate number-2 at 325 seconds, when flows are disturbed by the flow reversal in channel 24. This also causes flow in channel 4 to momentarily go to zero, resulting in a peak coolant channel temperature of 220.2 °F (104.6 °C). The

final time that liquid fractions in the coolant channels are momentarily below 0.8 occurs at the 325-second point.

After 80 seconds, the heat transfer modes are Mode 2 and 3, except for a momentary Mode 4 which occurs in channel 3 at 322 seconds. During the interval between the 70- and 325-second points (times when liquid fractions are less than 0.8), all of the coolant channels, except for channels 1, 24 and 25, frequently vary in liquid fraction between 1.0 and 0.96 to 0.98 due to nucleate boiling in the upper two volumes. The top two volumes of most coolant channels are in heat transfer Mode 3 from 60 to a little more than 500 seconds. Some Mode 4 occurs in the top volume during this same period. The five inches (12.7 cm) below core centerline (volume 3), the region where fuel plate peak power densities occur, is mainly in Mode 3 from 60 to around 400 seconds. In volume 4, coolant channel 3 is in Mode 3 about half the time from 60 to 240 seconds while channel 23 is in Mode 3 from around 100 to 300 seconds. Otherwise, heat transfer is in Mode 2 during this period.

After 500 seconds into the LOCA, no significant changes occur. Flow rates through channels 1 and 25, the only channels that maintained downward flow throughout the transient, slightly decrease over this time period. Channel 24 continually oscillates between slight upward flow to about three times greater downward flow. Because of a reduction in the generation of decay heat, coupled with the large heat sink that the reactor pool provides, coolant channel and fuel plate temperatures steadily begin to decrease after 500 seconds. At the end of 2,500 seconds, the peak coolant channel and fuel plate centerline temperatures are 175.3 °F (79.6 °C) and 188.9 °F (87.2 °C), respectively.

During the cold-leg break LOCA, more than 6 feet (1.8 m) of coolant is maintained above the top of the fuel plates. Additionally, the reactor has two small check valves (V550C/D) installed in parallel on the horizontal primary inlet piping between check valve V502 and the top of the pressure vessel. These check valves allow pool water (approximately 5.3 gpm (20 lpm) at 17 feet of static head (50.8 kpa)) to be admitted into the coolant piping when pool water pressure is greater on the inlet side of the valves than on the primary coolant system outlet side. Operation of these check valves is conservatively not included within the MURR RELAP5 model.

13.2.3.2.2 Rupture of Piping Between Valve V507A and the Reactor Pool

During a rupture of the primary coolant piping between isolation valve V507A and the reactor pool (hot-leg break), the decrease in primary coolant circulation pump discharge pressure and coolant flow is much slower than what occurs during a cold-leg break, and actually closer to what happens during a loss of flow accident. With the rupture occurring just upstream of valve V507A, a path is created which allows coolant to flow from the inverted loop down the piping and out the open rupture. This, combined with the air pressure that is admitted to the top of the inverted loop by the anti-siphon system, causes the section of primary piping from the top of the inverted loop to isolation valve V507A to quickly drain (volumes 1020100-1020700 within Appendix C).

The highest peak center line temperature of 281.2 °F (138.4 °C) occurs in fuel plate number-1, 0.2 seconds after the LOCA begins (see Figure 13.19). This peak centerline temperature is caused by the

combined effect of plate number-1 having the highest power peaking factor and the reduction in coolant flow which starts a fraction of a second before the control rods begin to drop. After this initial peak temperature at the start of the transient, the next highest fuel plate centerline temperature of 231.7 °F (110.9 °C) occurs in plate number-22 at 22 seconds. As shown in Figure 13.20, the highest coolant channel temperature of 219.0 °F (138.4 °C) occurs in channel 7 at 123.3 seconds and in channel 6 at 123.4 seconds.

The RELAP5 analysis shows that during the hot-leg break LOCA, heat transfer Modes 2 and 3 predominate. An occasional Mode 4 occurs in the top volume of some channels from 90 to 340 seconds. The only heat transfer modes that occur which are greater than Mode 4, Modes 10, 22 and 23. Modes 22 and 23 only occur in volume 1 of channels 2 and 3 between 12 to 13 seconds when flow reversals take place in these channels. Occasionally a few Mode 10 heat transfers occur in various channels during the accident.

13.2.3.3 Conclusions

The consequences of a loss of coolant due to a break in a section of primary coolant piping, including a double-ended rupture of the largest diameter coolant pipe, have been analyzed. Ruptures in sections of in-pool primary coolant piping, as well as ruptures at different locations relative to the reactor pool and the inlet and outlet isolation valves, have been evaluated. None of the postulated scenarios would result in the uncovering of the core or core damage, including the most serious accident which is a double-ended rupture of the primary coolant piping between isolation valve V507B and the reactor pool (cold-leg break).

Another part of the safety analysis relating to the LOCA involved the consideration of events that would occur once coolant flow (and a loss of coolant) was stopped, and decay heat was being dissipated by the remaining coolant in the partially drained reactor pressure vessel. The conclusions from the analysis of decay heat rejection indicated that, after a reactor shutdown from full power with an accompanying LOCA resulting in more than 6 feet (1.8 m) of water above the core, decay heat can safely be dissipated to the reactor pool with no core damage. Because of the large heat sink that is created by the reactor pool, any steam postulated from accident boiling will quickly condense in the empty piping and in-pool heat exchanger and drain back into the reactor pressure vessel.

Therefore, MURR possesses sufficient redundant safety features to prevent core damage as a result of the double-ended rupture of the largest diameter primary coolant piping and requires no additional emergency core cooling system for core protection in the event of a LOCA.

13.2.4 Loss of Primary Coolant Flow

13.2.4.1 Accident-Initiating Events and Scenarios

A loss of flow (LOF) accident for the primary coolant system can be initiated by any one, or a combination, of the following anomalies:

(a) Loss of facility electrical power (or coolant circulation pump power);
(b) Inadvertent closure of coolant loop isolation valve(s);
(c) Inadvertent loss of pressurizer pressure;
(d) Locked rotor in a coolant circulation pump; and
(e) Failure of a coolant circulation pump coupling.

Any of these five anomalies is considered to be possible for the MURR facility, but a loss of flow (by any means) without an accompanying reactor scram is not considered credible because of the redundancy in the reactor safety system. Because MURR is a downward flow reactor, the analysis must consider such challenging design features as flow stagnation and reversal that will occur during the accident.

A simplified schematic diagram of the MURR primary coolant system is shown in Figure 13.21. The system actually contains two (2) pumps and two (2) heat exchangers operating in parallel. For simplicity, only one (1) pump and one (1) heat exchanger are shown. There is also an additional air-operated in-pool heat exchanger isolation valve installed in parallel to the one depicted.

While the five types of LOF accidents listed above were analyzed, only the results of the worst-case accident [accident (c)] will be discussed in this report. LOF accidents (a) and (b), "loss of facility electrical power" and "inadvertent closure of coolant loop isolation valves," respectively, were also analyzed but are not discussed herein because they were found to be less serious accidents from the standpoint of reactor safety. LOF accidents (d) and (e), "locked rotor in a coolant circulation pump" and "failure of a coolant circulation pump coupling," will not result in a total loss of primary coolant flow. These accidents affect only one (1) of the pumps and the final flow will be approximately one-half of the initial flow. This reduced flow will result in a scram as sensed by any one of the five primary coolant flow detectors and no hazard exists. (Reference 13.52)

Table 13-17 provides a comparison of the normal reactor operating parameters and the conservative assumptions that were used in the loss of pressure LOF accident analysis when the transient starts.

13.2.4.2 Accident Analysis and Consequences

The worst-case LOF accident was analyzed using the MURR RELAP5 model, which is described in Appendix C. The LOF accident caused by a loss of pressure in the primary coolant system pressurizer can be initiated by anomalies such as a break in any of the piping penetrations near the top of the pressurizer, a failure of the pressurizer relief valve, or pressurizer nitrogen vent valve V545 failing in the open position. As pressurizer pressure decreases, primary coolant system pressure will also decrease as sensed by the following four pressure transmitters and sensors: PT 944A, PT 944B, PT 943 and PS 938. Each pressure transmitter or sensor will initiate a reactor scram when system pressure decreases to approximately 95% of the normal operating value.

In addition to initiating a reactor scram, PT 944A and PT 944B will de-energize relays 2K13 and 2K28; either of which will cause the following actions to occur:

1. Primary coolant circulation pumps P501A and P501B will stop;
2. Primary coolant isolation valves V507A and V507B will close; and
3. Anti-siphon system isolation valves V543A and V543B will open.

As primary coolant isolation valves V507A and V507B leave their fully-open position, a limit switch on each valve actuator will cause relays 2K10, 2K11, and 2K17 to de-energize, and, in addition to causing a reactor scram (which has already been initiated), the following actions to occur:

- Actions 1 and 3 listed above;
- In-pool heat exchanger isolation valves V546A and V546B will open; and
- Pressurizer surge line isolation valve V527C will close.

A reduction in primary coolant flow caused by pumps P501A and P501B stopping will also cause valves V546A and V546B to open when flow decreases to approximately 90% of the normal operating value as sensed by differential pressure sensors DPS 929, DPS 928A and DPS 928B. The MURR RELAP5 model of the loss of pressure LOF accident has valves V546A and V546B opening due to the reduction in flow as sensed by the above listed differential pressure sensors, which is before isolation valves V507A and V507B come off their open seats.

The sequence of events described above will cause the reactor to go from operating at full power to a shutdown condition, with an accompanying loss of forced primary coolant circulation, within a few seconds. Figure 13.22 depicts the RELAP5 derived flow rate through the core as it transitions from forced to natural circulation with the primary coolant isolation valves closed. As stated above, when primary coolant system pressure decreases to approximately 90% of the normal operating value (a pressure that corresponds to 75 psia (517 kPa above atmosphere) in the pressurizer), the following actions will occur: a reactor scram, primary coolant circulation pumps stop, primary coolant isolation valves close, anti-siphon system isolation valves open, in-pool heat exchanger isolation valves open, and pressurizer surge line isolation valve closes. As experimentally determined, primary coolant isolation valve V507B will fully close in 8.4 seconds after initiation of the trip, securing the forced circulation path. Primary coolant isolation valve V507A will fully close in 9.1 seconds (after initiation of the trip) and isolate the reactor core to only the in-pool portion of the primary coolant system.

For the loss of pressure LOF accident analysis, the MURR RELAP5 code was conservatively modeled with both primary coolant isolation valves fully closed 9.5 seconds after initiation of the reactor scram. With RELAP5, it was determined that downward flow will still exist through fuel element coolant channels 14 through 25 when the isolation valves are fully closed. Flow through coolant channels 1 through 13 will almost immediately reverse, and flow in the upward direction as the valves go fully closed. Curves depicting the upward, downward, and net coolant flows through the core immediately after the isolation valves are fully closed are shown in Figure 13.23. Positive values along the Y-axis indicate downward flow, whereas negative values indicate upward flow, or flow reversal, through the core. A maximum flow rate of 61 gpm (3.85 lps) through the in-pool heat exchanger occurs when net flow, which is in the upward direction, peaks at 69 seconds after the LOF transient begins. At this point in

time, downward flow through coolant channel 25 is about 7 gpm (0.5 lps), while the combined flow rate through the other 24 channels is in the upward direction at about 68 gpm (4.3 lps).

The individual flow rates through all twenty-five coolant channels are shown in Figure 13.24. Channel 1, with a coolant gap of 0.095 inches (2.413 mm), is located between the outside diameter of the inner pressure vessel wall and the concave side of fuel plate number-1. Channel 25, with a coolant gap of 0.075 inches (1.905 mm), is located between the convex side of fuel plate number-24 and the inside diameter of the outer pressure vessel wall. The remainder of the coolant channel gaps are 0.080 inches (2.032 mm) wide. As previously mentioned, downward flow will still exist in channels 14 through 25 even after the isolation valves are fully closed but will first start reversing in channel 4 about 12 seconds after initiation of the transient and end in channel 19 about 10 seconds later. The following is the order in which flow reversal occurs (through the channels): 14, 15, 23, 16, 22, 21, 17, 24, 20, 18 and 19. Flow reversal is initiated by the increase in coolant temperature within those channels. Coolant flow through channel 25 never reverses direction but reaches a minimum downward flow rate when the isolation valves close and then starts increasing, peaking about 24 seconds after the transient begins. Flow through channel 1, due to the cooler pool water inside the inner pressure vessel, is in the upward direction for only a few seconds and then reverses again to downward flow at 15 seconds. Flow through channel 1 continues in the downward direction until it peaks at 26 seconds, then decreases until it reverses for a third time in the upward direction at 55 seconds. As previously stated, flow through channel 25 never reverses and remains in the downward direction because of the heat that is being transferred from the primary coolant through the outer pressure vessel wall to the bulk pool water. The peak flow rates through coolant channels 1 and 25 occur around the 22- to 24-second mark, while flow through channels 14 through 24 peak in their flow reversal at around 19 to 26 seconds.

Figure 13.25 shows the centerline temperatures in the third section of all twenty-four fuel plates during the loss of pressure LOF accident. Because primary coolant flow starts decreasing a fraction of a second before reactor power starts to decrease, the highest fuel plate centerline temperature of 280.3 °F (137.9 °C) occurs in fuel plate number-1, 0.3 seconds into the transient. Fuel plate centerline temperatures then decrease as reactor power decreases from the insertion of the control blades. After the first second of the transient, the highest centerline temperature of 277.9 °F (136.6 °C) occurs in fuel plate number-22 during a flow reversal event, 17 seconds after the loss of pressure transient starts. When this peak centerline temperature is reached, plate number-22 cladding temperatures are 277.6 °F (136.4 °C) and 277.7 °F (136.5 °C). These temperatures are slightly higher than the cladding temperatures (269.9 °F (132.2 °C) and 271.9 °F (133.3 °C)) of fuel plate number-1, when the initial peak centerline temperature of 280.3 °F (137.9 °C) was reached during the first second of the transient. After the loss of forced primary coolant circulation, the peak fuel plate centerline temperatures occur during the flow reversal periods for the adjacent coolant channels.

Figure 13.26 shows the temperatures of all twenty-five coolant channels. A peak coolant temperature of 237.5 °F (114.2 °C) occurs in the third volume of channel 19, which is the area from core horizontal centerline to 5 inches below centerline. The fluid saturation temperature at the top of the reactor (which is the area of lowest pressure in the core) is 277 °F (136 °C). At a temperature of 237.5 °F (113.2

°C), the coolant is subcooled by approximately 40 °F (22 °C). The only heat transfer mode that occurs during the accident is Mode 2 (single-phase liquid convection), except for a fraction of second at the beginning of the transient when total coolant flow is decreasing before a reactor scram causes power to decrease. Momentary Mode 3 heat transfer (subcooled nucleate boiling) occurs during this first second in the peak heat flux regions of fuel plates number-1 and number-24. Although the highest temperature that a fuel plate reaches during the transient is 280.3 °F (137.9 °C), that is still more than 500 °F (277.8 °C) below the 900 °F (482 °C) temperature at which MURR fuel plates are tested to during fabrication to verify that no blistering will occur. The benchmarking performed on the RELAP model for the LOCA and LOFA analyses is described in Reference 13.55.

13.2.4.3 Conclusions

The accident discussed in this section represents the worst-case LOF accident that could be realized at MURR. For each scenario that was analyzed, the initial conditions used were conservative, with some of the coolant system parameters at the limiting safety system setting and with reactor power at 11 MW. For accident (b), "inadvertent closure of coolant loop isolation valve(s)," it was assumed that the reactor scram function from the closure of the coolant isolation valves had failed, and a scram did not occur until initiated by any one of five primary coolant flow sensors. Since there is no cladding failure predicted for any of the accidents under these conservative conditions, it is concluded that reactor safety is not jeopardized by any type of LOF accident. Finally, as discussed in Section 13.2.5.1, any fuel element damage that may occur from the partial flow blockage of a fuel element channel is bounded within the Fuel Failure during Reactor Operation analysis.

13.2.5 Mishandling or Malfunction of Fuel

Events or scenarios which could cause an accident in this category include 1) a fuel handling event where an element is dropped while in a cask, or dropped underwater and damaged severely enough to breach the cladding, 2) simple failure of the fuel cladding due to a manufacturing defect or corrosion, and 3) overheating of a fuel element with subsequent potential cladding failure due to a loss of primary coolant or coolant flow.

13.2.5.1 Fuel Failure during Reactor Operation

13.2.5.1.1 Accident-Initiating Events and Scenarios

Many types of accidents have been considered in conjunction with the operation of MURR. In all cases, safety systems have been designed such that the likelihood of an accident involving the release of a significant amount of fission products has essentially been eliminated. The safety systems take the form of automatic reactor shutdown circuits and process systems designed to ensure, through redundancy, that the reactor will shut down upon a significant deviation from normal operating conditions. The reactor is housed within a containment building which provides further protection against a significant release of radioactive material to the environment.

In the "Fuel Failure during Reactor Operation" accident for MURR, it is assumed that an accident condition has caused the melting of the number-1 fuel plate in four (4) separate fuel elements (Reference 13.8, Reference 13.11, and Reference 13.49). It is further assumed that the four (4) number-1 fuel plates are in the peak power region of the core.

While one might postulate that this accident could result from a partial flow blockage to the fuel, mitigating features such as the primary coolant system strainer, the fuel element end-fittings, and the pre-operational visual inspection of the reactor pressure vessels and core region following any fuel handling evolution, all prevent an accident of this type from occurring. In addition, it has been shown that a 75% blockage of coolant flow to the hot channel is insufficient to cause cladding failure (Reference 13.2).

13.2.5.1.2 Accident Analysis and Consequences

The fuel failure during reactor operation accident postulates partial fuel melting with an associated release of fission products into the primary coolant system. The accident is assumed to occur with the primary coolant system operating, resulting in a quick dispersal of the fission products throughout the system. With the design of the primary coolant system and its associated systems, particulate activity will remain in the coolant, and the gaseous activity that comes out of solution will collect in the reactor loop vent system and be retained there. Therefore, the primary coolant system relief valves and pressurizer are the only paths for the release of significant quantities of fission products to the environment.

The potential energy release from the melting of four (4) number-1 fuel plates could occur as a possible metal-water reaction (Reference 13.3). While hydrogen would be formed, it is highly unlikely that in a water environment a hydrogen deflagration reaction would occur. The amount of material which would be involved in a metal-water reaction under the conditions of four (4) number-1 fuel plates melting is not predictable as the amount depends upon many conditions. For purposes of calculation, it is conservatively assumed that all the fuel plate aluminum cladding exposed in the area is involved in the reaction. The uranium-aluminide dispersion fuel core contains a total of 33.56 Kg of aluminum. Of this, 1.3% or 436 grams is assumed to react according to the following equation:

$$Al + nH_2O \rightarrow AlO_n + nH_2 + heat$$

The energy release per Kg of aluminum is 18 MW-sec, for a total energy release of:

$$7.9 \text{ MW-sec} = 7.5 \times 10^3 \text{ BTU}$$

This amount of heat would easily be transferred to the adjacent fuel elements and primary coolant in the reactor core. Additionally, any steam that would form in the vicinity of the molten area would also assist in dissipating the heat. Since the fuel failure would result in a negligible release of energy to the primary coolant system, the introduction of pressure surges, which could lift the primary relief valves,

are not considered credible. The pressurizer is an isolated system, and since no significant pressure surges are anticipated, it will not be subject to mixing with the primary coolant system.

Any significant gaseous radioactivity entrapped in the reactor loop vent tank will cause a reactor scram and actuation of the containment building isolation system by action of the pool surface radiation monitor. Additionally, following the actuation of the anti-siphon system when the primary coolant system is secured, gases could also collect in the anti-siphon pressure tank. The location of these tanks under the pool surface, and the shielding provided by the water and the biological shield, will significantly reduce any radiation exposure to the reactor staff, visitors, or researchers.

Fission products entrapped in the primary coolant system can be removed by the reactor coolant cleanup system. This cleanup procedure would be undertaken under closely monitored and controlled conditions.

The primary coolant system does experience some coolant leakage into the reactor pool through the pressure vessel head packing and flange gasket. This leakage is typically less than 40 gallons (151 l) per week; an almost imperceptible leakage rate of approximately 4×10^{-3} gallons of primary coolant per minute into the pool. However, for purposes of calculation, a leakage rate of 80 gallons (303 l) per week is used. Based on this assumed conservative leakage rate, the radiation exposure to personnel in the containment building following fuel failure is calculated below.

Due to the complex nature of the MURR mixed core fuel cycle, the source terms for the "Fuel Failure during Reactor Operation" and "Fuel Handling Accident" were determined using the computer program MONTEBURNS instead of simplistic ORIGEN runs (Reference 13.42). MONTEBURNS is a coupled MCNP-ORIGEN code system developed by Los Alamos National Laboratory (LANL). Within the MONTEBURNS program, MCNP calculations are used to obtain accurate one-group fluxes and one-group cross sections that are then utilized by ORIGEN for fuel depletion and fission product activity calculations. The use of MONTEBURNS and the flow diagram utilized for various neutronic computations are detailed in reference 13.47.

TABLE 13-23
RADIOIODINE AND NOBLE GAS ACTIVITIES IN THE CORE FROM FUEL FAILURE

I-131 – 2.20×10^5 Ci	Kr-85 – 4.63×10^2 Ci	Xe-133 – 3.85×10^5 Ci
I-132 – 3.08×10^5 Ci	Kr-85m – 1.31×10^5 Ci	Xe-135 – 7.56×10^4 Ci
I-133 – 5.42×10^5 Ci	Kr-87 – 2.05×10^5 Ci	Xe-135m – 3.62×10^4 Ci
I-134 – 6.11×10^5 Ci	Kr-88 – 2.91×10^5 Ci	Xe-137 – 4.81×10^5 Ci
I-135 – 5.06×10^5 Ci	Kr-89 – 3.69×10^5 Ci	Xe-138 – 5.01×10^5 Ci
	Kr-90 – 3.68×10^5 Ci	Xe-139 – 4.07×10^5 Ci

An unirradiated fuel plate number-1 contains, on average, 19.26 grams of U-235, so four (4) unirradiated number-1 fuel plates contain 77.04 grams of U-235. The four number-1 fuel plates that melt correspond to 1.41% of the total U-235 in the Week 58 Core that was used to determine the high-power peaking

factor for the revised Safety Limits. The Week 58 Core has a total power history of 576 MWd. This power history results in a total reduced core mass of 5,474 grams of U-235 due to the previous fuel consumption. This 1.41% of U-235 melting releases 3.42% of the core fission products due to the highest power density fuel plate number-1 overall power peaking factor of 2.423, which is conservatively assumed to apply to all four (4) number-1 fuel plates (1.41% x 2.423 = 3.42%).

A conservative value of a 100% release of the radioiodine and noble gas fission products from the fuel is assumed in calculating the fission product inventory in the primary coolant system. It is also assumed that fission products released into the primary coolant are quickly and uniformly dispersed within the 2,000-gallon (7,571 l) primary coolant system volume and, during a normal week's operation, 80 gallons (7.9×10^{-3} gpm) of coolant leaks from the primary coolant system into the pool water. The radioactivity released into the reactor pool in 10 minutes (determined to be the maximum personnel occupancy time in the containment building after the accident for necessary operational personnel) is as follows:

It takes approximately five (5) minutes for Operations personnel to verify that the containment building has been evacuated following a containment building isolation. For the purposes of the fuel failure calculations, a conservative assumption of 10 minutes is used.

Example calculation of I-131 released into the reactor pool during the 10-minute leakage period:

= I-131 in fuel x 0.0342 x 1/2,000 gal x (7.9×10^{-3} gpm) x 10 min

= 2.20×10^{5} Ci x 0.0342 x 1/2000 gal x 0.079 gal

= 0.297 Ci

= 2.97×10^{5} µCi

The same calculation is used for the isotopes listed in Table 13-24.

TABLE 13-24
RADIOIODINE AND NOBLE GAS ACTIVITIES RELEASED INTO THE POOL
AFTER 10 MINUTES FOLLOWING FUEL FAILURE

I-131 – 2.97×10^{5} µCi	Kr-85 – 6.25×10^{2} µCi	Xe-133 – 5.20×10^{5} µCi
I-132 – 4.16×10^{5} µCi	Kr-85m – 1.77×10^{5} µCi	Xe-135 – 1.02×10^{5} µCi
I-133 – 7.32×10^{5} µCi	Kr-87 – 2.77×10^{5} µCi	Xe-135m – 4.89×10^{4} µCi
I-134 – 8.25×10^{5} µCi	Kr-88 – 3.93×10^{5} µCi	Xe-137 – 6.50×10^{5} µCi
I-135 – 6.84×10^{5} µCi	Kr-89 – 4.98×10^{5} µCi	Xe-138 – 6.77×10^{5} µCi
	Kr-89 – 4.97×10^{5} µCi	Xe-139 – 5.50×10^{5} µCi

Fission products released into the reactor pool will be detected by the pool surface and ventilation system exhaust plenum radiation monitors. For the purposes of this analysis, it is assumed that a

reactor scram and actuation of the containment building isolation system occurs by action of the pool surface radiation monitor.

The radioiodine released into the reactor pool over a 10-minute interval is conservatively assumed to be instantly and uniformly mixed into the 20,000 gallons (75,708 l) of bulk pool water, which then results in the following pool water concentrations for the radioiodine isotopes. The water solubility of the krypton and xenon noble gases released into the pool over this same time period are ignored and they are assumed to pass immediately through the pool water and evolve directly into the containment building air volume where they instantaneously form a uniform concentration in the isolated structure.

TABLE 13-25
RADIOIODINE CONCENTRATIONS IN THE POOL WATER AT 10 MINUTES FOLLOWING FUEL FAILURE

I-131 – 1.49×10^1 µCi/gal	I-133 – 3.66×10^1 µCi/gal	I-135 – 3.42×10^1 µCi/gal
I-132 – 2.08×10^1 µCi/gal	I-134 – 4.13×10^1 µCi/gal	

When the reactor is at 10 MW and the containment building ventilation system is in operation, the evaporation rate from the reactor pool is approximately 80 gallons (302.8 L) of water per day. For the purposes of this calculation, it is assumed that a total of 40 gallons (151.4 L) of pool water containing the previously listed radioiodine concentrations evaporates into the containment building over the 10-minute period. This assumption results in about seventy times more radioiodine in the containment building air than would be present at the end of the 10 minutes of evaporation. In addition, air with a temperature of 75 °F (23.9 °C) and 100% relative humidity contains H_2O vapor equal to 40 gallons (151.4 L) of water. Since the air in containment is normally at about 50% relative humidity, containing approximately 20 gallons (75.7 L) of water vapor, the assumed addition of 40 gallons (151.4 L) of water vapor overestimates by a factor of two the amount of water that would make the containment air saturated. It is also conservatively assumed that all of the radioiodine activity in the 40 gallons (151.4 L) of pool water that evaporates instantaneously forms a uniform concentration in the containment building air. When distributed into the containment building, this would result in the following radioiodine concentrations in the 225,000 ft^3 (6,371.3 m^3) air volume:

Example calculation of average I-131 in containment air during the tenth minute:

= I-131 concentration in pool water at 10 min x (9.5 min/10 min) x 40 gal x (9.5 min/10 min) x 1/225,000 ft^3 x 1 ft^3/28,317 cc x e^[(-0.693 x 9.5 min) / (8.02 day x 1440 min/day)]

= (1.49×10^1 µCi/gal) x 0.95 x 38 gal x 1/225,000 ft^3 x 1 ft^3/28,317 ml x 0.9994

= 8.41×10^8 µCi/ml

The average radioiodine concentrations are determined for the first and last minute and four 2-minute concentrations for the other eight (8) minutes. The overall average concentration is then calculated.

TABLE 13-26
AVERAGE RADIOIODINE CONCENTRATIONS IN THE CONTAINMENT BUILDING AIR
DURING THE 10 MINUTES FOLLOWING FUEL FAILURE

I-131 – 3.08×10^{-8} µCi/ml	I-133 – 7.56×10^{-8} µCi/ml	I-135 – 7.00×10^{-8} µCi/ml
I-132 – 4.16×10^{-8} µCi/ml	I-134 – 7.77×10^{-8} µCi/ml	

As noted previously, the krypton and xenon noble gases released into the reactor pool from the primary coolant system during the assumed 10-minute interval following fuel failure are assumed to pass immediately through the pool water (ignoring their solubility in water) and enter the containment building air volume where they instantaneously form a uniform concentration in the isolated structure. (Within 10 minutes the primary coolant system is shut down and the leakage driving force is stopped.) Based on the 225,000-ft^3 (6,371.3 m^3) volume of containment building air and the previously listed quantities of these gases released into the reactor pool, the average noble gas concentrations in the containment building for each time interval of the 10 minutes would be as follows:

Example calculation of average Kr-85 released into containment air during the 2-minute time interval between three (3) to five (5) minutes:

= Total Kr-85 activity x (4 min/10 min) x 1/ (225,000 ft^3 x 28,317 cc/ft^3) x e^[(-0.693 x 4 min) / (10.76 yr x 365.25 days/yr x 1,440 min/day)]

= 6.25×10^2 uCi x 0.4 /(6.371×10^9 ml) x 1.0

= 3.93×10^{-8} µCi/ml

The 10-minute average noble gas concentrations are calculated from the average of the first and last 1-minute time intervals and the other four 2-minute time intervals.

TABLE 13-27
AVERAGE NOBLE GAS CONCENTRATIONS IN THE CONTAINMENT BUILDING
AIR DURING THE 10 MINUTES FOLLOWING FUEL FAILURE

Kr-85 – 4.91×10^{-8} µCi/ml	Xe-133 – 4.08×10^{-5} µCi/ml
Kr-85m – 1.37×10^{-5} µCi/ml	Xe-135 – 7.95×10^{-6} µCi/ml
Kr-87 – 2.05×10^{-5} µCi/ml	Xe-135m – 2.86×10^{-6} µCi/ml
Kr-88 – 3.00×10^{-5} µCi/ml	Xe-137 – 1.69×10^{-5} µCi/ml
Kr-89 – 1.05×10^{-5} µCi/ml	Xe-138 – 3.86×10^{-5} µCi/ml
Kr-90 – 4.84×10^{-7} µCi/ml	Xe-139 – 8.08×10^{-7} µCi/ml

Based on the source term data provided, it is possible to determine the radiation dose to the thyroid from radioiodine and the dose to the whole body resulting from submersion in the airborne noble gases and radioiodine inside the containment building.

Because the airborne radioiodine source is composed of five (5) different iodine isotopes, it will be necessary to determine the dose contribution from each individual isotope and then sum the results. Dose Multiplication Factors were established using the Derived Air Concentrations (DACs) for the "listed" isotopes in Appendix B of 10 CFR 20 and calculated values for the four (4) "unlisted" submersion isotopes (Kr-89, Kr- 90, Xe-137, and Xe-139). The submersion DAC values that were calculated were done in accordance with the data and methodology as supplied in Federal Guidance Report (FGR) No.12 (Reference 13.49, Attachment 4).

Example calculation of thyroid dose due to I-131:

The DAC can also be defined as 50,000 mrem (thyroid target organ limit)/2,000 hrs, or 25 mrem/DAC-hr. Additionally, 10 minutes of one DAC-hr is 1.67×10^{-1} DAC-hr.

I-131 concentration in containment = 3.08×10^{-8} µCi/ml

I-131 DAC (10 CFR 20) = 2.00×10^{-8} µCi/ml

Dose Multiplication Factor = (I-131 concentration) / (I-131 DAC)

= (3.08×10^{-8} µCi/ml) / (2.00×10^{-8} µCi/ml)

= 1.54

A 10-minute thyroid exposure from I-131 is:

= Dose Multiplication Factor x DAC Dose Rate x 10 minutes

= 1.54 x (25 mrem/DAC-hr) x (1.67×10^{-1} DAC-hr)

= 6.42 mrem

The same calculation is used for the radioiodines listed in Table 13-28.

TABLE 13-28
DERIVED AIR CONCENTRATION VALUES AND 10-MINUTE EXPOSURES
– RADIOIODINE – FOLLOWING FUEL FAILURE

Radionuclide	**Derived Air Concentration**	**10-Minute Exposure**
I-131	2.00×10^{-8} µCi/ml	6.42×10^{0} mrem
I-132	3.00×10^{-6} µCi/ml	5.78×10^{-2} mrem
I-133	1.00×10^{-7} µCi/ml	3.15×10^{0} mrem
I-134	2.00×10^{-5} µCi/ml	1.62×10^{-2} mrem
I-135	7.00×10^{-7} µCi/ml	4.17×10^{-1} mrem
Total = 10.07 mrem		

Doses from the kryptons and xenons present in the containment building are assessed in much the same manner as the radioiodines, and the dose contribution from each individual radionuclide must be calculated and then added together to arrive at the final noble gas dose. Because the dose from the noble gases is only an external dose due to submersion, and because the DACs for these radionuclides are based on this type of exposure, the individual noble gas doses for 10 minutes in containment were based on their average concentration in the containment air and the corresponding DAC.

Example calculation of whole body dose due to Kr-85:

The DAC can also be defined as 5,000 mrem / 2,000 hrs, or 2.5 mrem/DAC-hr. Additionally, 10 minutes of one DAC-hr is 1.67×10^{-1} DAC-hr.

Kr-85 concentration in containment = 4.91×10^{-8} µCi/ml

Kr-85 DAC (10 CFR 20) = 1.00×10^{-4} µCi/ml

Dose Multiplication Factor = (Kr-85 concentration) / (Kr-85 DAC)

= (4.91×10^{-8} µCi/ml) / (1.00×10^{-4} µCi/ml)

= 4.91×10^{-4}

A 10-minute whole body exposure from Kr-85 is:

= Dose Multiplication Factor x DAC Dose Rate x 10 minutes

= 4.91×10^{-4} x (2.5 mrem/DAC-hr) x (1.67×10^{-1} DAC-hr)

= 2.05×10^{-4} mrem

The same calculation is used for the noble gases listed in Table 13-29.

TABLE 13-29
DERIVED AIR CONCENTRATION VALUES AND 10-MINUTE EXPOSURES
– NOBLE GASES – FOLLOWING FUEL FAILURE

Radionuclide	Derived Air Concentration	10-Minute Exposure
Kr-85	1.00×10^{-4} µCi/ml	2.05×10^{-4} mrem
Kr-85m	2.00×10^{-5} µCi/ml	2.85×10^{-1} mrem
Kr-87	5.00×10^{-6} µCi/ml	1.71×10^{0} mrem
Kr-88	2.00×10^{-6} µCi/ml	6.26×10^{0} mrem
Kr-89	1.90×10^{-6} µCi/ml	2.31×10^{0} mrem
Kr-90	2.80×10^{-6} µCi/ml	7.20×10^{-1} mrem
Xe-133	1.00×10^{-4} µCi/ml	1.70×10^{-1} mrem
Xe-135	1.00×10^{-5} µCi/ml	3.31×10^{-1} mrem
Xe-135m	9.00×10^{-6} µCi/ml	1.32×10^{-1} mrem
Xe-137	2.00×10^{-5} µCi/ml	3.53×10^{-1} mrem
Xe-138	4.00×10^{-6} µCi/ml	4.03×10^{0} mrem
Xe-139	3.70×10^{-6} µCi/ml	9.10×10^{-2} mrem
Total = 16.38 mrem		

To finalize the occupational dose in terms of Total Effective Dose Equivalent (TEDE) for a 10-minute exposure in the containment building after fuel failure, the doses from the radioiodines and noble gases must be added together, and result in the values listed in Table 13-30.

TABLE 13-30
10-MINUTE DOSE FROM RADIOIODINES AND NOBLE GASES IN
THE CONTAINMENT BUILDING FOLLOWING FUEL FAILURE

Total Iodine – Committed Dose Equivalent (CDE)	**10.07 mrem**
Iodine – Committed Effective Dose Equivalent (CEDE) (CEDE = CDE x 0.03)	**0.302 mrem**
Noble Gas – Committed Effective Dose Equivalent (CEDE)	**16.38 mrem**
Total Dose – Total Effective Dose Equivalent (TEDE)	**16.69 mrem**

By comparison of the maximum TEDE and CDE for those occupationally exposed during fuel failure to applicable NRC dose limits in 10 CFR 20, the final values are shown to be well within the published regulatory limits and, in fact, lower than 1% of any occupational limit.

Radiation shine through the containment structure was also evaluated when considering accident conditions and dose consequences to the public and MURR staff. Calculation of exposure rate from a fuel failure was performed using the computer program MicroShield 8.02 with a Rectangular Volume – External Dose Point geometry for the representation of the containment structure (Reference 13.42).

MicroShield 8.02 is a product of Grove Software and is a comprehensive photon/gamma ray shielding and dose assessment program that is widely used by industry for designing radiation shields.

The exposure rate values provided in Table 13-31 represent the radiation fields at 1 foot (30.5 cm) from a 12-inch (30.5 cm) thick ordinary concrete containment wall and at the Emergency Planning Zone (EPZ) boundary of 150 meters (492.1 ft). The airborne concentration source terms used to develop the exposure rate values are similar to those used for determining the dose to a worker within containment from noble gases. For radioiodine, the total iodine activity that leaked to the pool system in the 10-minute period was used for the dose calculations. The source term also assumes a homogenous mixture of nuclides within the containment free volume.

TABLE 13-31
RADIATION SHINE THROUGH THE CONTAINMENT BUILDING FOLLOWING FUEL FAILURE

Exposure Rate at 1-Foot from Containment Building Wall:	**1.374 mrem/hr**
Exposure Rate at Emergency Planning Zone Boundary (150 meters):	**0.0094 mrem/hr**

The containment building overpressure situation described within Section 13.2.1.2 is utilized to determine containment building and exhaust stack leakage in the following equations.

Example calculation of I-131 released through the exhaust stack during the first hour:

The average cubic foot per hour leakage rate can be calculated for each time interval.

= (I-131 activity in containment x leakage rate in scf/hr x 0.50) / (30,500 ft^3/min x 60 min/hr x 28,317 ml/ft^3)

= (2.58 x 10^{-3} µCi/ft^3 x 595.6 scf/hr x 0.50) / (5.18 x 10^{10} ml)

= 1.48 x 10^{-11} µCi/ml

The same calculation is used for the radioiodines listed in Table 13-32.

TABLE 13-32
RADIOIODINE CONCENTRATIONS IN AIR LEAKING FROM CONTAINMENT AND EXITING THE EXHAUST STACK DURING THE FURST HOUR FOLLOWING FUEL FAILURE

I-131 – 1.48 x 10^{-11} µCi/ml	I-133 – 3.60 x 10^{-11} µCi/ml	I-135 – 3.24 x 10^{-11} µCi/ml
I-132 – 1.788 x 10^{-11} µCi/ml	I-134 – 2.78 x 10^{-11} µCi/ml	

Example calculation of Kr-85 released through the exhaust stack in the first hour:

= (Kr-85 activity in containment x leakage rate in scf/hr) / (30,500 ft^3/min x 60 min/hr x 28,317 ml/ft^3)

= (2.72 x 10^{-3} µCi/scf x 595.6 scf/hr) / (5.18 x 10^{10} ml)

= 3.13 x 10^{-11} µCi/ml

The same calculation is used for the noble gases listed in Table 13-33.

TABLE 13-33
NOBLE GAS CONCENTRATIONS IN AIR LEAKING FROM CONTAINMENT AND EXITING THE EXHAUST STACK IN THE FIRST HOUR FOLLOWING FUEL FAILURE

Kr-85 – 3.13 x 10^{-11} µCi/ml	Kr-87 – 1.05 x 10^{-8} µCi/ml	Kr-89 – 3.39 x 10^{-11} µCi/ml
Kr-85m – 8.19 x 10^{-9} µCi/ml	Kr-88 – 1.74 x 10^{-8} µCi/ml	Kr-90 – 4.20 x 10^{-25} µCi/ml
Xe-133 – 2.59 x 10^{-8} µCi/ml	Xe-135m – 6.29 x 10^{-10} µCi/ml	Xe-138 – 7.75 x 10^{-9} µCi/ml
Xe-135 – 4.92 x 10^{-9} µCi/ml	Xe-137 – 1.41 x 10^{-10} µCi/ml	Xe-139 – 6.22 x 10^{-22} µCi/ml

The same type of calculation is used for all the radioiodines during the five (5) 1-hour intervals and the three (3) 4-hour intervals to determine the average concentration in the exhaust stack, which is listed in Table 13-34.

TABLE 13-34
AVERAGE RADIOIODINE CONCENTRATIONS IN AIR LEAKING FRON CONTAINMENT AND EXITING THE EXHAUST STACK FOLLOWING FUEL FAILURE

I-131 – 7.56 x 10^{-12} µCi/ml	I-133 – 1.59 x 10^{-11} µCi/ml	I-135 – 1.06 x 10^{-11} µCi/ml
I-132 – 3.37 x 10^{-12} µCi/ml	I-134 – 2.91 x 10^{-12} µCi/ml	

The same calculation is used for the noble gases listed in Table 13-35.

TABLE 13-35
NOBLE GAS CONCENTRATIONS IN AIR LEAKING FROM CONTAINMENT AND EXITING THE EXHAUST STACK FOLLOWING FUEL FAILURE

Kr-85 – 1.62 x 10^{-11} µCi/ml	Kr-87 – 1.38 x 10^{-9} µCi/ml	Kr-89 – 2.06 x 10^{-12} µCi/ml
Kr-85m – 2.26 x 10^{-9}µCi/ml	Kr-88 – 3.75 x 10^{-9} µCi/ml	Kr-90 – 2.55 x 10^{-26} µCi/ml
Xe-133 – 1.31 x 10^{-8}µCi/ml	Xe-135m – 4.06 x 10^{-11} µCi/ml	Xe-138 – 4.94 x 10^{-10} µCi/ml
Xe-135 – 1.80 x 10^{-9} µCi/ml	Xe-137 – 8.53 x 10^{-12} µCi/ml	Xe-139 – 3.77 x 10^{-23} µCi/ml

Assuming, as stated earlier, that (1) the average exhaust gas concentration due to the containment leakage rate applies for the 16.5-hour period in order to equalize the containment building pressure with atmospheric pressure, (2) the flow rate through the facility's ventilation exhaust stack is 30,500 scfm (863.7 m^3/min), (3) the reduction in concentration from the point of discharge at the exhaust stack to the point of maximum concentration in the unrestricted area is a factor of 292, and (4) after exiting the ventilation exhaust stack, conservatively there is assumed no decay of any radioiodines or noble gases. The following concentrations of radioiodines and noble gases with their corresponding

radiation doses are calculated to occur in the unrestricted area. The values listed are for the point of maximum concentration in the unrestricted area assuming a uniform, semi-spherical cloud geometry for noble gas submersion and assuming that the most conservative (worst-case) meteorological conditions exist for the entire 16.5-hour period of containment leakage following fuel failure. Radiation doses are calculated for the entire 16.5-hour period. Dose values for the unrestricted area were obtained using the same methodology that was used to determine doses inside the containment building, and it was assumed that an individual was present at the point of maximum concentration for the full 16.5 hours that the containment building was leaking.

A worst-case scenario effluent dilution factor of 292 using the Pasquill-Guifford Model for atmospheric dilution is used in this analysis. It is assumed that all offsite (public) dose occurs under these atmospheric conditions at the site of interest, i.e., 760 meters north of MURR. This limiting dilution factor occurs only during Stability Class F conditions for the location of interest. Stability Class F conditions occur 11.4% of the time when the wind blows from the south. Thus the calculation is conservative.

10 CFR 20 Appendix B Effluent Concentration Limits are used for the "listed" isotopes. Effluent Concentration Limits were calculated for each of the four (4) "unlisted" noble gases (Kr-89, Kr-90, Xe-137 and Xe-139) using the data and methodology contained in FGR No. 12 for submersion isotopes. The DAC value was first calculated and then a factor of 219 was applied using 10 CFR 20, Appendix B, as a reference point for DAC values from submersion isotopes. Exposure at 1 DAC equates to 5000 mrem per year whereas the Effluent Concentration Limit is 50 mrem per year (for noble gases, the Effluent Concentration Limit is determined by a submersion dose of 100 mrem per year). This results in a factor of 100 times lower allowable dose for the Effluent Concentration Limit as compared to the DAC. Exposure at the Effluent Concentration Limit assumes you are in that effluent concentration for 8,760 hours per year. Therefore, the time assumed to be exposed to the Effluent Concentration Limit is a factor of 4.38 times longer than the 2000 hours per year that defines a DAC. No credit is taken for transit time from the exhaust stack to the receptor point nor is credit taken for decay inside the containment building until release. In the case of Kr-89 and Xe-137, the transit time alone would be approximately one (1) half-life while the transit time for Kr-90 and Xe-139 would be at least four (4) half-lives. The DAC values were then divided by 219 to arrive at the effluent concentration limits for the calculated isotope limits, consistent with the methodology used in Appendix B of 10 CFR 20.

Example calculation of whole-body dose in the unrestricted area due to I-131:

Conversion Factor: (Public dose limit of 50 mrem/yr) x (1 yr/8760 hrs) = 5.71×10^{-3} mrem/hr

I-131 concentration	= 7.56×10^{-12} µCi/ml
I-131 effluent concentration limit	= 2.00×10^{-10} µCi/ml
I-131 Conversion Factor	= 5.71×10^{-3} mrem/hr

A 16.5-hour whole body exposure from I-131 is:

= I-131 concentration / (I-131 Effluent Concentration Limit x Conversion Factor x 16.5 hrs)

= 7.56×10^{-12} µCi/ml / (2.00×10^{-10} µCi/ml x 5.71×10^{-3} mrem/hr x 16.5 hrs /292)

= 1.22×10^{-5} mrem

The same calculation is used for the isotopes (radioiodines and noble gases) listed in Table 13-36.

TABLE 13-36
EFFLUENT CONCENTRATION LIMITS, CONCENTRATIONS AT POINT OF MAXIMUM CONCENTRATION AND RADIATION DOSES IN THE UNRESTRICTED AREA – RADIOIODINE – FOLLOWING FUEL FAILURE

Radionuclide	**Effluent Limit**	**Maximum Concentration[1]**	**Radiation Dose**
I-131	2.00×10^{-10} µCi/ml	2.59×10^{-14} µCi/ml	1.22×10^{-5} mrem
I-132	2.00×10^{-8} µCi/ml	1.16×10^{-14} µCi/ml	5.44×10^{-8} mrem
I-133	1.00×10^{-9} µCi/ml	5.44×10^{-14} µCi/ml	5.13×10^{-6} mrem
I-134	6.00×10^{-8} µCi/ml	9.97×10^{-15} µCi/ml	1.57×10^{-8} mrem
I-135	6.00×10^{-9} µCi/ml	3.63×10^{-14} µCi/ml	5.70×10^{-7} mrem
Total = 1.80E-5 mrem			

Note 1: Maximum Concentrations are radioiodine and noble gas concentrations leaking from the containment building and exiting the exhaust stack reduced by a dilution factor of 292.

TABLE 13-37
EFFLUENT CONCENTRATION LIMITS, CONCENTRATIONS AND POINT OF MAXIMUM CONCENTRATION, AND RADIATION DOSES IN THE UNRESTRICTED AREA – NOBLE GASES – FOLLOWING FUEL FAILURE

Radionuclide	Effluent Limit	Maximum Concentration[1]	Radiation Dose[2]
Kr-85	7.00×10^{-7} µCi/ml	5.56×10^{-14} µCi/ml	4.38×10^{-6} mrem
Kr-85m	1.00×10^{-7} µCi/ml	7.75×10^{-12} µCi/ml	4.26×10^{-3} mrem
Kr-87	2.00×10^{-8} µCi/ml	4.71×10^{-12} µCi/ml	1.29×10^{-2} mrem
Kr-88	9.00×10^{-9} µCi/ml	1.28×10^{-11} µCi/ml	7.86×10^{-2} mrem
Kr-89	8.60×10^{-9} µCi/ml	7.04×10^{-15} µCi/ml	4.50×10^{-5} mrem
Kr-90	1.20×10^{-8} µCi/ml	8.72×10^{-29} µCi/ml	4.00×10^{-19} mrem
Xe-133	5.00×10^{-7} µCi/ml	4.48×10^{-11} µCi/ml	4.94×10^{-3} mrem
Xe-135	7.00×10^{-8} µCi/ml	6.17×10^{-12} µCi/ml	4.84×10^{-3} mrem
Xe-135m	4.00×10^{-8} µCi/ml	1.39×10^{-13} µCi/ml	1.91×10^{-4} mrem
Xe-137	9.10×10^{-8} µCi/ml	2.92×10^{-14} µCi/ml	1.77×10^{-5} mrem
Xe-138	2.00×10^{-8} µCi/ml	1.69×10^{-12} µCi/ml	4.66×10^{-3} mrem
Xe-139	1.60×10^{-8} µCi/ml	1.29×10^{-25} µCi/ml	4.44×10^{-16} mrem
Total = 1.11E-1 mrem			

Note 1: Maximum Concentrations are radioiodine and noble gas concentrations leaking from the containment building and exiting the exhaust stack reduced by a dilution factor of 292.

To finalize the unrestricted dose in terms of Total Effective Dose Equivalent (TEDE), the doses from the radioiodines and noble gases must be added, resulting in the values shown in Table 13-38.

TABLE 13-38
DOSE FROM RADIOIODINES AND NOBLE GASES IN THE UNRESTRICTED AREA FOLLOWING FUEL FAILURE

Total Iodine – Committed Effective Dose Equivalent (CEDE)	**1.80E-5 mrem**
Noble Gas – Committed Effective Dose Equivalent (CEDE)	**1.11E-1 mrem**
Total Dose – Total Effective Dose Equivalent (TEDE)	**1.11E-1 mrem**

As shown by analysis, the doses are low; less than 0.1% of the Part 20 limits for allowable dose to the general public. Additionally, leakage in mechanical equipment room 114, from such items as valve packing, flange gaskets, pump mechanical seals, etc., was also considered in the fuel failure analysis. A realistic leakage rate of 60 milliliters within the 10-minute time interval was used. The additional contaminated water vapor and associated isotopes added to the facility ventilation exhaust system made a minimal (<1%) contribution to the total dose of an individual located in the facility. Therefore, the dose contribution to the unrestricted area would be expected to be approaching zero.

The fuel failure during reactor operation accident does not postulate a mechanism resulting in fuel failure only the resultant melting of fuel plates (See Section 13.2.5.1). This is the only accident which hypothesizes the release of fission products from the primary coolant. The analyzed consequences of

fuel failure during reactor operation accident show no occupational radiation doses or doses in the unrestricted area that exceed any regulatory limits and are less than the consequences of the MHA.

Reference 13.50 documents the NRC's review of this analysis and their own confirmatory analyses to ensure the doses from fuel failure and subsequent containment leakage remain within applicable regulatory limits.

13.2.5.2 Fuel Handling Accident (FHA)

13.2.5.2.1 Accident-Initiating Events and Scenarios

All fuel handling is performed in accordance with Special Nuclear Material (SNM) Control and Accounting Procedures and as outlined in the Operations Procedures. Irradiated fuel is handled with a specially designed remote tool. The fuel handling tool is designed to provide a positive indication of latching prior to movement of a fuel element. This feature is tested prior to any fuel handling sequence. Fuel elements are always handled one at a time so that they are maintained in a criticality-safe configuration. New or irradiated fuel may be stored in any one of the 88 in-pool fuel storage locations (not including the core). These storage locations are designed to (1) ensure a geometry such that the calculated K-effective is less than 0.9 under all conditions of moderation, (2) allow sufficient convection cooling, and (3) provide sufficient radiation shielding. 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.

All cask lifting equipment, including the 15-ton (13,608-kg) capacity overhead rectilinear crane, is rigorously maintained, including preventive maintenance and testing, as appropriate. Hence, no specific accidents regarding the handling of fuel have been identified for MURR. However, the probability of dropping a fuel element while underwater and damaging it severely enough to breach the fuel cladding was considered. A conservative potential radionuclide release and calculation of the occupational and public exposures from a Fuel Handling Accident are included below.

13.2.5.2.2 Accident Analysis and Consequences

The following calculations determining the postulated dose from a potential release of radioactivity from a fuel element during a handling accident closely follow the "Fuel Failure during Reactor Operation" calculations and methodology for personal exposure due to a release of fission products. The objective of these calculations is to present a worst-case dose assessment for a person who remains in the containment building for five (5) minutes following the release from a breached fuel element. MURR's fuel cycle averages having approximately 40 fuel elements in the cycle – divided into 20 pairs of elements. Paired elements are always loaded opposite each other in the core. All eight (8) fuel elements are replaced every refueling. MURR has averaged refueling the core more than 52 times a year since 1977. This type of accident has never occurred at MURR during any of these fuel handlings.

The two outer fuel plates of a fuel element, number-1 and -24, are the plates most likely to be damaged during fuel handling. The number-1 fuel plate contains 19.26 grams of U-135 before irradiation. The highest peak power density in the various MURR core configurations occurs in fuel plate number-1 of a fresh fuel element, which has a power peaking factor of 4.116 – located between 13.75 to 14.75 inches (34.93 to 37.47 cm) down from the top of the fuel plate. Fuel plate number-24 has the most surface area to be damaged; however, it has a lower peak power density and contains 45.32 grams of U-235. To be conservative, the analysis assumes that 0.125 grams of U-235 is exposed from plate number-1 during the FHA, which corresponds to removing a section of fuel meat from a plate that is 1-inch (2.54-cm) square and 5 mils (0.127 cm) thick. A power peaking factor of 4.116 is also applied.

Due to the complex nature of the MURR mixed core fuel cycle, the source terms for the "Fuel Failure during Reactor Operation" and "Fuel Handling Accident" were determined using the computer program MONTEBURNS instead of simplistic ORIGEN runs (Reference 13.42). MONTEBURNS is a coupled MCNP-ORIGEN code system developed by Los Alamos National Laboratory (LANL). Within the MONTEBURNS program, MCNP calculations are used to obtain accurate one-group fluxes and one-group cross sections that are then utilized by ORIGEN for fuel depletion and fission product activity calculations. The use of MONTEBURNS and the flow diagram utilized for various neutronic computations at MURR were described in detail in the response to Request for Additional Information (RAI) Question 3.a (MURR letter dated October 1, 2015).

The following radioiodine, krypton and xenon activities will be present in the MURR core 30 minutes after shutdown from 10 MW full power operation. Refueling typically occurs no sooner than one (1) hour after shutdown. This takes into account the time required to shut down the reactor, to secure the primary coolant system (required to stay in operation a minimum of 15 minutes after the control blades are fully inserted), and to remove the reactor pressure vessel head. For the purposes of the FHA calculations, a conservative assumption of 30 minutes is used.

TABLE 13-39
RADIOIODINE AND NOBLE GAS ACTIVITIES IN THE CORE AFTER 30-MINUTE DECAY FROM FHA

I-131 – 2.20×10^5 Ci	Kr-85 – 4.63×10^2 Ci	Xe-133 – 3.85×10^5 Ci
I-132 – 3.07×10^5 Ci	Kr-85m – 1.23×10^5 Ci	Xe-135 – 9.11×10^4 Ci
I-133 – 5.39×10^5 Ci	Kr-87 – 1.58×10^5 Ci	Xe-135m – 3.62×10^4 Ci
I-134 – 5.49×10^5 Ci	Kr-88 – 2.58×10^5 Ci	Xe-137 – 2.24×10^3 Ci
I-135 – 4.80×10^5 Ci	Kr-89 – 5.28×10^2 Ci	Xe-138 – 1.16×10^5 Ci
	Kr-90 – 6.31×10^{-12} Ci	Xe-139 – 7.89×10^{-9} Ci

Fission products released into the reactor pool will be detected by the pool surface and ventilation system exhaust plenum radiation monitors. For the purposes of this analysis, it is assumed that an actuation of the containment building isolation system occurs by action of the pool surface radiation monitor. Actuation of the isolation system will prompt Operations personnel to ensure that a total evacuation of the containment building is accomplished promptly, usually within two (2) to two and a

half (2.5) minutes. A conservative 5-minute evacuation time is used as the basis for the stay time in the dose calculations for personnel that are in containment during the FHA.

The following radioiodine and noble gas activities from the peak power position of fuel plate number-1 in the peak power density fuel element are assumed to instantaneously and homogenously distribute in the reactor pool.

Example calculation of I-131 released into the reactor pool:

= (I-131 in fuel / U-235 in core) x U-235 exposed x Power Peaking Factor x (1×10^6 µCi/Ci)

= (2.20×10^5 Ci / 5,474 grams) x 0.125 grams x 4.116 x (1×10^6 µCi/Ci)

= 2.07×10^7 µCi

Example calculation of Kr-85 released into the reactor pool:

= (Kr-85 in fuel / U-235 in core) x U-235 exposed x Power Peaking Factor x (1×10^6 µCi/Ci)

= (4.63×10^2 Ci / 5,474 grams) x 0.125 grams x 4.116 x (1×10^6 µCi/Ci)

= 4.35×10^4 µCi

The same calculations are used for the isotopes listed in Table 13-40.

TABLE 13-40
RADIOIODINE AND NOBLE GAS ACTIVITIES RELEASED INTO THE POOL FROM FHA

I-131 – 2.07×10^7 µCi	Kr-85 – 4.35×10^4 µCi	Xe-133 – 3.62×10^7 µCi
I-132 – 2.89×10^7 µCi	Kr-85m – 1.16×10^7 µCi	Xe-135 – 8.56×10^6 µCi
I-133 – 5.07×10^7 µCi	Kr-87 – 1.49×10^7 µCi	Xe-135m – 3.40×10^6 µCi
I-134 – 5.16×10^7 µCi	Kr-88 – 2.42×10^7 µCi	Xe-137 – 2.11×10^5 µCi
I-135 – 4.51×10^7 µCi	Kr-89 – 4.96×10^4 µCi	Xe-138 – 1.09×10^7 µCi
	Kr-90 – 5.93×10^{-10} µCi	Xe-139 – 7.42×10^{-7} µCi

The radioiodine released into the reactor pool over a 5-minute interval is conservatively assumed to be instantly and uniformly mixed into the 20,000 gallons (75,708 l) of bulk pool water, which then results in the pool water concentrations for the radioiodine isotopes shown in Table 13-41. The water solubility of the krypton and xenon noble gases released into the pool over this same time period are ignored and they are assumed to pass immediately through the pool water and evolve directly into the containment building air volume where they instantaneously form a uniform concentration in the isolated structure.

TABLE 13-41
RADIOIODINE CONCENTRATIONS IN THE POOL WATER FROM FHA

I-131 – 1.03 x 10^3 µCi/gal	I-133 – 2.53 x 10^3 µCi/gal	I-135 – 2.26 x 10^3 µCi/gal
I-132 – 1.44 x 10^3 µCi/gal	I-134 – 2.58 x 10^3 µCi/gal	

When the reactor is at 10 MW and the containment building ventilation system is in operation, the evaporation rate from the reactor pool surface is approximately 80 gallons (302.8 L) of water per day. For the purposes of this calculation, it is assumed that a total of 20 gallons (75.7 L) of pool water containing the previously listed radioiodine concentrations evaporates into the containment building over the 5-minute period. Containment air with a temperature of 75 °F (23.9 °C) and 100% relative humidity contains H_2O vapor equal to 40 gallons (151.4 L) of water. Since the air in containment is normally at about 50% relative humidity, containing approximately 20 gallons (75.7 L) of water vapor, the assumed addition of 20 gallons (75.7 L) of water vapor over five (5) minutes will not cause the containment air to be supersaturated. It is also conservatively assumed that all of the radioiodine activity in the 20 gallons (75.7 L) of pool water that evaporates instantaneously forms a uniform concentration in the containment building air. When distributed into the containment building, this would result in the following radioiodine concentrations in the 225,000 ft^3 (6,371.3 m^3) air volume:

Example calculation of I-131 released into containment air to determine the average concentration during the first minute:

= (I-131 concentration in pool x 4 gal/min x 0.5 min) x e^[(-0.693 x 0.5 min) / (8.02 day x 1440 min/day)] / (225,000 ft^3 x 28,317 ml/ft^3)

= (1.03 x 10^3 µCi/gal x 2 gal) x 0.9997 / (6.371 x 10^9 ml)

= 3.25 x 10^{-7} µCi/ml

The same calculation is used for the isotopes listed in Table 13-42 and was performed for each one (1) minute interval. The average radioiodine concentration over the 5-minute interval is the average of the five (5) 1-minute intervals.

TABLE 13-42
AVERAGE RADIOIODINE CONCENTRATIONS IN THE CONTAINMENT BUILDING AIR DURING THE 5-MINUTE PERIOD FROM FHA

I-131 – 1.62 x 10^{-6} µCi/ml	I-133 – 3.97 x 10^{-6} µCi/ml	I-135 – 3.52 x 10^{-6} µCi/ml
I-132 – 2.23 x 10^{-6} µCi/ml	I-134 – 3.88 x 10^{-6} µCi/ml	

As noted previously, the krypton and xenon noble gases released into the reactor pool during the 5-minute interval following the FHA are assumed to pass immediately through the pool water and enter the containment building air volume where they instantaneously form a uniform concentration in the isolated structure. This assumption is conservative since it ignores the known solubility of krypton and

xenon noble gases in the 100 °F (37.8 °C) pool water, which would reduce their release into the containment building. Based on the 225,000-ft^3 (6,371.3 m^3) volume of containment building air, and the previously listed Curie quantities of these gases released into the reactor pool, the maximum noble gas concentrations in the containment structure at the end of five (5) minutes would be as follows:

Example calculation of Kr-85 average concentration in containment air during the first minute:

= Kr-85 activity x e^[(-0.693 x 0.5 min) / Kr-85 $T_{1/2}$] / (225,000 ft^3 x 28,317 ml/ft^3)

= (4.35 x 10^4 µCi x e^[(-0.693 x 0.5 min) / (10.76 yr x 365.25 day/yr x 1440 min/day)] / 6.371 x 10^9 ml)

= 6.83 x 10^{-6} µCi/ml

Likewise, the average noble gas concentration of the five (5) 1-minute interval concentrations is the average concentration of the five (5) minute interval.

TABLE 13-43
AVERAGE NOBLE GAS CONCENTRATIONS IN THE CONTAINMENT BUILDING AIR DURING THE 5-MINUTE PERIOD FROM FHA

Kr-85 – 6.83 x 10^{-6} µCi/ml	Xe-133 – 5.68 x 10^{-3} µCi/ml
Kr-85m – 1.80 x 10^{-3} µCi/ml	Xe-135 – 1.34 x 10^{-3} µCi/ml
Kr-87 – 2.28 x 10^{-3} µCi/ml	Xe-135m – 4.78 x 10^{-4} µCi/ml
Kr-88 – 3.77 x 10^{-3} µCi/ml	Xe-137 – 2.17 x 10^{-5} µCi/ml
Kr-89 – 4.71 x 10^{-6} µCi/ml	Xe-138 – 1.52 x 10^{-3} µCi/ml
Kr-90 – 1.35 x 10^{-20} µCi/ml	Xe-139 – 2.11 x 10^{-17} µCi/ml

Based on the source term data provided, it is possible to determine the radiation dose to the thyroid, Committed Dose Equivalent (CDE), from radioiodine and the dose to the whole body, Committed Effective Dose Equivalent (CEDE), resulting from submersion in the airborne noble gases and radioiodine inside the containment building.

Because the airborne radioiodine source is composed of five (5) different iodine isotopes, it will be necessary to determine the dose contribution from each individual isotope and then sum the results. Dose Multiplication Factors were established using the Derived Air Concentrations (DACs) for the "listed" isotopes in Appendix B of 10 CFR 20 and calculated values for the four (4) "unlisted" submersion isotopes (Kr-89, Kr- 90, Xe-137 and Xe-139). The submersion DAC values that were calculated were done in accordance with the data and methodology as supplied in Federal Guidance Report (FGR) No.12. (Reference 13.49, Attachment 4)

Example calculation of thyroid dose due to I-131:

The DAC can also be defined as 50,000 mrem [thyroid target organ limit-(CDE)]/2,000 hrs, or 25 mrem/DAC-hr. Additionally, five (5) minutes of one DAC-hr is 8.33×10^{-2} DAC-hr.

I-131 concentration in containment = 1.62×10^{-6} µCi/ml

I-131 DAC (10 CFR 20) = 2.00×10^{-8} µCi/ml

Dose Multiplication Factor = (I-131 concentration) / (I-131 DAC)

= (1.62×10^{-6} µCi/ml) / (2.00×10^{-8} µCi/ml)

= 81.0

A 5-minute thyroid exposure from I-131 is:

= Dose Multiplication Factor x DAC Dose Rate x 5 minutes

= 81.0 x (25 mrem/DAC-hr) x (8.33×10^{-2} DAC-hr)

= 169 mrem

The same calculation is used for the radioiodines listed in Table 13-44.

TABLE 13-44
DERIVED AIR CONCENTRATION VALUES AND 5-MINUTE EXPOSURES – RADIOIODINES – FROM FHA

Radionuclide	**Derived Air Concentration**	**5-Minute Exposure**
I-131	2.00×10^{-8} µCi/ml	1.69×10^{2} mrem
I-132	3.00×10^{-6} µCi/ml	1.55×10^{0} mrem
I-133	1.00×10^{-7} µCi/ml	8.26×10^{1} mrem
I-134	2.00×10^{-5} µCi/ml	4.04×10^{-1} mrem
I-135	7.00×10^{-7} µCi/ml	1.05×10^{1} mrem
Total = 264.00 mrem		

Doses from the kryptons and xenons present in the containment building are assessed in much the same manner as the radioiodines, and the dose contribution from each individual radionuclide must be calculated and then added together to arrive at the final noble gas dose. Because the dose from the noble gases is only an external dose due to submersion, and because the DACs for these radionuclides are based on this type of exposure, the individual noble gas doses for five (5) minutes in containment were based on their average concentration in the containment air and the corresponding DAC.

Example calculation of whole-body dose (CEDE) due to Kr-85:

The DAC can also be defined as 5,000 mrem/2,000 hrs, or 2.5 mrem/DAC-hr. Additionally, five (5) minutes of one DAC-hr is 8.33×10^{-2} DAC-hr.

Kr-85 concentration in containment = 6.83×10^{-6} µCi/ml

Kr-85 DAC (10 CFR 20) = 1.00×10^{-4} µCi/ml

Dose Multiplication Factor = (Kr-85 concentration) / (Kr-85 DAC)

= (6.83×10^{-6} µCi/ml) / (1.00×10^{-4} µCi/ml)

= 0.0683

A 5-minute whole-body exposure from Kr-85 is:

= Dose Multiplication Factor x DAC Dose Rate x 5 min

= 0.0683 x (2.5 mrem/DAC-hr) x (8.33×10^{-2} DAC-hr)

= 1.42×10^{-2} mrem

The same calculation is used for the noble gases listed in Table 13-45.

TABLE 13-45
DERIVED AIR CONCENTRATION VALUES AND 5-MINUTE EXPOSURES – NOBLE GASES – FROM FHA

Radionuclide	**Derived Air Concentration**	**5-Minute Exposure**
Kr-85	1.00×10^{-4} µCi/ml	1.42×10^{-2} mrem
Kr-85m	2.00×10^{-5} µCi/ml	1.88×10^{1} mrem
Kr-87	5.00×10^{-6} µCi/ml	9.49×10^{1} mrem
Kr-88	2.00×10^{-6} µCi/ml	3.92×10^{2} mrem
Kr-89	1.90×10^{-6} µCi/ml	5.17×10^{-1} mrem
Kr-90	2.80×10^{-6} µCi/ml	1.00×10^{-14} mrem
Xe-133	1.00×10^{-4} µCi/ml	1.18×10^{1} mrem
Xe-135	1.00×10^{-5} µCi/ml	2.79×10^{1} mrem
Xe-135m	9.00×10^{-6} µCi/ml	1.11×10^{1} mrem
Xe-137	2.00×10^{-5} µCi/ml	2.26×10^{-1} mrem
Xe-138	4.00×10^{-6} µCi/ml	7.90×10^{1} mrem
Xe-139	3.70×10^{-6} µCi/ml	1.19×10^{-12} mrem
Total = 636.49 mrem		

To finalize the occupational dose in terms of Total Effective Dose Equivalent (TEDE) for a 5-minute exposure in the containment building after an FHA, the doses from the radioiodines and noble gases must be added together and result in the values in Table 13-46.

TABLE 13-46
5-MINUTE DOSE FROM RADIOIODINES AND NOBLE GASES IN THE CONTAINMENT BUILDING FROM FHA

Total Iodine – Committed Dose Equivalent (CDE)	**264.00 mrem**
Iodine – Committed Effective Dose Equivalent (CEDE) (CEDE = CDE x 0.03)	**7.92 mrem**
Noble Gas – Committed Effective Dose Equivalent (CEDE)	**636.49 mrem**
Total Dose – Total Effective Dose Equivalent (TEDE)	**644.41 mrem**

By comparison of the maximum TEDE and CDE for those occupationally exposed during an FHA to applicable dose limits in 10 CFR 20, the final values are shown to be well within the published regulatory limits and lower than 15% of any occupational limit.

Radiation shine through the containment structure was also evaluated when considering accident conditions and dose consequences to the public and MURR staff. Calculation of exposure rate from an FHA was performed using the computer program MicroShield 8.02 with a Rectangular Volume – External Dose Point geometry for the representation of the containment structure (Reference 13.42). MicroShield 8.02 is a product of Grove Software and is a comprehensive photon/gamma ray shielding and dose assessment program that is widely used by industry for designing radiation shields.

The exposure rate values provided in Table 13-47 represent the radiation fields at 1 foot (30.5 cm) from a 12-inch (30.5 cm) thick ordinary concrete containment wall and at the Emergency Planning Zone (EPZ) boundary of 150 meters (492.1 ft). The airborne concentration source terms used to develop the exposure rate values are identical to those used for determining the dose to a worker within containment from noble gases. For radioiodine, the total iodine activity from the FHA was used for the dose calculations, not the amount that evaporated in five (5) minutes. The source term also assumes a homogenous mixture of nuclides within the containment free volume.

TABLE 13-47
RADIATION SHINE THROUGH THE CONTAINMENT BUILDING FROM FHA

Exposure Rate at 1-Foot (0.3 m) from Containment Building Wall:	**59.77 mrem/hr**
Exposure Rate at Emergency Planning Zone Boundary (150 meters):	**0.41 mrem/hr**

The containment building overpressure situation described within Section 13.2.1.2 is utilized to determine containment building and exhaust stack leakage in the following equations.

Example calculation of I-131 released through the exhaust stack during the first hour:

= (I-131 concentration in pool x 20 gal) x e^[(-0.693 x 0.5 hr) / I-131 $T_{1/2}$] / (scf in containment building)

= (1.03 x 10^3 µCi/gal x 20 gal) x e^[(-0.693 x 0.5 hr) / (8.02 days x 24 hr/day)] / (229,800 scf)

= (1.03 x 10^3 µCi/gal x 20 gal) x 0.9982 / (229,800 scf)

= 8.98 x 10^{-2} µCi/scf

The same calculation is used for the radioiodines listed in Table 13-48.

TABLE 13-48
RADIOIODINE CONCENTRATIONS IN CONTAINMENT AIR DURING THE FIRST HOUR FROM FHA

I-131 – 8.98 x 10^{-2} µCi/scf	I-133 – 2.17 x 10^{-1} µCi/scf	I-135 – 1.86 x 10^{-1} µCi/scf
I-132 – 1.08 x 10^{-1} µCi/scf	I-134 – 1.51 x 10^{-1} µCi/scf	

Example calculation of the Kr-85 concentration released into containment:

= Kr-85 activity x e^[(-0.693 x 0.5 hr) / (10.76 yr x 365.25 days/yr x 24 hr/day)] / 229,800 scf

= 4.35 x 10^4 µCi x 0.999996 / 229,800 scf

= 1.89 x 10^{-1} µCi/scf

The same calculation is used for the noble gases listed in Table 13-49.

TABLE 13-49
AVERAGE NOBLE GAS CONCENTRATIONS IN THE CONTAINMENT AIR DURING THE FIRST HOUR FROM FHA

Kr-85 – 1.89 x 10^{-1} µCi/scf	Xe-133 – 1.57 x 10^2 µCi/scf
Kr-85m – 4.66 x 10^1 µCi/scf	Xe-135 – 3.59 x 10^1 µCi/scf
Kr-87 – 4.92 x 10^1 µCi/scf	Xe-135m – 3.80 x 10^0 µCi/scf
Kr-88 – 9.34 x 10^1 µCi/scf	Xe-137 – 3.97 x 10^{-3} µCi/scf
Kr-89 – 2.94 x 10^{-4} µCi/scf	Xe-138 – 1.09 x 10^1 µCi/scf
Kr-90 – 4.36 x 10^{-32} µCi/scf	Xe-139 – 7.29 x 10^{-26} µCi/scf

The average containment building leakage rate was calculated for each of the first five (5) hours and for the following three (3) 4-hour intervals:

Hours:	0 - 1	1 - 2	2 - 3	3 - 4	4 - 5	5 - 9	9 - 13	13 - 16.5
scf/hr:	595.6	558.7	521.8	485.0	448.1	355.8	207.9	67.9

The average concentration of the radioactive iodine's and noble gases in the facility's exhaust stack was calculated based on the average isotope concentration in containment during each of the time periods and the average leakage rate for the time interval. The iodine activities are reduced to 50% due to the previous stated 50% deposition in the containment structure in their leakage path. This leakage of activity out of containment is drawn into the facility exhaust ventilation system. The facility exhaust ventilation system has a flow rate of 30,500 scfm and results in the following average exhaust activities:

Example calculation of the average concentration of I-131 released through the exhaust stack during the first hour:

= (I-131 activity concentration in containment µCi/scf x scf leakage rate x 0.5) / (30,500 scfm x 60 min/hr x 1 hr x 28,317 ml/ft^3)

= (8.98 x 10^{-2} µCi/scf x 595.6 scf x 0.5) / 5.18 x 10^{10} ml

= 5.16 x 10^{-10} µCi/ml

The same calculation is used for the radioiodines listed in Table 13-50.

TABLE 13-50
AVERAGE RADIOIODINE CONCENTRATIONS IN AIR LEAKING FROM CONTAINMENT AND EXITING THE EXHAUST STACK DURING THE FIRST HOUR FROM FHA

I-131 – 5.16 x 10^{-10} µCi/ml	I-133 – 1.25 x 10^{-9} µCi/ml	I-135 – 1.07 x 10^{-9} µCi/ml
I-132 – 6.20 x 10^{-10} µCi/ml	I-134 – 8.69 x 10^{-10} µCi/ml	

Example calculation of the average concentration of Kr-85 released through the exhaust stack during the first hour:

= (Kr-85 activity concentration in containment µCi/scf x scf leakage rate) / (30,500 scfm x 60 min/hr x 1 hr x 28,317 ml/ft^3)

= (1.89 x 10^{-1} µCi/scf x 595.6 scf/hr) / (30,500 scfm x 60 min/hr x 1 hr x 28,317 ml/ft^3)

= 2.18 x 10^{-9} µCi/ml

TABLE 13-51
AVERAGE NOBLE GAS CONCENTRATIONS IN AIR LEAKING FROM CONTAINMENT AND EXITING THE EXHAUST STACK DURING THE FIRST HOUR FROM FHA

Kr-85 – 2.18 x 10^{-9} µCi/ml	Kr-87 – 5.65 x 10^{-7} µCi/ml	Kr-89 – 3.38 x 10^{-12} µCi/ml
Kr-85m – 5.35 x 10^{-7} µCi/ml	Kr-88 – 1.07 x 10^{-6} µCi/ml	Kr-90 – 5.01 x 10^{-40} µCi/ml
Xe-133 – 1.80 x 10^{-6} µCi/ml	Xe-135m – 4.37 x 10^{-8} µCi/ml	Xe-138 – 1.25 x 10^{-7} µCi/ml
Xe-135 – 4.12 x 10^{-7} µCi/ml	Xe-137 – 4.56 x 10^{-11} µCi/ml	Xe-139 – 8.38 x 10^{-34} µCi/ml

The containment building leakage rate, duration, and unrestricted area dilution factors and stay times described within Section 13.2.1.2 are utilized to determine dose impacts in the following equations.

Example calculation of whole-body dose in the unrestricted area due to I-131:

Conversion Factor: (Public dose limit of 50 mrem/yr) x (1 yr/8760 hrs) = 5.71 x 10^{-3} mrem/hr

I-131 concentration = 2.65 x 10^{-10} µCi/ml

I-131 effluent concentration limit = 2.00 x 10^{-10} µCi/ml

I-131 Conversion Factor = 1.325 mrem/hr

A 16.5-hour whole body exposure from I-131 is:

= [(I-131 concentration / I-131 Effluent Concentration Limit) x Conversion Factor x 16.5 hrs] / Dilution Factor

= [[(2.63 x 10^{-10} µCi/ml) / (2.00 x 10^{-10} µCi/ml)] x 1.325 mrem/hr x 16.5 hrs] / 292

= 4.24 x 10^{-4} mrem

The same calculation is used for the isotopes (radioiodines and noble gases) listed in Tables 13-53 and 13-54.

TABLE 13-52
EFFLUENT CONCENTRATION LIMITS, CONCENTRATIONS AT POINT OF MAXIMUM CONCENTRATION, AND RADIATION DOSES IN THE UNRESTRICTED AREA – RADIOIODINE – FROM FHA

Radionuclide	Effluent Limit	Maximum Concentration[1]	Radiation Dose
I-131	2.00 x 10^{-10} µCi/ml	9.01 x 10^{-13} µCi/ml	4.24 x 10^{-4} mrem
I-132	2.00 x 10^{-8} µCi/ml	4.01 x 10^{-13} µCi/ml	1.89 x 10^{-6} mrem
I-133	1.00 x 10^{-9} µCi/ml	1.88 x 10^{-12} µCi/ml	1.77 x 10^{-4} mrem
I-134	6.00 x 10^{-8} µCi/ml	3.12 x 10^{-13} µCi/ml	4.89 x 10^{-7} mrem
I-135	6.00 x 10^{-9} µCi/ml	1.20 x 10^{-13} µCi/ml	1.88 x 10^{-5} mrem
Total = 6.23E-4 mrem			

Note 1: Maximum Concentrations are radioiodine and noble gas concentrations leaking from the containment building and exiting the exhaust stack reduced by a dilution factor of 292.

TABLE 13-53
EFFLUENT CONCENTRATION LIMITS, CONCENTRATIONS AT POINT OF MAXIMUM CONCENTRATION, AND RADIATION DOSES IN THE UNRESTRICTED AREA – NOBLE GASES – FROM FHA

Radionuclide	Effluent Limit	Maximum Concentration[1]	Radiation Dose[2]
Kr-85	7.00×10^{-7} µCi/ml	3.87×10^{-12} µCi/ml	1.04×10^{-6} mrem
Kr-85m	1.00×10^{-7} µCi/ml	5.06×10^{-10} µCi/ml	9.54×10^{-4} mrem
Kr-87	2.00×10^{-8} µCi/ml	2.53×10^{-10} µCi/ml	2.38×10^{-3} mrem
Kr-88	9.00×10^{-9} µCi/ml	7.93×10^{-10} µCi/ml	1.66×10^{-2} mrem
Kr-89	8.60×10^{-9} µCi/ml	7.01×10^{-16} µCi/ml	1.53×10^{-8} mrem
Kr-90	1.20×10^{-8} µCi/ml	1.04×10^{-43} µCi/ml	1.63×10^{-36} mrem
Xe-133	5.00×10^{-7} µCi/ml	3.12×10^{-9} µCi/ml	1.17×10^{-3} mrem
Xe-135	7.00×10^{-8} µCi/ml	5.17×10^{-10} µCi/ml	1.39×10^{-3} mrem
Xe-135m	4.00×10^{-8} µCi/ml	9.68×10^{-12} µCi/ml	4.56×10^{-5} mrem
Xe-137	9.10×10^{-8} µCi/ml	9.46×10^{-15} µCi/ml	1.96×10^{-8} mrem
Xe-138	2.00×10^{-8} µCi/ml	2.72×10^{-11} µCi/ml	2.56×10^{-4} mrem
Xe-139	1.60×10^{-8} µCi/ml	1.74×10^{-37} µCi/ml	2.04×10^{-30} mrem
Total = 2.28E-2 mrem			

Note 1: Maximum Concentrations are radioiodine and noble gas concentrations leaking from the containment building and exiting the exhaust stack reduced by a dilution factor of 292.

To finalize the unrestricted dose in terms of Total Effective Dose Equivalent (TEDE), the doses from the radioiodines and noble gases must be added together, and result in the values displayed in Table 13-54:

TABLE 13-54
DOSE FROM RADIOIODINES AND NOBLE GASES IN THE UNRESTRICTED AREA FROM FHA

Total Iodine – Committed Effective Dose Equivalent (CEDE)	**6.23E-4 mrem**
Noble Gas – Committed Effective Dose Equivalent (CEDE)	**2.28E-2 mrem**
Total Dose – Total Effective Dose Equivalent (TEDE)	**2.34E-2 mrem**

Summing the doses from the noble gases and the radioiodines substantiates the low levels in the unrestricted area should an FHA occur and should the containment building leak following such an event. Because the dose values are so low, well below the applicable 10 CFR 20 regulatory limit for the unrestricted area, MURR does not anticipate any issues regarding offsite dose during a FHA scenario.

13.2.5.3 Fuel Element Malfunction Due to Cladding Deformation or Corrosion

The MURR core was re-analyzed in the mid-1980s to also operate with fuel assemblies which contain a maximum U-235 loading of 1,270 grams per assembly (Extended Life Aluminide Fuel or ELAF). The ELAF Program, conducted by EG&G Idaho for the DOE, 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 established by reactor licensing restrictions. Studies of deformation mechanisms for aluminide fuel elements have looked at fuel plate swelling and blistering. These potential sources of cladding damage have been addressed, and it was concluded, based on test data (Reference 13.26), that these failure mechanisms are not expected to occur even with increased burnup limits. Failure of aluminide fuel elements by these mechanisms has not been a problem in the over 4,500 aluminide fuel elements used at MURR and ATR since 1971.

Corrosion can result in aluminum cladding failure in two ways: pitting and oxide film formation. Oxide film forms as a result of essentially uniform corrosion, while pitting occurs when the corrosion rate is accelerated in a local area. MURR has experienced no fuel element failures due to pitting (Reference 13.27), but retired one fuel element early after it had been used for 126 MWD of the planned 150-MWD-usage because of a suspected manufacturing defect which caused a slight increase in I-131 level in the primary coolant (Reference 13.40). Pitting corrosion is not catastrophic in nature and can be detected by conventional monitoring techniques in place at MURR as demonstrated in this example.

To place an upper limit on a pitting event, a worst-case pit release scenario was analyzed. The Fuel Failure during Reactor Operation Accident addresses the release of fission products by assuming the melting of four number-1 fuel plates, which results in the melting of 1.41% of the fuel and a theoretical release of 3.42% of the fission products if a 100% fuel element fission product release fraction is assumed (which is conservative). This 1.41% of U-235 melting releases 3.42% of the core fission products due to the highest power density fuel plate number-1 overall power peaking factor of 2.423, which is conservatively assumed to apply to all four (4) number-1 fuel plates (1.41% x 2.423 = 3.42%).

Based on the statement in Reference 13.26 that the pit diameter is six times the depth, the surface area of a pit where it penetrates the 0.015-inch (0.038-cm) thick fuel plate cladding would be 0.090 inches (0.229 cm) in diameter. To be conservative, the analysis assumes a pit diameter of 0.20 inches (0.51 cm), and an instantaneous release of fission products (100% release fraction) from the fuel meat volume equal to the pit area times the fuel meat thickness (0.020 inches (0.051 cm)). This volume contains 0.015 grams of U-235 compared to the 77.04 grams contained in the number-1 plates of four fuel elements. Based upon these assumptions, the release would be equivalent to 1.95×10^{-4} times the release assumed in the FFA.

I-131 activity concentration in the primary coolant:

$= [(2.2 \times 10^{11}\ \mu Ci) \times (1.95 \times 10^{-4}) \times 0.0342] / [(2{,}000\ gal) \times (3{,}785\ ml/gal)]$

$= 1.94 \times 10^{-1}\ \mu Ci/ml$

where:

$2.2 \times 10^{11}\ \mu Ci$ = I-131 activity in the core from fuel failure (Table 13-23);

1.95×10^{-4} = 0.015 gm/77.04 gm, the ratio of the amount of fission product activity assumed released by "the pit" compared to the FFA release; and

0.0342 = the fraction of the total fuel melted (including peaking factor) and the total fission product activity released in the Fuel Failure During Reactor Operation Accident assuming a 100% fission product release fraction.

Aluminum film corrosion studies (Reference 13.28, Reference 13.43, and Reference 13.44) report that the oxide film rate of formation is dependent on coolant pH, operating time, surface temperature and the magnitude of the heat flux. MURR has also been able to measure and predict oxide thickness (Reference 13.27). In this respect, the maximum oxide thickness on a new 1,270-gram fuel element with a 300-MWD power history at 10 MW is predicted to be 0.000854 inches (0.217 mm). The 300-MWD power history corresponds to a peak burnup of slightly greater than 2.3×10^{21} fissions per cm^3. The three standard deviations value for oxide thickness at 300 MWD is 1.75 mils (0.045 mm). The probability of having an oxide thickness greater than 1.75 mils (0.045 mm) is less than 0.14% (Reference 13.11). MURR's measured and predicted values have not been judged to indicate the likelihood of cladding failure or significant fission product release. Thus, the FFA is still the most significant accident related to the potential risks from fuel corrosion or cladding deformation.

13.2.5.4 Conclusions

Generally, the most severe condition which is analyzed with regard to reactor accidents is either a loss of primary coolant or a loss of primary coolant flow during reactor operation. Both of these accidents are analyzed in this chapter and the results show no core damage. In addition, there are no other accidents that will result in a release of fission products from the reactor fuel, which is assumed in the fuel failure analysis. Even if such an event were to occur, the anti-siphon and reactor loop vent systems are designed such that any released radioactivity would be contained in the primary coolant system.

System design and operational procedures reduce the likelihood of any foreign material being introduced into the reactor core that could cause a partial flow blockage. Calculations have been performed which indicate that even partial flow blockage to a fuel element will not result in cladding failure (Reference 13.2). A considerable margin of safety has been designed into the system in this regard. Also, considering the results of the analyses which show no core damage in the event of a loss of primary coolant or a loss of primary coolant flow accident (see Sections 13.2.3 and 13.2.4), and in view of the design of the anti-siphon and reactor loop vent systems, it is concluded that there is no radiation risk to personnel in the reactor containment building or in the unrestricted area should one of these events occur.

The NRC documented their review of this analysis and performed their own confirmatory analyses to ensure the MHA remained within applicable regulatory limits, in the Safety Evaluation Report documented under Reference 13.50.

13.2.6 Experiment Malfunction

13.2.6.1 Accident-Initiating Events and Scenarios

Experiments conducted at MURR are subdivided into two general classifications: neutron beam, and neutron irradiation and/or isotope production. The neutron beam experiments are those research projects which utilize one of the beamports. The neutron irradiation and isotope production experimental facilities include the center test hole, the graphite reflector region, the pneumatic tube system, and in-pool locations external to the graphite reflector (bulk pool). The experimental facilities and their utilization program are described in greater detail in Chapter 10, "Experimental Facilities and Utilization."

Improperly controlled experiments could potentially result in damage to the reactor, unnecessary radiation exposure to the facility staff and members of the general public, and an inadvertent release of radioactivity into the unrestricted environment. Mechanisms for these events include 1) the production of excessive amounts of radionuclides with subsequent unexpected radiation levels, 2) the generation of pressure within a sample canister to a level where failure of the experiment could occur, and 3) a large, unplanned addition of positive reactivity due to improper placement of a sample in the reactor. Other mechanisms for damage, such as corrosion or excessive temperatures, are also possible.

13.2.6.2 Accident Analysis and Consequences

Because of the potential for experiments to cause damage to the reactor if not properly controlled, there are strict procedural and regulatory requirements (Reference 13.29) addressing the review and approval of an experiment to be placed in the reactor. These requirements are focused on ensuring that experiments remain safe, and, in that respect, incorporate requirements designed to reduce the likelihood of damage to the reactor and the possibility of radioactivity releases or radiation doses which exceed the limits of 10 CFR 20, should some type of failure occur. For example, specific requirements in MURR administrative procedures, such as the Reactor Utilization Request (RUR), establish detailed administrative procedures, technical requirements, and the need for safety reviews for all types of proposed reactor experiments.

Safety reviews of proposed experiments require the performance of specific safety analyses to assess such considerations as criticality and/or reactivity, heat generation, off-gassing and/or chemical reactions, and shielding. This review process ensures the safety of reactor experiments and has been successfully used for many years at MURR and other research reactors.

The MHA at MURR is the complete failure of a 5-gram LEU fueled experiment and is discussed in detail in Section 13.2.1. Limiting the generation of certain fission products in a fueled experiment helps ensure that occupational radiation doses as well as doses to the general public will be within the limits of 10 CFR 20 should there be an experiment failure involving fission-product release. A limit of 150 curies of I-131 through I-135 for each fueled experiment ensures that the occupational doses and doses to the general public in the unrestricted area due to radioiodine are well within 10 CFR 20 limits. The following

requirements on fueled experiments will ensure that projected doses are bounded by the MHA analyses:

- Fueled experiments must be designed and operated so that identifiable accidents such as a loss of primary coolant flow, loss of experiment cooling, etc. will not result in a release of fission products or radioactive materials from the experiment;
- The maximum temperature of a fueled experiment shall be restricted to at least a factor of two (2) below the melting temperature of any material in the experiment; and
- Fueled experiments containing inventories of I-131 through I-135 greater than 1.5 curies or strontium-90 (Sr-90) greater than 5 millicuries shall be vented to the facility ventilation exhaust stack through high efficiency particulate air (HEPA) and charcoal filters which are continuously monitored for an increase in radiation levels.

The generation of Sr-90 in a fueled experiment is limited to 300 millicuries, which is within the limits of the MHA analyses. Since no dose limits in the unrestricted area will be exceeded by the MHA, doses from similar fueled experiments where the Sr-90 inventory is limited to 300 millicuries will be safely within the limits of 10 CFR 20.

In order to reduce the likelihood of damage to the reactor or pool from explosive materials, explosive materials are not allowed to be irradiated nor are they allowed to be generated in quantities over 25 mg TNT-equivalent. The irradiation container for this material shall be designed and tested for a pressure exceeding the maximum expected pressure by at least a factor of two (2). Such containment will eliminate potential damage to reactor components or other experiments. (An example calculation for encapsulation pressurization was presented to the NRC in the written response to Request for Additional Information letter, dated September 3, 2010, Question 13.9.)

Reactivity limits placed on experiments ensure (1) that the rate of change of any movable experiment be 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 be less than the value of reactivity which would cause a violation of a safety limit. Each movable experiment or movable parts of any individual experiment is limited to a maximum worth of 0.001 $\Delta k/k$. The magnitude of the reactivity worth of each unsecured experiment is limited to 0.0025 $\Delta k/k$. MURR can withstand a positive reactivity step insertion of 0.007 $\Delta k/k$ with no core damage. This value is then used to establish the more restrictive limit of 0.006 $\Delta k/k$ for the reactivity worth of each secured removable experiment, for the absolute total reactivity worth of all experiments in the center test hole, and for the total reactivity worth of all unsecured experiments in the reactor. Section 13.2.2 provides the step reactivity insertion analysis for determining the reactivity limits for all MURR experiments.

13.2.6.3 Conclusions

As shown by the preceding analyses, limitations placed on the MURR experimental facilities help prevent accidents which would jeopardize the safe operation of the reactor or create a hazard to the safety of the facility staff and/or general public.

Consequently, limitations on experiments greatly reduce the possibility of experiment failure and minimize the consequences of postulated accidents such that the consequences are less than that of the MHA. The Reactor Utilization Request (RUR) outlines important criteria which are considered during the review and approval of any reactor experiment. The most important section of the RUR is the safety analysis section. This section analyzes all possible accidents and transients to determine if the experiment introduces a question pursuant to 10 CFR 50.59 or if there are any other safety-related issues associated with the experiment that need to be resolved.

13.2.7 Loss of Electrical Power

Normal electrical power is supplied by the University of Missouri at Columbia Power Plant and/or the City of Columbia through an electrical power distribution system which serves the entire campus. Should the facility suffer a loss of normal electrical power, the emergency electrical power system would provide electrical power to essential reactor components in order to allow continued operation of selected monitoring systems and to assure personnel safety. Emergency electrical power is supplied by a 275-kW diesel generator through an Automatic Transfer Switch (ATS) which transfers source power from the normal electrical power system to the emergency power system. The consequences of a loss of normal electrical power with the emergency electrical power system operable are insignificant. If operating, the reactor would automatically scram and an orderly shutdown of process equipment would be performed by the reactor operator. The potentially more challenging event, a loss of normal electrical power with the subsequent failure of the emergency power system (i.e., a complete loss of electrical power to the facility), is analyzed below.

The normal and emergency electrical power systems are discussed in detail in Chapter 8, “Electrical Power Systems.” Included therein is a description of the transfer of electrical power from the normal source to the emergency power bus.

13.2.7.1 Accident-Initiating Events and Scenarios

A loss of normal electrical power could occur due to the many events and scenarios which routinely affect the distribution of commercial electrical power. A loss of emergency electrical power implies a malfunction of the emergency power system. The most probable cause would be the failure of the emergency diesel generator to start. However, the exact circumstances that lead to a complete loss of electrical power to the facility are immaterial to the analysis. Therefore, a complete loss of electrical power will be analyzed for the following two scenarios: 1) a complete power loss with the reactor operating at 10 MW, and 2) a complete power loss with the reactor shutdown.

13.2.7.2 Accident Analysis and Consequences

The electrical loads powered by the emergency electrical power system are listed in Section 8.2.4. The accident analysis contained herein will describe how each reactor system or selected load is affected by a complete loss of electrical power.

13.2.7.2.1 Power Loss with the Reactor Operating

The reactor is operating at 10 MW. Should there be a loss of normal electrical power, with a corresponding failure of the emergency electrical power system, the following reactor facility responses would occur.

13.2.7.2.1.1 Reactor Control System

The reactor would scram due to the interruption of current to the electromagnets which hold the shim blades in position. The shim blades would drop by gravitational force into the core region and the reactor would be shut down.

13.2.7.2.1.2 Instrumentation and Control (I&C) Systems

Reactor and process instrumentation is powered by two 120-VAC distribution panels located in the reactor control room. The 15-kVA Uninterruptible Power Supply (UPS) provides 120-VAC electrical power to these panels until the discharge limit of the UPS 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). At this time, the UPS would automatically secure to prevent a low voltage transient on the system. No control console information would then be available to the operators. All subsequent information regarding the status of the reactor (e.g., process equipment, control blade position, etc.) would have to be obtained locally by visual observation. A reactor shutdown may be confirmed by visually observing 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 “bluish glow” (Cerenkov radiation) in the region of the reactor core. Examination of the valve operators in the reactor pool upper bridge area would indicate the position of these valves and confirm their proper operation. The Area Radiation Monitoring System (ARMS) would no longer be operable. However, health physics personnel would be able to monitor radiation levels with portable instruments.

13.2.7.2.1.3 Reactor Process Systems

Reactor process equipment (e.g., primary and pool coolant circulation pumps, isolation valves, etc.) would fail to their shutdown positions due to their fail-safe design. A loss of electrical power would cause a cessation of flow in the primary and pool coolant systems and a closure of their isolation valves. Decay heat removal isolation valves V546A and V546B would automatically open, allowing a flow path for primary coolant through the in-pool heat exchanger. The decay heat removal system complies with the single failure criterion of IEEE-279 (Reference 13.31) and requires no electrical power to function as

designed (see Section 5.8). In addition, the large reserve of coolant in the pool provides a significant heat sink for heat removal during a loss of power event.

13.2.7.2.1.4 Containment Building Ventilation Isolation Doors 504 and 505

Electrical power to the drive motors of Doors 504 and 505 would be lost and they would fail to close (or open) in response to any electrical signal. The control system which inflates the gaskets is actuated only when the isolation doors are in the closed position and therefore, the sealing gaskets would not inflate. The backup isolation doors, however, fail closed on a loss of electrical power to their solenoids.

13.2.7.2.1.5 Emergency Air Compressor

The emergency air compressor would fail to automatically start in response to a pressure decrease in its accumulator tank. The volume of the tank is 10.5 ft^3 (0.297 m^3) at a nominal pressure of 100 psig (689 kPa above atmosphere), which is sufficient to inflate all sealing gaskets. However, the ability to recharge the accumulator tank to normal operating pressure would be lost. The primary function of the emergency air compressor is to provide compressed air to the sealing gaskets of all containment isolation doors. Since the doors would not be operable, there would be little demand on the emergency air system.

13.2.7.2.1.6 Truck Entry Door 101

Door 101 is maintained in the closed position with the sealing gasket inflated during reactor operation. The status of this door would be unaffected by a loss of electrical power.

13.2.7.2.1.7 Personnel Air Lock Doors 276 and 277

The entry control system for Doors 276 and 277 is designed and interlocked such that one door is always closed and sealed, ensuring maintenance of containment integrity. A manually operated throw-out clutch allows manual operation of these doors if the ability to operate them electrically is lost. This allows an individual to exit or enter the containment building through these doors in the event of a power failure. However, the ability to maintain at least one door in the closed position with its sealing gasket inflated is no longer available. Since the reactor would be shutdown and no release scenario would be credible, containment integrity would not be a primary consideration.

13.2.7.2.1.8 Facility Ventilation Exhaust Fans (EF-13 and EF-14)

EF-13 and EF-14 would secure during a complete loss of electrical power. The operation or inoperation of these fans would have no consequence on the status of the reactor since the reactor is shutdown.

13.2.7.2.1.9 Reactor and Laboratory Corridor and Exit Lights

The corridor and exit lights in both the laboratory and reactor containment building would extinguish on a complete loss of electrical power. Battery operated emergency lights strategically positioned

throughout the facility would provide sufficient lighting in all critical locations, particularly along emergency escape routes. Each light has a self-charging battery pack and a switching circuit that actuates the emergency light upon electrical power failure.

13.2.7.2.1.10 Fan Failure Alarm System

The alarm system for EF-13 and EF-14 would no longer be operable. However, as stated in Section 13.2.7.2.1.8, a loss of electrical power would secure the facility ventilation exhaust fans. Therefore, the fan failure alarm system would not be required.

13.2.7.2.1.11 Intercommunication System

The reactor facility utilizes two principal communication systems: a computerized telephone system and a multiple-station, two-way intercommunication system. A loss of electrical power would result in the inability to communicate using the two-way system. However, the telephone system would still allow communication to each laboratory and to various areas inside the reactor containment building. Available portable hand-held radio transmitter-receivers ("walkie-talkies") also provide a method of communication throughout the facility.

13.2.7.2.1.12 Evacuation/Isolation Alarm System

All audible and visual facility evacuation and reactor isolation alarms would be lost. The facility emergency plan provides, as needed, alternate communication methods for evacuation.

13.2.7.2.1.13 Diesel Generator Room Distribution Panel

Power to the emergency diesel generator control panel, room lighting and room temperature control is provided by this distribution panel. A complete loss of electrical power would de-energize these loads. However, failure of the emergency power system would negate the need for electrical power to this panel.

13.2.7.2.1.14 Nitrogen Station

The loss of electrical power to the solenoid-operated valves of the nitrogen station would prevent the nitrogen station from being able to supply nitrogen gas to the pressurizer. Since the reactor is shutdown due to the loss of electrical power and decay heat is being removed by the in-pool heat exchanger, the loss of the nitrogen station would have no effect on the status of the reactor.

13.2.7.2.1.15 Fire Protection System

Normal supply power would be lost to the fire detection system. However, the system is equipped with a battery backup that would provide power for the entire system for a period of twenty-four (24) hours. Additionally, fire protection is not required to accomplish a safe shut down of the reactor or to maintain a safe shutdown condition.

13.2.7.2.2 Power Loss with the Reactor Shutdown

If a complete loss of electrical power occurs with the reactor in a shutdown condition, the status of the reactor systems or loads would be the same as discussed above (Section 13.2.7.2.1) with the following exceptions.

13.2.7.2.2.1 Reactor Control System

Since the reactor is shutdown with all process equipment secured, it is not critical that the status of the reactor or reactor systems be monitored on a continuous basis. Alternate means, such as direct visual inspection, could be used to assess reactor status.

13.2.7.2.2.2 Personnel within the Reactor Containment Building

Personnel who are granted unescorted access have been instructed on how to manually operate the personnel airlock doors. Simple instructions are posted next to each door so that personnel can exit the containment building in an orderly fashion.

13.2.7.2.2.3 Reactor Containment Building

Since Door 101 would be closed and sealed, and the 16-inch (40.64-cm) ventilation exhaust isolation valves and the backup doors would have failed to the closed position, and the reactor containment building would be isolated. Access out of the containment building would be via personnel air lock Doors 275 and 276, which would not be resealed.

13.2.7.3 Conclusions

The MURR's design does not require electrical power to safely shut down the reactor or to maintain an acceptable shutdown condition. In the event of a complete loss of electrical power to the facility, the reactor would be shut down and the core would be cooled by natural convection circulation through the in-pool heat exchanger. In addition, the emergency electrical power system is not required for protection of the integrity of the fuel elements.

13.2.8 External Events

13.2.8.1 Accident-Initiating Events and Scenarios

Meteorological disturbances, such as hurricanes, tornadoes, or floods, were considered as potential external accident-initiating events at MURR. 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 local rainstorm activity. However, the reactor facility is situated about 200 feet (61 m) from the nearest 100-year floodplain. Severe thunderstorms and tornadoes do

occur in the region. Between 1950 and 2025, fewer than 1% of all tornadoes recorded in Missouri occurred within Boone County. Thunderstorms are observed during every month of the year. MURR is designed to withstand the extreme wind speeds associated with any thunderstorm activity. Section 2.3.1 provides detail on the overall meteorology of the area. Sections 2.4.1 and 2.4.2 describe surface water and water drainage details for the area. These sections provide no basis for a significant meteorological accident scenario.

Section 2.5.2 provides a detailed assessment of the region's seismology. Evaluation of the extensive data allows a conclusion that Boone County shows little evidence of past seismic activity. Local seismology, as summarized in Section 2.5.2.3, and seismological conclusions in Section 2.5.2.6, state that 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.

Accidents caused by human controlled events, such as an explosion, toxic release, or any other unusual occurrence which could damage the reactor, are of low probability. There are no nearby industrial, transportation, or military facilities with the potential of causing a credible accident scenario which prevents a safe reactor shutdown or results in the release of radioactive material exceeding the general public exposure limits of 10 CFR 20.

13.2.8.2 Accident Analysis and Consequences

The basic design and structure of the facility provide significant protection for the reactor against any external events. The reactor is housed in a five-level poured concrete building with 12-inch (30.48-cm) thick reinforced exterior walls. The containment building has been determined to be structurally adequate to resist the Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) (see Section 2.5.2.5). Additionally, the reactor core is located below grade level and surrounded by a massive reinforced concrete edifice (biological shield) approximately 31 feet (9.4 m) in height and varying in thickness from 3.0 to 7.5 feet (0.9 to 2.3 m). 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 provides excellent protection against natural phenomena that could result in damage to the reactor core assembly. In addition, the reactor is located in the Hinkson Creek valley with a high bluff directly to the west. This location helps protect the reactor from severe weather phenomena, such as high winds and tornadoes.

13.2.8.3 Conclusions

Based on meteorological, seismic and other characteristics of the region, it can be concluded with reasonable assurance that there are no geographic or demographic features that render the MURR site unsuitable for operation of the facility, and no accidents with consequences even approaching the MHA will be caused by external events related to the site or the region.

13.2.9 Mishandling or Malfunction of Equipment

This class of accidents represents occurrences that do not fall into one of the other eight postulated accident events or categories.

13.2.9.1 Leak in the Pool Coolant System

The severity of a leak in the pool coolant system would, in general, be dependent upon its location. A leak in a section of the system which cannot be immediately isolated is more severe than a leak developing in a section that can be isolated. For the purposes of this analysis, a leak in the pool coolant system can originate in any one of the four following areas:

- In the section of piping between the reactor pool and return line isolation valve V509;
- In the section of piping between the reactor pool and supply line check valve V519A;
- In a section of the system not specifically mentioned as one of the two previous locations; and
- In the beamports.

The accident of greatest consequence is a leak in the pool coolant piping between the reactor pool and valve V509 or valve V519A. A leak in this section of the system could potentially lower the water level in the reactor pool to a point where the reflector tank would become completely drained. A leak developing in another section of the pool coolant system could lead to some water loss from the reactor pool; however, once detected, the leak would be isolated and the pool level stabilized. A leak through one of the beamports is analyzed and discussed in detail in Section 13.2.9.2.

13.2.9.1.1 Accident-Initiating Events and Scenarios

A leak in the pool coolant system would most likely occur through either corrosion of the piping or a mechanical failure. Leaks caused by corrosion should be small and easily detected before the water level in the reactor pool has lowered significantly. In such a case, make-up water would be supplied by the reactor plant make-up water system (see Section 9.12.3) until the leak could be repaired. Any leakage would be collected and diverted to the radioactive liquid waste retention and disposal system.

A leak due to a mechanical failure (e.g., complete shearing of a pipe) could potentially cause a major loss of water from the pool. In the event of a leak on the isolatable portion of the pool coolant system, an automatic reactor scram would occur from either a reduction in coolant flow, pressure, or pool water level. If not already initiated automatically, sufficient time would be available for the reactor operator to secure the pool coolant circulation pumps and close isolation valve V509 before a significant loss of pool water could occur. If the pool is drained to a level where the in-pool heat exchanger is exposed, operation of the decay heat removal system would be impaired. However, the primary coolant system can continue to operate and dissipate any core decay heat to the secondary system. The more serious accident, a leak on the un-isolable portion of the pool coolant system, is discussed below.

13.2.9.1.2 Accident Analysis and Consequences

The primary consideration when discussing an un-isolable leak in the pool coolant system and the potential reactivity effect of voiding the reflector region is calculating the depth to which the pool would drain if either the reactor pool 6-inch (15.24-cm) supply or return line is completely sheared and emergency pool fill (see Section 13.2.9.2.2) is initiated. The separation of the two piping sections must be complete, otherwise there would be no significant loss of water. The assumption is also made that the reflector plenum natural convection valve V547 is closed. This valve is normally maintained in the open position during reactor operation and would be closed by the reactor operator as required by the emergency procedures. If the reactor is operating, an automatic scram will occur from either a reduction in coolant flow, pressure, or pool water level.

If the piping break were to occur in the 6-inch (15.24-cm) return line, the reactor pool would drain through the reflector tank. The resistance to flow presented by the reflector region is such that 13 feet (4 m) of water would remain above the top of the reflector tank at equilibrium. In this scenario the reactivity effect would be zero.

If the piping break were to occur in the 6-inch (15.24-cm) supply line, drainage would be through the pool diffuser. The diffuser is a vertical section of pipe with 252 5/8-inch (1.59-cm) diameter holes (36 rows of seven holes per row), which allows the return water to discharge to the reactor pool at a minimal velocity. The reactor pool would drain to a level approximately 10 rows below the top of the diffuser. This level would be lower than the top of the reflector tank. However, since the reactor pool would not be draining through the reflector tank, the reflector region would remain flooded, and the reactivity effect of this accident would also be zero.

The worst-case scenario is a simultaneous break in both the supply and return lines, in which case the reflector tank may become completely drained. With the reactor shutdown (shim blades fully inserted), K-eff would increase to a maximum value of 0.93 with the reflector region approximately 50% voided and decrease to less than 0.88 when the reflector region is completely voided. If operating, the reactor would scram on either low pool coolant flow or low system pressure and K-eff would respond as indicated in the previous sentence.

In the event of a piping break, it is important to consider how rapidly water loss would occur and how much time would be available to take corrective actions. If the reactor pool is draining through the diffuser, the calculated time for the water surface to reach the level of the top of the reflector tank is approximately six minutes. With raw water being supplied to the pool at the rate of 1,000 gpm (3,785 lpm) by the emergency pool fill system, the water level in the reactor pool would reach the top of the reflector tank in about ten minutes.

13.2.9.1.3 Conclusions

Depending upon its size, a leak in a section of the pool coolant system which can be isolated would result in some water loss from the reactor pool. The consequences of a slow leak would be minimal and

would require collection and containment of the leakage. If operating, the reactor would be shut down, the pool coolant circulation pumps secured, and pool coolant isolation valve V509 closed. If the leak is large enough that the water level in the pool decreases to a point where a higher than normal radiation level is created on the upper bridge, demineralized water from the reactor plant make-up water system would be added. Any leakage would be contained in the mechanical equipment room (Room 114) and pumped to the radioactive liquid waste retention and disposal system.

If a leak developed in an un-isolable section of the pool coolant system, the reactor would be shut down, the pool coolant circulation pumps secured, and the pool coolant isolation valve V509 closed. If the leak could not be quickly secured, the emergency pool fill system would supply raw water to the reactor pool in excess of 1,000 gpm (3,783 lpm). This would allow the facility to be filled with water to the ground level, ensuring the reactor core remained covered.

13.2.9.2 Shearing of a Beamport

The six beamports are arranged in groups of three, on opposite sides of the reactor. Each beamport assembly consists of three major components: a fixed beamport liner, a removable beamport liner (beamtube), and a removable collimator liner. When fully inserted, each beamtube penetrates the graphite reflector region and terminates adjacent to the beryllium reflector. The beamtube is attached to the fixed liner by means of a bolt ring located in a recessed vestibule. A packing gland assembly seals the beamtube to the fixed liner. Typically, the removable collimator liner is installed within the beamtube. The removable collimator liner is attached and sealed to the beamtube in the same way the beamtube is attached and sealed to the fixed liner. This arrangement allows the beamtube to be filled with helium or demineralized water.

13.2.9.2.1 Accident-Initiating Events and Scenarios

A heavy object dropped from the 15-ton (13,608-kg) capacity overhead rectilinear crane or from either the upper or lower reactor pool bridge could potentially shear one of the beamtubes extending into the pool. This would create a path for water to leak from the reactor pool if the removable collimator liner was not installed in the beamtube. However, it is standard practice at MURR to have the collimator liner installed whether or not a beamport is being utilized. Therefore, the complete shearing of a beamtube would not produce a condition where pool water would be lost since the packing gland seal between the collimator liner and the beamtube is designed to withstand the static head of the water in the reactor pool with no leakage.

Further precautions to minimize the possibility of damage to the reactor or fuel are undertaken while moving the shipping cask into or out of the reactor pool. This evolution is performed in an area of the pool called the weir floor, which is well away from the reactor structure and above the elevation of all fuel storage areas. Movement of the cask is performed by an experienced operator and multiple spotters. Additionally, all cask lifting equipment, including the 15-ton (13,608-kg) capacity crane, is rigorously maintained, including preventive maintenance. (Reference 13.53)

13.2.9.2.2 Accident Analysis and Consequences

In the case of shearing a beamtube without the collimator liner installed, there would be a loss of water from the reactor pool. Closure of the beamport shield door would provide some restriction in water flow. However, this can only be accomplished if no experimental equipment is positioned within the line of the shield door. Any obstruction in the flow path would serve to decrease the rate of water loss from the reactor pool. If the reactor is operating, a decrease in the pool water level would cause an automatic rod run-in. The rod run-in ensures that the radiation level above the pool surface from direct core radiation will remain less than 20 mrem/hr. Should the pool water level continue to decrease, a reactor scram would occur.

Two 7,000-gallon (26,498-l) steel tanks provide storage of demineralized make-up water for the reactor. The available contents of both tanks can be gravity-fed or rapidly pumped to the reactor pool. If the loss of pool water cannot be compensated for by this method, the emergency pool fill system can supply raw water in excess of 1,000 gpm (3,785 lpm) to the reactor pool. This rate of water addition is adequate to maintain greater than three feet (0.9 m) of water above a completely severed 6-inch (15.24-cm) beamport with no impediments in the port. The emergency pool fill system is actuated by the opening of a 4-inch (10.16-cm) ball valve, located in a recessed box immediately adjacent to the control room. This valve operation requires only a quarter turn from closed to full-open. The MU water supply system provides a virtually unlimited source of raw water for the emergency pool fill system. Five deep wells, each with varying flow rates, supply water to a 10-inch (25.4-cm) fire main that services the campus. The Southwest well, which is located approximately two hundred feet 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 (25.4-cm) main to provide a portion of the campus water supply. The 8-inch (20.32-cm) wet fire line that provides the flow path for the emergency pool fill system is connected to this 10-inch (25.4-cm) main. To ensure a continuous supply of water to the 10-inch (25.4-cm) main, a 1,000-kW diesel generator provides emergency electrical power to the supply pumps upon a loss of normal electrical power. In addition to the five deep wells, a 10-inch (25.4-cm) 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. All isolation valves between the 10-inch (25.4-cm) fire main and the 4-inch (10.16-cm) ball valve are locked open, ensuring that the water supply to the emergency pool fill system cannot be inadvertently isolated.

13.2.9.2.3 Conclusions

The shearing of a beamtube due to a dropped object is non-credible. Due to the physical constraints surrounding the beamtubes, such as the nuclear instrumentation drywells and the upper and lower bridge assemblies, it would be difficult to suspend a heavy object above a beamtube. It is standard practice at MURR to have the removable collimator liner installed in the beamtube, thereby preventing a loss of pool water should a beamtube be sheared. If a sheared beamtube causes a loss of reactor pool

water, an unlimited amount of raw water is available through the emergency pool fill system to maintain pool water at a level of at least three feet (0.9 m) above the sheared beamtube. If pool water decreases to a level which exposes the in-pool heat exchanger and impairs its operation, reactor core decay heat may still be removed by operating the primary coolant system. Therefore, there will be no fuel damage or release of fission products, and the impact of this accident is bounded by the MHA.

13.2.9.3 Failure of In-Pool Heat Exchanger Isolation Valves to Open

In the event of a loss of primary coolant flow, a primary loop isolation or a loss of electrical power, isolation valves V546A and V546B automatically open, providing a flow path for primary coolant through the in-pool heat exchanger. Heat from the core will be transferred to the in-pool heat exchanger and then dissipated to the reactor pool. These valves are arranged in parallel so that the operation of either valve will allow the decay heat removal system to perform its intended function. Two solenoid-operated valves, installed in series, control the air supply to each valve actuator. A closure signal or a loss of electrical power will de-energize both solenoid valves, vent the air from the actuator, and allow a spring to open the valve. Actuation of either solenoid valve will open its associated in-pool heat exchanger isolation valve. The 546 valves may also be manually operated from the reactor pool upper bridge, if required.

13.2.9.3.1 Accident-Initiating Events and Scenarios

The decay heat removal system satisfies the single failure criterion of IEEE-279 (Reference 13.31). Therefore, the failure of both 546 isolation valves to open when required is not considered credible. This event has, been analyzed using the MURR RELAP5 model, which demonstrates that even without an engineered decay heat removal system, core decay heat can be adequately 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. A detailed discussion of this analysis is not included, as the more severe loss of primary coolant accident concludes that no fuel damage will occur (see Section 13.2.3).

13.2.9.3.2 Conclusions

The likelihood that both in-pool heat exchanger isolation valves V546A and V546B would fail to open when required is not credible. However, as shown by the loss of primary coolant accident analysis in Section 13.2.3, even without operation of the in-pool heat exchanger, sufficient heat will be transferred through the reactor pressure vessels and associated isolated piping. The reactor core would remain covered with water and fuel plate temperatures would remain well below the temperature at which fission product release from the fuel could occur. Therefore, this postulated accident is less significant than the MHA.

13.2.9.4 High-Pressure Transient

This accident is analyzed in order to demonstrate that the primary coolant system pressure boundary is protected from any postulated high-pressure transient. Protection against a high-pressure transient is

not required for fuel cladding integrity. However, minimizing the likelihood of propagating a fracture in the pressure boundary of the primary coolant system is important in that the primary coolant system provides a barrier of protection against the release of fission products should a fuel element failure occur. Multiple equipment malfunctions, in addition to a lack of response by the reactor operator, would have to occur in order for primary coolant system pressure to increase to the Technical Specification limit of 110 psig (758 kPa above atmosphere).

The pressurizer system maintains primary coolant system pressure within the LSSSs for both 5- and 10-MW operation. Reactor inlet pressure is maintained at 85 psia (586 kPa) by nitrogen gas admitted to, or released from, the pressurizer tank. The pressurizer also provides a path for the addition of primary grade water lost during normal operational evolutions such as primary coolant sampling.

The pressurizer and water make-up systems are described in detail in Chapter 5, "Reactor Coolant Systems." The nitrogen supply system is described in Chapter 9, "Auxiliary Systems." The instrumentation which actuates the automatic features of the pressurizer and makeup water systems is discussed in Section 7.6.5.

13.2.9.4.1 Accident-Initiating Events and Scenarios

An increase in the primary coolant system pressure would be caused by any one of the following three scenarios:

- The addition of nitrogen gas through nitrogen addition valve V526;
- A reactor plant heat-up from 70 to 160 °F (21 to 71 °C); or
- The continuous addition of make-up water by primary coolant charging pump P533.

In all three postulated scenarios, the assumption is made that pressurizer vent valve V545 will fail to open when required. Valve V545 is an air-operated-to-open, spring-to-close diaphragm valve which vents the pressurizer to the facility ventilation exhaust system should pressurizer pressure increase to approximately 105% of normal operating pressure.

13.2.9.4.2 Accident Analysis and Consequences

Nitrogen addition valve V526 is an air-operated-to-open, spring-to-close diaphragm valve (fail-safe) which admits nitrogen gas to the pressurizer when system pressure decreases to about 95% of normal operating pressure. If valve V526 should malfunction and stick in the open position, the primary coolant system pressure would increase to a maximum of about 95 psia [(655 kPa) nitrogen regulator set point]. If the reactor is operating, a high-pressure scram (at about 115% of normal operating pressure) would initiate a shutdown prior to reaching this value. Should the nitrogen regulator also fail open, nitrogen at a pressure of 140 psig (965 kPa above atmosphere) would be supplied to the pressurizer. In order to make the postulated scenario realistic (i.e., only two failures), the assumption made now is that nitrogen pressurizer vent valve V545 would respond and open as required. With valve V545 open, system pressure would attain a maximum of 73.5 psig (507 kPa above atmosphere) (Reference 13.32).

A reactor plant heat-up from an initial primary coolant temperature of 70 °F (21 °C) to a final temperature of 160 °F (71 °C) would cause approximately 43 gallons (163 l) of water to expand into the pressurizer tank. Assuming that the pressurizer tank was initially half full of water, the addition of 43 gallons (163 l) of primary coolant would compress the nitrogen bubble and increase system pressure to a maximum of 103 psig (710 kPa above atmosphere). This transient would be terminated prior to this point by operator action when the "Pressurizer Hi Pressure" annunciator alarm is received (at about 110% of normal operating pressure). If no operator action is taken, a "Pressurizer High Pressure Scram" would also automatically occur by at least 80 psig (552 kPa above atmosphere) prior to reaching the 103 psig (710 kPa above atmosphere) pressure (Reference 13.32).

Finally, the most severe high-pressure transient analyzed is the continuous addition of make-up water by the primary coolant charging pump P533. The assumption made is that the coolant charging pump starts at an initial pressurizer pressure of 73.5 psig (507 kPa above atmosphere) (upper level of the normal operating band). The primary coolant system would increase to a maximum pressure of about 100 psig (689 kPa above atmosphere). At this pressure, even if the nitrogen pressurizer vent valve V545 did not open, one of the two primary coolant system relief valves would lift and relieve pressure. In reality, however, this transient would be terminated prior to this point by operator action when the "Pressurizer Hi Pressure" annunciator alarm is received. If the reactor is operating and no operator action is taken, a "Pressurizer High Pressure Scram" would automatically initiate a shutdown prior to reaching the relief valve set point.

13.2.9.4.3 Conclusions

As shown by the preceding analyses, MURR is adequately protected from postulated high-pressure transients. A system high pressure alarm alerts the operator to an abnormal high-pressure condition, allowing for corrective action prior to reaching a protective system set point. Should primary coolant system pressure increase to 115% above normal operating pressure, but not in excess of 80 psig (552 kPa above atmosphere), a "Pressurizer High Pressure Scram" will automatically shut down the reactor. The primary coolant system pressure boundary is further protected from overpressure by pressure relief valves installed on the nitrogen pressurizer and the primary coolant system. The relief valve set points are lower than the Technical Specification limit of 110 psig (758 kPa above atmosphere), providing sufficient margin to ensure that the primary coolant system design pressure of 125 psig (862 kPa above atmosphere) is not exceeded. Since protection against high pressure transients is not required for fuel cladding integrity, fission product release is not a factor in this postulated accident and the MHA is still the accident presenting the greatest overall impact.

13.2.9.5 Failure of the Neutron Startup Source

An antimony-beryllium neutron source was initially used in the startup program. However, gamma photons (largely from activated structural components) now initiate (γ, n) and (n, 2n) reactions in the beryllium reflector that far surpass the startup neutron population which could be introduced by the antimony-beryllium source, and this source 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 is described in greater detail in Section 4.2.4.

13.2.9.5.1 Accident-Initiating Events and Scenarios

A failure of the neutron source could occur due to a small leak or a sudden rupture of the source capsule. The inadvertent removal of the neutron source from the reactor pool is also discussed.

13.2.9.5.2 Accident Analysis and Consequences

The neutron source consists of a mixture of compressed antimony and beryllium powder doubly encapsulated in 304 stainless steel with all seams tungsten inert gas (TIG) fusion welded. The inner container was leak-tested using a standard bubble test procedure. The outer capsule was leak-tested by helium mass spectrograph methods at an overpressure of 500 psig (3.45 MPa above atmosphere). The sensitivity of the latter leakage test is 10^{-8} cm^3/sec. If a small leak should develop, it is highly unlikely that the leak could go undetected for more than one day due to the monitoring practices observed by the control room staff as well as the weekly pool coolant sample analysis performed at MURR.

The surface of the reactor pool is monitored by two radiation monitors with remote readouts in the control room. The readouts from these monitors are observed and recorded in the control room logs every two hours by the control room staff. A normal baseline activity and surface dose rate exists in the pool from the presence of activation products in the pool coolant water. This baseline varies little from day to day except when samples are being handled in the pool. Antimony-124 is not a routine isotope found in the pool coolant water based on weekly sampling analysis. A sample breach of the antimony-beryllium source with as little as 1% of the source leaking into the pool would cause a dose rate increase of approximately 33%. This would result in an activity concentration of 1.32×10^{-2} µCi/ml in the pool coolant water, well above the minimum detectable activity (MDA) for gamma spectral analysis of pool coolant water. An increase of this magnitude over the normal baseline activity in the pool would be readily observable to the control room staff via the above pool radiation monitors and would raise questions among the staff as to the cause of the increase. Once solid samples and other sources in the pool have been ruled out as the cause, the Reactor Manager would request that a pool coolant water sample be taken and analyzed. Once the analysis is completed, the presence of antimony-124 would be confirmed, and steps would be taken to mitigate the situation.

The detection limits for antimony-122 (Sb-122) and antimony-124 (Sb-124) are approximately 5×10^{-6} µCi/ml and 6×10^{-6} µCi/ml, respectively. Therefore, a leaking source would be discovered well before the allowable water effluent limits of 10 CFR 20, Appendix B are reached. The volume of the reactor pool, including the pool coolant system, is approximately 28,000 gallons (105,992 l). Therefore, an antimony concentration of 1×10^{-6} µCi/ml would equate to an inventory of only 106 µCi in the total pool system volume. (Reference 13.54)

As discussed above, a small leak in the neutron source would be detected before a hazardous condition developed. However, a sudden rupture of the capsule could potentially release a large amount of

activity into the reactor pool. The scenario analyzed is a rupture due to an external force, e.g., capsule damage due to the impact of a large object. Due to the physical constraints (e.g., sample holder extension rods, offset mechanisms, etc.) surrounding the neutron source while it is being irradiated in the graphite reflector region, it is highly improbable that a large object could inadvertently fall on the source. For this reason, and the fact that the outer capsule was subjected to a 500-psig (3.45-MPa above atmosphere) overpressure with no loss of integrity during its fabrication, a large release of activity due to an external force is not considered credible. Also, careful sample handling techniques ensure that the neutron source will not be damaged when in use. In addition, Tipton (Reference 13.33) points out that beryllium has been irradiated to neutron fluences of 1.8×10^{20} n.v.t. (neutrons/cm^2s) with no dimensional changes. To achieve this n.v.t., the neutron source would have to be irradiated for approximately 13 weeks. This length of irradiation far exceeds the time required to produce a source strength of 100 curies of Sb-124 (the license limit). Additionally, there are no direct gaseous products from antimony's decay, and its expansion should be no greater than that of the beryllium. Consequently, pressure buildup due to volumetric expansion is negligible.

Strict administrative controls prevent the inadvertent removal of the neutron source from the reactor pool. The normal storage location is in the deep section of the pool. An identification tag is attached to the end of its handling line. Radiation surveys are conducted by the health physics staff any time radioactive material is moved in the reactor pool which may cause a radiation level of greater than 100 mrem/hr (exception: routine sample handling). In addition, radiation area monitors in the reactor pool area will alarm well before the antimony source approaches the pool surface, thereby alerting personnel in the area and allowing time for corrective action. Radiation surveys are also conducted any time the water level in the reactor pool is lowered. These controls serve to adequately limit the possibility of exposure to personnel from the neutron source.

13.2.9.5.3 Conclusions

As shown by the preceding analyses, a sudden rupture of the neutron source caused by an internal or external force is not considered credible. Should a small leak develop, it would be detected well before the allowable water effluent limits of 10 CFR 20 are reached. Administrative controls limiting the irradiation time of the neutron source prevent the activity level from exceeding the facility license limit.

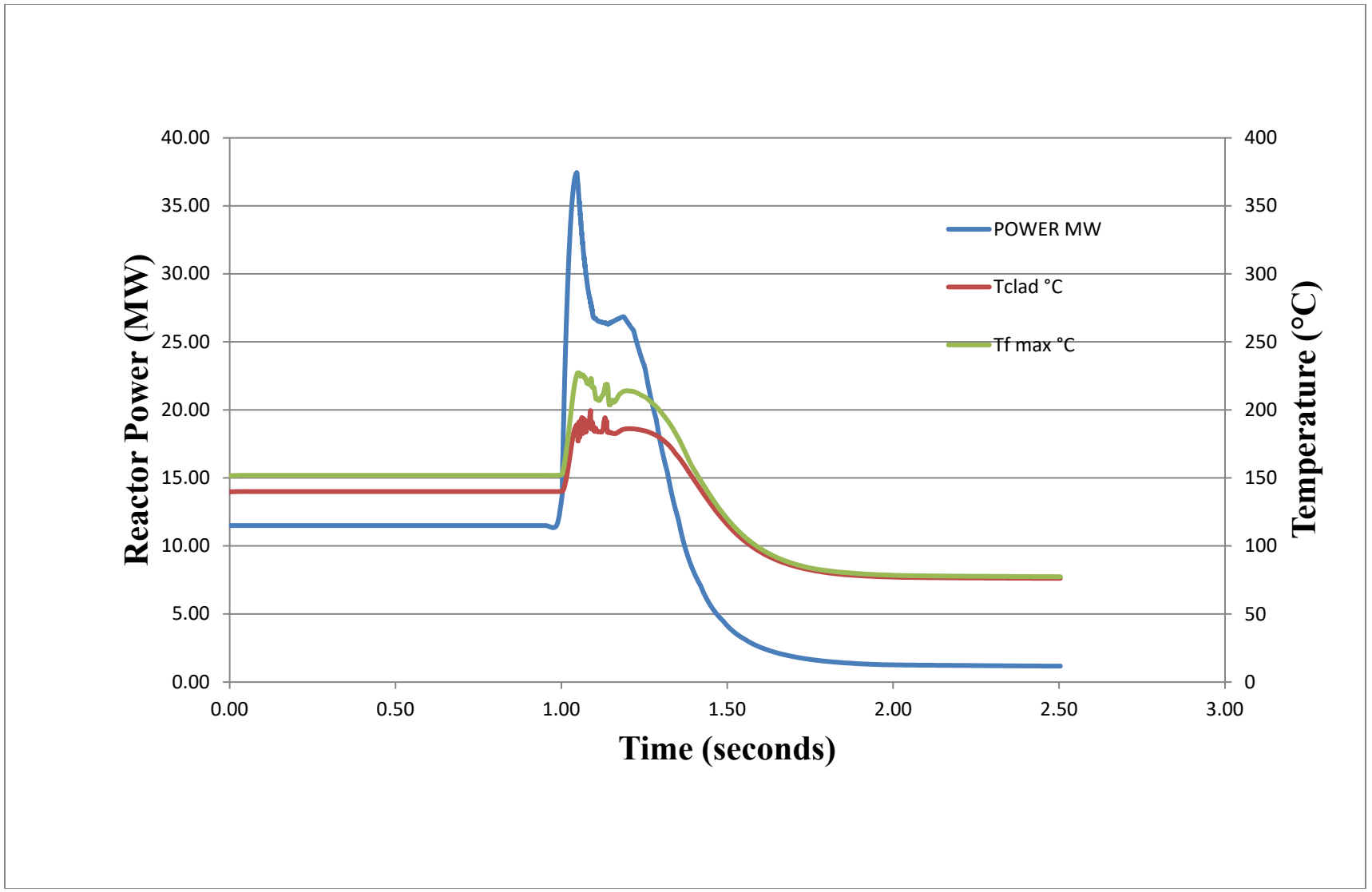


FIGURE 13.1 – REACTOR POWER, FUEL, AND CLADDING TEMPERATURES VS. TIME FOR A POSITIVE STEP INSERTION OF 0.006 ΔK/K

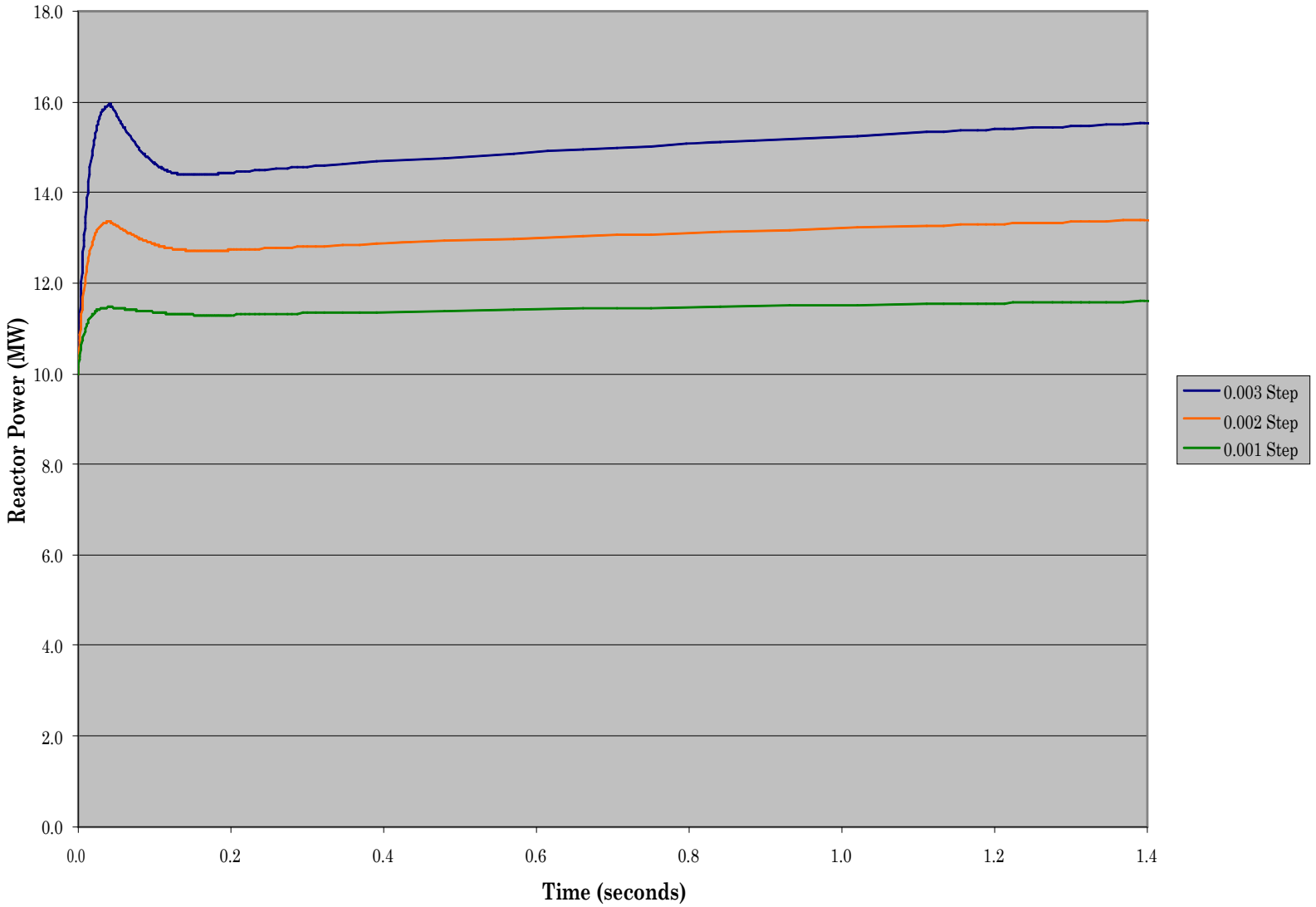


FIGURE 13.2 – REACTOR POWER VERSUS TIME FOR VARIOUS SMALLER POSITIVE REACTIVITY STEP INSERTIONS

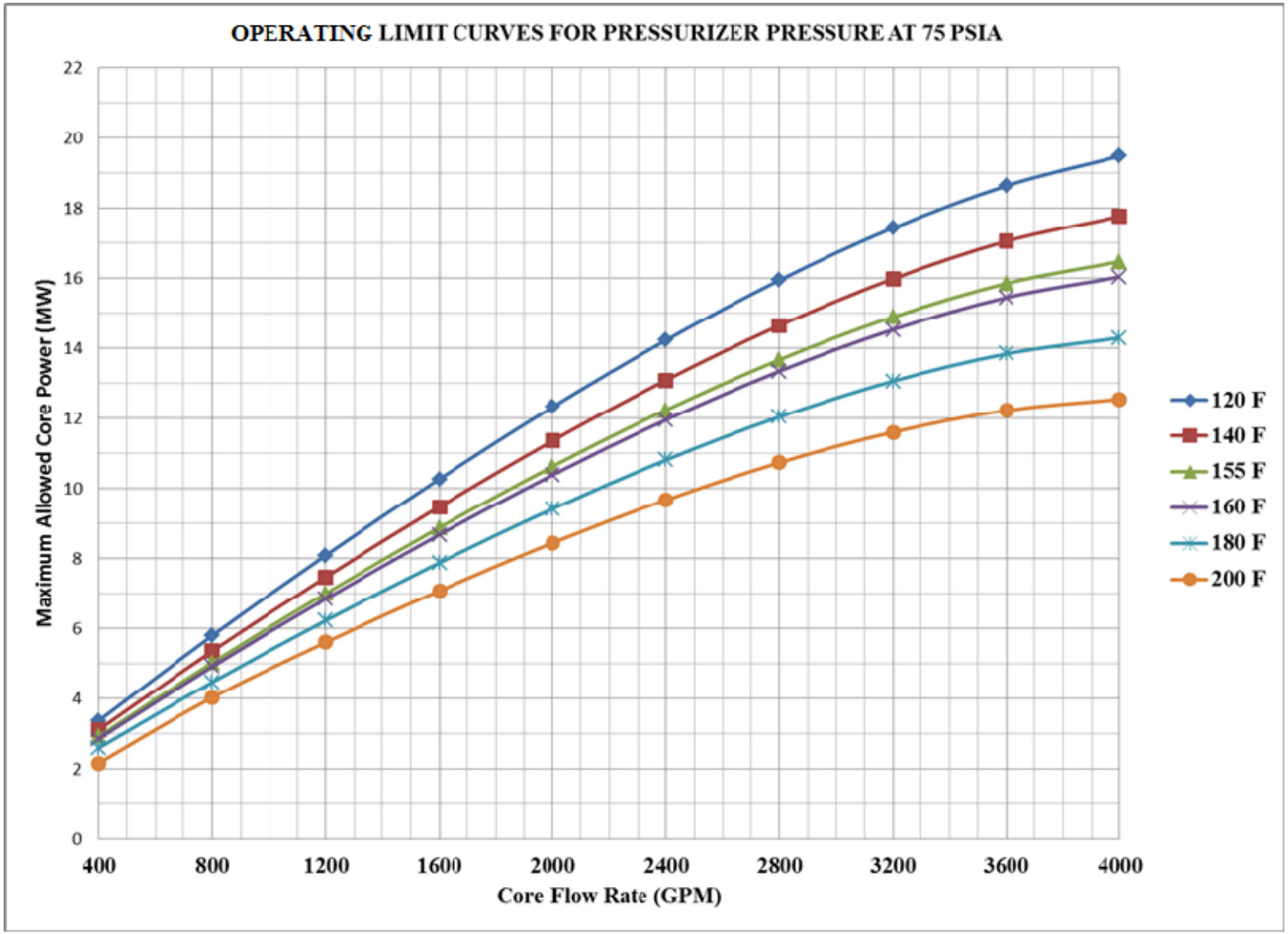


FIGURE 13.3 – MURR OPERATING LIMIT CURVES (PRESSURIZER PRESSURE AT 75 PSIA)

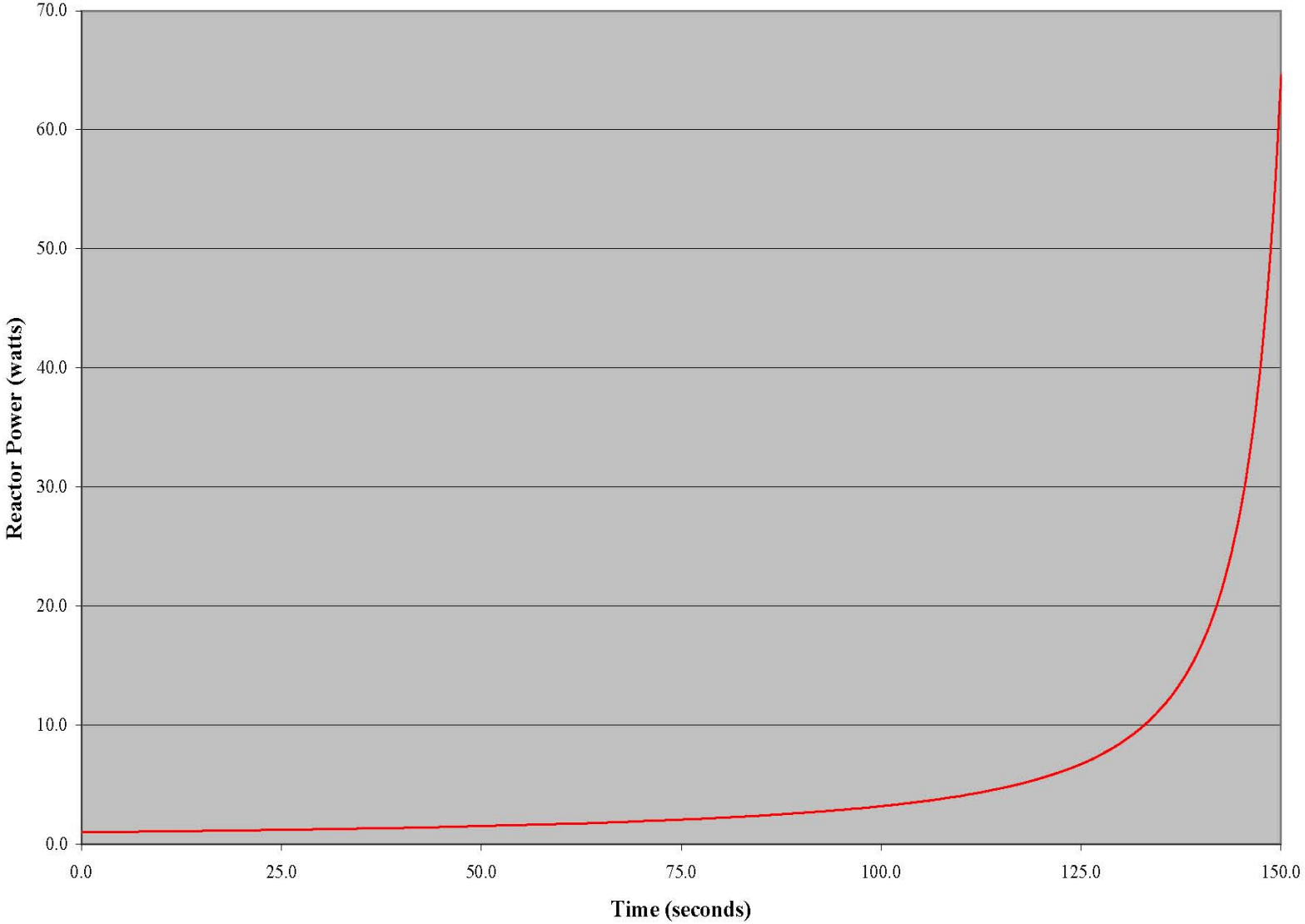


FIGURE 13.4 – REACTOR POWER VERSUS TIME FOR CONTINUOUS CONTROL BLADE WITHDRAWAL

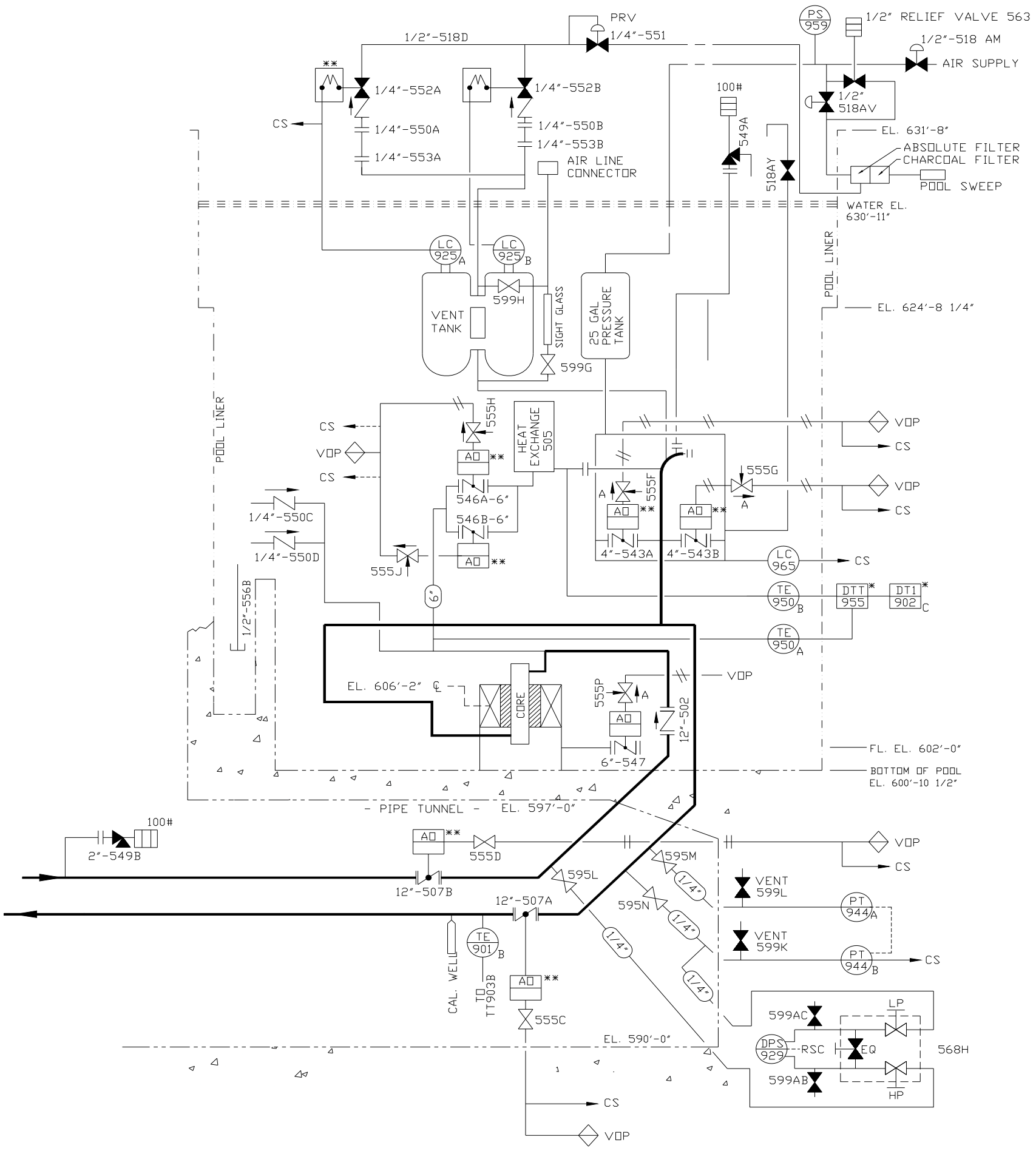


FIGURE 13.5 – SCHEMATIC OF IN-POOL PORTION OF PRIMARY COOLANT SYSTEM

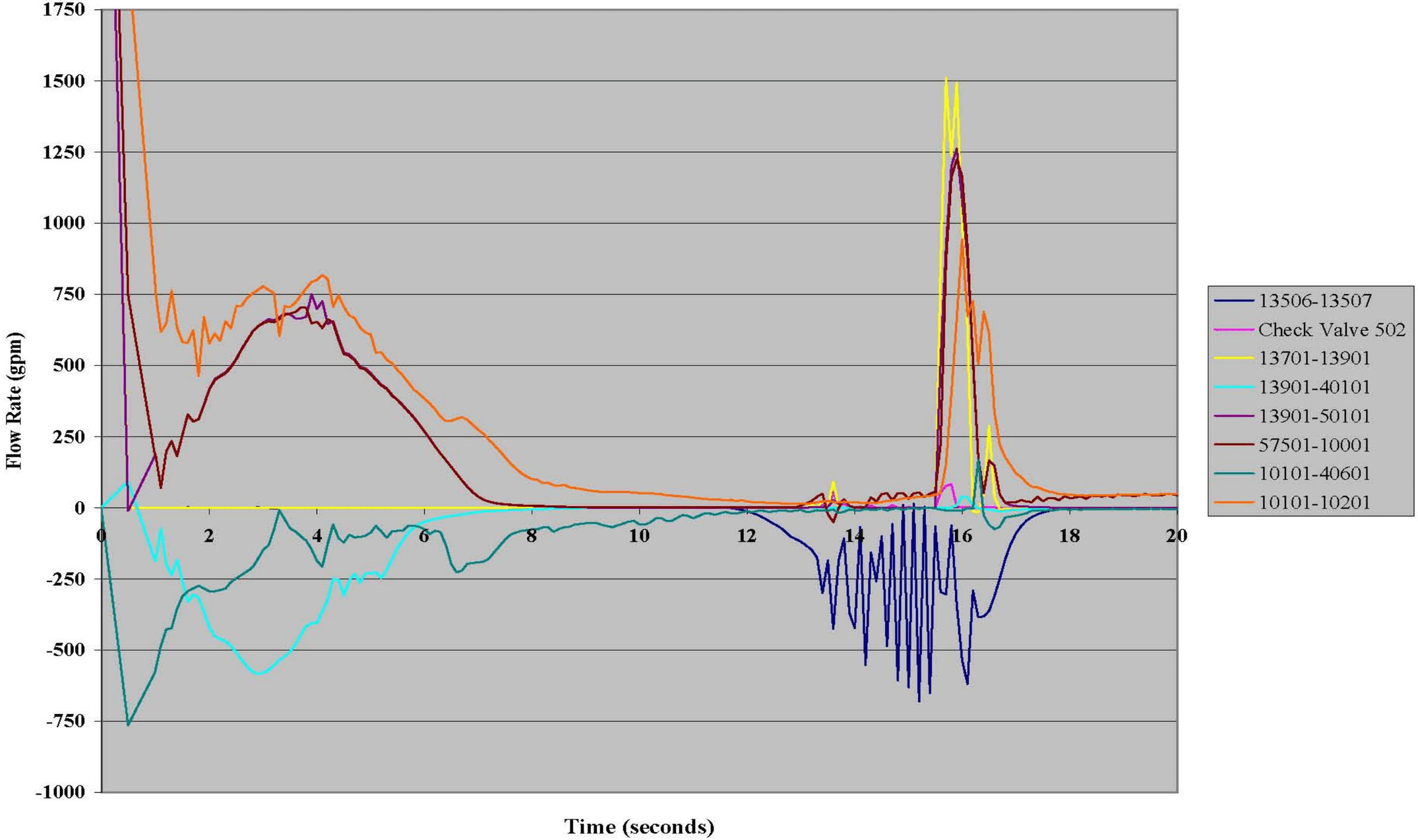


FIGURE 13.6 – JUNCTION FLOW RATES DURING THE FIRST 20 SECONDS OF THE COLD-LEG LOCA

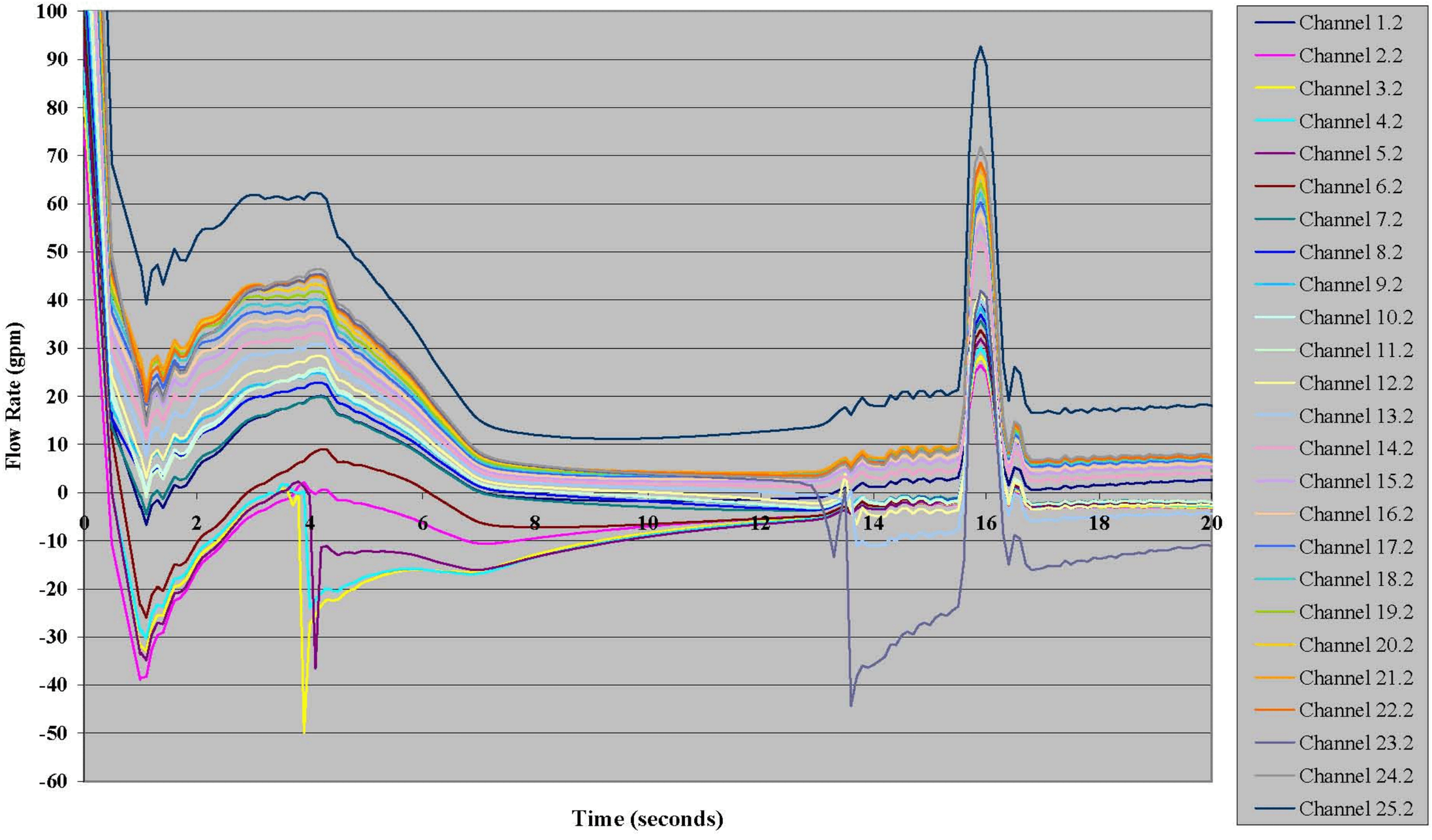


FIGURE 13.7 – COOLANT FLOW THROUGH THE 25 INDIVIDUAL CHANNELS DURING THE FIRST 20 SECONDS OF THE COLD-LEG LOCA

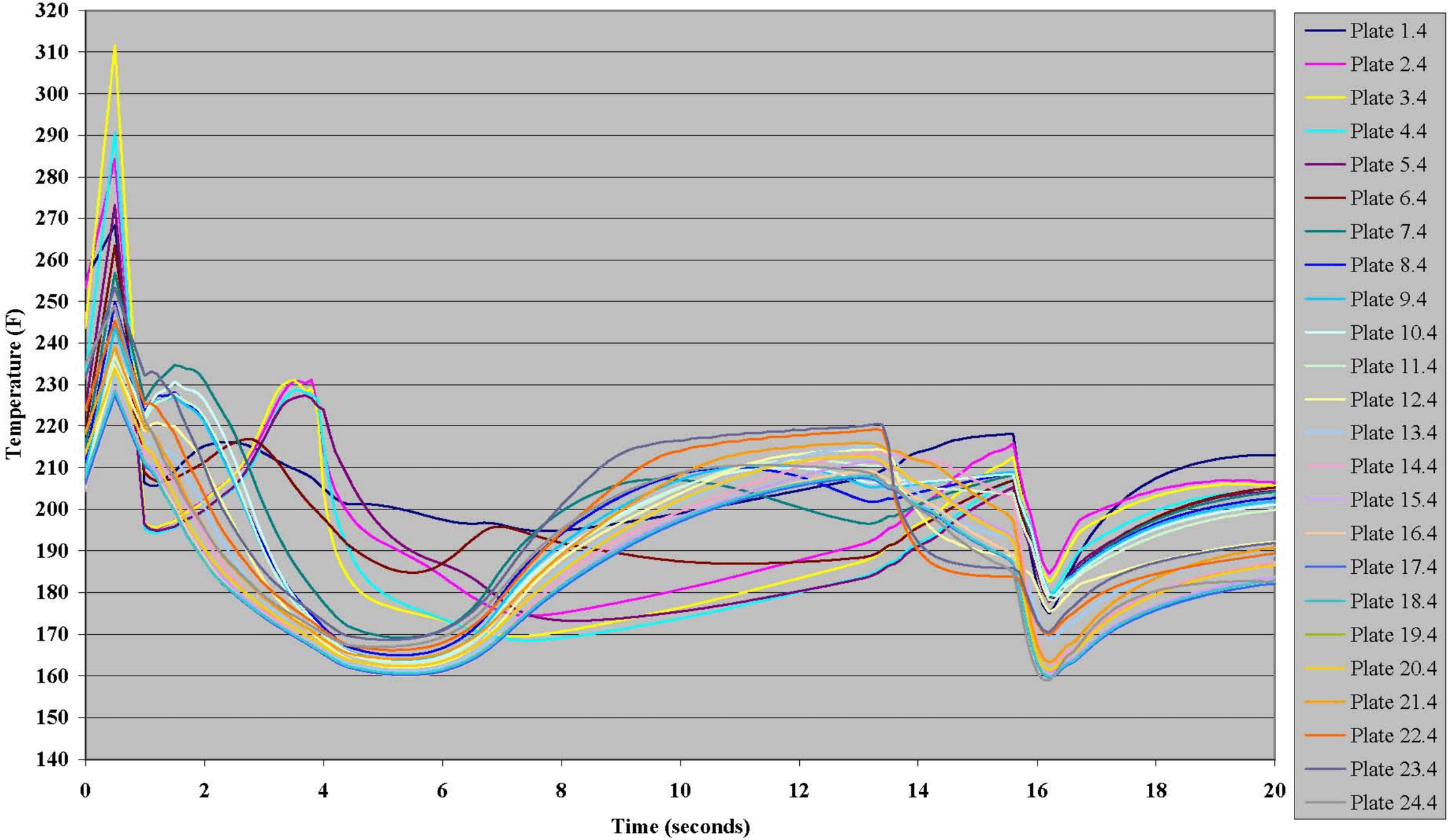


FIGURE 13.8 – CENTERLINE TEMPERATURE OF THE 24 FUEL PLATES (SECTION 4) DURING THE FIRST 20 SECONDS OF THE COLD-LEG LOCA

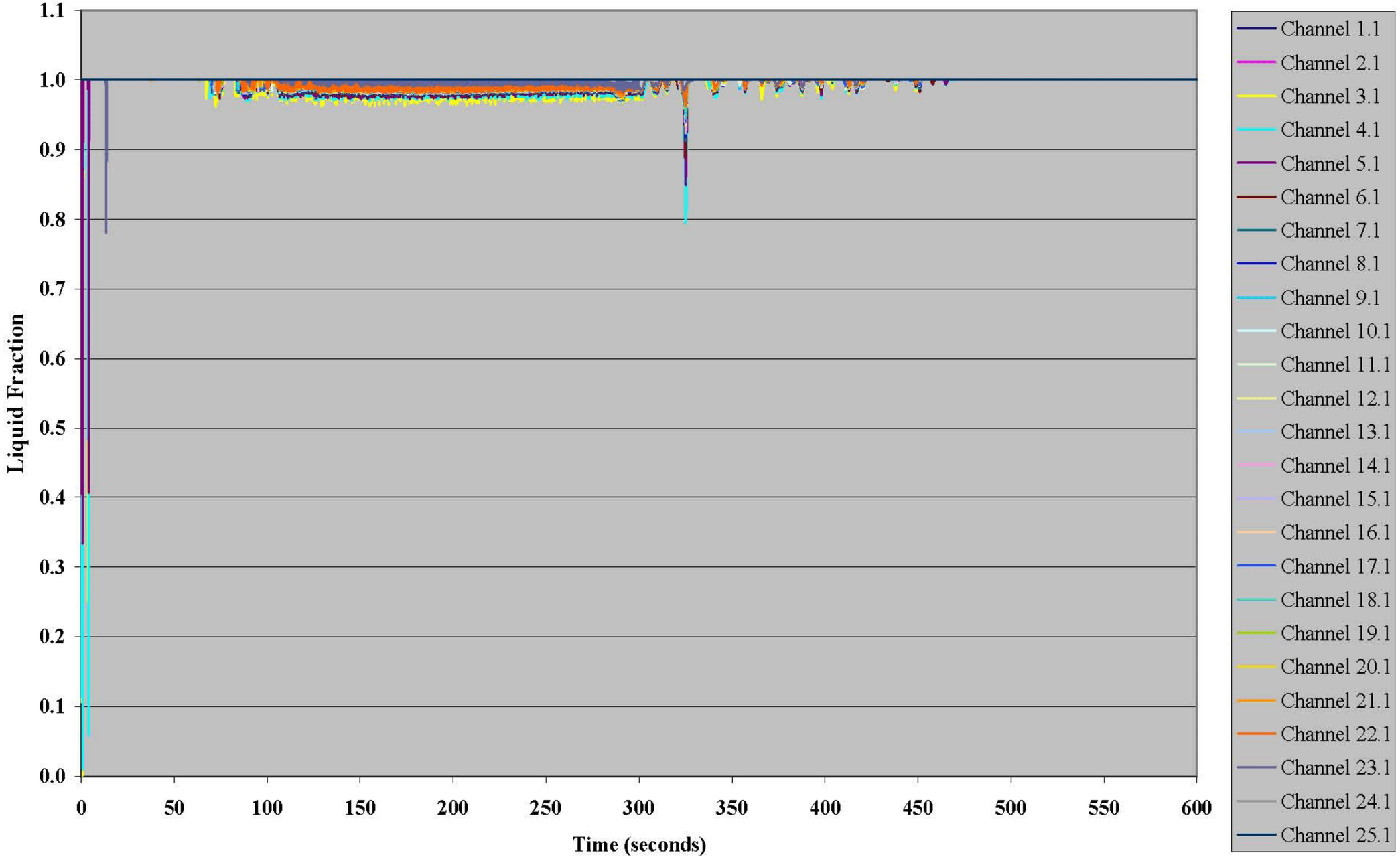


FIGURE 13.9 – LIQUID FRACTION OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 1) DURING THE FIRST 600 SECONDS OF THE COLD-LEG LOCA

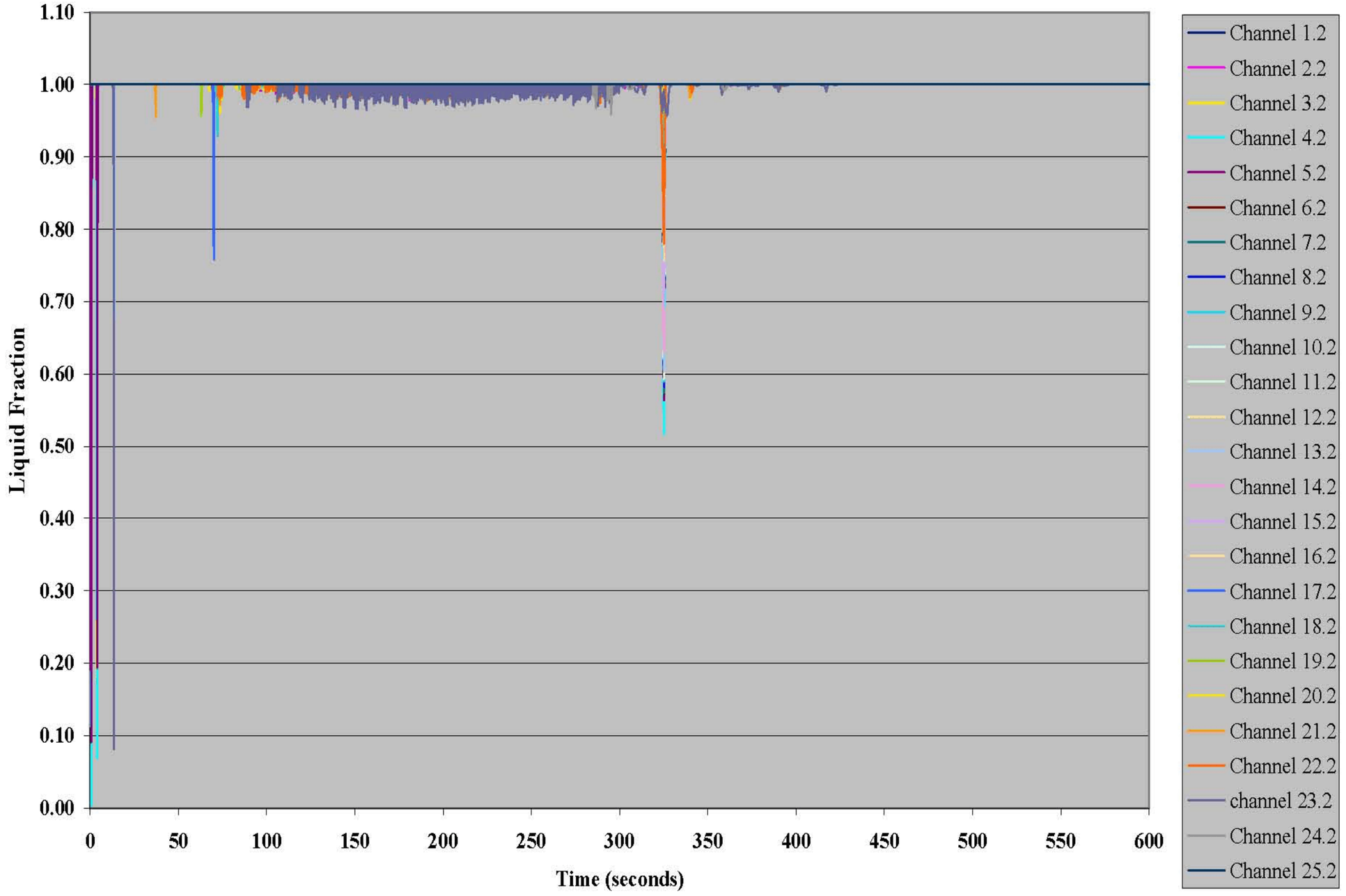


FIGURE 13.10 – LIQUID FRACTION OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 2) DURING THE FIRST 600 SECONDS OF THE COLD-LEG LOCA

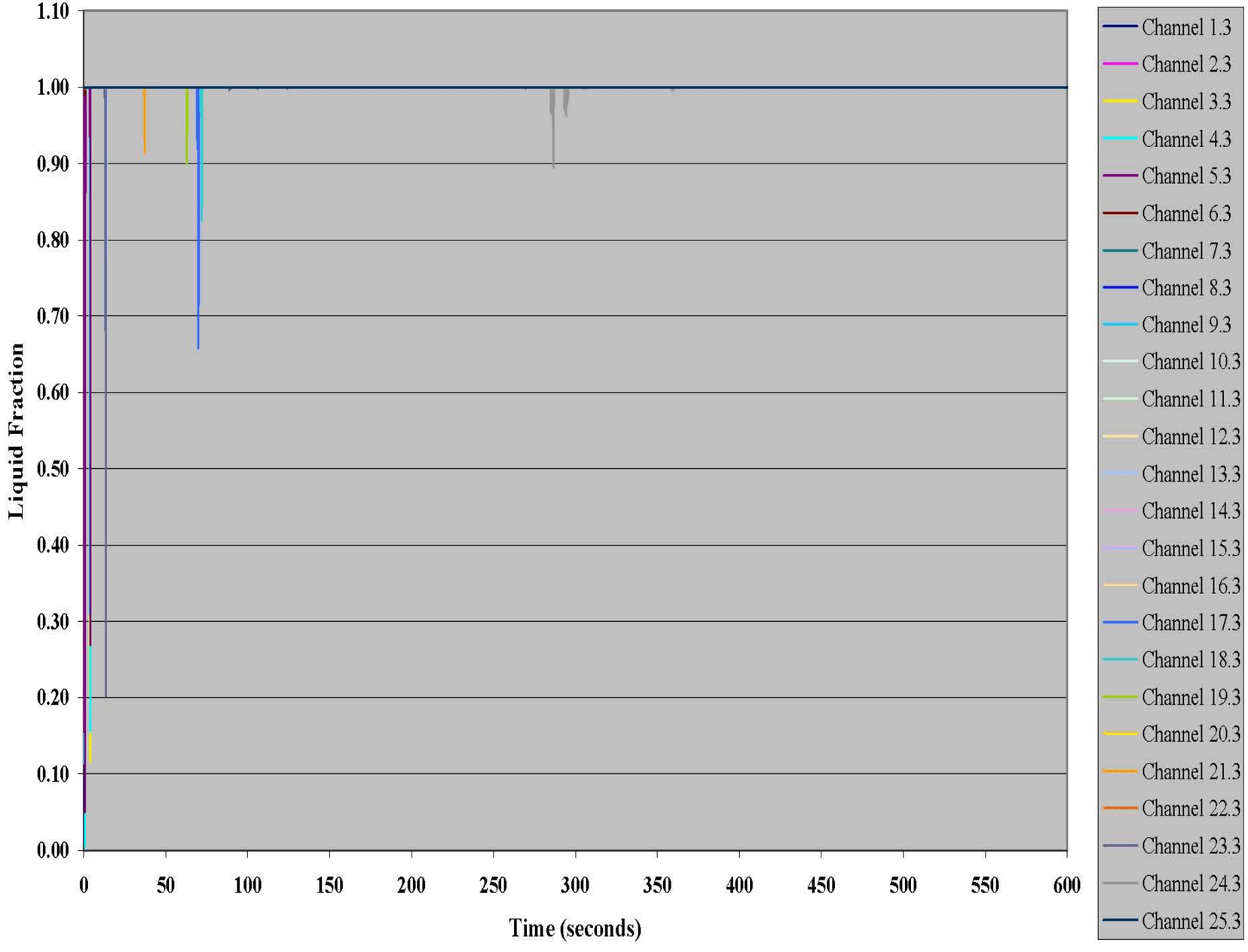


FIGURE 13.11 – LIQUID FRACTION OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 3) DURING THE FIRST 600 SECONDS OF THE COLD-LEG LOCA

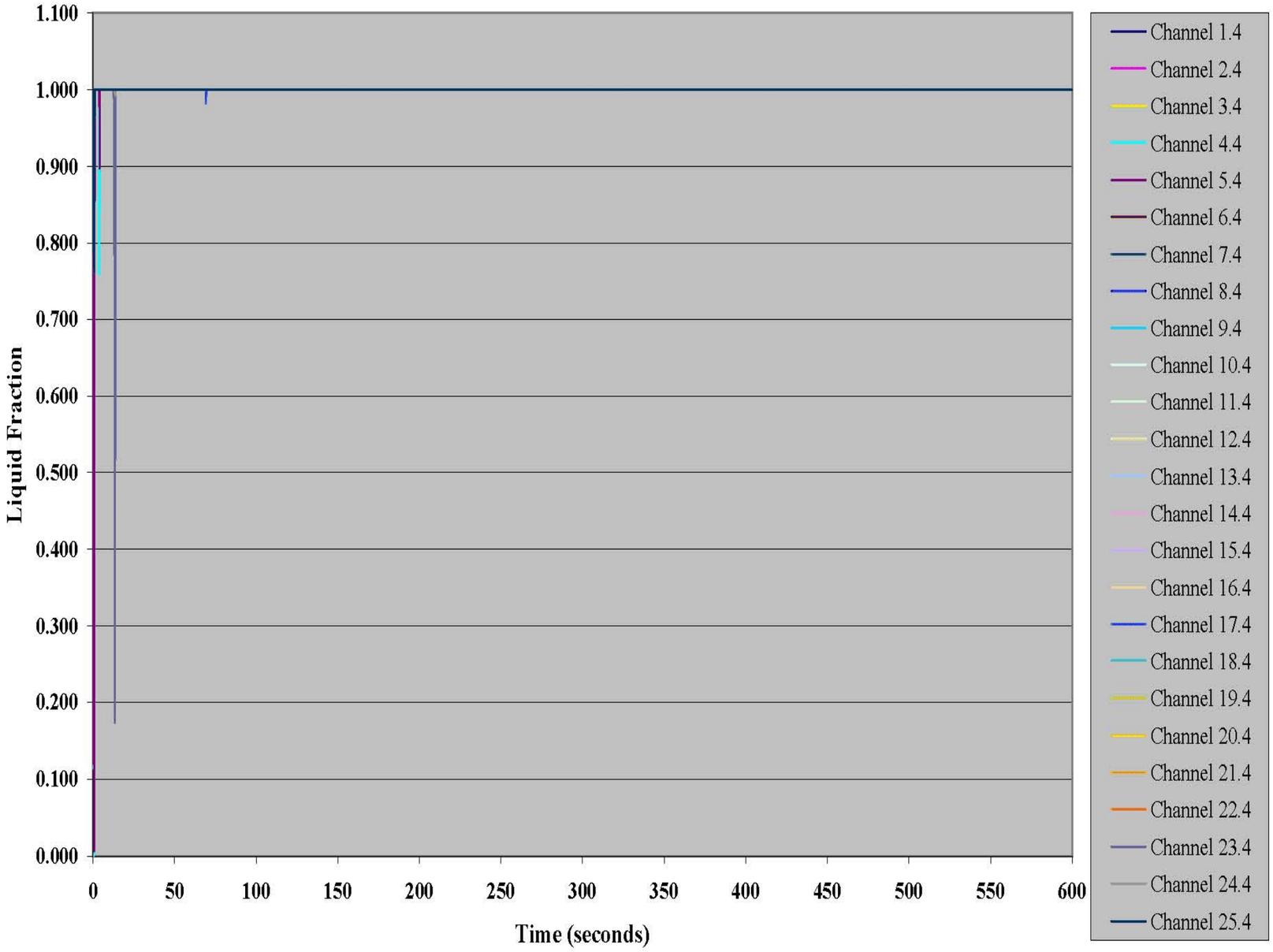


FIGURE 13.12 – LIQUID FRACTION OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 4) DURING THE FIRST 600 SECONDS OF THE COLD-LEG LOCA

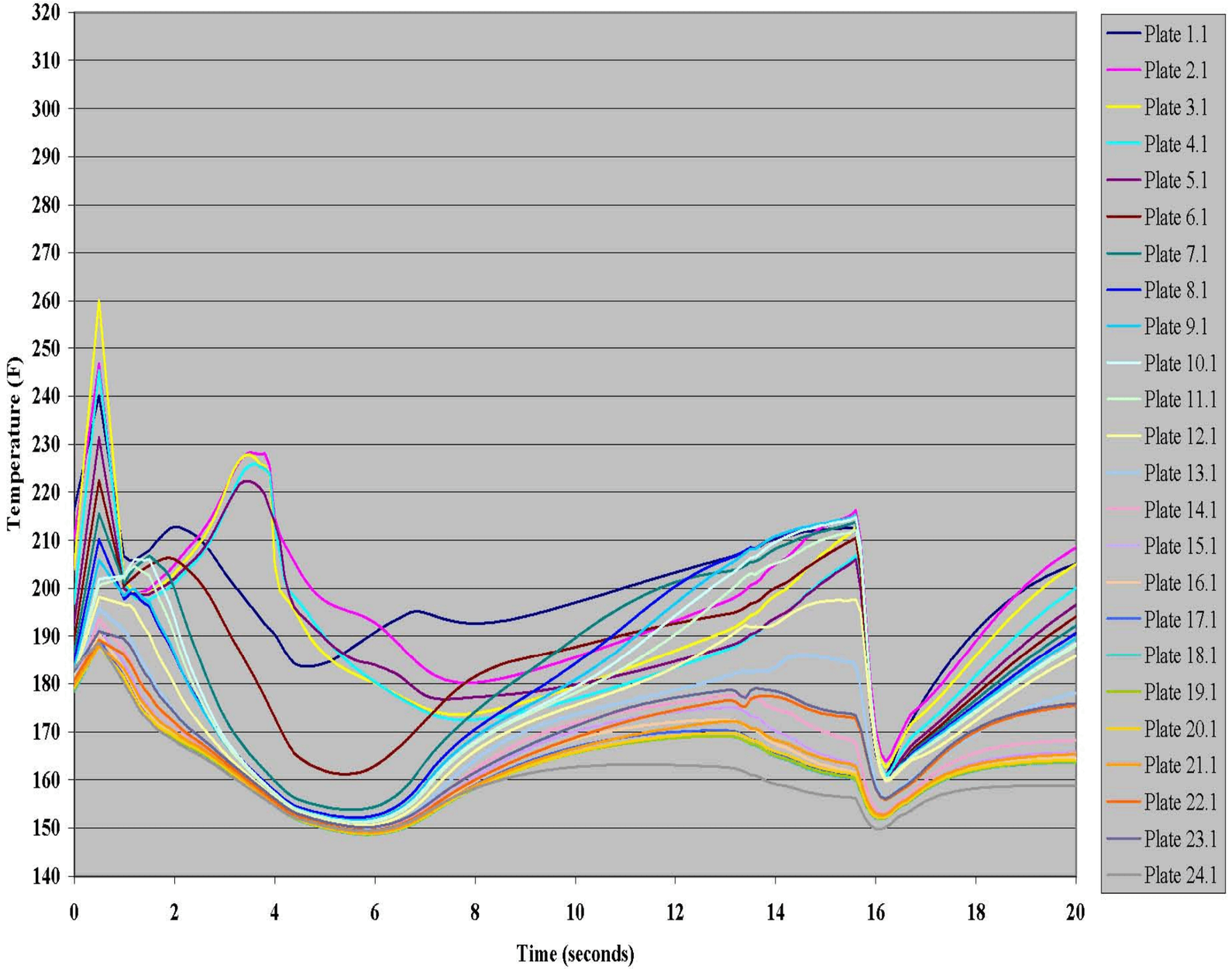


FIGURE 13.13 – CENTERLINE TEMPERATURE OF THE 24 FUEL PLATES (SECTION 1) DURING THE FIRST 20 SECONDS OF THE COLD-LEG LOCA

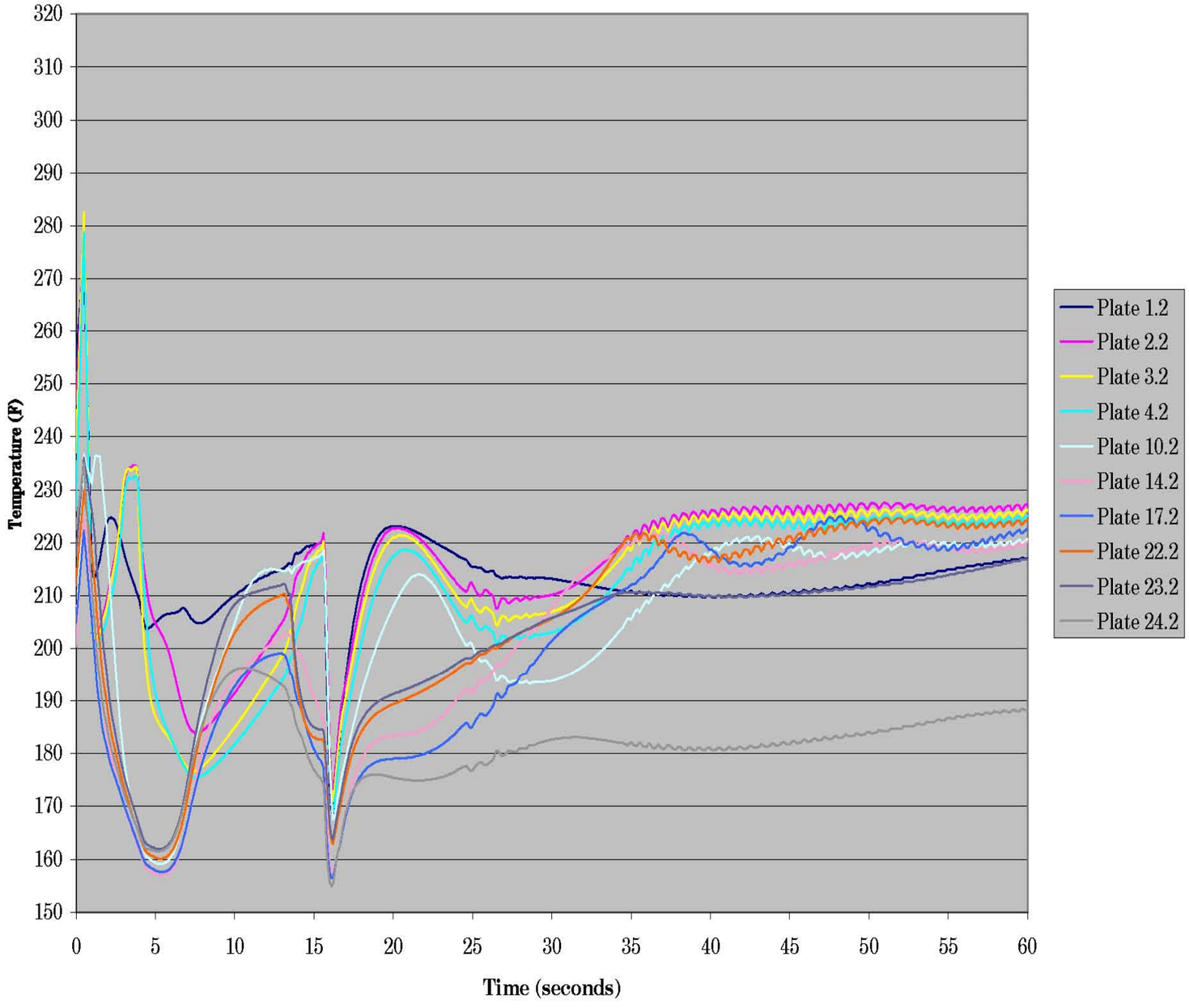


FIGURE 13.14 – CENTERLINE TEMPERATURE OF THE 24 FUEL PLATES (SECTION 2) DURING THE FIRST 20 SECONDS OF THE COLD-LEG LOCA

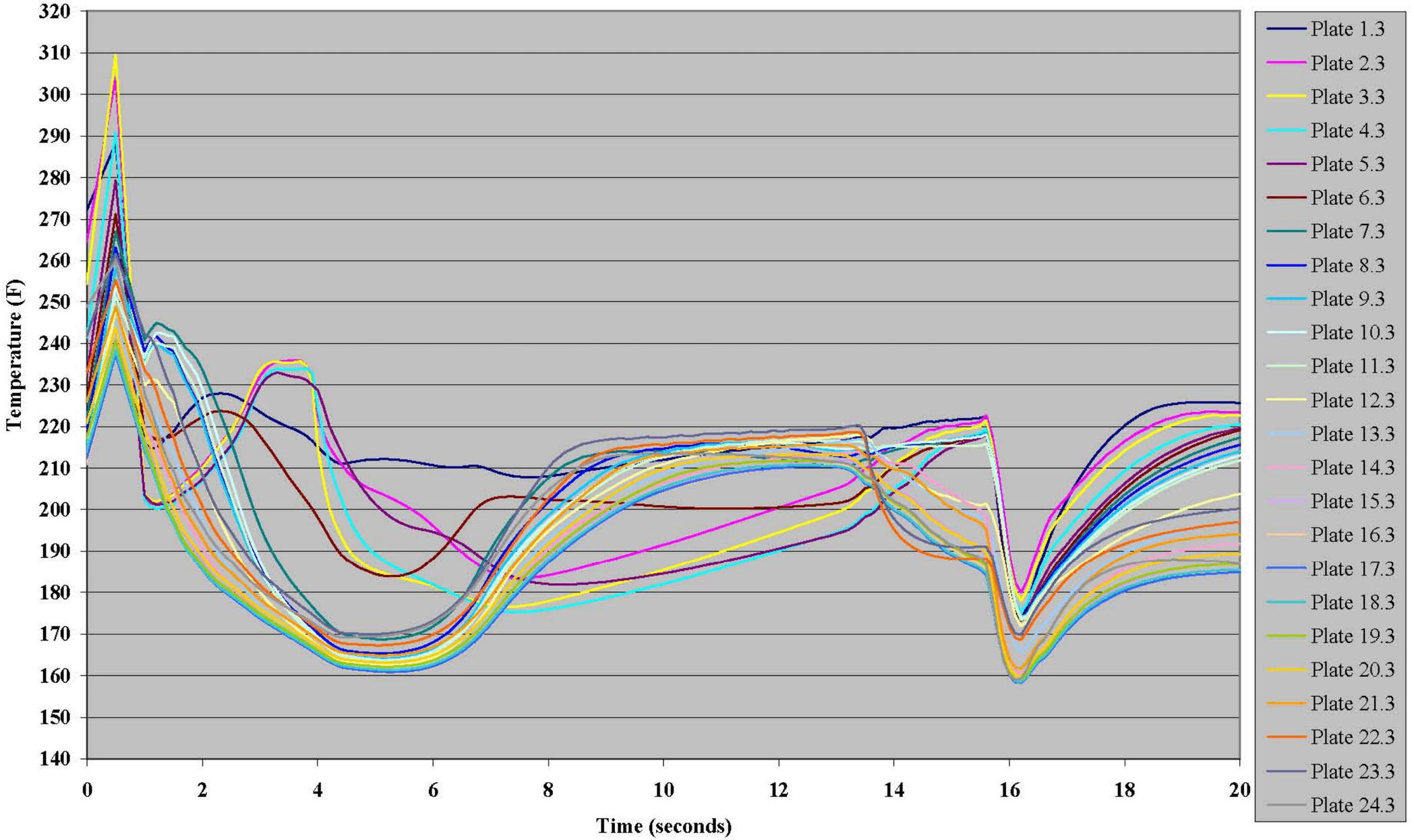


FIGURE 13.15 – CENTERLINE TEMPERATURE OF THE 24 FUEL PLATES (SECTION 3) DURING THE FIRST 20 SECONDS OF THE COLD-LEG LOCA

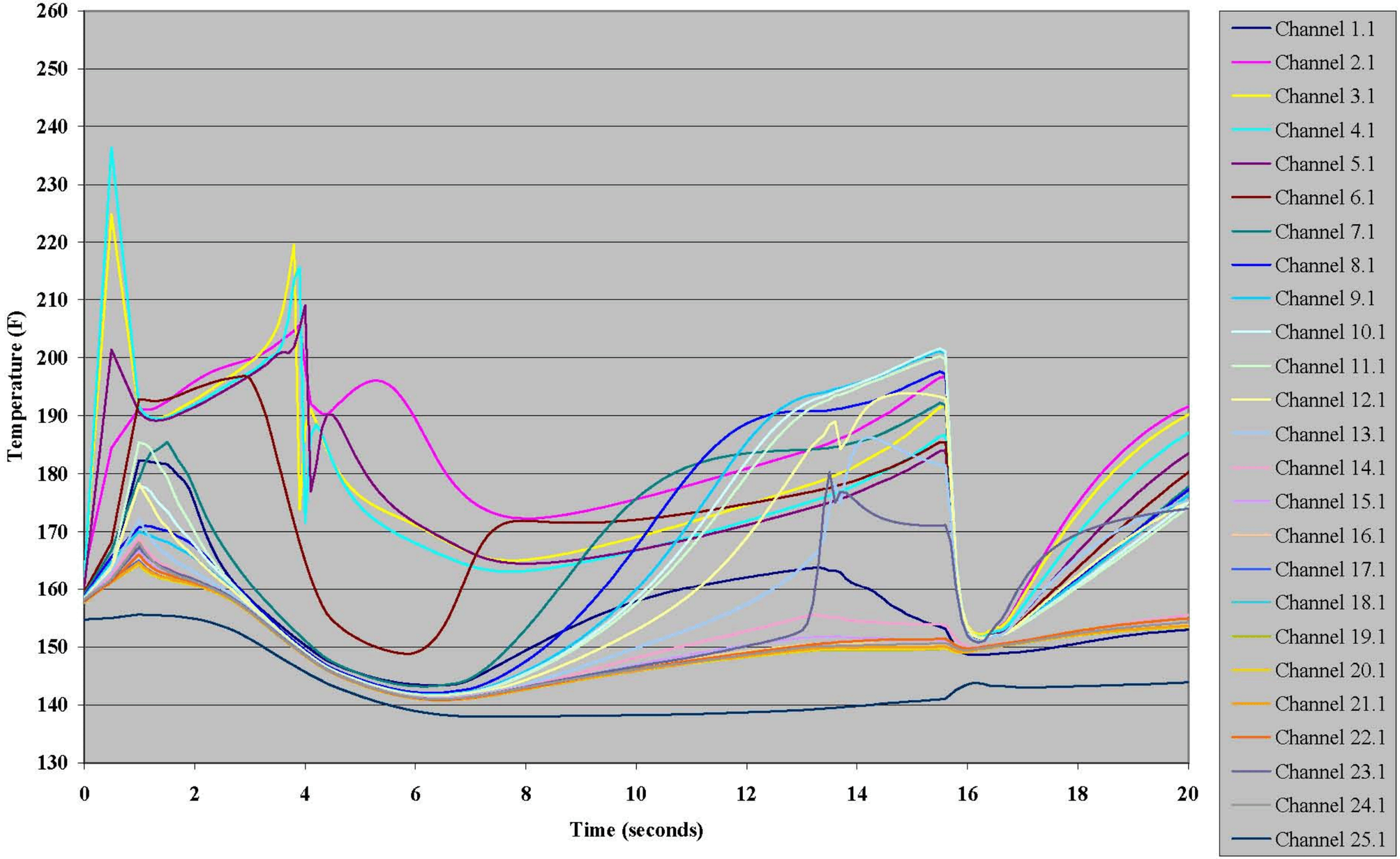


FIGURE 13.16 – TEMPERATURE OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 1) DURING THE FIRST 20 SECONDS OF THE COLD-LEG LOCA

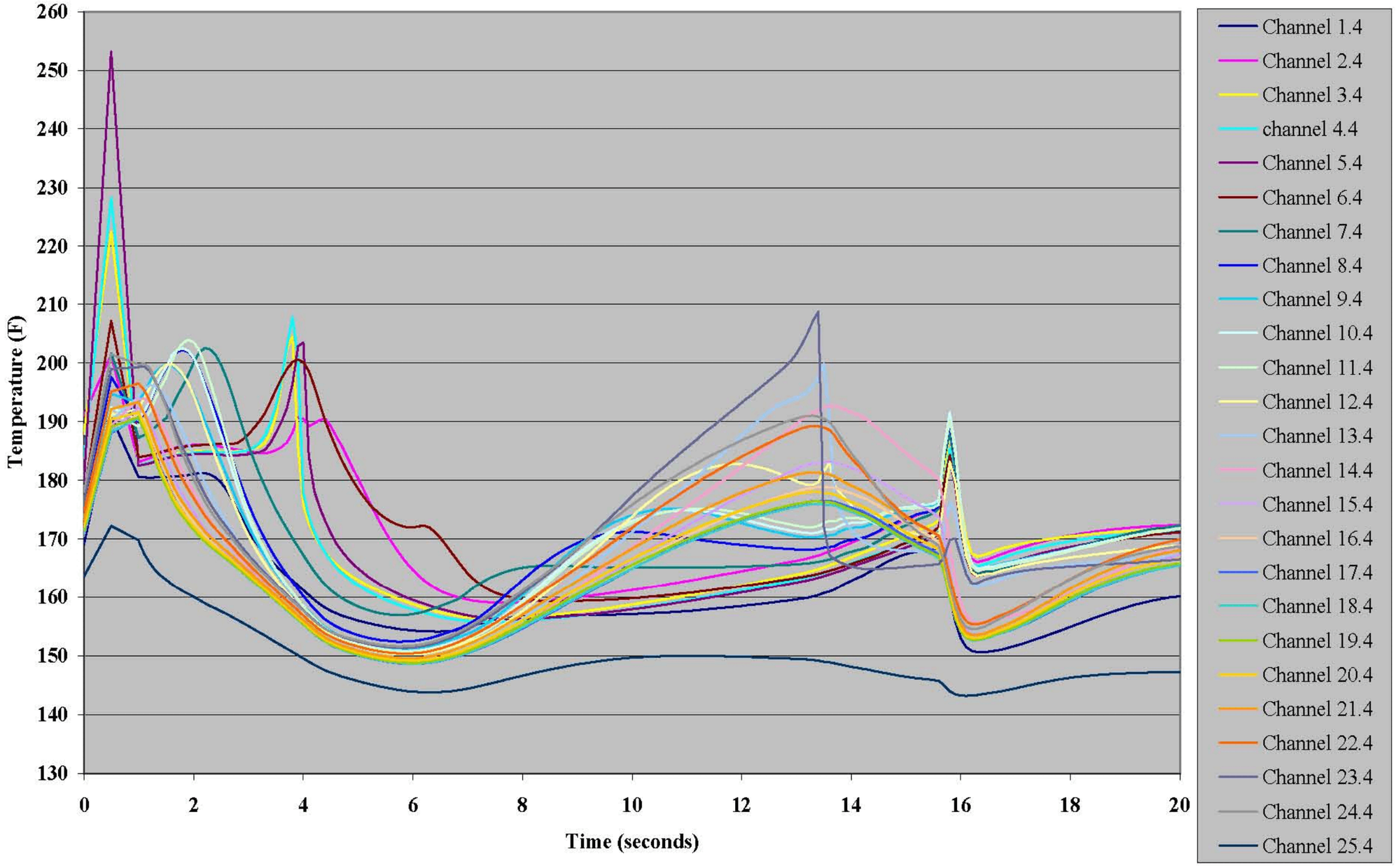


FIGURE 13.17 – TEMPERATURE OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 4) DURING THE FIRST 20 SECONDS OF THE COLD-LEG LOCA

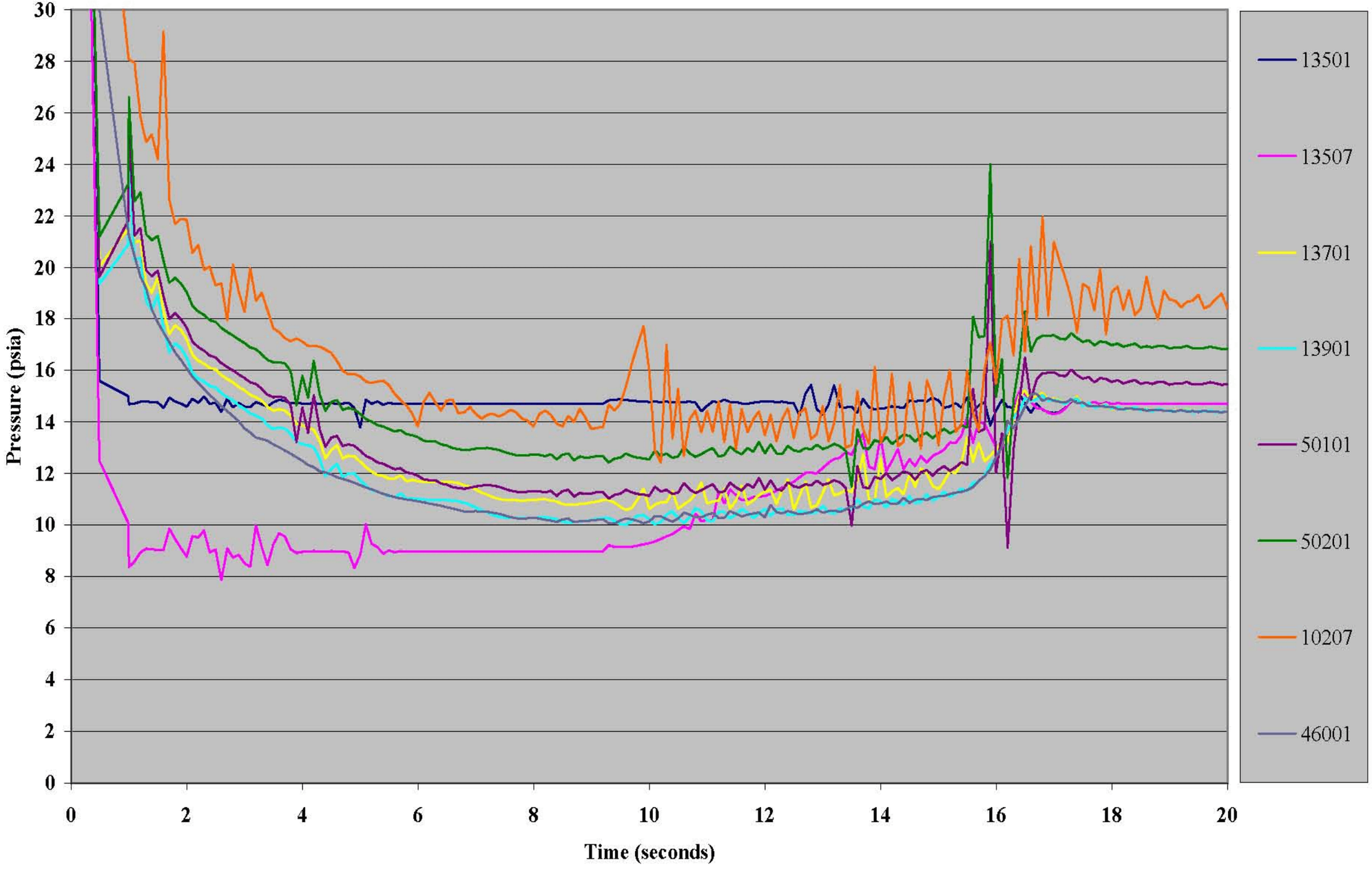


FIGURE 13.18 – VOLUME PRESSURES DURING THE FIRST 20 SECONDS OF THE COLD-LEG LOCA

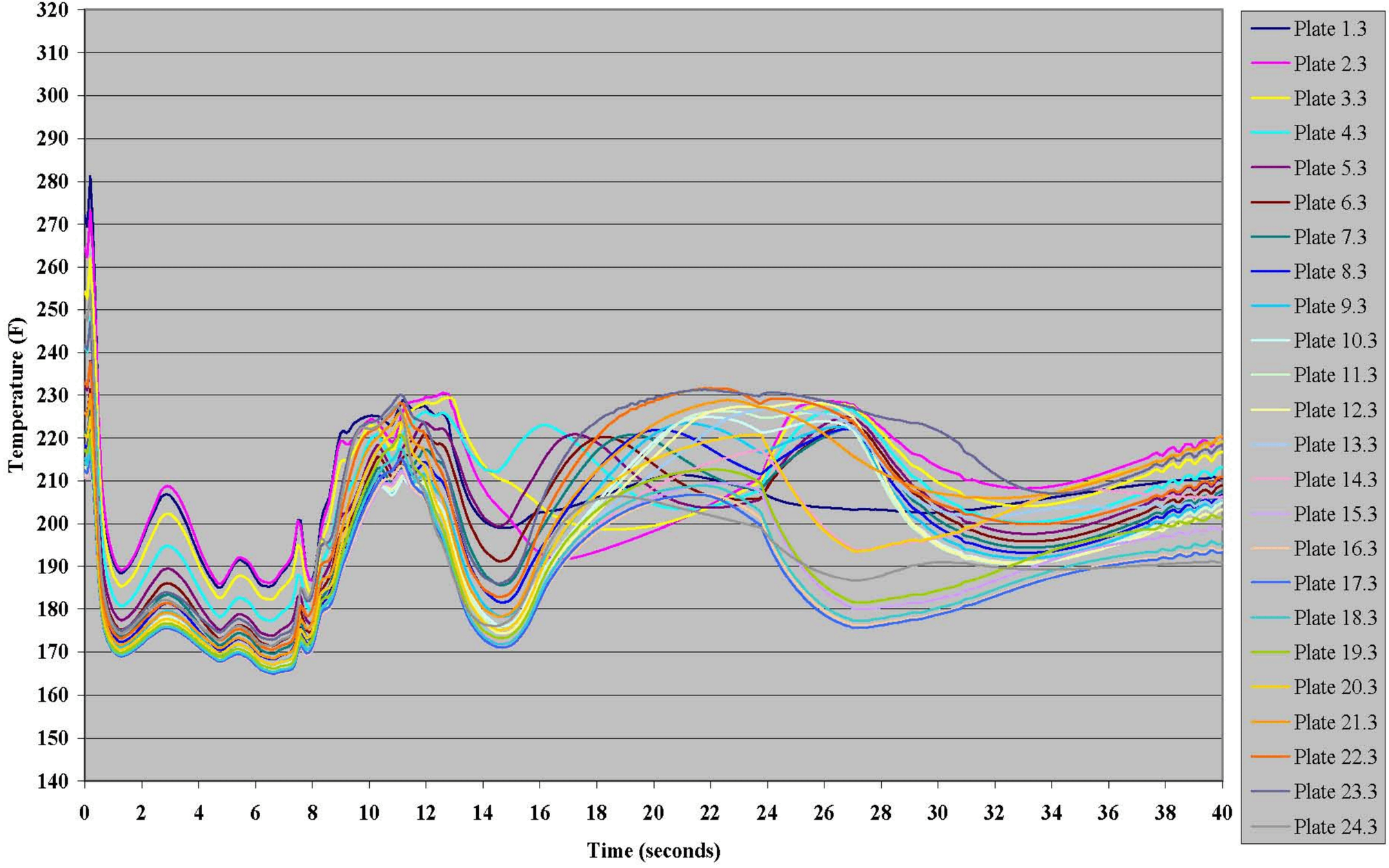


FIGURE 13.19 – CENTERLINE TEMPERATURE OF THE 24 FUEL PLATES (SECTION 3)DURING THE FIRST 40 SECONDS OF THE HOT-LEG LOCA

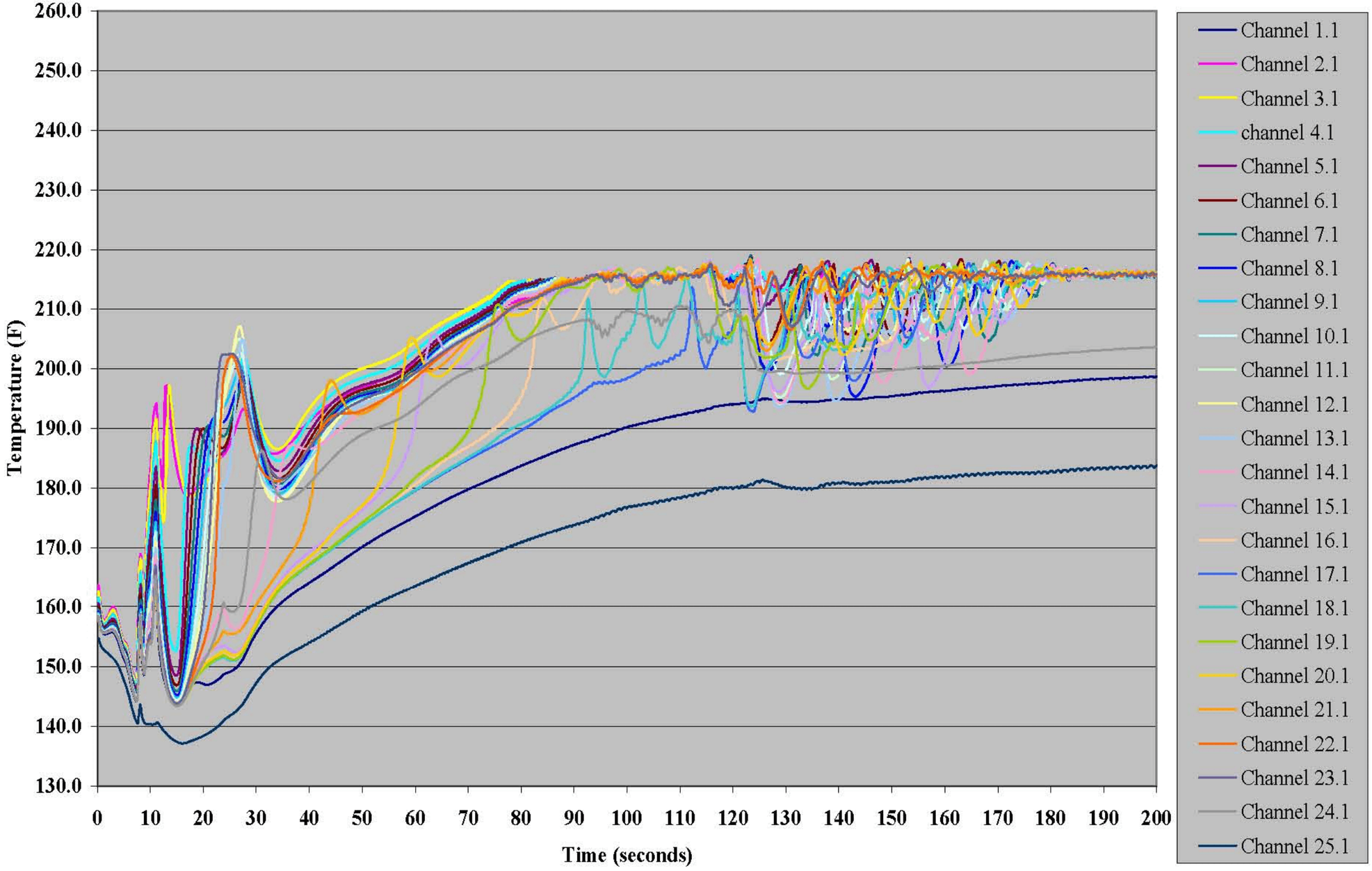


FIGURE 13.20 – TEMPERATURE OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 1) DURING THE FIRST 200 SECONDS OF THE HOT-LEG LOCA

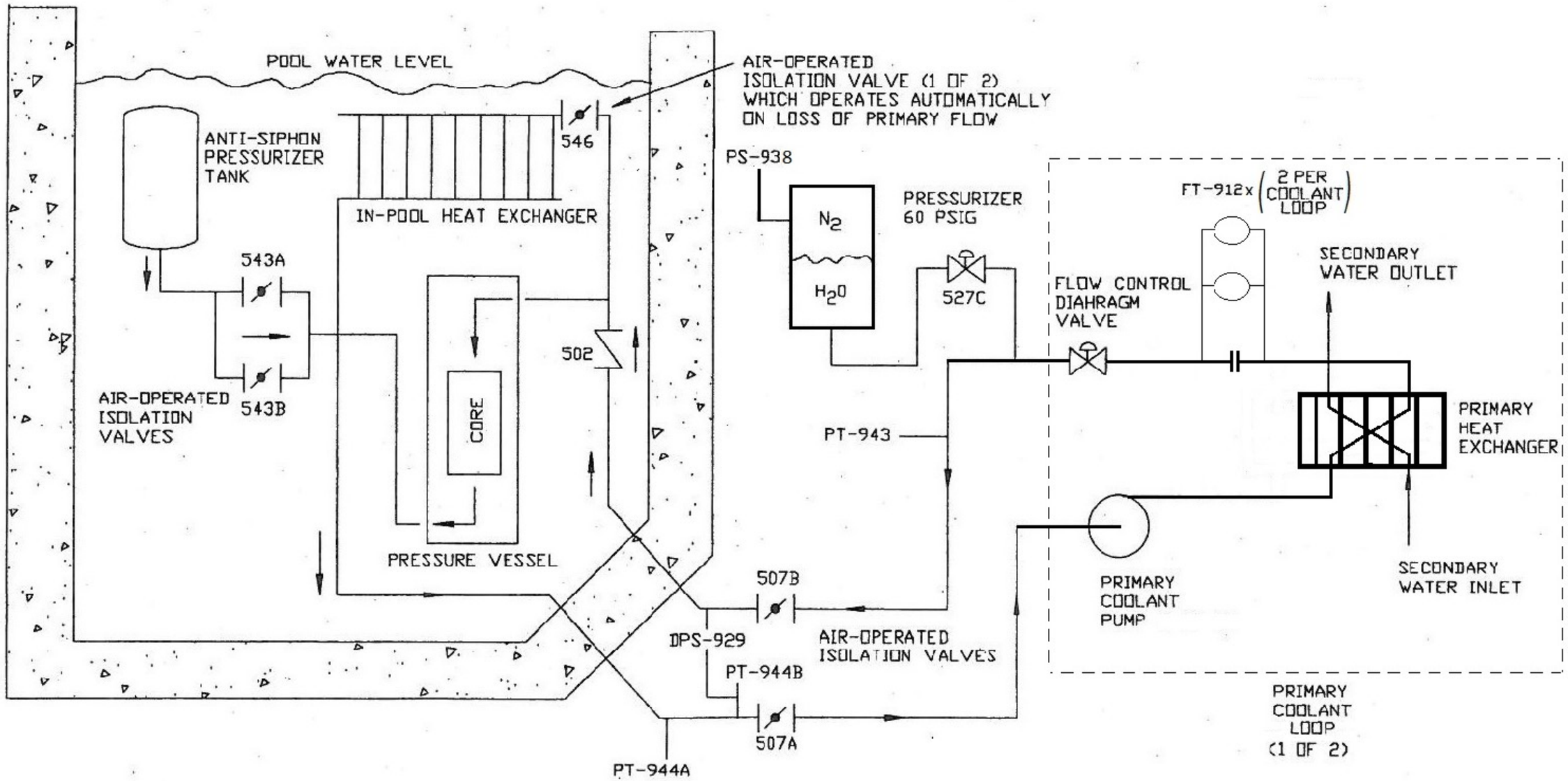


FIGURE 13.21 – SIMPLIFIED SCHEMATIC OF THE PRIMARY COOLANT SYSTEM

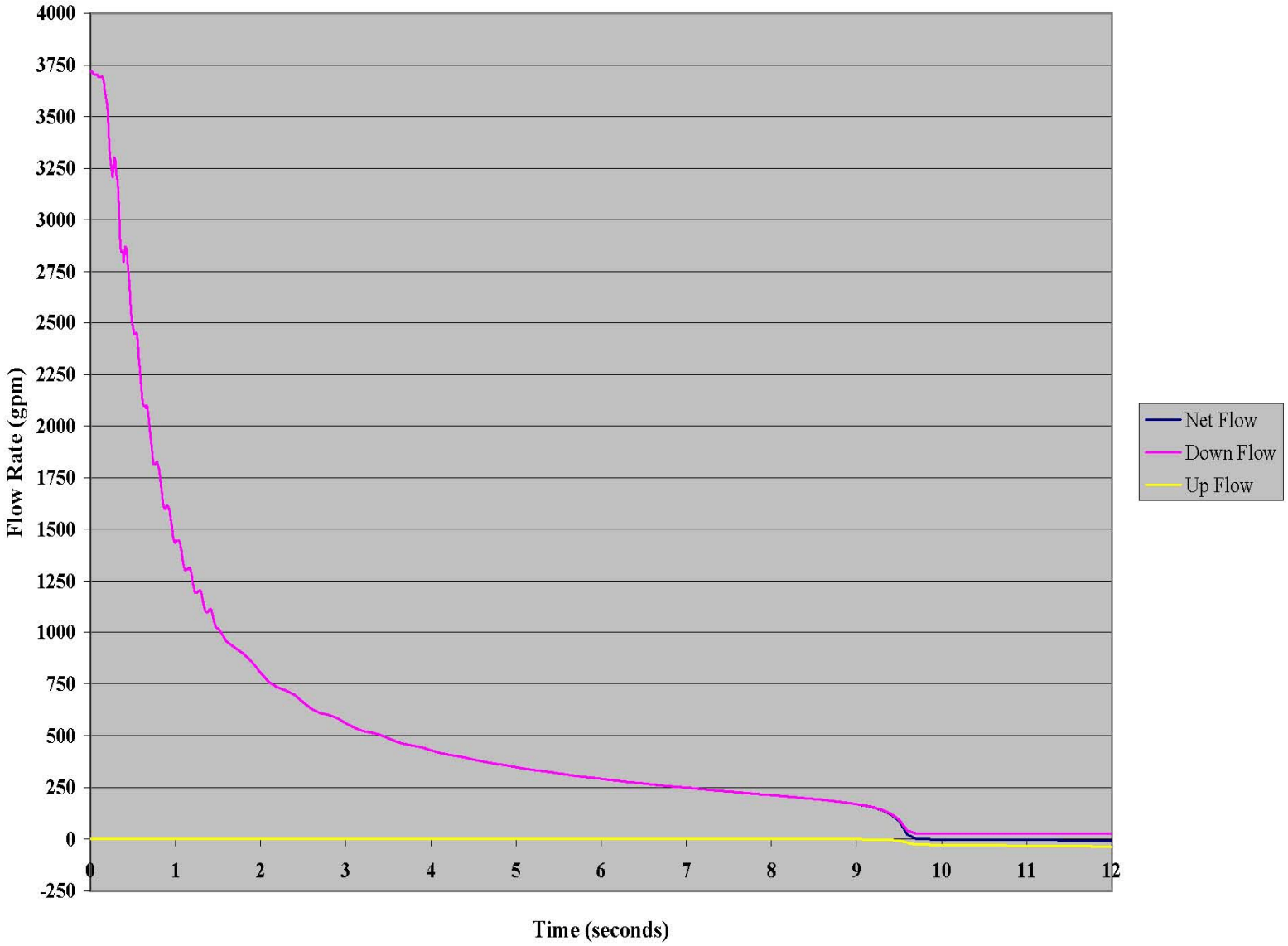


FIGURE 13.22 – DOWNWARD, UPWARD, AND NET COOLANT FLOW THROUGH THE CORE DURING THE FIRST 12 SECONDS OF THE LOF ACCIDENT

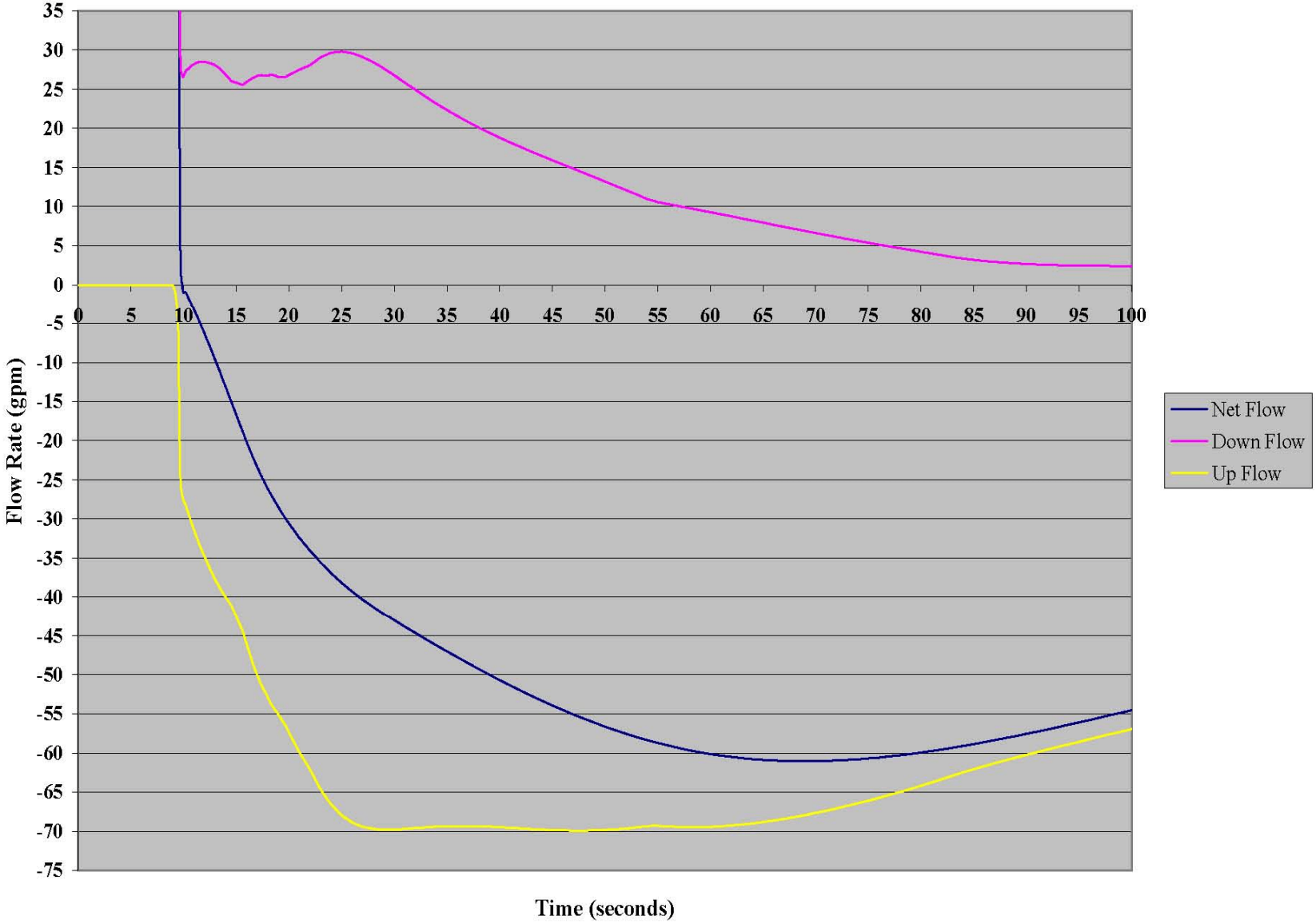


FIGURE 13.23 – DOWNWARD, UPWARD, AND NET COOLANT FLOW THROUGH THE CORE DURING THE FIRST 100 SECONDS OF THE LOF ACCIDENT

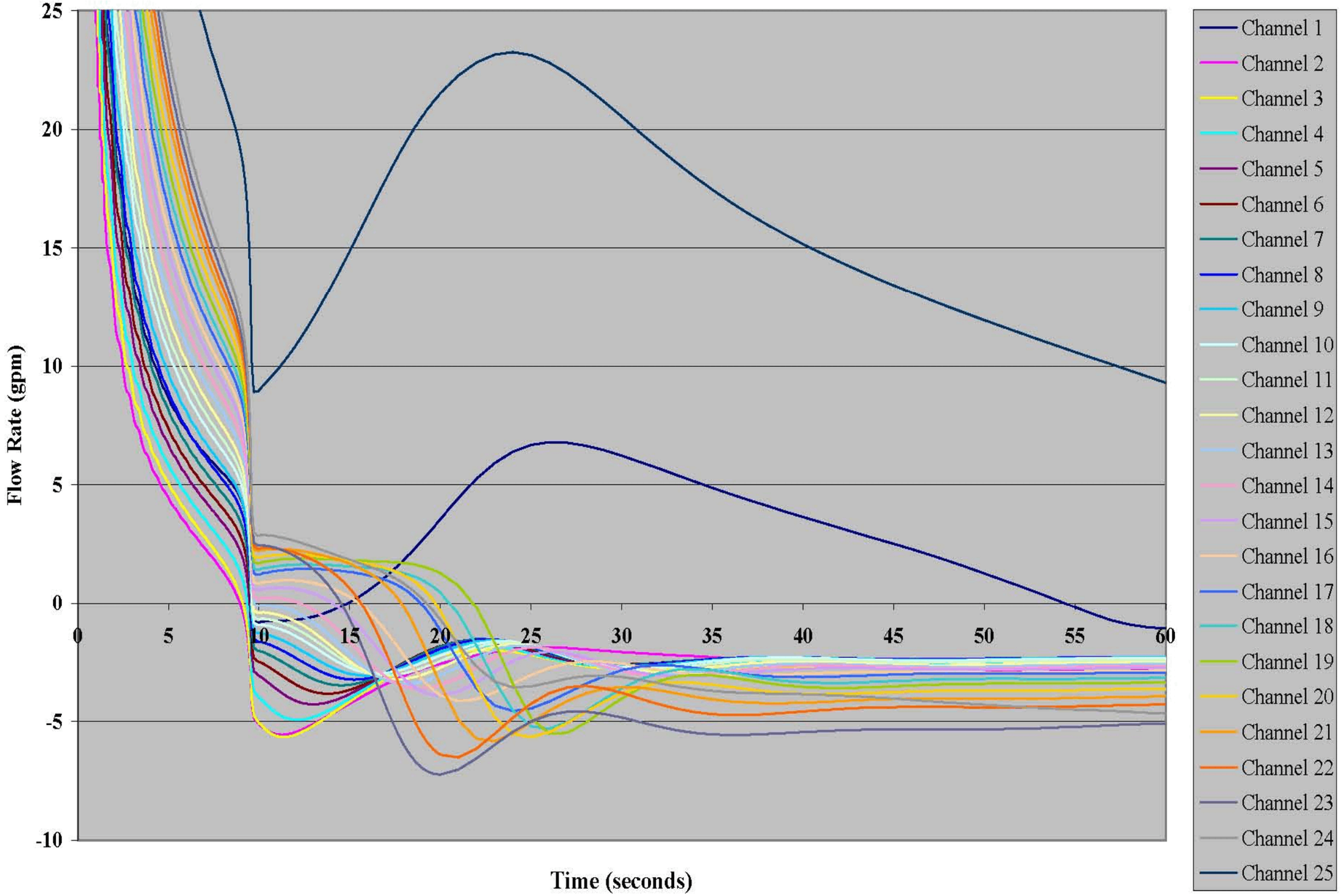


FIGURE 13.24 – COOLANT FLOW THROUGH THE 25 INDIVIDUAL COOLANT CHANNELS DURING THE FIRST 60 SECONDS OF THE LOF ACCIDENT

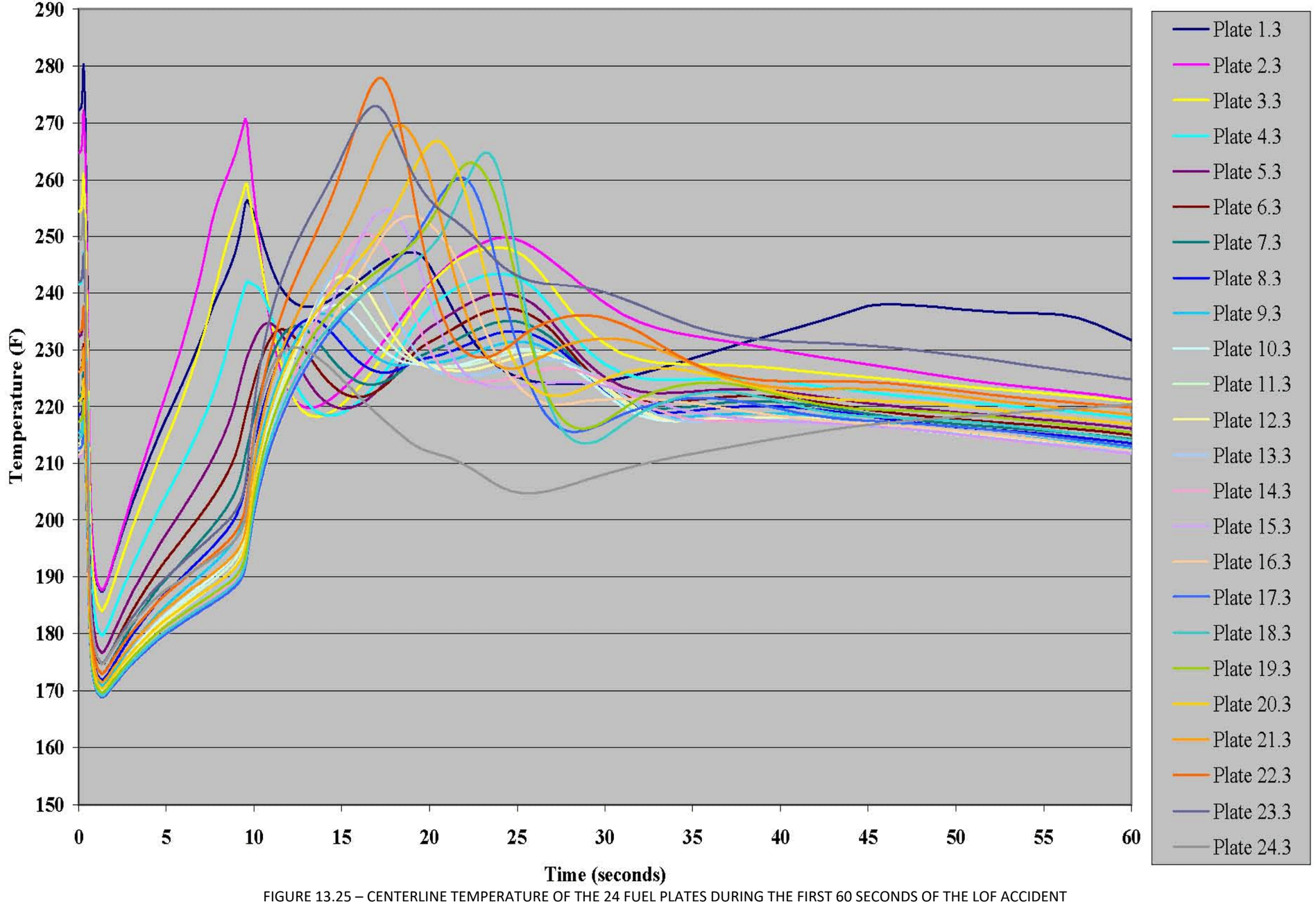


FIGURE 13.25 – CENTERLINE TEMPERATURE OF THE 24 FUEL PLATES DURING THE FIRST 60 SECONDS OF THE LOF ACCIDENT

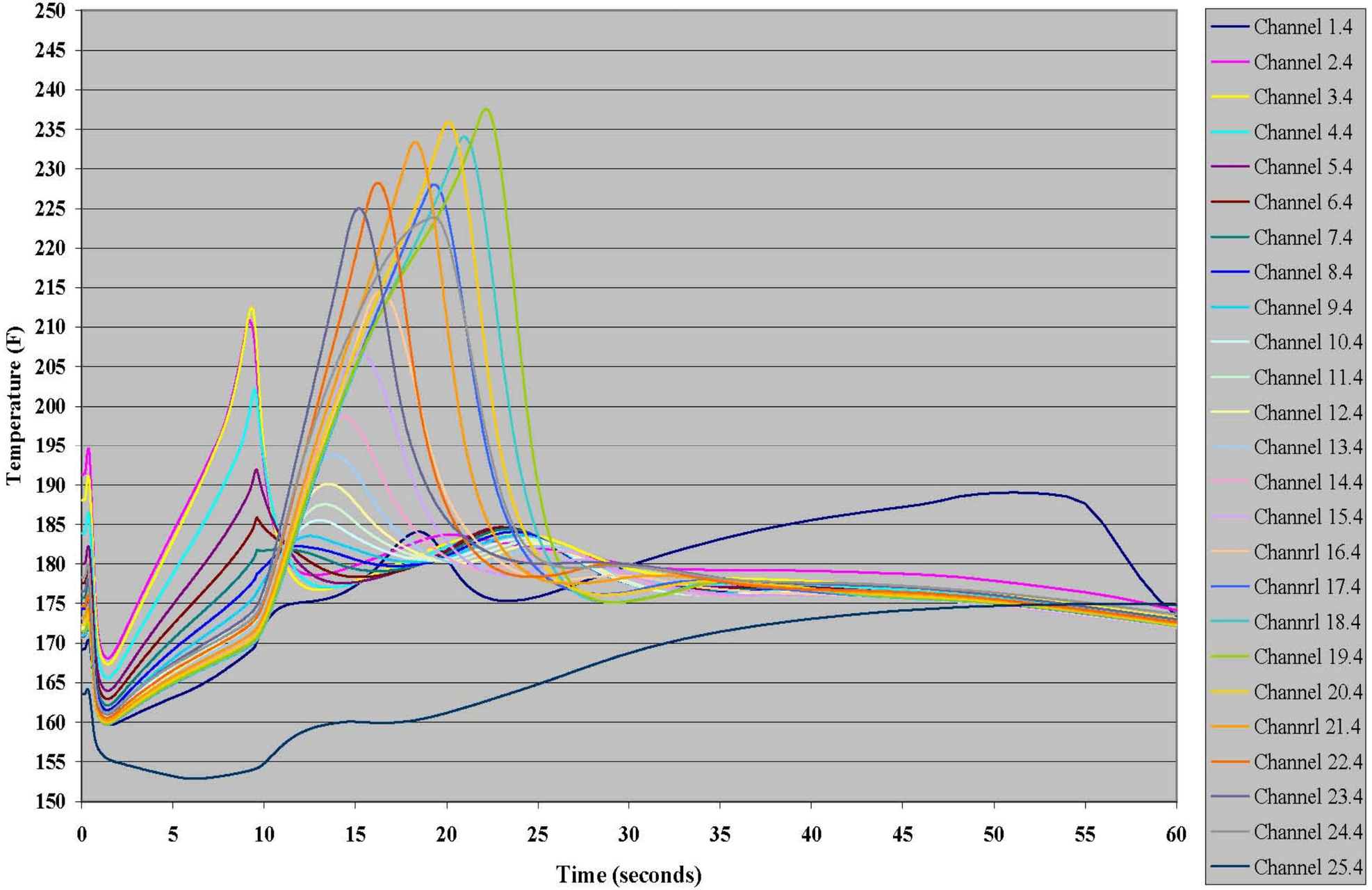


FIGURE 13.26 – TEMPERATURE OF THE 25 INDIVIDUAL COOLANT CHANNELS DURING THE FIRST 60 SECONDS OF THE LOF ACCIDENT

CHAPTER 14

TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

14.0 TECHNICAL SPECIFICATIONS 2
14.1 Introduction 2
14.2 Format and Content 2
14.2.1 Definitions 2
14.2.2 Safety Limits and Limiting Safety System Settings 2
14.2.2.1 Safety Limits 2
14.2.2.2 Limiting Safety System Settings 3
14.2.3 Limiting Conditions for Operations 3
14.2.4 Surveillance Requirements 3
14.2.5 Design Features 4
14.2.6 Administrative Controls 4

REFERENCES

14.1 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," ANS, LaGrange Park, Illinois, 1990.

14.2 Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, NUREG-1537, February 1996.

14.3 Safety Evaluation Report Related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-Power Reactors, NUREG-1313, July 1988.

14.0 TECHNICAL SPECIFICATIONS

This chapter discusses the development, format, and contents of the reactor facility Technical Specifications (TSs).

14.1 Introduction

The TSs represent an agreement between the licensee and the U.S. Nuclear Regulatory Commission (NRC) on administrative controls, equipment availability, operational conditions and limits, and other requirements imposed on reactor facility operation in order to protect the environment and the health and safety of the facility staff and the general public in accordance with 10 CFR 50.36.

Specific limitations and equipment requirements for safe reactor operation and for dealing with abnormal situations are called specifications. These specifications, typically derived from the facility descriptions and safety considerations contained in this document, represent a comprehensive envelope of safe operation. Only those operational parameters and equipment requirements directly related to preserving that safe envelope are listed in the TSs. Procedures or actions employed to meet the requirements of these TSs are not included in the TSs. Normal operation of the reactor within the limits of the TSs will not result in off-site radiation exposure in excess of 10 CFR 20 guidelines.

14.2 Format and Content

The format and content of the University of Missouri Research Reactor (MURR) TSs that were submitted as part of the application for license renewal follow the guidance of the 1990 revision to American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, “The Development of Technical Specifications for Research Reactors” (Reference 14.1). For areas where Reference 14.1 might require modification or clarification in order to provide acceptable TS, NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” provides additional guidance (Reference 14.2).

14.2.1 Definitions

This section provides a uniform interpretation of terms and phrases used in Reference 14.1 and other associated standards. The applicable definitions listed in Reference 14.1 are typically stated verbatim. Modifications and additional definitions presented in Reference 14.2 have been used to clarify the meaning of terms used in ANSI/ANS 15.1. Definitions specific to MURR are included to clarify terms in the TSs.

14.2.2 Safety Limits and Limiting Safety System Settings

14.2.2.1 Safety Limits

All reactor licensees are required by 10 CFR 50.36(c) to specify safety limits in the TSs. These safety limits are limits on important process variables that are found to be necessary to reasonably protect the

integrity of the primary barrier that guards against the uncontrolled release of fission products from the reactor fuel. Important process variables are measurable parameters that individually or in combination reflect the physical condition of the primary barrier. Reference 14.1 provides a list of parameters that may be acceptable as process variables. However, Reference 14.2 states that not all safety limits for non-power reactors must be monitored and actually measurable. Safety limits could also be inferred from limitations on other process variables. For MURR, the primary barrier that protects against the uncontrolled release of fission products is the aluminum cladding of the fuel plates. Maintaining the integrity of the fuel cladding requires that the cladding remain below the blistering temperature of 986 °F (530 °C) (Reference 14.3). Since there is no mechanism at MURR to directly monitor fuel cladding temperature, the measurable parameters used to infer this safety limit include reactor power, primary coolant flow, reactor inlet water temperature, and pressurizer pressure.

The MURR safety limit analysis, as presented in Section 4.6.3, provides a four-dimensional safety 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 the departure from nucleate boiling (DNB).

14.2.2.2 Limiting Safety System Settings

For each measurable parameter used to infer the safety limit in the SAR has not been exceeded, a protective channel has been identified that prevents the value of the parameter, i.e. reactor power, primary coolant flow, reactor inlet water temperature, or pressurizer pressure, from exceeding the safety limit analysis. The calculated set point for this protective action is defined as the Limiting Safety System Setting (LSSS). The LSSS provides the minimum acceptable safety margin considering process uncertainty, the overall measurement uncertainty, and transient phenomena of the process instrumentation. The LSSS is chosen such that automatic protective action will terminate the most severe anticipated transient from reaching a safety limit. Because the LSSSs are analytical limits, the protective channels are typically set to actuate at more conservative values, thus providing greater operational flexibility. Section 4.6.4 provides the bases for the LSSSs for MURR.

14.2.3 Limiting Conditions for Operations

Limiting Conditions for Operations (LCOs) are those administratively established constraints on equipment and operational characteristics that shall be adhered to during operation of the facility. The LCOs are the lowest functional capability or minimum performance level which ensures that the reactor will not be damaged, that the reactor will be capable of performing its intended function, and that no one will suffer undue radiological exposures because of reactor operations.

14.2.4 Surveillance Requirements

Surveillance Requirements prescribe the frequency and scope of surveillance to demonstrate the functional capability or performance level corresponding with that specified by the Limiting Conditions

for Operation. In general, three types of surveillance requirements are specified: operability checks, calibrations, and system inspections.

Maximum allowable surveillance intervals provide operational flexibility and are not used to reduce frequency. Established frequencies are maintained over the long term. Any time that a reactor system or component is modified or repaired, the surveillance for that system will be performed as part of the operability check of the system or component. This should be done regardless of when the surveillance was last performed or when it is next due.

14.2.5 Design Features

This section provides information regarding the design features of the reactor facility which are necessary to ensure that major alterations to safety-related components or equipment are not made prior to appropriate safety reviews. The SAR contains the details necessary for establishing criteria for these specifications. Therefore, only those design features of MURR describing materials of construction and geometric arrangements, which if altered or modified would significantly affect safety and are not included in Sections 2 or 3, are included in this section.

14.2.6 Administrative Controls

This section provides information and controls on staffing and operations of the reactor facility to ensure that facility management and staff are acceptably knowledgeable and aware of the technical requirements to operate a safe facility and compliance with all applicable regulations and license conditions.

CHAPTER 15

FINANCIAL QUALIFICATIONS

15.0 FINANCIAL QUALIFICATIONS

On December 30, 2024, the U.S. NRC issued a final rule, titled “Non-Power Production or Utilization Facility License Renewal” effective January 29, 2025 (ML24241A113) which eliminated the requirement for non-power production or utilization facility licensees to submit financial qualification information with license renewal applications under § 50.33(f)(2). This ruling additionally stated that the NRC found that the financial qualification information did not meaningfully contribute to the NRC’s safety determination on the license renewal application. As a result, the information contained in Chapter 15 on financial qualifications, which was submitted in the application for license renewal and supplemented by responses to NRC Requests for Additional Information, has been removed from the SAR.

CHAPTER 16

OTHER LICENSE CONSIDERATIONS

TABLE OF CONTENTS

16.0 OTHER LICENSE CONSIDERATIONS 2
16.1 Prior Use of Reactor Components 2
16.1.1 Fuel and Fuel Cladding 2
16.1.2 Primary Coolant System Pressure Boundary 3
16.1.3 In-Pool Components Receiving High Neutron Fluence 5
16.1.4 Reactor Pool Liner 6
16.1.5 Reactor Containment Structure and Isolation System 6
16.1.6 Reactor Safety and Engineered Safety Features Actuation Systems 7
16.1.7 Area Radiation Monitoring System 7
16.2 Medical Use of Non-Power Reactors 7

REFERENCES

16.1 Letter from MURR to NRC, "Slight Elevation of Fission Products Detected in Primary Coolant Water Analyses," December 1997.

16.2 Technical Data Report, TDR-0101, "Nondestructive Testing of RPV Spool Piece," University of Missouri Research Reactor, April 2006.

16.3 Farrell, K., "Assessment of Aluminum Structural Materials for Service within the ANS Reflector Vessel," ORNL, 1995

16.4 Written communication dated August 31, 2010, regarding responses to "University of Missouri at Columbia – Request for Additional Information Re: License Renewal, Safety Analysis Report, 45-Day Response Questions dated June 1, 2010. [ML120050315]

16.5 Yahr, G.T., "Fatigue Design Curves for 6061 T-6 Aluminum," ORNL, 1993.

16.6 Sargent and Lundy, LLC., "Condition Evaluation of Aluminum Reactor Pool Liner," July 2001.

16.7 Sargent and Lundy, LLC., "Containment Structure Condition Assessment," July 2001.

16.8 MURR Letter (and enclosure) to U.S. NRC, Response to NRC Request for Additional Information for Issuance of Amendment No. 32, Facility License No. R-103, University of Missouri Research Reactor, April 12, 2001. [ML011070569]

16.9 NRC issued Renewed Reactor Facility Operating License by letter dated January 4, 2017

16.0 OTHER LICENSE CONSIDERATIONS

This chapter discusses license considerations not addressed elsewhere in the SAR. One of these considerations is the prior use of reactor components. Additionally, the University of Missouri Research Reactor (MURR) has no history or expertise in the use of the facility for medical purposes; therefore, no detailed discussion of medical use is included here.

16.1 Prior Use of Reactor Components

MURR first attained criticality in October 1966. The facility was originally licensed to operate at a thermal power of 5 MW even though the reactor design, except for the coolant systems, allowed for 10-MW operation. In 1974, the reactor was upgraded and licensed to operate at 10 MW. A general overview of the facility and a list of the major facility modifications that have been performed since 1966 are provided in Chapter 1, "The Facility." All systems, structures, and components that comprise the facility will continue to be utilized in the same manner as originally designed. Assessment of prior use components is based solely on MURR's history of operation, and no components discussed have a history of prior use at facilities other than MURR.

Prior use components and systems that are evaluated for continued operation and are significant to safety include the following: fuel and fuel cladding, Primary Coolant System pressure boundary, Reactor Pool liner, Reactor Containment Building Structure and Isolation System, Safety System, Engineered Safety Features, and Radiation Monitoring System. The following discussion shows that each of the prior use components and systems considered will continue to perform their respective functions for a time well in excess of the licensing period of the Renewed Reactor Facility License R-103 issued by the NRC on January 4, 2017.

16.1.1 Fuel and Fuel Cladding

The performance and limitations of the MURR fuel elements are discussed in Chapter 4, "Reactor Description." The current and all reasonable fuel cycles exempt the fuel elements from consideration as prior use.

The existing surveillance method of visually inspecting fuel elements upon receipt, prior to use, during each refueling, and at the end-of-life provides adequate confidence in the continued performance of the fuel elements and allows detection of any cladding failure or defect as early as possible. MURR has used over 1000 fuel elements since 1971 with no failures but retired one (1) fuel element early after it had been used for 126 MWD of the planned 150 MWD-usage because of a suspected manufacturing defect which caused a slight increase in I-131 level in the primary coolant (Reference 16.1). This slight increase was easily detected by the online Fission Product Monitor and subsequently validated by radiochemical analysis of a water sample.

16.1.2 Primary Coolant System Pressure Boundary

The Primary Coolant System is discussed in detail in Chapter 5, "Reactor Coolant Systems," but generally is comprised of two (2) sides - process and reactor - and divisible by Primary Coolant Isolation Valves V507A and V507B.

For the Primary Coolant System as a whole, the existing maintenance and surveillance systems provide adequate confidence in the continued performance of the Primary Coolant System and allow detection of any condition that may require corrective actions. A Water Clean-Up Loop provides adequate control of corrosion, and no significant deterioration mechanisms exist for the majority of the Primary Coolant System.

For the reactor side of the Primary Coolant System, the pressure boundary is rarely accessible for inspection. In addition to maintenance and surveillance systems, the in-pool piping and components of the Primary Coolant System are rigorously inspected during the infrequent periods that they are accessible. During the beryllium reflector replacements performed in January 2006, March 2014, and August 2022, when a large portion of the reactor core support structure was disassembled to gain access to the reflector, all accessible portions were thoroughly inspected by reactor staff, including a visual inspection using a radiation tolerant, underwater digital camera with pan-zoom-tilt capabilities. The specific areas examined included: the inner and outer pressure vessels (PV); fuel support spider; upper reflector tank (with 6 graphite elements removed); beryllium reflector alignment dowel pins; terminal ends of the pneumatic tube system; and terminal ends of beamports A, B and C. The digital camera inspection was recorded to DVD format for future reference. A liquid dye penetrant test was performed by an independent testing company on representative welds on a section of In-Pool Primary System piping (Reference 16.2). The results of the inspection revealed no signs of deterioration and no specific areas of concern. Ultrasonic wall thickness measurements taken at various locations of In-Pool Primary Coolant piping indicate that structural integrity has not been degraded by the mechanisms of corrosion or flow erosion. Additionally, weekly refueling evolutions provide a limited opportunity to inspect the reactor pressure vessels. With the reactor pressure vessel cover removed, the Operator removing and installing the fuel elements has an excellent visual and tactile estimate of the condition of the vessel surfaces.

The inner and outer reactor pressure vessels are, by design, the only Primary Coolant System components subject to high neutron fluence. The pressure vessels are considered serviceable well in excess of the planned operating period of the reactor. Both pressure vessels are constructed from aluminum alloy 6061-T6. The vessels are designed with a significant margin between the maximum design stress and the allowed stress limit for aluminum 6061-T6. The pressure vessels have operated in a temperature and neutron environment that have either maintained or increased their material strength. These conclusions are supported by the following paragraphs.

The reactor pressure vessels separate the pressurized Primary Coolant System from the open Pool System. The pressure vessels are located completely inside the Reactor Pool. A break in either pressure vessel would cause a Primary Coolant System leak into the Pool System and a Primary Coolant low

pressure reactor scram. There are four (4) independent Primary Coolant low pressure scrams, and each independently can safely shut down the reactor. Therefore, a break in the pressure vessels does not cause a reactor safety issue, but only prevents continued operation of the reactor. The failed pressure vessel would have to be replaced before the reactor could be restarted. Spare inner and outer pressure vessels are on hand in case of this event.

The following environmental parameters in which the pressure vessels operate can affect their material condition: stress, temperature, neutron fluence, and neutron spectrum. The reactor pressure vessel design pressure is 100 psig (689 kPa above atmosphere). The Limiting Safety System Setting (LSSS) for Primary Coolant System pressure is 75 psia (517 kPa) as measured at the pressurizer. The pressurizer connects to the Primary Coolant System before the Primary System piping enters the Reactor Pool. Therefore, the pressure vessels are at pressurizer pressure during the time when the Primary Coolant pumps are not operating, and at a slightly lower pressure during pump operation in which the approximately 3,800 gpm (14,385 lpm) of primary coolant flow produces a slight pressure drop between the pressurizer and the inlet to the reactor pressure vessels.

The design pressure of 100 psig [hydrostatic pressure of 150 psig (1034 kPa above atmosphere)] is used to calculate the pressure vessel stress. The outer pressure vessel has an outside diameter (OD) of 12.55 inches (31.88 cm) and an inside diameter (ID) of 11.925 inches (30.29 cm). Therefore, the wall thickness is 0.3125 inches (0.79 cm). The inner pressure vessel has an OD of 5.06 inches (12.85 cm) [in the vertical grooves is the smallest OD] and an ID of 4.50 inches (11.43 cm). Therefore, the wall thickness is 0.280 inches (0.71 cm). The stress on the pressure vessels can be calculated from these values.

max stress on outer PV = internal pressure x radius/thickness

= 0.689 MPa × 5.963 inches/0.3125 inches

= 13.1 MPa (1,900 psia)

max stress on inner PV = external pressure x OD/thickness

= 0.689 MPa × 5.06 inches/0.280 inches

= 12.5 MPa (1,813 psia)

The original design calculations gave 8,500 psia (58.6 MPa) as the allowed stress limit for unwelded aluminum 6061-T6. There are no welds on the portions of the pressure vessels located in any significant neutron flux. Both pressure vessels have the Primary Coolant System on the high-pressure side and the Pool Coolant System on the low-pressure side. During normal reactor operations, the temperature range for the Primary Coolant System falls within 48 to 60 °C (118 to 140 °F) and 38 to 48 °C (100 to 118 °F) for the Pool Coolant System. Therefore, the pressure vessel temperatures stay below 100 °C (212 °F). This maintains the tempered strength of aluminum 6061-T6 and, as calculated above, provides a significant margin from the allowed stress limit.

As of December 31, 2024, the reactor pressure vessels have been in service approximately 167,078 MWD of operation. This has resulted in the following:

- A peak fast fluence [>0.1 MeV] of 3.65 × 1027 n/m2 to the inner pressure vessel;
- A peak thermal fluence of 2.83 × 1027 n/m2 to the inner pressure vessel;
- A peak fast fluence of 2.88 × 1027 n/m2 to the outer pressure vessel; and
- A peak thermal fluence of 2.51 × 1027 n/m2 to the outer pressure vessel.

A report on the effect of the radiation environment on material properties of aluminum 6061-T6 describes the variation of strength and elongation as a function of thermal neutron fluence (Reference 16.3). This indicates that the ultimate strength and yield strength both increase with thermal neutron fluence above 10^{24} to 10^{25} n/m^2. The maximum thermal fluence measured in that report is 4.00×10^{27} n/m^2. Assuming the current operating schedule continues, the thermal neutron fluence does not put the pressure vessels at risk of failure due to the stress until at least December of 2044.

Thermal neutron flux has generated approximately 6.9 wt.% silicon in the inner pressure vessel by the transmutation of aluminum. Silicon is insoluble in aluminum at temperatures below 200 °C (392 °F). The silicon precipitates are responsible for most of the radiation strengthening discussed above, and contribute somewhat to swelling. Fast neutron flux causes additional swelling in aluminum due to the production of microscopic voids. The total swelling is the sum contribution due to both void formation and silicon production. Total neutron flux will cause a conservative peak of 2.7% swelling by the end of 2030. Therefore, swelling will not cause a significant increase in stress through the proposed operating period. (Reference 16.4)

A report on the effect of fatigue stress on aluminum 6061-T6 shows the infinite [>10^7 cycles] lifetime fatigue stress to be 50 MPa (7,251 psia) (Reference 16.5). The maximum cyclic stress for the MURR pressure vessels is the transition between being pressurized and depressurized as part of starting up and shutting down the Primary Coolant System. This results in a pressure change of approximately 60 psia (410 kPa) causing a stress of approximately 8 MPa (1,160 psia), less than 20% of the infinite lifetime stress limit. If it were assumed that this occurs 200 times per year (more typical is around 70-80), there would be 12,850 cycles over a 64 1/4-year operating period through December 2030. Therefore, the accumulated fatigue stress does not limit the service life of the inner and outer pressure vessels. (Reference 16.4)

16.1.3 In-Pool Components Receiving High Neutron Fluence

MURR was designed such that the five following components/regions that receive a high neutron fluence could be replaced: the inner and outer reactor pressure vessels, the center test hole canister (flux trap sample holder), the control blades, the beryllium reflector, and the graphite reflector elements.

The material condition of the inner and outer pressure vessels is discussed in detail in Section 16.1.2,. The remaining four (4) components are replaced periodically due to material condition or prior to

reaching predicted performance limitations. The existing monitoring and scheduled component replacements preclude these components from consideration as prior use components.

16.1.4 Reactor Pool Liner

In April and June 2000, a detailed assessment of the Reactor Pool liner was performed by the engineering firm Sargent & Lundy LLC. (Reference 16.6). Since the welds and adjacent areas of the aluminum pool liner are where corrosion is most likely to occur, the inspection of the liner focused on the welds and the aluminum plate and components around the welds. No evidence of a number of potential corrosion mechanisms, and forms of linear distress, including cracks, deformations (including bulges), buckling, and tears (at anchorages or attachments) was found on the inspected welds and plates. The conclusion from the inspection was that, based on the condition after 34 years of reactor operations, an additional 34 years of good performance by the aluminum pool liner is expected, with the operating conditions and operating procedures continuing as they have been.

The Pool Skimmer System provides a secondary function of corrosion prevention. The system returns water to the otherwise relatively stagnant area between the fixed and moveable beamport liners. Periodic operation of the Skimmer System thus aids in preventing the formation of a concentration cell at the interface of the beamport liners and the pool proper.

Both the Primary and Pool Coolant Systems have a Clean-Up System for demineralization and corrosion control. These ion-exchange systems maintain a low conductivity and maintain the pH in a range around 5 to 6. Aluminum aqueous corrosion resistance is high in slightly acidic water. In addition to routine monitoring, water samples are taken weekly from the Primary and Pool Coolant Systems and are analyzed for pH, conductivity, and contained radioisotopes. Thus, pH and conductivity are monitored and maintained in an appropriate range to minimize corrosion and degradation of aluminum piping, components and the pool liner.

The pool liner has adequate integrity and sufficient maintenance systems in place to provide confidence in its continued performance beyond the proposed operating period.

16.1.5 Reactor Containment Structure and Isolation System

In June 2000, a detailed assessment of the Reactor Containment Building was performed by the engineering firm Sargent & Lundy LLC. (Reference 16.7). The report concluded that the Containment Building was structurally sound, could resist the Safe Shutdown Earthquake (see Section 2.5.2.5), and had only minor concrete and coating deterioration of the walls. The report further stated that, following recommended repairs, the structure would continue to perform its function through the proposed licensing period. The applicable recommended repairs - minor concrete crack repairs and seal coatings - were subsequently performed. The Containment Building compliance test, which measures the leakage rate of the structure, has shown no indication of degradation or a notable trend toward such degradation.

The remainder of the Isolation System components, such as the Entry Doors and door gaskets, Utility Entry Water Seal, and Ventilation System Doors and Valves are discussed in detail in Chapter 6, "Engineered Safety Features." The existing maintenance and surveillance systems provide adequate confidence in the continued performance of the Reactor Containment Structure and Isolation System.

16.1.6 Reactor Safety and Engineered Safety Features Actuation Systems

As discussed in Chapter 7, "Instrumentation and Control Systems," the Reactor Safety and Engineered Safety Features Actuation Systems must be able to affect a reactor scram, initiate a Containment Building isolation or activate the Anti-Siphon System. All Safety System and Engineered Safety components have inspection, maintenance, and surveillance items performed regularly. The various detectors, channels, and circuit components have been thoroughly reviewed and upgraded where applicable to ensure that suitable parts are available and/or spare parts are on hand (Reference 16.8). Furthermore, the mechanical components associated with the Reactor Safety and Engineered Safety Features Actuation Systems such as the control blades, offset mechanisms, isolation doors, valves and gaskets, and Anti-Siphon System isolation valves and actuators are also adequately monitored through the inspection, maintenance and surveillance systems in place. In all, the Reactor Safety and Engineered Safety Features Actuation Systems and their components which can be considered prior use provide sufficient confidence in their continued performance through the proposed operating period.

16.1.7 Area Radiation Monitoring System

The Area Radiation Monitoring System is discussed in detail in Chapter 7, "Instrumentation and Control Systems." All detector, alarm, and monitoring components are regularly inspected and maintained. Surveillance is performed on those portions which initiate the Isolation System. Rare electronic failures have occurred in these modules over time, and adequate spares are on hand or available to ensure that the system as a whole will perform as designed through the proposed operating period.

16.2 Medical Use of Non-Power Reactors

MURR has not been utilized for medical purposes and there are no immediate future plans for such use.

CHAPTER 17

DECOMMISSIONING

TABLE OF CONTENTS

17.0 DECOMMISSIONING 2
17.1 Introduction 2
17.2 Decommissioning Alternatives 3
17.3 Revisions to the Original Cost Estimate 3
17.4 Decommissioning Cost Estimate 4
17.4.1 Adjustment Factor 4
17.4.2 Adjusted Decommissioning Cost Estimate 6

LIST OF TABLES

Table 17-1 ANNUAL COST DURING SAFSTOR 3
Table 17-2 SUMMARY OF COST OVER 30-YEAR PERIOD (1989 Dollars) 4

REFERENCES

17.1 Written communication dated May 10, 2024, as required by 10 CFR 30.35(e)(2), regarding an update to the University of Missouri Research Reactor decommissioning cost estimate for calendar year 2024. [ML24135A068]

17.2 NUREG-0586, “Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities,” August 1988.

17.3 NUREG-1307, “Report on Waste Burial Charges,” Rev. 19, February 2023.

17.4 U.S. Department of Labor - Bureau of Labor Statistics, Employment Cost Index - Midwest Region.

17.5 U.S. Department of Labor - Bureau of Labor Statistics, Producer Price Index - Commodities [Series ID: wpu0543].

17.6 U.S. Department of Labor - Bureau of Labor Statistics, Producer Price Index - Commodities [Series ID: wpu0573].

17.0 DECOMMISSIONING

This chapter of the SAR describes in detail the University of Missouri's financial ability and choice of alternative to decommission the reactor facility.

17.1 Introduction

In a letter dated June 29, 1990, the University of Missouri provided the U.S. Nuclear Regulatory Commission (NRC) a report required by 10 CFR 50.33(k)(2) providing reasonable assurance that funds would be available to decommission the University of Missouri-Columbia Research Reactor (MURR). As required by 10 CFR 50.75(e)(2)(iv) (The corresponding regulation as of 2024 is 10 CFR 50.75(e)(1)(iv)), a cost estimate for decommissioning was provided. 10 CFR 50.75(d) further specifies that the cost estimate be adjusted periodically over the life of the facility. For Renewed Facility Operating License No. R-103, MURR adjusts the decommissioning costs every five years. However, as per 10 CFR 30.35(e)(2), adjusting the decommissioning costs for byproduct material licenses is required every three years. MURR has a Broad Scope Material License (No. 24-00513-39). Therefore, MURR will adjust the cost of decommissioning the facility every three years. (Reference 17.1)

The following is a chronological history of when the decommissioning costs were updated, the final cost estimate, and the reason for the update.

Year	Cost Estimate	Reason for Updated Estimate
2024	$65.14 Million	Material License 10 CFR 30.35(e)(2)
2021	$61.02 Million	Material License 10 CFR 30.35(e)(2) and Reactor License 10 CFR 50.75(d)
2018	$58.31 Million	Material License 10 CFR 30.35(e)(2)
2016	$58.41 Million	Relicensing RAI dated April 8, 2016
2015	$58.99 Million	Material License 10 CFR 30.35(e)(2)
2013	$58.90 Million	Relicensing RAI dated February 11, 2013
2012	$54.70 Million	Material License 10 CFR 30.35(e)(2)
2009	$47.30 Million	Relicensing RAI dated July 10, 2009
2005	$39.95 Million	Reactor License 10 CFR 50.75(d)
2000	$34.10 Million	Reactor License 10 CFR 50.75(d)
1995	$11.80 Million	Reactor License 10 CFR 50.75(d)
1990	$9.00 Million	Reactor License 10 CFR 50.75(d)

17.2 Decommissioning Alternatives

Three decommissioning alternatives are defined in NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities" (Reference 17.2):

- In decontamination (DECON), the equipment, structures, and portions of a facility and site containing radioactive materials are removed or decontaminated to a level permitting release of the property by the NRC shortly after operations cease;
- In safe storage (SAFSTOR), the nuclear facility is placed and maintained in a condition that allows it to be safely stored and subsequently decontaminated to a level permitting release of the property by the NRC; and
- In entombment (ENTOMB), radioactive materials are encased in a structurally long-lived material such as concrete. The entombed structure is appropriately maintained and surveillance is continued until the radioactivity decays to a level permitting release of the property by the NRC.

17.3 Revisions to the Original Cost Estimate

Table 17-1 was revised in 1995 to delete the 30-year annuity method of determining the present value of the annual costs associated with SAFSTOR. Uncertainty in future inflation and interest rates could have had the potential of introducing a significant under-estimation of costs using the annuity method.

Table 17-2 was also revised in 1995 to incorporate the revised annual cost of SAFSTOR from Table 17-1. These revisions were made to the original tables and are in 1989 dollars.

TABLE 17-1
ANNUAL COST DURING SAFSTOR

SAFSTOR Items	Cost
Security	15,000.00
Minor Maintenance and Repair	10,000.00
Major Repair	10,000.00
Offsite Laboratory Work and Equipment Repair	6,000.00
Reactor Facility Services	50,000.00
Laboratory Samples, EPA Reports, and Surveillance	30,000.00
Total	$121,000.00a

a. The 30-year total cost estimate is 3.6 million dollars.

TABLE 17-2
SUMMARY OF COST OVER 30-YEAR PERIOD (1989 DOLLARS)

Item	Cost
Labor	4.9 Million
Equipment and Supplies	0.27 Million
Radioactive Shipments	0.6 Million
Termination Survey	0.06 Million
Annual Storage Cost	3.6 Million
Subtotal	$9.43 Million
Contingency (25%)	$2.36 Million
Total	$11.8 Million

17.4 Decommissioning Cost Estimate

17.4.1 Adjustment Factor

The adjustment factor was designed for updating reference Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) decommissioning estimates, but serves as a convenient method to adjust estimated costs over time. The variables are relevant to research and test reactor decommissioning estimates, although coefficients may vary slightly. Typically, an average of the PWR and BWR costs is used.

Decommissioning costs are divided per 10 CFR 50.75(c)(2) into three general areas that tend to escalate similarly: (1) labor, materials, and services, (2) energy and waste transportation, and (3) radioactive waste burial/disposition. A relatively simple equation is used to update the estimate of cost given a cost estimate in base-year dollars (1986 dollars) and the fractional escalation of these three categories of cost over the time period of interest. That equation is:

$$\text{Estimated Cost (2024)} = [1986\ \$\ \text{Cost}] \times [A\ L_x + B\ E_x + C\ B_x];$$

Where:

Estimated Cost (2024) = estimated decommissioning costs in 2024 dollars

[1986 $ Cost] = estimated decommissioning costs in 1986 dollars

A = fraction of the [1986 $ Cost] attributable to labor, materials, and services (0.65)

B = fraction of the [1986 $ Cost] attributable to energy and transportation (0.13)

C = fraction of the [1986 $ Cost] attributable to waste burial (0.22)

L_x = labor, materials, and services cost adjustment, January 1986 to March 2024

E_x = energy and waste transportation cost adjustment, January 1986 to November 2023

B_x = LLW burial/disposition cost adjustment, January 1986 to February 2023.

The coefficients in the adjustment factor of 10 CFR 50.75 (c)(2) are established as A = 0.65, B = 0.13, and C = 0.22. The escalation formula becomes:

Estimated Cost (2024) = [1986 $ Cost] × [0.65 L_x + 0.13 E_x + 0.22 B_x]

17.4.1.1 Determination of L_x, E_x, and B_x

These ratios are determined using the information supplied by NUREG-1307, "Report on Waste Burial Charges," Rev. 19, February 2023 (Reference 17.3), and by the U.S. Department of Labor - Bureau of Labor Statistics (References 17.4, 17.5, 17.6).

A. Labor Adjustment Factors

L_x is calculated using the Employment Cost Index (ECI) taken from Table 6 of the current BLS data entitled "Employment Cost Index for total compensation, for private industry workers, by bargaining status, census region, and census divisions.." The ECI is taken for the Midwest Region, West North Central (March 2024). The Base L_x is derived using the methodology of Section 3.2, of NUREG-1307, therefore:

L_x = [(ECI), March 2024) × (Base L_x)] ÷ 100

= [(163.4) × (2.08)] ÷ 100

= 3.40

B. Energy Adjustment Factors

The adjustment factors for energy, E_x, is a weighted average of two components, namely, industrial electrical power, P_x, and light fuel oil, F_x.

For the reference PWR: $E_x(PWR) = 0.58P_x + 0.42F_x$

For the reference BWR: $E_x(BWR) = 0.54P_x + 0.46F_x$

P_x and F_x are the values of current Producer Price Indexes (PPI) (Reference. 17.5, 17.6) divided by the corresponding indexes for January 1986.

P_x = 294.670 (November 2023 value for code wpu0543) / 114.2 (January 1986 value for code wpu0543)

= 2.58

F_x = 369.728 (November 2023 value for code wpu0573) / 82.0 (January 1986 value for code wpu0573)

= 4.51

Therefore:

E_x(PWR) = (0.58 × 2.58) + (0.42 × 4.51)

= 3.39

E_x(BWR) = (0.54 × 2.58) + (0.46 × 4.51)

= 3.47

E_x for MURR is calculated as an average of E_x(PWR) and E_x(BWR), therefore:

E_x(average) = (3.39 + 3.47)/2,

= 3.43

C. Waste Burial Adjustment Factors

The adjustment factor for waste burial/disposition, B_x, is taken directly from data on the appropriate LLW burial location as given in Table 2-1, "Values of Bx as a Function of LLW Burial Site, Waste Vendor and Year," in NUREG-1307 (Reference 17.3) . For facilities that have no disposal site available for LLW, the NUREG provides B_x values in the last column of Table 2-1.

B_x(PWR) = 13.711

B_x(BWR) = 12.296

B_x for MURR is calculated as an average of the PWR and BWR, therefore:

B_x(average) = (13.711 + 12.296)/2,

= 13.0

17.4.2 Adjusted Decommissioning Cost Estimate

The following estimate is for the SAFSTOR option with 30-year storage and includes a 25% contingency. For future consideration, Reference 17.2 indicated the DECON option may be advantageous from an economic standpoint (DECON estimated costs are about 80% of 30-year SAFSTOR option); however, the DECON option would result in about 260% higher occupational dose to accomplish. In keeping with the current ALARA principles of dose reduction and the uncertainty of future directions of ALARA, the 30-year SAFSTOR option still appears to be a good compromise.

Reactor Type: Research (Plate-Type Fuel)

Thermal Power Rating: 10 MW_{th}

Location of Facility: Midwest Region of the U.S.

LLW Disposition Preference: Contract with Waste Vendors

LLW Burial Location: Determined by Vendor Specifications (non-compact agreement)

Decommissioning Cost (2024 $)

= [1986 $ Cost] × [A L_x + B E_x + C B_x]

= ($11.8 Million) [(0.65) × (3.40) + (0.13) × (3.43) + (0.22) × (13.00)]

= (11.8 Million) [5.52]

= $65.14 Million

CHAPTER 18

HIGH-ENRICHED TO LOW-ENRICHED URANIUM CONVERSIONS

TABLE OF CONTENTS

18.0 HIGH-ENRICHED TO LOW-ENRICHED URANIUM CONVERSIONS 3
18.1 Introduction 3
18.2 Background 3
18.3 Continuing Efforts by MURR 4

REFERENCES

18.1 Rest, J., Kim, Y.S., Hofman, G.L., Meyer, M.K., and Hayes, S.L., U-Mo Fuels Handbook - Version 1.0, RERTR Program, Argonne National Laboratory, June 2006.

18.2 Letter Request to US NRC, Application for Unique Purpose Exemption from Conversion from HEU Fuel, Facility License No. R-103, University of Missouri Research Reactor, September 26, 1986 [ML20214W702]; and supplemented by Letter Response to US NRC Request for Additional Information Supporting the Unique Purpose Exemption Request, February 27, 1987 [ML20212C964].

18.3 Written communication dated August 18, 2017 from MURR to the NRC regarding a Preliminary Safety Analysis Report for converting MURR from HEU to LEU. [ML17132A370].

18.4 Letter from the NRC dated April 21, 2020, that the NRC staff would discontinue review of the PSAR submitted due to the preliminary status of LEU fuel suitable for the conversion from HEU. [ML20085G507].

18.0 HIGH-ENRICHED TO LOW-ENRICHED URANIUM CONVERSIONS

This chapter provides some background material on the feasibility of converting U.S. high-performance research and test reactors from high-enriched to low-enriched uranium fuel and the continuing efforts by staff of the Missouri University Research Reactor (MURR) within this conversion process. By letter dated September 26, 1986, the University of Missouri requested that a determination be made by the U.S. Nuclear Regulatory Commission (NRC) that MURR has a "Unique Purpose," as defined by 10 CFR 50.2, and is, therefore, exempt from the conversion from HEU to LEU fuel (Reference 18.2).

18.1 Introduction

The conversion of high-performance research and test reactors in the U.S. and abroad to the use of low-enriched uranium (LEU) fuel requires large increases in uranium densities in the fuel meat of their various fuel plates. In addition, conversion of some lower-power research reactors to the use of LEU fuels requires uranium densities substantially higher than those possible with U_3Si_2 dispersion fuel. Because the high-uranium-content compounds (i.e., U_3Si and U_6Fe) previously have shown to be unstable under irradiation in fuel plates, the emphasis for US-RERTR (Reduced Enrichment for Research and Test Reactors) advanced fuel development has been on metallic uranium of low alloy content for both the monolithic and dispersion fuel designs (Reference 18.1).

The RERTR Program, initiated by the U.S. Department of Energy (DOE) in 1978, helps develop the necessary technology to enable the conversion of civilian facilities from high-enriched uranium (HEU) to LEU fuels.

18.2 Background

Since 1997, several tests of U-Mo dispersion fuels have been conducted by the Canadian, French, Korean, and U.S. research reactor fuel development programs. These tests have shown that in terms of irradiation behavior, U-Mo alloys are the best candidates for the dispersed fuel phase.

U-Mo alloys were extensively studied in the 1960s as fast reactor fuels and for use in fast burst reactors. Since fast reactor fuel irradiation experiments were conducted at high temperature, the irradiation performance database generated as a result of this work is only marginally applicable to the current issue of research reactor fuel development. In addition to irradiation testing, a large amount of work was completed on the determination of phase equilibrium, transformation kinetics, and physical, thermal, and mechanical properties. Since the properties of aluminum alloys are generally well known, the combined database for aluminum and U-Mo provide a starting point from which values for U-Mo dispersions can be estimated by the development and application of appropriate correlations.

Some data cannot easily be estimated, thus requiring new data to be generated. The largest deficiency is in the area of the properties of $(U\text{-}Mo)Al_x$ compounds that form as a result of fuel-matrix interaction (for dispersion fuel) or fuel-cladding interaction (for solid U-Mo, or monolithic, fuel). For example, the thermal conductivity of these compounds has a large bearing on dispersion fuel behavior but has not

been measured. The same situation applies in higher aluminide phases, since the nature of the compounds that form as a result of this reaction is not well known. As another example, the mechanical strength of fuel is a strong function of interface properties and processing technique and requires measurement to establish properties.

Although the use of U-Mo alloys in dispersion fuel enables high-densities to be achieved, a major issue with this fuel is the reaction between U-Mo and matrix Al. It has been shown that under certain irradiation conditions, this reaction product exhibits unstable swelling behavior, resulting in excessive and unpredictable fuel plate swelling. However, the irradiation behavior of the U-Mo fuel particles themselves has been shown to be stable. Several potential fixes to fuel performance problems associated with the interaction phase are currently being irradiation-tested. The U-Mo monolithic fuel provides the highest possible densities and eliminates the problem of the fuel-matrix reaction; however, a similar problem may arise in the interaction layer formed between the U-Mo and the aluminum alloy cladding.

18.3 Continuing Efforts by MURR

Because of its compact design, which requires a high loading density of uranium-235, the MURR core cannot presently perform its intended function without the use of HEU fuel. The use of LEU fuel requires even higher uranium-235 densities because of the non-fissioning absorption effect of uranium-238. No currently qualified LEU fuel-type exists that can provide the uranium loading densities required for MURR to operate.

However, MURR is actively collaborating with the RERTR Program and four other U.S. high-performance research reactor facilities that use HEU fuel to find a suitable LEU fuel replacement. Although each one of the five high-performance research reactors is responsible for its own feasibility and safety studies, regulatory interactions, fuel procurement, and conversion, there are common interests and activities among all five reactors that will benefit from a coordinated, working-group effort.

The following reports and letters demonstrate MURR's ongoing efforts to convert from HEU to LEU when qualified LEU-type fuel is developed and tested:

- Technical Data Report (TDR)-0103, "MURR HEU/LEU Conversion Feasibility Study," July 2006
- TDR-0125, "Feasibility Analysis for HEU to LEU Conversion of MURR," April 6, 2011
- TDR-0137, "Draft Chapter 4, Reactor Description, Safety Analysis Report for HEU to LEU Conversion"
- TDR-0139, "HEU to LEU Conversion Fuel Handling Accident Analysis"
- TDR-0151, "MURR LEU Fuel Element Storage Plan and Criticality Analysis"
- TDR-0152, "MURR LEU Low-Power Physics Test Plan, Procedures and Prediction Analysis"

By letter dated August 18, 2017, MURR submitted to the NRC a Preliminary Safety Analysis Report (PSAR) for conversion of MURR from HEU to LEU. (Reference 18.3)

By letter dated April 21, 2020, the NRC notified MURR that due to the preliminary status of LEU fuel development, the NRC decided it was not prudent to continue their review of the conversion PSAR at that time. (Reference 18.4)

MURR staff continue to demonstrate their commitment to conversion of MURR to LEU when a qualified fuel is developed and tested by their participation in ongoing RERTR meetings.

APPENDIX B

RADIOLOGICAL IMPACT OF AR-41 DURING NORMAL OPERATIONS

TABLE OF CONTENTS

B.0 RADIOLOGICAL IMPACT OF AR-41 DURING NORMAL OPERATION 3
B.1 Introduction 3
B.1.1 Purpose 3
B.1.2 Commitment to ALARA 3
B.1.3 Radiological Standards 4
B.2 Data and Assumptions 4
B.2.1 Stack Release Point 4
B.2.2 Meteorological Data 5
B.3 Calculations 13
B.3.1 Effective Stack Height 13
B.3.2 Argon-41 Concentration 15
B.3.3 Annual Dose 16
B.4 Maximum Individual Dose Estimates 16
B.5 Normal Operational Releases 18
B.6 Comparison of Risk 19
B.7 Conclusions 20

LIST OF TABLES

Table B-1 METEOROLOGICAL DATA FOR COLUMBIA, MISSOURI STABILITY CLASS INFORMATION, 1960 to 1969 6
Table B-2 MAXIMUM ANNUAL INDIVIDUAL DOSE AT 150 METERS 17
Table B-3 MAXIMUM ANNUAL INDIVIDUAL DOSE AT 760 METERS 17
Table B-4 OPERATIONAL RADIATION DOSE ESTIMATES 19
Table B-5 ESTIMATED DOSES IN TERMS OF PERCENT OF NATURAL BACKGROUND 19

LIST OF FIGURES

Figure B.1 – UNIVERSITY OF MISSOURI TOPOGRAPHICAL MAP 21

REFERENCES

B.1 U.S. Nuclear Regulatory Commission Amendment No. 12 for Facility Operating License No. R-103 (NRC Docket No. 50-186), July 5, 1979.

B.2 Callaway Environmental Report, Operational License Stage, Volume 1, Tables 2.3-19 and 2.3-20.

B.3 Cember, Herman, Introduction to Health Physics, Second Edition, Pergamon Press, 1983, pp. 340-352.

B.4 DeGroot, Morris H., Probability and Statistics, Addison-Wesley Publishing Company, Inc., 1975, pp. 49-50.

B.5 Regulatory Guide 1.109, "Calculations of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluation Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977.

B.6 U.S. Nuclear Regulatory Commission Amendment No. 8 for Facility Operating License No. R-103 (NRC Docket No. 50-186), February 24, 1978.

B.7 National Commission on Radiological Protection (NCRP) Report No. 94, "Exposure of the Population in the United States and Canada from Natural Background Radiation," December 1987.

B.8 Written communication to the NRC, dated October 1, 2015, regarding responses to requests for additional information related to the renewal of Operating License No. R-103 for the University of Missouri-Columbia Research Reactor, [ML15275A314].

B.0 RADIOLOGICAL IMPACT OF AR-41 DURING NORMAL OPERATION

B.1 Introduction

B.1.1 Purpose

The licensing of a nuclear facility requires that the dose rates in the unrestricted areas proximate to the facility be calculated for normal as well as accident conditions. The purpose of this appendix is to document the methodology and calculations that were used to predict the dose rates at the Emergency Planning Zone (EPZ) boundary and the nearest residence as a result of the normal operation of the University of Missouri Research Reactor (MURR). The source term for normal operation is the noble gas argon-41 (Ar-41), which has a half-life of 1.8 hours. As previously presented in the MURR Safety Analysis Report (SAR), the reactor facility does not constitute an undue hazard or risk to the health and safety of the general public. The engineered safety features of the facility, and the conservative safety limits and factors will prevent any significant radiological releases. Nevertheless, MURR must comply with the federal requirements for calculating offsite doses. The radiological consequences of the Maximum Hypothetical Accident (MHA) are discussed in Chapter 13 of this report. The source term for the MHA is derived from the complete failure of a fueled experiment and the subsequent release of fission products into the pool coolant system.

Ar-41 is produced when the argon-40 (Ar-40) in air (about 1.0% of atmospheric content) is activated by thermal neutrons. The principal production areas within the reactor facility include the pneumatic tube (p-tube) system, the thermal column, and the beamports. The Ar-41 produced in these areas is then subsequently released to the atmosphere through the facility ventilation exhaust stack.

B.1.2 Commitment to ALARA

MURR management is committed to keeping releases of radioactive materials as low as is reasonably achievable (ALARA). In addition to reducing the amount of liquid waste released to the sanitary sewer, emphasis has been placed on minimizing the production of Ar-41, which accounts for about 92% of the radioactivity released through the facility ventilation exhaust stack between 2018 through 2022. Over the years, these efforts have included design modifications to the thermal column such as: the reduction in air volume being irradiated on the sides of the graphite stack, the addition of a neutron absorbing material on the top of the graphite stack (thereby preventing neutrons from reaching the air volume above the thermal column), and sealing the thermal column collimator to prevent the release of gaseous activity. These modifications reduced the amount of Ar-41 being released during 10-MW operation to levels less than those previously released during 5-MW operation. Greater than 98% of the remaining Ar-41 production occurs within the p-tube system. The high gamma-ray and neutron heating rates in the p-tube terminals require that cooling air be continuously circulated past the sample carriers to prevent damage to the sample material. Since the system flow rate is about 175 cfm (4.96 m^3/min) when in operation, it would be impractical to use bottled gas not containing Ar-40 or to hold up, or delay, the exhaust air for a sufficient amount of time to allow for appreciable radioactive decay. A closed-loop circulating system, using either nitrogen or carbon dioxide, has the disadvantage that a

small amount of air contamination into that system presents an exposure risk to personnel within the laboratory from high concentrations of Ar-41 since the system is pressurized.

Several alternatives exist to reduce the production of Ar-41 in the p-tube system. These alternatives include a reduction in the number of sample carrier tubes, the replacement of the existing sample carrier tubes with smaller diameter tubes, and the relocation of the carrier tubes to areas of lower neutron flux. All of these alternatives, however, reduce the experimental capabilities of MURR. Nevertheless, beginning in August 2020 (Standing Order 20-07), the practice of shutting down and isolating the p-tube system when not scheduled for use (typically during weekends) has further reduced the release of Ar-41 from the p-tube system.

B.1.3 Radiological Standards

The International Commission on Radiological Protection (ICRP) has been a principal organization studying the effects of ionizing radiation for many years. In 1959, Committee II of the ICRP published recommendations for maximum permissible concentrations (MPCs) of radionuclides in air and water. These recommendations became the technical bases for radionuclide concentration limits published in Title 10, Chapter I of the Code of Federal Regulations, Part 20 (10 CFR 20). Subsequently, Committee II reviewed the current state of knowledge and published updated recommendations in Publication 30 (1978/99), which superseded Publication 2. In Publication 30, the ICRP recommended that the concentration limit in air for radioactive noble gases be based only on the total effective dose equivalent (TEDE) computed for a person immersed in a large cloud of gamma-ray emitters. This guidance is justified in Publication 30, where it is shown that the internal and skin doses from beta particles would add less than 1% of the TEDE.

In 1994, the U.S. Nuclear Regulatory Commission (NRC) implemented a major revision to 10 CFR 20 which incorporated many of the new dosimetry concepts published by the ICRP over the past several years. Since the new revision of 10 CFR 20 is applicable to non-power reactors licensed by the NRC and is widely used as a basis for regulatory limits for similar reactors not under NRC jurisdiction, the calculations and interpretations in this appendix are based on the requirements of 10 CFR 20. The current 10 CFR 20 concentration limits for Ar-41 are:

1. For accessible areas inside the operations boundary: 3×10^{-6} µCi/ml; and

2. For accessible areas outside the operations boundary: 1×10^{-8} µCi/ml.

B.2 Data and Assumptions

B.2.1 Stack Release Point

The following data and calculations describe the physical information of the ventilation exhaust stack release point.

Elevation above sea level = 687 feet (209 m);

Diameter = 40 inches (1.02 m);

Maximum Flow Rate = 30,500 ft^3/min (863.7 m^3/min);

Cross Sectional Area = $\pi \times r^2$;

$$= \pi \times \left(\frac{40 \text{ inches}}{2 \times 12 \text{ inches / ft}} \right)^2;$$

= 8.73 ft^2 (0.82 m^2);

Air Velocity

$$= \frac{30{,}500 \text{ ft}^3 \text{ / min}}{8.73 \text{ ft}^2} \times \frac{0.304 \text{ m}}{\text{ft}} \times \frac{1 \text{ min}}{60 \text{ sec}};$$

= 17.7 m/sec.

The Technical Specification limit for Ar-41 release from MURR is 350 times the Air Effluent Concentration (AEC) listed in Appendix B, Table II, of 10 CFR 20, or:

Q = 350 × AEC × Flow Rate;

= 350 × (1 × 10^{-8} µCi/ml) × (30,500 ft^3/min) × (2.831 × 10^4 ml/ft^3);

= (3.0 × 10^3 µCi/min) × (1 × 10^{-6} Ci/µCi) × (1 min/60 sec);

= 5.0 × 10^{-5} Ci/sec.

B.2.2 Meteorological Data

In the previous environmental assessment (Ref. B.1), the NRC used meteorological data collected at the Callaway Nuclear Power Plant near Fulton, Missouri. This data was collected between May 5, 1973 and May 4, 1975, and judged by the NRC to be "reasonably representative of long-term conditions expected at the MURR site." This current radiological assessment utilizes meteorological data gathered in Columbia, Missouri during the period 1960 to 1969 (Ref. B.2).

The Columbia data was judged to be more appropriate for use in assessing airborne releases from the reactor facility because of the longer data period and the proximity of the data site to the MURR site. Additionally, this data contains a more detailed presentation of observations than what is currently available from the unmanned weather stations that exist today. Table B-1 lists wind data (stability, class, speed, and frequency) for each of the sixteen compass points.

TABLE B-1
METEOROLOGICAL DATA FOR COLUMBIA, MISSOURI STABILITY CLASS INFORMATION, 1960 TO 1969

Direction - North by Northeast (NNE)				
Class[a]	**% Class[b]**	**Wind Speed[c] (m/sec)**	**% NNE Wind[d]**	**%s Comb.[e]**
A	0.4	2.3	3.4	1.4×10^{-4}
B	4.7	2.8	2.7	1.3×10^{-3}
C	11.5	4.0	3.5	4.0×10^{-3}
D	53.6	5.7	4.2	2.3×10^{-2}
E	17.6	3.8	3.2	5.6×10^{-3}
F	12.2	2.4	4.9	6.0×10^{-3}
Direction - Northeast (NE)				
Class[a]	**% Class[b]**	**Wind Speed[c] (m/sec)**	**% NE Wind[d]**	**%s Comb.[e]**
A	0.4	2.1	1.7	6.8×10^{-5}
B	4.7	2.7	2.6	1.2×10^{-3}
C	11.5	3.7	2.7	3.1×10^{-3}
D	53.6	5.2	3.9	2.1×10^{-2}
E	17.6	3.6	2.8	4.9×10^{-3}
F	12.2	2.5	4.9	6.0×10^{-3}
Direction - East by Northeast (ENE)				
Class[a]	**% Class[b]**	**Wind Speed[c] (m/sec)**	**% E Wind[d]**	**%s Comb.[e]**
A	0.4	2.0	7.8	3.1×10^{-4}
B	4.7	2.8	5.1	2.4×10^{-3}
C	11.5	3.9	4.3	4.9×10^{-3}
D	53.6	4.9	4.7	2.5×10^{-2}
E	17.6	3.4	4.2	7.4×10^{-3}
F	12.2	2.5	6.8	8.3×10^{-3}

TABLE B-1
METEOROLOGICAL DATA FOR COLUMBIA, MISSOURI STABILITY CLASS INFORMATION, 1960 TO 1969
(CONT.)

Direction - East (E)				
Class[a]	**% Class[b]**	**Wind Speed[c] (m/sec)**	**% E Wind[d]**	**%s Comb.[e]**
A	0.4	2.0	4.3	1.7×10^{-4}
B	4.7	2.9	5.3	2.5×10^{-3}
C	11.5	3.8	4.4	5.1×10^{-3}
D	53.6	4.9	4.4	2.4×10^{-2}
E	17.6	3.5	5.0	8.8×10^{-3}
F	12.2	2.5	7.9	9.6×10^{-3}
Direction - East by Southeast (ESE)				
Class[a]	**% Class[b]**	**Wind Speed[c] (m/sec)**	**% ESE Wind[d]**	**%s Comb.[e]**
A	0.4	2.0	3.4	1.4×10^{-4}
B	4.7	2.9	4.7	2.2×10^{-3}
C	11.5	3.9	4.8	5.5×10^{-3}
D	53.6	5.3	6.1	3.3×10^{-2}
E	17.6	4.0	6.1	1.1×10^{-2}
F	12.2	2.6	4.5	5.5×10^{-3}
Direction - Southeast (SE)				
Class[a]	**% Class[b]**	**Wind Speed[c] (m/sec)**	**% SE Wind[d]**	**%s Comb.[e]**
A	0.4	2.2	2.6	1.0×10^{-4}
B	4.7	2.9	4.6	2.2×10^{-3}
C	11.5	4.1	6.4	7.4×10^{-3}
D	53.6	5.7	7.8	4.2×10^{-2}
E	17.6	4.1	8.2	1.4×10^{-2}
F	12.2	2.5	4.3	5.2×10^{-3}

TABLE B-1
METEOROLOGICAL DATA FOR COLUMBIA, MISSOURI STABILITY CLASS INFORMATION, 1960 TO 1969 (CONT.)

Direction - South by Southeast (SSE)				
Class[a]	**% Class[b]**	**Wind Speed[c] (m/sec)**	**% SSE Wind[d]**	**%s Comb.[e]**
A	0.4	2.3	4.3	1.7×10^{-4}
B	4.7	3.0	6.5	3.1×10^{-3}
C	11.5	4.1	8.7	1.0×10^{-2}
D	53.6	5.6	9.3	5.0×10^{-2}
E	17.6	4.1	12.0	2.1×10^{-2}
F	12.2	2.7	7.2	8.8×10^{-3}
Direction - South (S)				
Class[a]	**% Class[b]**	**Wind Speed[c] (m/sec)**	**% S Wind[d]**	**%s Comb.[e]**
A	0.4	2.1	6.0	2.4×10^{-4}
B	4.7	3.0	10.8	5.1×10^{-3}
C	11.5	4.2	14.4	1.7×10^{-2}
D	53.6	5.6	11.8	6.3×10^{-2}
E	17.6	4.0	17.6	3.1×10^{-2}
F	12.2	2.6	12.0	1.5×10^{-2}
Direction - South by Southwest (SSW)				
Class[a]	**% Class[b]**	**Wind Speed[c] (m/sec)**	**% SSW Wind[d]**	**%s Comb.[e]**
A	0.4	2.4	6.0	2.4×10^{-4}
B	4.7	3.1	8.6	4.0×10^{-3}
C	11.5	4.1	9.7	1.1×10^{-2}
D	53.6	5.6	5.5	2.9×10^{-2}
E	17.6	3.9	7.4	1.3×10^{-2}
F	12.2	2.6	6.3	7.7×10^{-3}

TABLE B-1
METEOROLOGICAL DATA FOR COLUMBIA, MISSOURI STABILITY CLASS INFORMATION, 1960 TO 1969
(CONT.)

Direction - Southwest (SW)				
Class[a]	**% Class[b]**	**Wind Speed[c] (m/sec)**	**% SW Wind[d]**	**%s Comb.[e]**
A	0.4	1.8	5.2	2.1×10^{-4}
B	4.7	3.0	9.2	4.3×10^{-3}
C	11.5	4.1	7.5	8.6×10^{-3}
D	53.6	5.4	3.5	1.9×10^{-2}
E	17.6	3.9	4.3	7.6×10^{-3}
F	12.2	2.5	6.0	7.3×10^{-3}
Direction - West by Southwest (WSW)				
Class[a]	**% Class[b]**	**Wind Speed[c] (m/sec)**	**% WSW Wind[d]**	**%s Comb.[e]**
A	0.4	2.2	6.0	2.4×10^{-4}
B	4.7	3.0	10.8	5.1×10^{-3}
C	11.5	4.3	9.0	1.0×10^{-2}
D	53.6	5.9	4.9	2.6×10^{-2}
E	17.6	3.9	5.7	1.0×10^{-2}
F	12.2	2.5	5.9	7.2×10^{-3}
Direction - West (W)				
Class[a]	**% Class[b]**	**Wind Speed[c] (m/sec)**	**% W Wind[d]**	**%s Comb.[e]**
A	0.4	1.8	3.4	1.4×10^{-4}
B	4.7	2.8	6.7	3.1×10^{-3}
C	11.5	3.9	6.2	7.1×10^{-3}
D	53.6	6.0	4.7	2.5×10^{-2}
E	17.6	3.7	5.3	9.3×10^{-3}
F	12.2	2.5	6.1	7.4×10^{-3}

TABLE B-1
METEOROLOGICAL DATA FOR COLUMBIA, MISSOURI STABILITY CLASS INFORMATION, 1960 TO 1969 (CONT.)

Direction - West by Northwest (WNW)				
Class[a]	**% Class[b]**	**Wind Speed[c] (m/sec)**	**% WNW Wind[d]**	**%s Comb.[e]**
A	0.4	2.1	4.3	1.7×10^{-4}
B	4.7	2.8	5.4	2.5×10^{-3}
C	11.5	4.3	5.1	5.9×10^{-3}
D	53.6	6.7	7.9	4.2×10^{-2}
E	17.6	4.0	5.5	9.7×10^{-3}
F	12.2	2.5	5.0	6.1×10^{-3}
Direction - Northwest (NW)				
Class[a]	**% Class[b]**	**Wind Speed[c] (m/sec)**	**% NW Wind[d]**	**%s Comb.[e]**
A	0.4	2.2	4.3	1.7×10^{-4}
B	4.7	2.9	4.4	2.1×10^{-3}
C	11.5	4.3	4.7	5.4×10^{-3}
D	53.6	7.1	8.8	4.7×10^{-2}
E	17.6	4.2	5.1	9.0×10^{-3}
F	12.2	2.5	3.6	4.4×10^{-3}
Direction - North by Northwest (NNW)				
Class[a]	**% Class[b]**	**Wind Speed[c] (m/sec)**	**% NNW Wind[d]**	**%s Comb.[e]**
A	0.4	2.3	1.7	6.8×10^{-5}
B	4.7	2.7	2.9	1.4×10^{-3}
C	11.5	4.1	3.0	3.5×10^{-3}
D	53.6	6.6	5.8	3.1×10^{-2}
E	17.6	4.0	3.6	6.3×10^{-3}
F	12.2	2.4	3.0	3.7×10^{-3}

TABLE B-1
METEOROLOGICAL DATA FOR COLUMBIA, MISSOURI STABILITY CLASS INFORMATION, 1960 TO 1969 (CONT.)

Direction - North (N)				
Class[a]	**% Class[b]**	**Wind Speed[c] (m/sec)**	**% N Wind[d]**	**%s Comb.[e]**
A	0.4	2.4	7.8	3.1×10^{-4}
B	4.7	2.7	4.8	2.3×10^{-3}
C	11.5	4.0	4.8	5.5×10^{-3}
D	53.6	6.0	6.2	3.3×10^{-2}
E	17.6	3.8	4.0	7.0×10^{-3}
F	12.2	2.5	5.8	7.1×10^{-3}

a. Stability class as defined by Pasquill's Categories (Ref. B.3).
b. Annual frequency distribution of stability class for all directions, or the total probability of occurrence for that class.
c. Average wind speed for stability class and wind direction.
d. Annual frequency distribution of wind direction for the specific stability class, or the probability of the wind direction given that the stability class exists.
e. %s comb. = (% class/100) × (% NNE/100), or the joint probability of the specific stability class and the specific direction occurring at the same time. Example: A conditional probability is one in which the probability of the events depends upon whether the other event has occurred (Ref. B.4).
P(A) = Probability of Class A Conditions = 0.4%.
P(N/A) = Probability of Wind Direction from N Given Class A Conditions = 7.8%.
P(AN) = Probability of having Class A Conditions and Wind Direction from N.
$P(AN) = P(A) \times P(N/A) = 3.1 \times 10^{-4}$.

B.3 Calculations

The following three equations were used to calculate the Ar-41 concentration and dose, along with the associated assumptions used for each case, at a distance x, downwind from the stack release point.

B.3.1 Effective Stack Height

The effluent exiting the stack carries momentum and buoyancy which potentially propels the effluent vertically beyond the stack. This leads to an effective release height which may be larger than the physical stack height. Calculations are based on the Pasquill-Gifford Model of determining stack release concentrations. This method is applicable for small volume releases (< 50 m3/sec), having a significant effluent exit velocity (> 10 m/sec), and a small temperature difference (< 50°C above ambient). The MURR gaseous effluent exit velocity at the normal ventilation flow rate of 14.4 m3/sec is about 17.7

m/sec. Buoyant forces due to a temperature difference between atmospheric and building air is considered negligible. The following equation was used to calculate the Effective Stack Height (H):

$$H = h + d\left(\frac{v}{\mu}\right)^{1.4}\left(1 + \frac{\Delta T}{T}\right);$$

where:

h = actual height (m), difference in elevation from release point to downwind site of dose calculation;

d = diameter of release point (m);

μ = average wind speed for specific stability class (m/sec);

v = exit velocity (m/sec);

ΔT = temperature difference between the stack air and the surrounding air, assumed to be 0; and

T = absolute temperature of the stack air.

Therefore,

$$H = h + d\left(\frac{v}{\mu}\right)^{1.4}.$$

MURR calculates effective stack height, for the purposes of determining dose from radionuclide emissions, as the difference in vertical elevation between the point of emission at the end of the MURR exhaust stack and the receptor height at the point of interest, plus the effective stack height calculated using the Davidson equation. This equation considers the stack diameter and exhaust velocity of the gases leaving MURR to calculate an injection height and thus an effective stack height into the atmosphere. Wind speed is also an input parameter into this formula as it is a function of the particular Pasquill atmospheric stability class that is being modeled for the general wind direction that is being used to calculate the offsite dose; thus, it is included in the equation.

G.A. Briggs notes on page 23 of his book, "Plume Rise," that the Davidson formula, "…often greatly underestimates observed rises…," this underestimation would cause the offsite dose calculations using the Pasquill-Guifford model to overestimate doses to the individual at the point of dose calculation interest. In fact, dose estimates generated using this model are not out of line with doses calculated using the COMPLY2 computer code which is used to determine annual doses (demonstrate compliance) to the nearest resident from MURR as part of the facility's annual National Emission Standards for Hazardous Air Pollutants (NESHAPS) compliance report. Additionally, using Briggs' own equations for

calculation of effective stack heights from the same reference book "Plume Rise," confirms that while the Davidson model underestimates effective stack heights, these underestimated effective stack heights lead to an overestimation of dose, thus providing a conservative approach to the offsite dose calculations.

B.3.2 Argon-41 Concentration

Data for the lateral (σ_Y) and the vertical (σ_Z) dispersion coefficients were obtained from Reference B.3. The following equation was used to calculate Ar-41 concentration:

$$\frac{\chi}{Q} = \frac{1}{\pi \times \sigma_Y \times \sigma_Z \times \mu} \times \exp\left[-\frac{1}{2} \times \left(\frac{Y^2}{\sigma_Y^2} + \frac{H^2}{\sigma_Z^2}\right)\right];$$

where:

χ = concentration at downwind site of dose calculation (µCi/ml or Ci/m^3);

Q = release rate (Ci/sec);

σ_Y = lateral dispersion coefficient at downwind site of dose calculation for specific stability class (m);

σ_Z = vertical dispersion coefficient at downwind site of dose calculation for specific stability class (m);

µ = average wind speed for specific stability class (m/sec);

Y = distance from plume centerline (m); and

H = effective stack height (m).

For maximum concentration:

$$\frac{\chi}{Q} = \frac{1}{\pi \times \sigma_Y \times \sigma_Z \times \mu} \times \exp\left[-\frac{1}{2} \times \left(\frac{H}{\sigma_Z}\right)^2\right].$$

Further, for the case of a ground release (i.e., H = 0):

$$\frac{\chi}{Q} = \frac{1}{\pi \times \sigma_Y \times \sigma_Z \times \mu}.$$

Considering decay, the equation becomes:

$$\frac{\chi}{Q} = \frac{e^{-\lambda t}}{\pi \times \sigma_Y \times \sigma_Z \times \mu};$$

where:

λ = decay constant for Ar-41 (sec^{-1}); and

t = time (sec);

= x/μ.

B.3.3 Annual Dose

The following equation was used to calculate the Annual Dose (D):

$$D = DCF \times \sum_i \chi_i \times (\%\ comb)_i;$$

where:

DCF = dose conversion factor (Ref. B.5);

= 8.84 x 10^{-3} (mrem-m^3/ρCi-Y) for Ar-41;

I = summation for overall stability classes; and

$(\%\ comb.)_i$ = relative frequency for stability class, i, and specific wind direction.

B.4 Maximum Individual Dose Estimates

In order to determine the maximum dose to an individual, the south wind direction was chosen as being the most probable wind direction. The annual exposure from Ar-41 was determined at the maximum release rate for two different distances: 150 meters north to the Emergency Planning Zone (EPZ) boundary (Ref. B.6), and at the nearest residence in relation to the facility (approximately 760 meters north). Elevations for these two sites were estimated from a University of Missouri topographical map (Figure B.1). Data and the maximum calculated dose estimates for these two locations are given in Tables B-2 and B-3, with an example calculation following the tables. The maximum annual dose at 150 and 760 meters was approximately 0.7 mrem/y and 4.2 mrem/y, respectively. The difference in relative plume height at these sites is what leads to the difference in dose rates.

TABLE B-2
MAXIMUM ANNUAL INDIVIDUAL DOSE AT 150 METERS

Location: 150 Meters Directly North Elevation at Man Height: 636 Feet (194 Meters)							
Class	**Eff. Height (m)**	**σ_Y (m)**	**σ_Z (m)**	**χ/Q (sec/m^3)**	**χ (μCi/ml or Ci/m^3)**	**%s Comb.**	**Dose with %s (mrem/y)**
A	35	33	23	6.27E-05	3.14E-09	2.40E-04	0.00
B	27	23	15	6.09E-05	3.04E-09	5.10E-03	0.08
C	23	17	11	4.55E-05	2.28E-09	1.70E-02	0.19
D	20	12	7	1.14E-05	5.71E-10	6.30E-02	0.18
E	23	8.5	5	4.76E-08	2.38E-12	3.10E-02	0.00
F	30	6	3.2	5.24E-22	2.62E-26	1.50E-02	0.00
						Total	0.46

TABLE B-3
MAXIMUM ANNUAL INDIVIDUAL DOSE AT 760 METERS

Location: 760 Meters Directly North Elevation at Man Height: 700 Feet (213 Meters)							
Class	**Eff. Height (m)**	**σ_Y (m)**	**σ_Z (m)**	**χ/Q (sec/m^3)**	**χ (μCi/ml or Ci/m^3)**	**%s Comb.**	**Dose with %s (mrem/y)**
A	16	170	270	3.30E-06	1.65E-10	2.40E-04	0.00
B	8	120	85	1.04E-05	5.18E-10	5.10E-03	0.01
C	4	85	52	1.71E-05	8.55E-10	1.70E-02	0.07
D	1	55	26	3.97E-05	1.99E-09	6.30E-02	0.63
E	4	42	18	1.03E-04	5.13E-09	3.10E-02	0.80
F	11	30	12	2.23E-04	1.12E-08	1.50E-02	0.84
						Total	2.35

Example Calculation:

Distance = 760 meters north;

Elevation = 700 feet (213 m);

Class E:

μ = 4.0 m/sec;

σ_Y = 41 meters;

σ_Z = 18 meters;

Effective Stack Height (H):

$$H = 209 + 1.02 \times \left(\frac{17.7}{4.0}\right)^{1.4} - 213;$$

= 4.2 meters.

Ar-41 Concentration:

$$\frac{\chi}{Q} = \frac{1}{\pi \times 41 \times 18 \times 4} \times \exp\left[-\frac{1}{2} \times \left(\frac{4.2}{18}\right)^2\right];$$

$= (1.08 \times 10^{-4}) \times (0.97);$

$= 1.05 \times 10^{-4}$ sec/m^3;

$\chi = (1.05 \times 10^{-4}$ sec/m$^3) \times (5.0 \times 10^{-5}$ Ci/sec);

$= 5.3 \times 10^{-9}$ Ci/m^3;

$= 5.3 \times 10^{-9}$ µCi/ml.

Class E occurs 17.6% of the time and, of that time, the wind blows from the South 17.6% of the time.

$$\%s\ comb = \left(\frac{17.6}{100}\right) \times \left(\frac{17.6}{100}\right);$$

= 0.031.

$$Dose_E = \left(5.3 \times 10^{-9} \frac{Ci}{m^3}\right) \times (0.031) \times \left(8.84 \times 10^{-3} \frac{mrem - m^3}{pCi - Y}\right) \times \left(10^{12} \frac{pCi}{Ci}\right);$$

= 1.4 mrem/y.

B.5 Normal Operational Releases

When this analysis was performed, the previous ten years of MURR operations had released approximately 1,100 Ci/y of Ar-41 with a stack flow rate of about 30,500 ft^3/ min. Production of Ar-41 is expected to remain relatively the same, and so the average Ar-41 concentration is predicted to be:

(30,500 ft^3/min) × (2.831 × 10^4 ml/ft^3) × (60 min/h) × (24 h/day) × (365 day/y)

= 4.54 × 10^{14} ml/y;

$(1{,}100\ \text{Ci/y}) \times (10^6\ \mu\text{Ci/Ci}) \times (1/(4.54 \times 10^{14})\ \text{y/ml})$

$= 2.42 \times 10^{-6}\ \mu\text{Ci/ml};$

which is approximately 69% of the Technical Specifications limit. Because the calculated dose estimates are proportional to the total amount of Ar-41 released, the dose estimates for actual operating conditions are easily calculated using the ratios of the stack release rates (given Ar-41 production remains constant). The actual operational radiation dose estimates are given in Table B-4.

TABLE B-4
OPERATIONAL RADIATION DOSE ESTIMATES

Person	Distance from Facility	Total Dose
Individual	150 meters	0.5 mrem/y
Individual	760 meters	3.0 mrem/y

B.6 Comparison of Risk

In the safety evaluation performed by the NRC in support of Amendment No. 12 (Ref. B.1), an individual located at the nearest residence was estimated to receive an annual average total body dose of 13 mrem/y based on the 1977/1978 release of 1,925 Ci/y and 29 mrem/y for the maximum estimate. Although the assumptions, data, and conditions for calculation are not fully described in NRC Amendment No. 12, estimated doses are greater than those predicted by the current assessment, which utilizes a more realistic model (effective stack height and stability class weighting) and better site-specific data (meteorological data). The NRC concluded "that there would be no significant environmental impact attributable" to an increase in the stack release limit to 350 AEC. With lower doses estimated for the current stack height and flow rate, it is also concluded that no significant environmental impact exists. The same conclusion applies to instantaneous release limits.

Another method of assessing risk from the estimated doses is to compare them to natural background dose rates. The average whole body dose to an individual in the United States is approximately 360 mrem/y (Ref. B.7). The estimated doses in terms of the percent of natural background are given in Table B-5.

TABLE B-5
ESTIMATED DOSES IN TERMS OF PERCENT OF NATURAL BACKGROUND

Person	Distance from Facility	Maximum Case	Normal Operation
Individual	150 meters	0.2 %	0.1 %
Individual	760 meters	1.2 %	0.8 %

Variations of this magnitude can be found in annual dose for populations living in different areas of the United States with no observable effects.

B.7 Conclusions

The intent of the analysis of normal operational releases is to indicate the maximum dose at two locations – the Emergency Planning Zone boundary and the nearest residence in relation to the facility. These two areas can reasonably be expected to be occupied on a permanent or semi-permanent basis. The Emergency Planning Zone boundary (150 meters north) is at a building occupied approximately 40 to 50 hours per week by the employees who work there. The location of the nearest resident is 760 meters due north.

The actual location of the highest calculated dose due to argon-41 emission occurs approximately 350 meters north of the facility in an unoccupied area that is under the control of the University of Missouri (within the site boundary). Near this location there are two maintenance sheds affiliated with the University of Missouri Golf Course. Using a conservative occupancy factor of 24% (40 hrs/week divided by 168 total hours per week) for this location a calculated annual dose due to argon-41 would be approximately 1.1 mrem [4.4 mrem (at 350 meters North) x 0.24]. If one were to assign occupancy factors of 24% to the occupant at the 150 meter point and 100% to the nearest resident located at the 760 meter point, to reflect the likelihood of a person being present at any one of the locations for a given amount of time, the highest likely dose anyone would receive would still be at the point of the nearest residence (760 meters north). Thus, this is where the highest offsite dose would most likely be received by a member of the public.

The estimated dose rates calculated using improved methods and data were no greater than those calculated from previous appraisals where impact was judged by the NRC to be not significant in environmental impact. Therefore, there was no significant reduction in safety as the result of the license renewal of continued operation at the current MURR stack release conditions.

The analysis of normal operational releases in Table B-4 assumed an annual average Ar-41 of 1100 Ci/y. The average concentration of Ar-41 over the period 2015 through 2024 was 979 Ci/y. Reduced use of the p-tubes would reduce this average annual concentration further. With the annual average Ar-41 release over the same 10-year period being less than the concentration used in the analysis above, impacts from normal operational releases do not require a reanalysis of Ar-41 releases.



FIGURE B.1 – UNIVERSITY OF MISSOURI TOPOGRAPHICAL MAP

APPENDIX C

THERMAL-HYDRAULIC TRANSIENT ANALYSIS

TABLE OF CONTENTS

C.0 THERMAL-HYDRAULIC TRANSIENT ANALYSIS 3
C.1 Introduction 3
C.2 Methodology for Transient Analysis of the Reactor System 3
C.2.1 RELAP5 Application 3
C.2.2 Modeling of MURR 4
C.2.3 RELAP5/MOD3.3 Input Data 10
ATTACHMENT 1 IN-POOL COMPONENTS OF THE PRIMARY COOLANT LOOP THERMAL-HYDRAULIC MODEL 13
Cold Leg Components 13
Core Components 13
Hot Leg Components 14
In-Pool Heat Exchanger Loop 14
ATTACHMENT 2 EX-POOL COMPONENTS OF THE PRIMARY COOLANT LOOP THERMAL-HYDRAULIC MODEL 16
ATTACHMENT 3 COMPONENTS OF THE POOL COOLANT LOOP THERMAL-HYDRAULIC MODEL 19
ATTACHMENT 4 CONTROL VARIABLES AND REACTOR TRIPS IN THE THERMAL-HYDRAULIC MODEL 20
Control Variables 20
Reactor Trips 21

LIST OF TABLES

Table C-1 POWER DENSITY PROFILE OF THE MODELED CORE (fissions per cm^3-sec) 7
Table C-2 POWER DENSITY PROFILE OF THE MODELED CORE (fissions per cm^3-sec) (cont.) 8
Table C-3 RELATIVE DECAY POWER VERSUS TIME POST TRIP 10

LIST OF FIGURES

Figure C.1 – DETAILED RELAP5 MODEL OF THE MURR PRIMARY COOLANT LOOP 22
Figure C.2 – IN-POOL PRIMARY COOLANT LOOP PIPING 23
Figure C.3 – EX-POOL PRIMARY COOLANT LOOP PIPING - HOT AND COLD LEGS 24
Figure C.4 – DETAILED RELAP5 MODEL OF THE MURR CORE 25
Figure C.5 – DETAILED RELAP5 MODEL OF THE MURR BULK REACTOR POOL AND POOL COOLANT LOOP 26
Figure C.6 – AVERAGE FISSIONS/CC-S IN THE FUEL PLATE SECTIONS 27
Figure C.7 – ACTUAL VERSUS MODEL PREDICTED PRIMARY COOLANT PUMP COASTDOWN 28

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C.0 THERMAL-HYDRAULIC TRANSIENT ANALYSIS

C.1 Introduction

The potential for fuel damage has been evaluated for various accidents and transients postulated for the University of Missouri Research Reactor (MURR). The criterion for 'no fuel damage' is that the fuel plate peak temperatures do not exceed the fuel plate minimum blister temperature [in excess of 900 °F (484 °C)]. All MURR fuel plates are tested at 900 °F (484 °C) during the fabrication process to verify that no cladding blisters will form. The computer code RELAP5/MOD3.3 (Reference C.1) was used to perform the thermal-hydraulic transient analyses needed to determine the peak fuel plate temperatures reached for each of the accidents analyzed. The MURR RELAP5/MOD3.3 Model includes all 24 fuel plates and 25 coolant channels. The pre-accident assumed steady-state power level is given in the accident analysis. The power distribution within the various fuel plates as represented in the RELAP5/MOD3.3 Model is presented in Section C.2.2.6. The complete RELAP5/MOD3.3 Model for MURR is described in Section C.2.2.

C.2 Methodology for Transient Analysis of the Reactor System

C.2.1 RELAP5 Application

RELAP5 is a light- and heavy-water reactor transient analysis code developed by the Idaho National Engineering and Environmental Laboratory (INEEL) for the U.S. Nuclear Regulatory Commission (NRC) and is now maintained by Information Systems Laboratories, Inc. (Reference C.1). It 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. RELAP5 is one of the most widely used system codes for analyzing reactor accidents/transients. The Department of Energy (DOE) has utilized RELAP5 to analyze design basis accidents for the Advanced Test Reactor (ATR) and High Flux Isotope Reactor (HFIR) in their respective Safety Analysis Reports (SARs). RELAP5 has also been applied to the analysis of the High Flux Beam Reactor (HFBR) and the National Institute of Standards and Technology Reactor (NBSR). The MURR fuel element and reactor design is closest in design to the ATR, but can also be related to other research reactor designs and fuel element geometries (MTR plate-type).

RELAP5/MOD3.3, the current version of the code system, was developed with the objective of creating a code version suitable for the analysis of postulated accidents in water reactor systems, including both large- and small-break loss of coolant accidents (LOCAs) as well as a full range of operational transients. The hydrodynamic model in RELAP5/MOD3.3 is a one-dimensional, transient, two-fluid model for flow of a two-phase steam-water mixture. The non-equilibrium transient two-fluid model is represented by the conservation equations of mass, momentum, and energy for each phase. The steam phase can contain non-condensable components and the water phase can have a solute component. Special process models are available to handle choked flow, abrupt area changes, and counter-current flow.

Metal components are modeled by heat structures with internal heat generation. Heat transfer within the structures is by one-dimensional heat conduction. A full boiling curve is implemented in the code

for modeling heat transfer between heat structures and the coolant. Reactor power and decay power are calculated by a point kinetics model with reactivity feedback. In RELAP5/MOD3.3, a hydraulic system is constructed by connecting fluid components, such as pipes, valves, pumps, etc., in series or in parallel. Geometric data and the initial thermodynamic state of the fluid are required for the interconnecting components. The initial flow rate is required at the junctions between two components. Heat structures are defined with the heat transfer surface facing the coolant in a hydraulic component. Time varying boundary conditions can be specified in terms of fluid flow rate or the thermodynamic state of the fluid. Control system components are available in RELAP5/MOD3.3 to model system dynamic behavior such as component trips and the evaluation of system variables. The benchmarking performed on the RELAP model for the LOCA and LOFA analyses is described in the following reference. (Reference C.4)

NRC Request for Additional Information, RAI 13.4.c., RELAP5 Application. (May 6, 2010)
RELAP5/MOD3.3 had a fundamental error in the point kinetics model that has recently been fixed. Does the version of the code used in the thermal-hydraulic analysis have the fixes implemented? If not, confirm that the analysis model is giving results consistent with the transient, e.g. by checking results as a function of time step or with another stand-alone point kinetics model. (Reference C.2)

MURR response to NRC RAI 13.4.c. (September 3, 2010)
In the Extended Life Aluminide Fuel (ELAF) analysis, MURR benchmarked both PARET and RELAP5 reactivity transient analyses against one of the SPERT-IV tests (Special Power Excursion Reactor Test). The -12/25 core of the SPERT-IV tests was selected for comparison because the test was performed under conditions very similar to that of MURR - forced coolant circulation in a low pressure and low temperature environment using plate-type fuel. The comparisons of the power and temperature transients for a $1.14 step reactivity insertion were modeled. The PARET code provided a response to the reactivity transient, which was conservative but close to measured values, while RELAP5 significantly over-predicted the fuel temperatures. The results were presented at the American Nuclear Society (ANS) 1989 Annual Meeting in a paper titled “PARET/ANL and RELAP5/MOD2 Benchmarking Comparison Against the SPERT-IV Test Data” by S. S. Kim and J. C. McKibben (University of Missouri Research Reactor Facility, June 1989). Based on this analysis, MURR has only used PARET for reactivity transient analyses.

The MURR RELAP analysis in the SAR used SCDAP RELAP5 MOD3.3, as noted in Section C.1. The use of RELAP5 in the Loss of Flow Accident (LOFA) and Loss of Coolant Accident (LOCA) analyses does not include any point kinetics analysis. RELAP is only used to model the thermal-hydraulic transient. The MURR RELAP5 models handled the reactor decay power as a function of time after a SCRAM is actuated. The modeling is based, conservatively, on a longer operating time than what occurs in the MURR core. This provides a higher than actual decay heat generation. (Reference C.3).

C.2.2 Modeling of MURR

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 RELAP5/MOD3.3 model of MURR simulates the transport of heat and coolant in both the primary and pool coolant systems. The reactor pressure vessels, the primary coolant loop, the bulk reactor pool, and the pool coolant loop are represented by a series of hydrodynamic volumes. Fuel plates in the core region are represented by heat structures. Steady-state and decay power are controlled as time-dependent variables in RELAP5/MOD3.3. A schematic block diagram showing the main components of the MURR primary coolant loop is shown in Figure C.1.

The discussion of the MURR model will be grouped into six sub-sections: the reactor pressure vessels, the primary coolant loop, the secondary coolant loop, the bulk reactor pool and pool coolant loop, valve V547/anti-siphon system, and the fuel plates. A component number, as defined in the RELAP5/MOD3.3 input deck, is used to identify each hydrodynamic volume and heat structure modeled.

C.2.2.1 Reactor Pressure Vessels

Two cylindrical reactor pressure vessels provide a fixed geometry for the reactor fuel region consisting of eight (8) fuel elements having identical physical dimensions placed vertically around the annulus formed between the pressure vessels.

The pressure vessels, including the reactor core assembly, 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.

C.2.2.2 Primary Coolant Loop

Parallel flow paths in the MURR primary coolant loop are modeled by combining them into a single effective flow path. This applies to the split in the primary coolant loop into two branches through the two primary coolant pumps and two primary coolant heat exchangers. This simplification does not have a significant effect on the RELAP5/MOD3.3 analysis since the parallel flow paths are thermally and hydraulically similar. Additionally, after the first few seconds of the transient, the primary coolant piping with parallel flow paths are no longer involved in flow through the core. Figures C.2 and C.3 depict the layout of the hot and cold legs of the primary coolant loop. The specific components used to define the primary coolant loop are described in Attachments 1 and 2.

C.2.2.3 Secondary Coolant Loop

The secondary coolant loop is modeled simply as a once-through circuit. At one end, a source supplies the cooling water to the primary coolant heat exchangers. After the heat exchangers, the secondary coolant (light water) flows to a sink. The specific components used to define the secondary coolant loop are described in Attachments 1 and 2.

C.2.2.4 Bulk Reactor Pool and Pool Coolant Loop

The bulk reactor pool is an aluminum-lined structure 10 feet (3 m) in diameter and 30 feet (9.1 m) deep containing approximately 20,000 gallons (75,708 l) of light water.

The pool coolant loop is modeled by a single effective flow path combining the two pool coolant pumps and pool coolant heat exchanger. Again, this simplification does not have a significant effect on the RELAP5 analysis since the parallel flow paths are thermally and hydraulically similar. The interaction of the pool with the reactor pressure vessels, control rod gap, and the graphite reflector is explicitly modeled. A detailed representation of the RELAP/MOD3.3 model of the bulk reactor pool and pool coolant loop is shown in Figure C.5. The specific components used to define the pool coolant loop are described in Attachment 3.

C.2.2.5 Valve V547/Anti-Siphon System

The reflector plenum natural convection valve V547 (Butterfly 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 ensure 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 V547 can be left in the open position while increasing normal pool flow to provide sufficient cooling to these regions. Since the Butterfly Valve is left open at all times, it is modeled in RELAP5/MOD3.3 as an open pipe.

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 primary coolant system piping external to the reactor pool.

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, breaking any potential siphon which may have been created by the pipe rupture. The RELAP5/MOD3.3 Model incorporates a pressurized air volume at the appropriate point in the primary coolant system for injection upon detection of low pressure in the primary coolant loop.

C.2.2.6 Fuel Elements

The eight (8) MURR fuel elements associated with a full-core loading are represented by alternating concentric annuli geometrically representing fuel plates and coolant channels, respectively. Rectangular heat structures are used to represent the MURR fuel element plates. Each fuel plate is associated with four heat structures (top 7.75 inches, the 5 inches above core centerline, the 5 inches below core centerline, and the bottom 7.75 inches). The RELAP5/MOD3.3 Model of the MURR core is shown in Figure C.4. The top section of the fuel plates are heat structures 601 through 624, the second section

are heat structures 626 through 649, the third section are heat structures 651 through 674, and the bottom section are heat structures 676 through 699. The average fissions per cm^3-second for each section of the 24 fuel plates are shown in Figure C.6. Tables C-1 and C-2 provide the power density profile of the modeled core and the RELAP5/MOD3.3 power profile of the modeled core, respectively. The tables present fission rate per cm^3 and percentage power per fuel plate section. The size varies between the plates and sections. As can be seen in Figure C.6, the third section is the peak section for all 24 fuel plates. The actual peak point within the third sections varies from 1.5% to 2.7% above the section average and averages 2.1% above. Figure C.6 and Tables C-1 and C-2 are for normal 10-MW steady-state operation. Table C-3 provides the total reactor power level as a function of time post reactor trip, assuming 120 days of full-power operation.

TABLE C-1
POWER DENSITY PROFILE OF THE MODELED CORE (FISSIONS PER CM^3-SEC)

Plate	Section 1 (Top 7.75" (19.685 cm))	Section 2 (Next 5" (12.7 cm))	Section 3 (Next 5" (12.7 cm))	Section 4 (Bot. 7.75" (19.685 cm))	Peak
1	9.57E+13	1.39E+14	1.60E+14	1.36E+14	1.63E+14
2	8.10E+13	1.19E+14	1.38E+14	1.17E+14	1.41E+14
3	7.25E+13	1.08E+14	1.25E+14	1.06E+14	1.28E+14
4	6.23E+13	9.39E+13	1.09E+14	9.17E+13	1.11E+14
5	5.61E+13	8.46E+13	9.79E+13	8.23E+13	9.97E+13
6	5.12E+13	7.79E+13	9.08E+13	7.65E+13	9.26E+13
7	4.76E+13	7.30E+13	8.54E+13	7.21E+13	8.72E+13
8	4.49E+13	6.94E+13	8.19E+13	6.85E+13	8.37E+13
9	4.32E+13	6.68E+13	7.83E+13	6.63E+13	8.01E+13
10	4.14E+13	6.45E+13	7.61E+13	6.41E+13	7.74E+13
11	4.18E+13	6.54E+13	7.68E+13	6.45E+13	7.83E+13
12	4.09E+13	6.45E+13	7.65E+13	6.45E+13	7.83E+13
13	4.00E+13	6.41E+13	7.61E+13	6.41E+13	7.74E+13
14	3.92E+13	6.36E+13	7.54E+13	6.32E+13	7.65E+13
15	3.83E+13	6.32E+13	7.54E+13	6.32E+13	7.65E+13

TABLE C-2
POWER DENSITY PROFILE OF THE MODELED CORE (FISSIONS PER CM^3-SEC) (CONT.)

Plate	Section 1 (Top 7.75" (19.685 cm))	Section 2 (Next 5" (12.7 cm))	Section 3 (Next 5" (12.7 cm))	Section 4 (Bot. 7.75" (19.685 cm))	Peak
16	3.83E+13	6.32E+13	7.65E+13	6.45E+13	7.83E+13
17	3.78E+13	6.41E+13	7.85E+13	6.63E+13	8.01E+13
18	3.74E+13	6.54E+13	8.17E+13	6.90E+13	8.37E+13
19	3.83E+13	6.85E+13	8.63E+13	7.34E+13	8.81E+13
20	3.92E+13	7.21E+13	9.21E+13	7.79E+13	9.43E+13
21	4.00E+13	7.65E+13	1.00E+14	8.54E+13	1.02E+14
22	4.18E+13	8.37E+13	1.11E+14	9.39E+13	1.14E+14
23	4.49E+13	9.26E+13	1.25E+14	1.06E+14	1.28E+14
24	4.98E+13	1.06E+14	1.45E+14	1.25E+14	1.49E+14

TABLE C-2
RELAP5 POWER PROFILE OF THE MODELED CORE

Plate	Section 1 (Top 7.75" (19.685 cm))	Section 2 (Next 5" (12.7 cm))	Section 3 (Next 5" (12.7 cm))	Section 4 (Bottom 7.75" (19.685 cm))
1	0.948%	0.986%	1.130%	1.349%
2	0.850%	0.894%	1.032%	1.228%
3	0.803%	0.855%	0.992%	1.173%
4	0.726%	0.782%	0.909%	1.069%
5	0.686%	0.739%	0.856%	1.008%
6	0.656%	0.713%	0.831%	0.982%
7	0.638%	0.699%	0.818%	0.967%
8	0.629%	0.694%	0.818%	0.959%
9	0.629%	0.695%	0.815%	0.966%

TABLE C-2
RELAP5 POWER PROFILE OF THE MODELED CORE (CONT.)

Plate	Section 1 (Top 7.75" (19.685 cm))	Section 2 (Next 5" (12.7 cm))	Section 3 (Next 5" (12.7 cm))	Section 4 (Bottom 7.75" (19.685 cm))
10	0.627%	0.698%	0.824%	0.971%
11	0.658%	0.735%	0.863%	1.016%
12	0.668%	0.752%	0.892%	1.053%
13	0.677%	0.774%	0.919%	1.083%
14	0.685%	0.795%	0.942%	1.105%
15	0.692%	0.815%	0.973%	1.142%
16	0.714%	0.842%	1.020%	1.204%
17	0.728%	0.880%	1.079%	1.276%
18	0.741%	0.926%	1.156%	1.367%
19	0.781%	0.998%	1.258%	1.498%
20	0.822%	1.080%	1.380%	1.634%
21	0.864%	1.179%	1.542%	1.843%
22	0.927%	1.323%	1.756%	2.080%
23	1.022%	1.503%	2.027%	2.418%
24	1.162%	1.771%	2.412%	2.906%

TABLE C-3
RELATIVE DECAY POWER VERSUS TIME POST TRIP

Time (sec)	Relative Power	Time (sec)	Relative Power
0.00	1.0000	70.00	0.0314
0.30	0.1792	80.00	0.0305
0.70	0.1145	90.00	0.0298
1.00	0.0943	100.00	0.0292
2.00	0.0846	120.00	0.0281
3.00	0.0771	180.00	0.0278
5.00	0.0668	200.00	0.0274
6.00	0.0630	240.00	0.0259
7.00	0.0598	300.00	0.0242
8.00	0.0572	400.00	0.0221
9.00	0.0549	420.00	0.0218
10.00	0.0529	540.00	0.0201
20.00	0.0410	600.00	0.0195
30.00	0.0373	800.00	0.0178
40.00	0.0352	1000.00	0.0166
50.00	0.0336	2000.00	0.0133
60.00	0.0324	4000.00	0.0106

C.2.3 RELAP5/MOD3.3 Input Data

C.2.3.1 Geometry

Much of the RELAP5/MOD3.3 input is for the dimensions of the hydrodynamic volumes. The geometric inputs for the MURR are based on plant drawings and on-site walk downs. Figures C.2 and C.3 summarize the dimensions of the primary coolant loop piping. All piping is dimensioned as it exists in the MURR system with the exception of the stated modification whereby parallel 8-inch (20.32-cm) piping is replaced in the RELAP5/MOD3.3 model with a single 12-inch (30.48-cm) pipe.

C.2.3.2 Primary Coolant Pump

The two primary coolant circulation pumps are horizontal, centrifugal, single-stage pumps that are direct-connected to 125-HP (93.2-kW) drive units through flexible couplings. These pumps are modeled as a single pump with the characteristics of the combined pumps supplying approximately 3,750 gpm (14,195 lpm) with sufficient discharge head to overcome system pressure drop losses. Pump coastdown curves as measured at MURR and as obtained from the RELAP5/MOD3.3 Model are presented in Figure C.7. These curves demonstrate that the modeled pump coastdown is conservative with respect to actual reactor measurements. This is especially true during the first 20 seconds after a pump trip which is demonstrated as the critical time post trip in which fuel integrity is demonstrated to be maintained.

C.2.3.3 Primary Coolant Heat Exchanger

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 one pass through the U-tube side of the heat exchanger with a velocity of no greater than 7 feet/second (2.13 m/s). 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 heat exchangers are modeled in RELAP5/MOD3.3 as a single heat transfer entity which transfers energy into an infinite sink at the design rate of the combined heat exchangers.

C.2.3.4 Core Flow Distribution

Flow distribution through the core is modeled in RELAP5/MOD3.3 initially as proportional to the effective cross-section of each of the channels through the core. These initial flows are then allowed to adjust to steady-state flow rates as established though the hydrodynamic properties established in RELAP5/MOD3.3.

C.2.3.5 Decay Heat Removal System

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, 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 4.5 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.

The decay heat removal system is modeled in RELAP5/MOD3.3 by allowing core thermal energy to be transferred to the bulk reactor pool and subsequently removed via circulation of the pool coolant loop.

C.2.3.6 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.), all of which ensure 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, breaking any potential siphon which may have been created by the pipe rupture.

The anti-siphon system is modeled by RELAP/MOD3.3 by establishing a pressurized air volume that is released into the outlet piping during sequencing of each modeled event.

ATTACHMENT 1
IN-POOL COMPONENTS OF THE PRIMARY COOLANT LOOP THERMAL-HYDRAULIC MODEL

Cold Leg Components

Component V135 - Cold Leg Pipe after Valve V507B to Check Valve V502
This piping component represents the primary coolant loop between the cold leg isolation valve and the check valve prior to the reactor pressure vessel inlet.

Component CV136 - Check Valve V502
This component represents check valve V502.

Component V137 - Cold Leg Pipe after Check Valve V502 to Reactor Pressure Vessel Inlet
A single volume representing the cold leg pipe after check valve V502 and prior to the reactor pressure vessel inlet.

Component V139 - Reactor Pressure Vessel Inlet
A branch element which represents the pressure vessel inlet. It has outputs both to the core and to the emergency in-pool heat exchanger loop.

Core Components

Component V501 - Upper Core Plenum
The upper plenum occupies the region above the fuel elements and channels and serves as the collection point of the individual flow channels through the core.

Components V502 through V526 - Core Flow Channels
Individual flow channel annuli are modeled that represent the flow between each of the individual fuel plates of all eight fuel elements along with the channels inside and outside the elements. V502 is the channel between the inner pressure vessel and fuel plate number-1. V526 is the channel between the outer pressure vessel and fuel plate number-24. The coolant channels are modeled to accept the heat output of the fuel elements which have been proportioned radially and axially in accordance with the core flux profile.

Components J530 through J545 - Upper Junctions
The inlets to the fuel element channels are represented by single junctions in conjunction with the branch junctions of the core upper plenum.

Components J550 through J565 - Lower Junctions
The outlets from the fuel element channels are represented by single junctions in conjunction with the branch junctions of the core lower plenum.

Component V575 - Lower Core Plenum
The lower plenum occupies the region below the fuel elements and channels and serves as the distribution point for the individual flow channels through the core.

Component J580 - Lower Plenum (Reactor Pressure Vessel) Outlet
Single junction which connects the lower plenum with the hot leg outlet pipe.

Hot Leg Components

Component V100 - Hot Leg Pipe from Reactor Pressure Vessel
Piping component exiting the lower plenum.

Component V101 - Hot Leg Branch to In-Pool Heat Exchanger Loop
Continuing piping component which connects the hot leg exiting the reactor with the emergency in-pool heat exchanger loop and anti-siphon system.

Component V102 - Hot Leg Pipe to Isolation Valve V507A
Piping component continuing to isolation valve V507A.

In-Pool Heat Exchanger Loop

Component V401 - In-Pool Heat Exchanger Inlet
Pipe component from the primary coolant loop up to the in-pool heat exchanger isolation valve.

Component IV402 - Motor Controlled In-Pool Heat Exchanger Isolation Valve V546
Spring-to-open, air-operated-to-close automatic valve modeled as a motor controlled isolation valve.

Component V405 - In-Pool Heat Exchanger
Piping component serving as the in-pool heat exchanger. An embedded heat structure serves to transfer heat from the loop to the bulk reactor pool water.

Component V406 Branch Connection to Anti-Siphon System
Branch component which continues the in-pool heat exchanger loop but allows connection to the anti-siphon system.

Component V407 - End Pipe
Single volume serves as primary coolant inlet path from the pool when the reactor is operating in natural circulation (Mode III). In Mode III, the primary outlet path is out the top of the pressure vessel with no pressure vessel head installed.

Component IV449 - Motor Controlled Anti-Siphon Isolation Valve V543
Spring-to-open, air-operated-to-close automatic valve modeled as a motor controlled isolation valve.

Component V460 - Anti-Siphon Pressure Tank

Single volume containing a pressurized air volume that is released into the circulating system upon activation of the anti-siphon isolation valve.

Component V104 - Motor Controlled Isolation Valve V507A

This component is a spring-to-close, air-operated-to-open automatic valve modeled as a motor controlled isolation valve in the hot leg. It is one of the two main isolation valves which isolates the in-pool primary coolant piping that contains the core from the rest of the primary piping in the event of a loss of coolant accident.

Component V105 - Hot Leg Pipe after Isolation Valve V507A and Before Two Loop Branch

This piping component represents the primary coolant loop between the hot leg isolation valve and the piping size reduction point prior to the pump inlet.

Component J106 – Junction

Junction which connects volume 105 to volume 107.

Component V107 - Pump Inlet

This component represents the inlet volume to the primary coolant pump.

Component P108 - Primary Coolant Pump

This pump component represents the two primary coolant pumps combined.

Component V109 - Pump Outlet

This component represents the outlet volume from the primary coolant pump.

Component CV110 - Check Valve V517

Check valve V517 after the primary coolant pump.

Component V111 - Primary Coolant Piping

This piping component represents the primary coolant loop between the hot leg check valve V517 and the primary heat exchanger inlet junction.

Component J112 - Primary Coolant Heat Exchanger Inlet Junction

Single junction connecting the primary piping to the modeled heat exchanger.

Component V113 - Primary Coolant Heat Exchanger

Pipe component with modeled heat structures which provide transfer of heat from the primary coolant to an ultimate heat sink.

Component J114 - Primary Coolant Heat Exchanger Outlet Junction

Single junction connecting the primary coolant piping to the modeled heat exchanger.

Component V115 - Primary Coolant Piping

Pipe component from the heat exchanger to the flow control valve.

Component MV116 - Flow Control Valve V420

Valve is modeled as a motor-controlled valve to throttle primary coolant in order to maintain flow within normal parameters. The valve is locked in its present position upon initiation of a transient.

Component V131 - Primary Coolant Piping

Pipe component from the flow control valve to the inlet point for the pressurizer.

Component V132 - Pressurizer Output Branch

Branch component which connects the pressurizer line to the primary coolant piping.

Component V133 - Primary Coolant Piping

Pipe component continuing to cold leg isolation valve V507B.

Component MV134 - Motor Controlled Isolation Valve V507B

This component is a spring-to-close, air-operated-to-open automatic valve modeled as a motor controlled isolation valve in the cold leg. It is one of the two main isolation valves which isolates the in-pool primary coolant piping that contains the core from the rest of the primary piping in the event of a loss of coolant accident.

Component V801 - Pressurizer Surge Line

Line from pressurizer control valve to the primary piping.

Component V802 - Pressurizer to Surge Line Junction

Pressurizer control valve.

Component V803 - Pressurizer Tank Piping

Pipe component from pressurizer to pressurizer control valve.

Component J804 - Pressurizer Junction

Junction component connecting pressurizer tank to the pressurizer tank piping.

Component V805 - Pressurizer Tank

Time dependent volume providing constant pressure to the primary coolant loop.

Component V898 - Branch Connecting

Branch allowing connection of non-physical valves for LOCA simulation.

Component TV899 & TV902 - Non-Physical Cold Leg Valves Modeled to Enable LOCA Simulation

Trip valves for initiation of cold leg LOCA.

Components V900 & V901 - Establishing Containment Volume for Cold Leg Break Simulation

Time dependent volumes to allow for fluid and pressure release during a modeled transient.

Component V903 - Branch Connecting

Branch allowing connection of non-physical valves for LOCA simulation.

Components TV904 & TV 906 - Non-Physical Hot Leg Valves Modeled to Enable LOCA Simulation

Trip valves for initiation of hot leg LOCA.

Components V905 & V907 - Establishing Containment Volume for Hot Leg Break Simulation

Time dependent volumes to allow for fluid and pressure release during a modeled transient.

ATTACHMENT 3
COMPONENTS OF THE POOL COOLANT LOOP THERMAL-HYDRAULIC MODEL

Component V700 - Pool Supply
Pipe component from pool coolant pump to the pool. The pool heat exchanger is modeled as a heat structure in a pipe section which transfers heat to an infinite heat sink.

Component V702 – Pool
Branch component as input for through-core components. This branch component is sized to represent the significant pool volume

Component V703 - Setting Boundary Condition for Open Pool
Time dependent volume which establishes the pool surface at atmospheric pressure.

Component J704 - Interface to Open Pool
Junction between the modeled open atmosphere and the pool surface.

Component V705 - Water Island
Pipe component representing flow through the center test hole.

Component V708 - Pipe to Pool Coolant Pump
Pipe component from the lower plenum to the pool coolant pump.

Component P710 - Pool Coolant Pump
Pump component establishing pool flow.

Component V715 - Control Blade Gap Flow Channel
Pipe component representing flow through the control blade gap external to the pressure vessels.

Component V720 - Lower Flow Plena for Pool Outlet
Branch collecting flow from the various through-core or bypass elements.

Component V725 - Graphite Flow Channels
Pipe component representing flow through the sample positions in the graphite reflector.

Component V735 - Butterfly Valve Flow Channel
Pipe component modeled as straight path since valve V547 is locked open.

ATTACHMENT 4
CONTROL VARIABLES AND REACTOR TRIPS IN THE THERMAL-HYDRAULIC MODEL

Control Variables

5	Overall Core Coolant Mass
7	Core Void Fraction
8	Core Void Fraction (Part of 7)
9	Mass in Upper Plena
10	Mass in Lower Plena
13	Total Heat Transfer to Coolant in the Core
14	Total Heat Transfer to Core Coolant (Part of 13)
15	Total Water Mass in Pressure Vessel
16	Total Heat Transfer Through Inner and Outer Pressure Vessel Walls
17	Total Heat Transfer into Pool Coolant Loop
18	Total Heat Transfer (Power)
20	Differential Pressure (DPS-929)
21	Differential Pressure V100 - V101
22	Core Differential Pressure
23	Vessel Differential Pressure
25	Differential Pressure V115 - V131
26	Differential Pressure (DPS-928)
30	Heat Transfer through Loop Heat Exchanger
31	Heat Transfer to Graphite
32	Heat Transfer to Pool
33	Heat Transfer to Anti-Siphon Pressure Tank
42	Heat Transfer to Water Island in Center Test Hole
43	Heat Transfer to Pool Water Through Hot Leg Piping Wall
44	Heat Transfer to Pool Water Through Cold Leg Piping Wall
45	Heat Transfer out of In-Pool Heat Exchanger
46	Total Heat Transfers to Pool Water (Not including in-pool HX and piping)
47	Total Heat Transfer to Pool Water
48	Temperature Differential Between Inlet and Outlet of In-Pool Heat Exchanger
51	Heat Transfer through Primary Coolant Heat Exchanger
52	Power Out
53 - 57	Total Coolant Channel Water Mass
109	Total Water Mass in the Hot Leg (Not including the down leg)
110	Total Water Mass in the Cold Leg (Not including those before check valve V502)

Reactor Trips

501	Simulation Initiation
503	Low Flow Indication
504	Forever False Control Condition
505	Low Pressure Indication
507 - 508	Primary Coolant Loop Flow Control
512	Closure Delay for Valves V543A/B
513	Closure Delay for Valves V546A/B
515	Control Rod Drop Delay
516	Closure Delay for Valves V507A and V507B
605	Detect Low Pressure after Simulation Start
607 - 608	Freeze Throttle Valve after Simulation Start
609	Low Flow or Valves V507A or V507B Off Open Seat
612	Valves V543A/B Open
613	Valves V546A/B Open
615	Control Rod Drop
616	Valves V507A and V507B Close
619	Prevent Valve Action When Not Part of Simulation

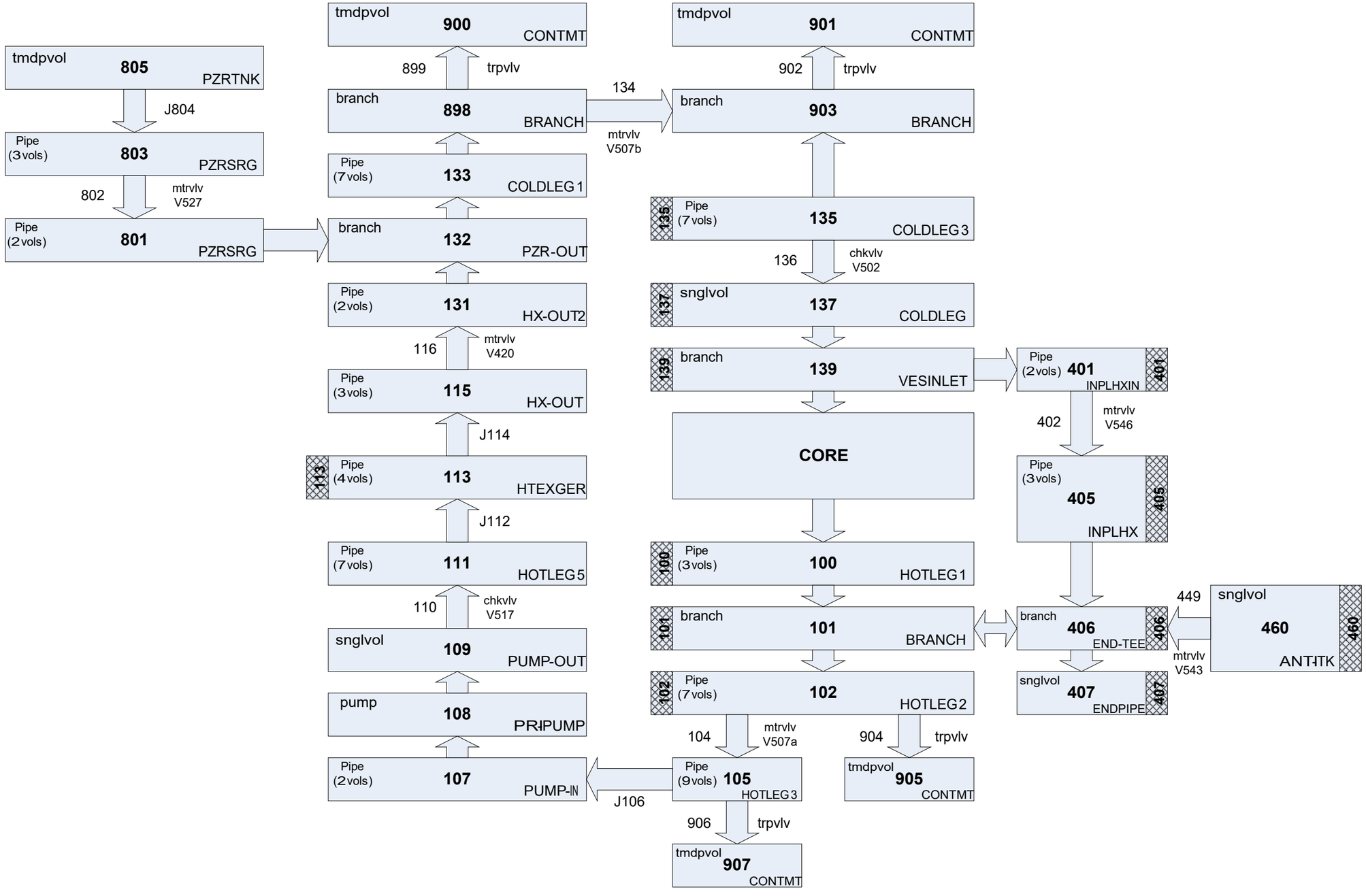


FIGURE C.1 – DETAILED RELAP5 MODEL OF THE MURR PRIMARY COOLANT LOOP

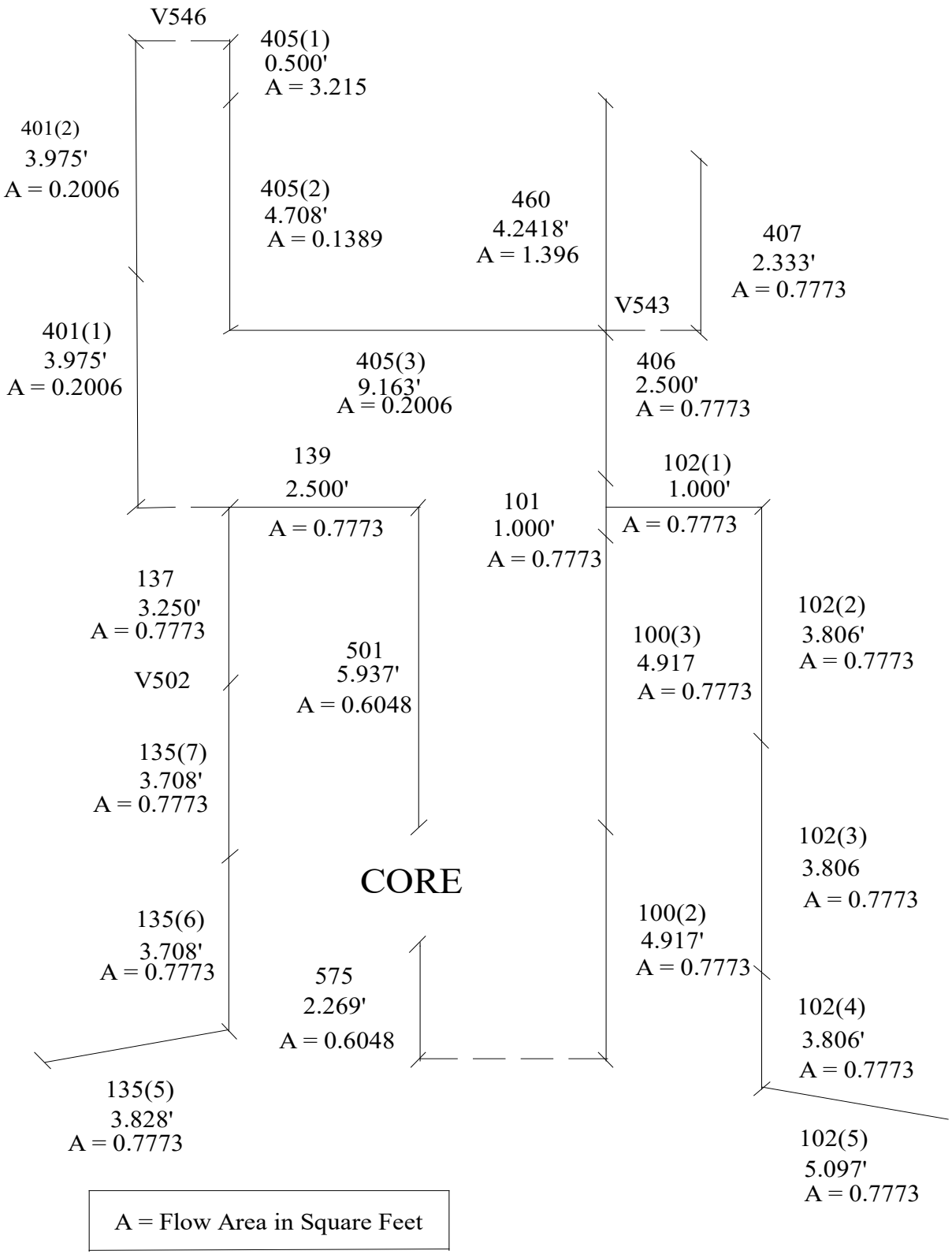


FIGURE C.2 – IN-POOL PRIMARY COOLANT LOOP PIPING

FIGURE C.3 – EX-POOL PRIMARY COOLANT LOOP PIPING - HOT AND COLD LEGS

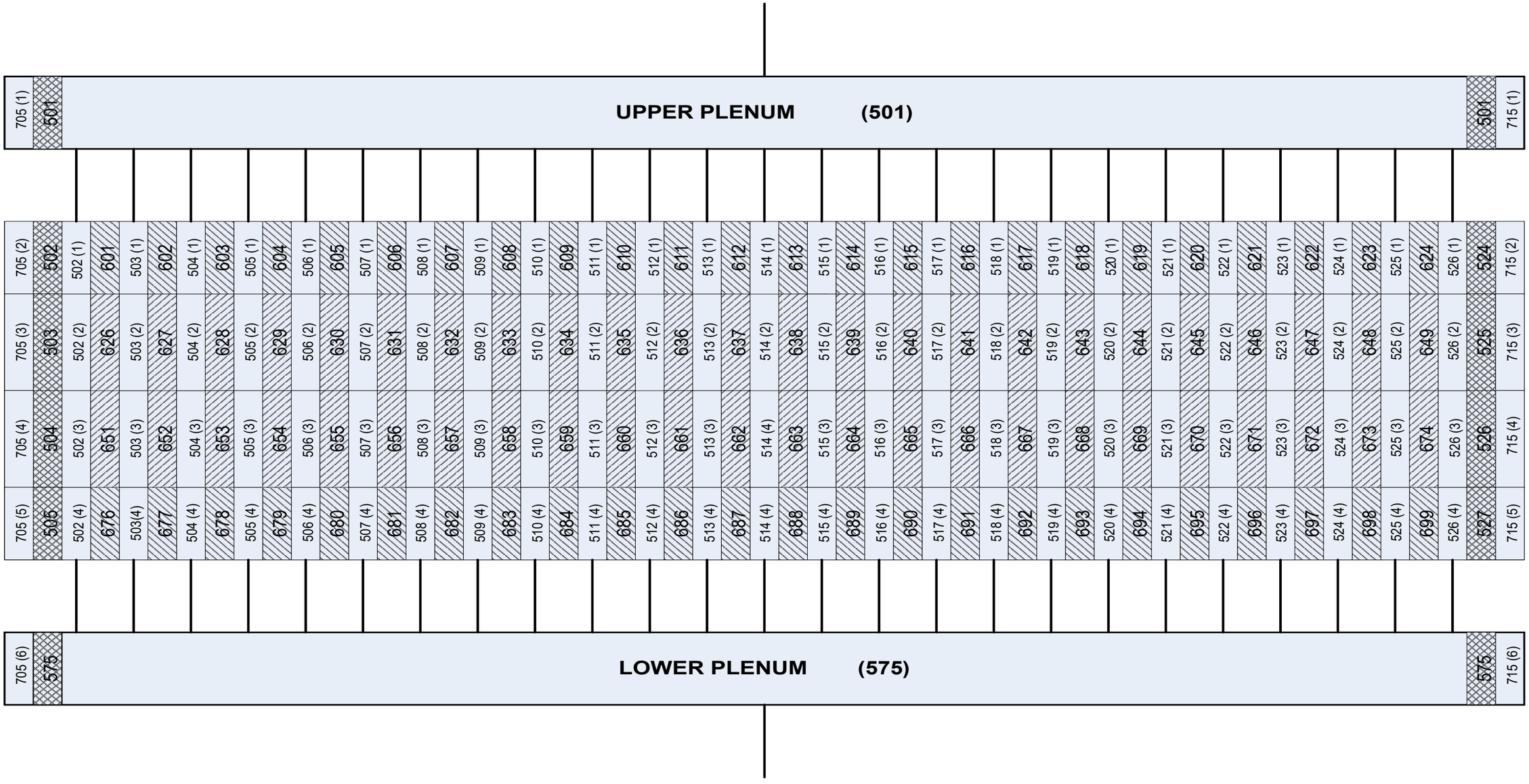


FIGURE C.4 – DETAILED RELAP5 MODEL OF THE MURR CORE

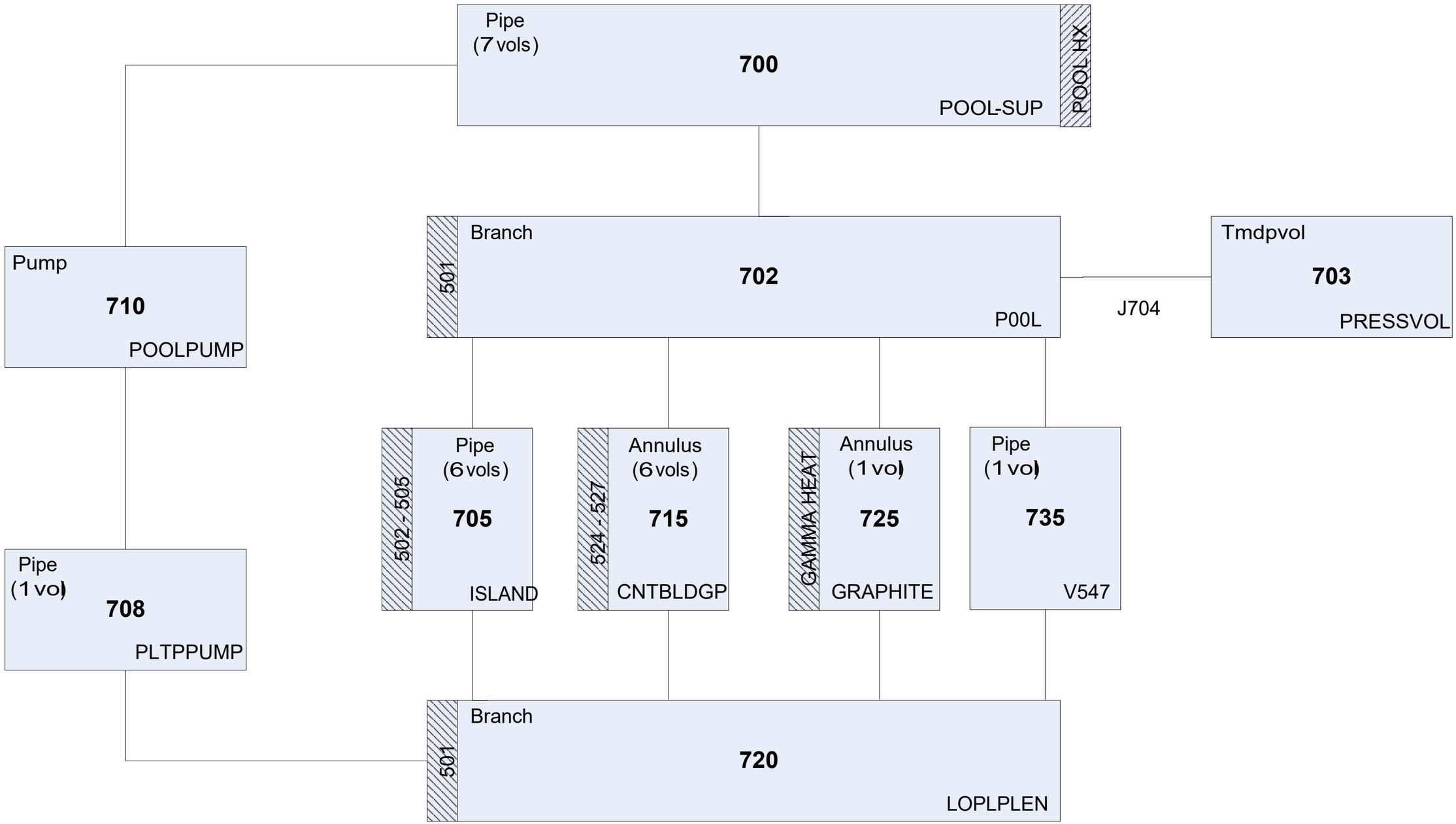


FIGURE C.5 – DETAILED RELAP5 MODEL OF THE MURR BULK REACTOR POOL AND POOL COOLANT LOOP

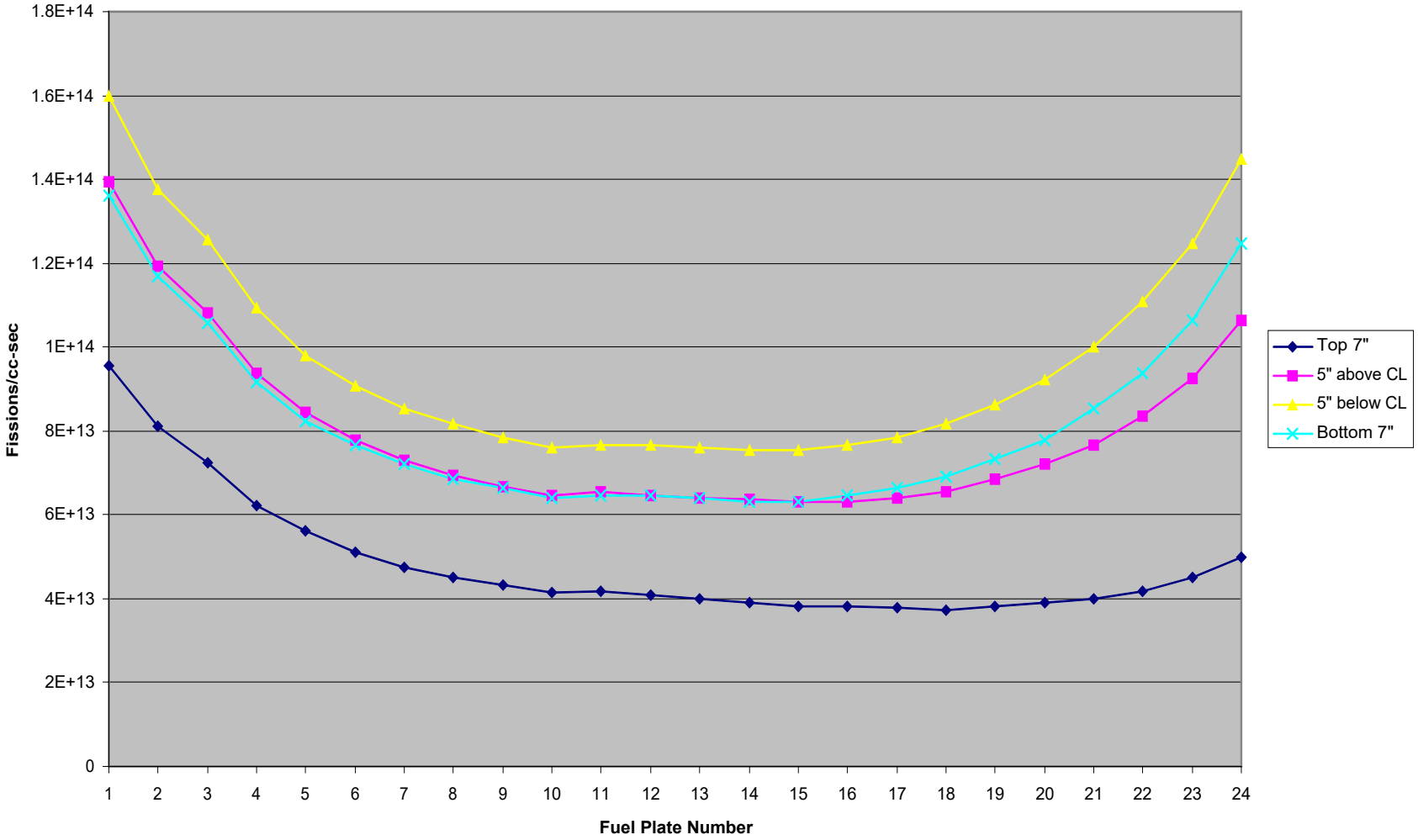


FIGURE C.6 – AVERAGE FISSIONS/CC-S IN THE FUEL PLATE SECTIONS

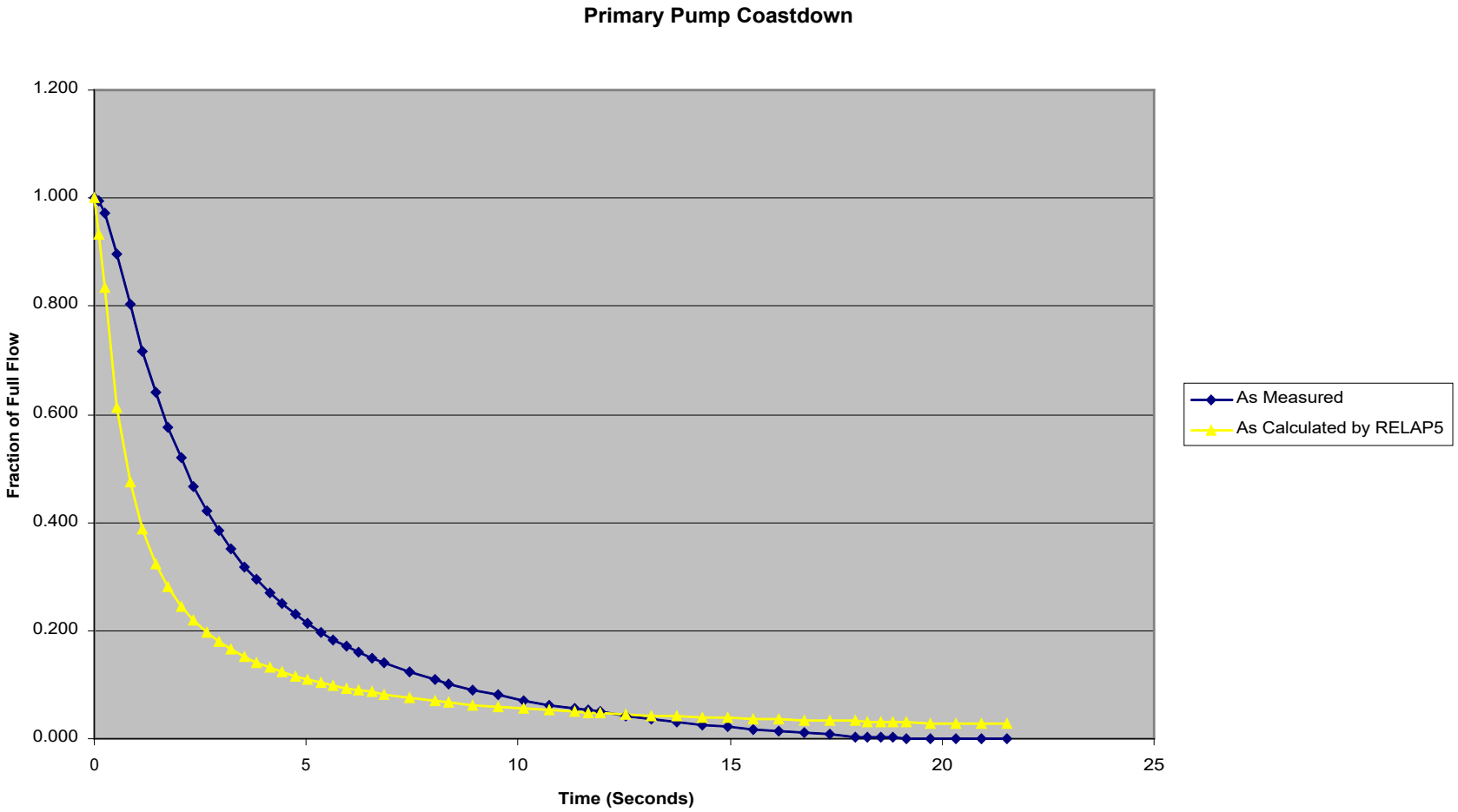


FIGURE C.7 – ACTUAL VERSUS MODEL PREDICTED PRIMARY COOLANT PUMP COASTDOWN