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CHAPTERS 10 - 18

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CHAPTER 10

EXPERIMENTAL FACILITIES AND UTILIZATION

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

REFERENCES

10.1 Preliminary Hazards Report, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, March 1961.

10.2 Hazards Summary Report, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, July 1965.

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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.

10.1 Introduction

The experiment program at the Missouri University Research Reactor (MURR) provides a broad range of analytical, radiographic, and irradiation services for use by the research community and the commercial sector. The 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 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 experiment 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 (approximately1 $\times 10^{14}$ 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, is (a) any device or material which is exposed to significant radiation from the reactor and is not a normal part of the reactor or (b) any operation designed to measure or monitor reactor characteristics or parameters. Experiments conducted at the MURR are subdivided into two general classifications: (1) neutron beam, and (2) 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, neutron activation analysis (e.g., archeological samples), and transmutation doping of silicon. 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. The 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 the 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.[1] A major safety feature of the reactor can be found in the design of the beryllium reflector,[1] 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 control as discussed in the following section.

10.3 Experimental Facilities

MURR experimental facilities are designed, operated, and utilized in a manner that will not exceed the reactor's limiting safety system settings or limiting conditions for operation of the Technical Specification requirements during normal operations. The radiological controls and ALARA programs ensure that personnel and public radiation doses do not exceed the requirements of 10 CFR 20 and are maintained as low as reasonably achievable. 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.

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[1]The original safety evaluation of the MURR is documented in the Preliminary Hazards Report (Ref. 10.1), the Hazards Summary Report (Ref. 10.2), and Hazards Summary Report, Addenda 1-5 (Ref. 10.3-10.7).

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 pressure vessel (island tube) and which extends 15 inches (38.1 cm) above and below the core centerline. A specially-designed 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 a high degree of administrative control 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, the test hole canister is inserted or removed only when the reactor is in a shutdown condition. A licensed operator verifies that the test hole canister is properly installed and secured. If the 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 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 test hole canister. Three center test hole assemblies are designed and approved for use at the MURR, those being a six-tube, a three-tube, and a single-tube.

The three-tube test hole canister (Figure 10.1) consists of three 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 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 test hole canister is 14 feet 2½ inches (4.3 m).

The six-tube test hole canister (Figure 10.2) is similar in design to the three-tube test hole canister with a few exceptions. There are an added four vertical inches of irradiation capacity in the three 1.334-inch (3.4-cm) I.D. tubes and the addition of three 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 test hole canister and in all sample loading documentation.

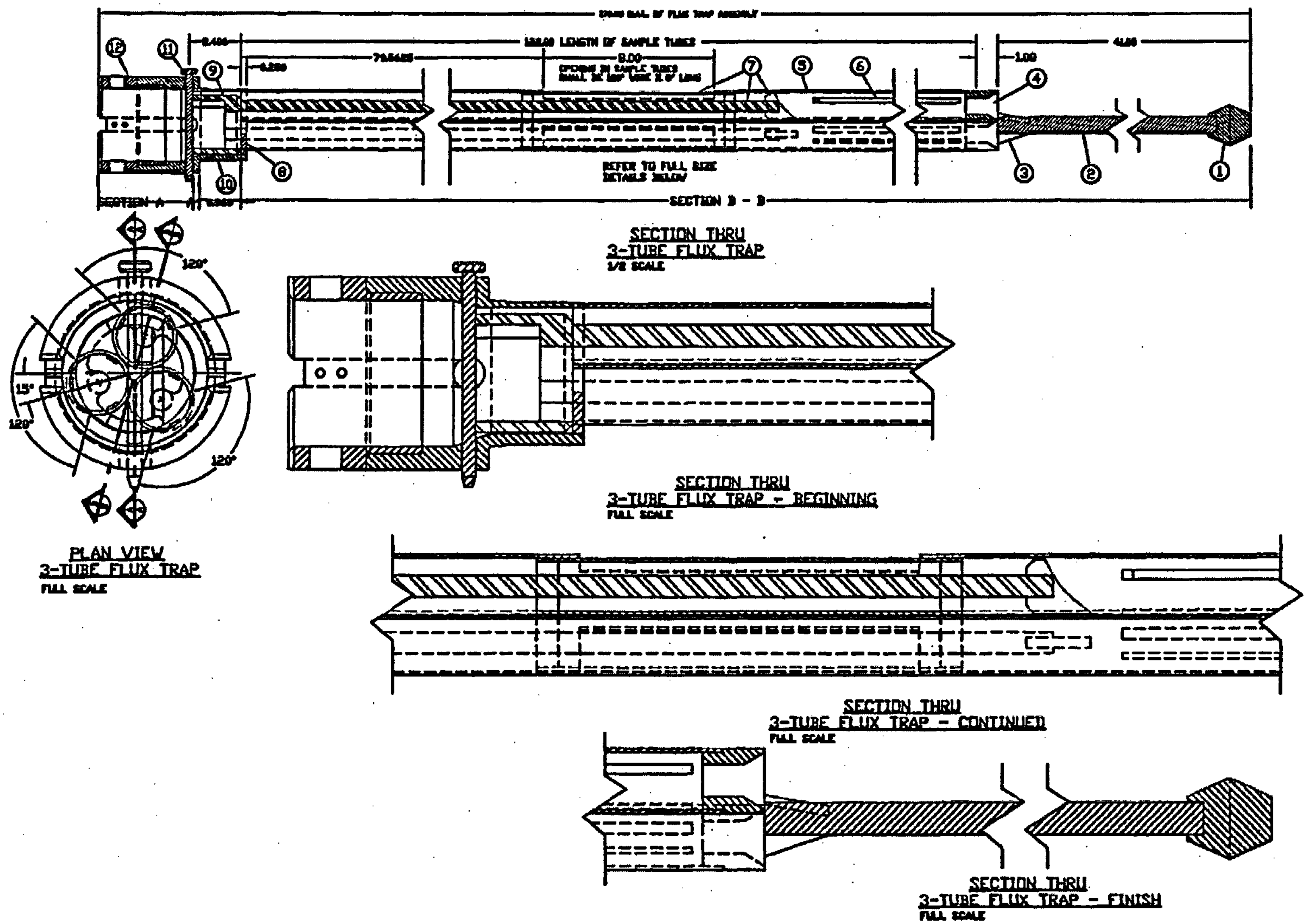

FIGURE 10.1
THREE-TUBE TEST HOLE CANISTER

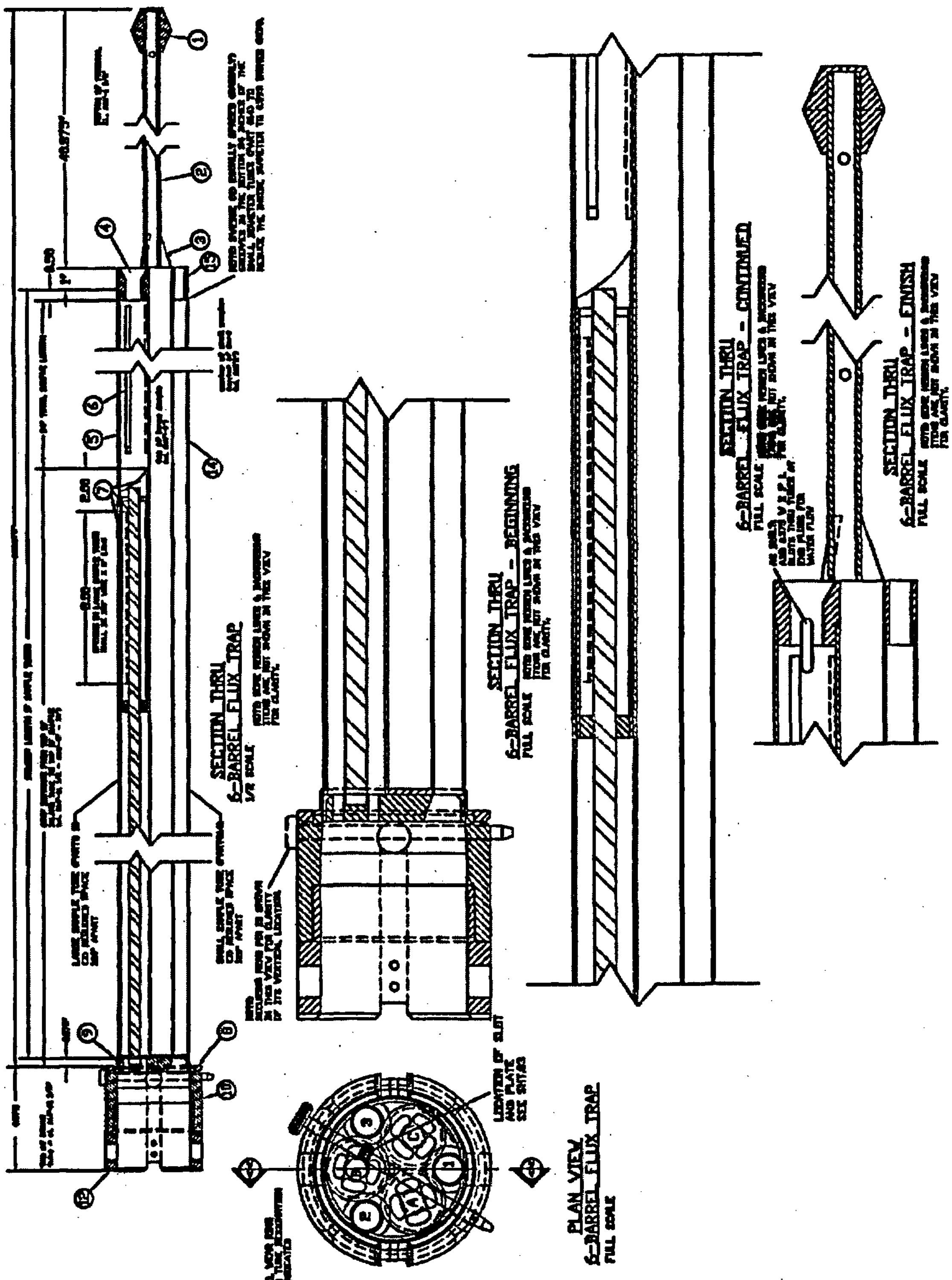

FIGURE 10.2
SIX-TUBE TEST HOLE CANISTER

When installed, the test hole canister position is positively determined by a latching mechanism located at the top of the assembly. Two stainless steel latching fingers secure the canister to the upper portion of the inner reactor pressure vessel (island tube). To provide additional vertical alignment and support, the base piece engages into a test hole slot which is welded to the reflector tank base flange. The canister top is designed to preclude the possibility of foreign objects entering the center test hole while the assembly is in place.

The experiment volume of the larger diameter tubes of both the three- and six-tube test hole canisters is filled at all times with either aluminum 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 hair pin 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 test hole canister may or may not contain experiment samples during operation.

All samples placed in the 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 test hole canister is also designed for use in the center test hole. It was used during initial operation of the MURR and replaced by the three-tube assembly when an increase in material irradiation capacity was required.

10.3.1.3 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 affects 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 5.2-Kg uranium alloy fuel core to the 6.2-Kg UAl_x aluminide fuel core, the temperature and void coefficients of the primary were carefully measured and found to be close to the original calculated values.[1] Two independent measurements

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[1]See Footnote on Page 10-2.

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[1] 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 test hole canister loading is reviewed and the reactivity worth of all samples is determined.

While the probability is highly unlikely, a scenario can be constructed in which all of the experiments in the center test hole are rapidly extracted. Therefore, a restriction is placed on the limit of the net reactivity worth for all the experiments in the center test hole in accordance with the analysis in Chapter 13.

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. 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).

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 experiment 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 beam tubes are the equivalent of 4-inch schedule 40 aluminum pipe and the 6-inch beam tubes are the equivalent of 6-inch 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 beam tubes increase to 8 inches 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	Elevation[a]	Description
Beamport "A"	4-inch I.D.	minus 2 inches (5.08 cm)	radial
Beamport "B"	6-inch I.D.	minus 7 inches (17.78 cm)	radial
Beamport "C"	6-inch I.D.	minus 14 inches (35.56 cm)	radial-tangential[b]
Beamport "D"	4-inch I.D.	minus 14 inches (35.56 cm)	radial-tangential[b]
Beamport "E"	6-inch I.D.	minus 7 inches (17.78 cm)	radial
Beamport "F"	4-inch I.D.	minus 2 inches (5.08 cm)	radial

[a]Elevation of the beamports is with respect to the reactor core centerline.
[b]The radial-tangential beamports are actually perpendicular to the beryllium reflector as are the radial beamports, however, their elevations are below the fuel region and therefore they do not view the reactor core directly.

Each beamport assembly (Figure 10.3) consists of three 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, therefore 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 ½-inch 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 the experiment 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

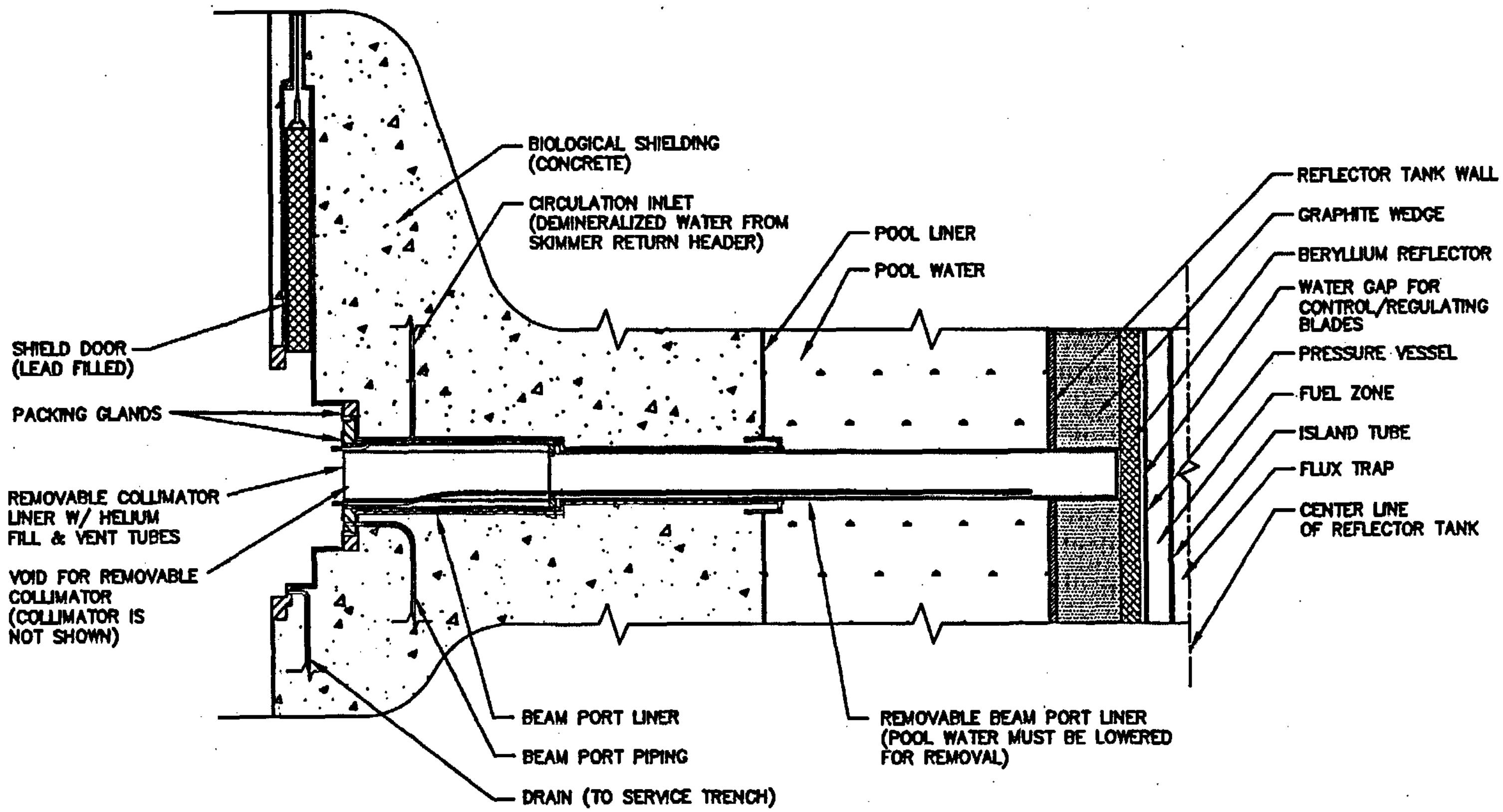

FIGURE 10.3
CROSS-SECTIONAL VIEW OF A BEAMPORT ASSEMBLY

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 off-gas vent line, a 1-inch drain line, and two 2-inch conduit sleeves. The conduit sleeves provide a path for instrumentation or equipment to be routed between the vestibule and the stepback (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 overhead rectilinear crane. When a neutron beam is to be accessed, the door is raised and two pins are installed to secure it in the open position.

Two 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 ½-inch vent line connected to each port. The vent lines are ducted to the reactor containment building 16-inch 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 <u>General Requirements</u>

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 effect 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 a 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 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 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 limiting conditions for operation 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 millirem/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 10.4). 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 ¼-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 1½-inch (0.95-m) square by 12¼-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 cooling slot of ¼ of an inch (6.35 mm) ± ⅛ of an inch (3.18 mm) exists between the canned lead shield and the aluminum housing.

The over-all 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 is a single block of graphite approximately 4 inches (10.2 cm) by 4 inches (12.2 cm) in cross section and may be removed individually.

Two 4-inch 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.

[redacted]

FIGURE 10.4
THERMAL COLUMN ASSEMBLY

The thermal column door moves on two level floor tracks and is driven by an electric motor through a gear reducer box. The inner surface of the door is lined with ¼-inch thick boral sheeting. To prevent a potential radiation hazard due to inadequate thermal column shielding, 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. The signal also initiates an annunciator alarm: "Thermal Column Door Open." The thermal column door is shown in Figure 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 x 10^{14} n/cm^2-sec) for the purpose of material irradiation.

10.3.4.2 Description

The P-Tube System is a standard 1½-inch-diameter, vacuum operated sample transfer system consisting of two turbo-compressors, a solenoid control valve cabinet, two electric-operated switch assemblies, four in-pool concentric tube terminals, six sending-receiving stations, and associated piping and tubing. The system is capable of simultaneously transferring four rabbits either into or out of the graphite reflector region. Only two terminals and three 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 photo-cell, and then to the terminus located in a graphite reflector element. It returns to the laboratory along the same path. The photo-cell 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 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, thus minimizing the possibility of fragments from a broken rabbit becoming trapped in the terminus. This arrangement also ensures air flow is into the pneumatic tube system should a leak develop in the sample carrier tubing.

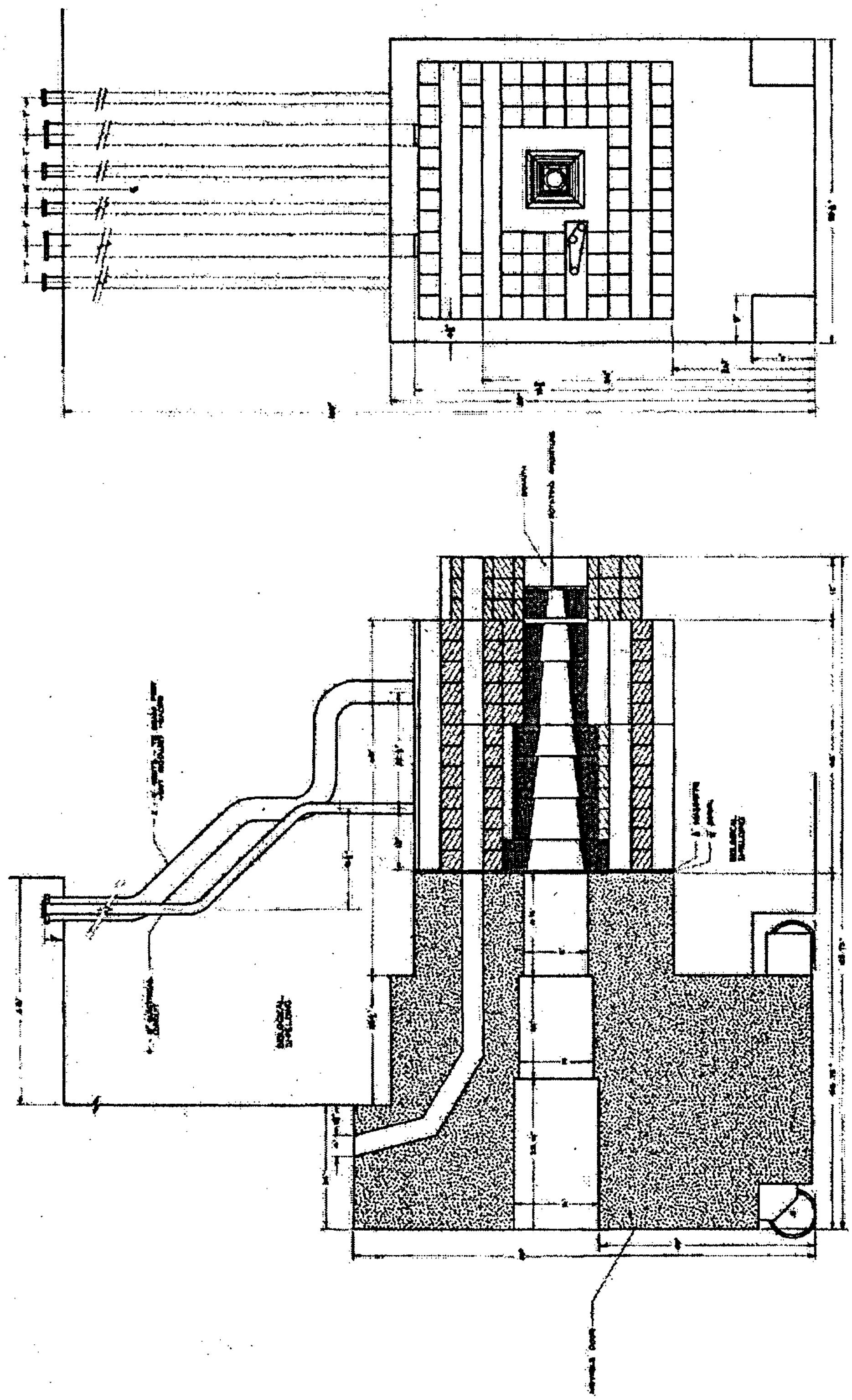

FIGURE 10.5
THERMAL COLUMN DOOR

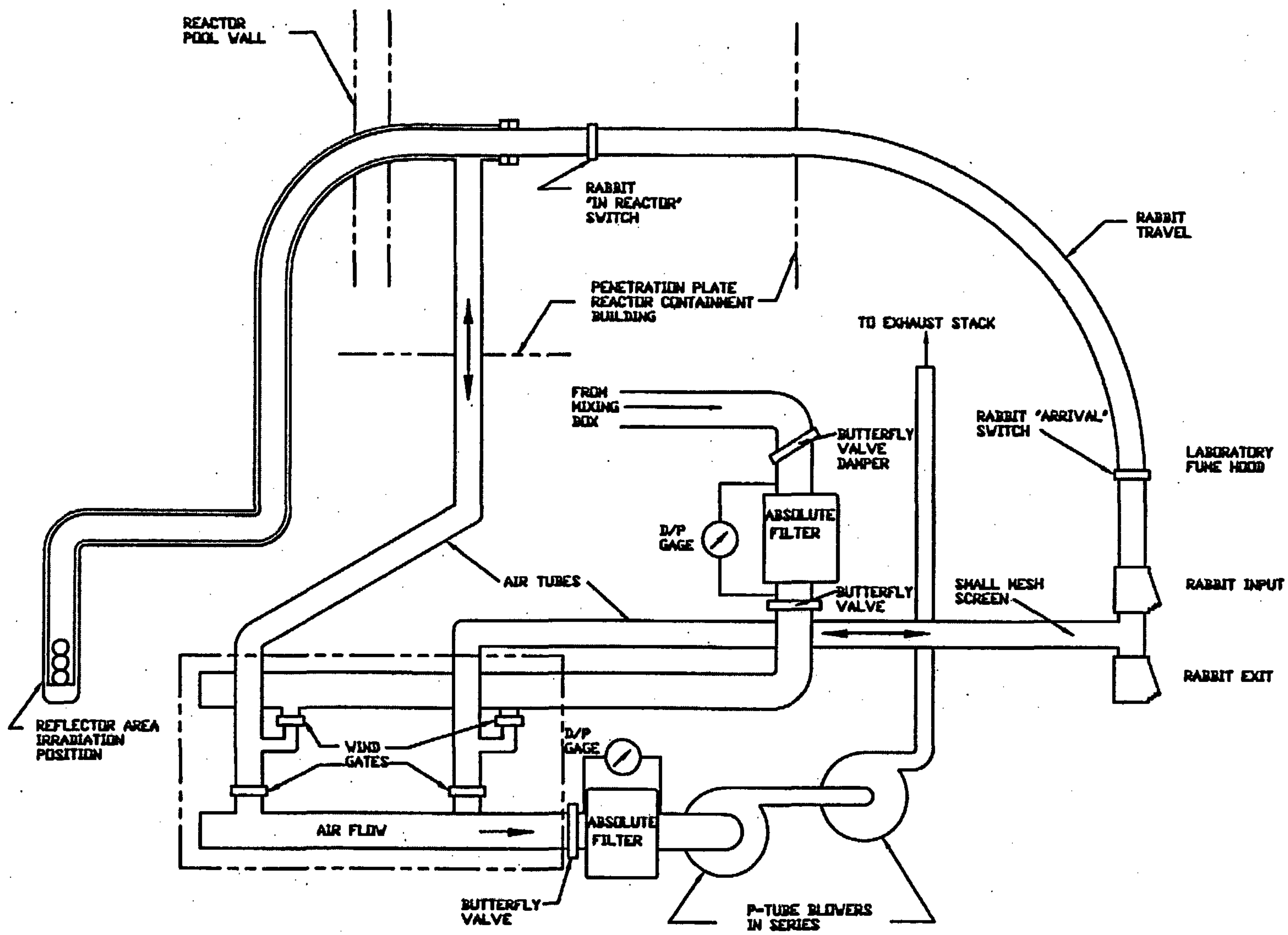

FIGURE 10.6
PNEUMATIC TUBE SYSTEM

The length of the sample carrier tube, 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 2¼-inch (5.7-cm) diameter and the inner tube (sample tube) has a 1½-inch (3.8-cm) diameter. Immediately on the water side of the reactor pool liner is an aluminum concentric flange which allows the removal and installation of the in-pool portion of the transfer tubing.

One sample carrier tube is designed to service at most two laboratories. The sending-receiving stations in these laboratories are electrically interlocked such that when a rabbit is dispatched from one station the other station becomes inoperable.

Two (2) 7½-HP, 190-cfm overhung-type turbo-compressors, designated the east and west p-tube blowers, circulate the required air flow through the pneumatic 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 pneumatic 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 (^{41}Ar) which is produced in the terminus, the pneumatic tube system incorporates two 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 ^{41}Ar in the terminus. Also, a time delay circuit starts the east p-tube blower approximately 15 seconds after the west p-tube blower, thus minimizing an air surge through the system.

10.3.4.3 General Requirements

The following are the general requirements (Ref. 10.8) of the pneumatic 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 air flow 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 x 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 Pneumatic 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 will 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 x 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, Graphite Reflector Irradiation Positions.

10.4 Experiment Review

10.4.1 Introduction

All experiments conducted at the 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 (RSSC) and the Reactor Advisory Committee (RAC).

10.4.2 Reactor Utilization Request

The Reactor Utilization Request (RUR) Summary Sheet is prepared by the principle experimenter with the assistance of the facility staff (Figure 10.7 provides an example of a typical summary sheet). 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. The complete safety analysis consists of the following seven individual analyses.

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.

DATE SAMPLE MATERIAL

SAMPLE NAME:

DESCRIPTION:

FORM:

WEIGHT LIMITS:

OTHER:

ENCAPSULATION:

POSITIONS:

MAXIMUM FLUX:

MAXIMUM FLUENCE:

SPECIAL FLUENCE:

POST IRRADIATION:

SOURCE DOCUMENTS:

CROSS REFERENCES:

Reactor Manger Date

FIGURE 10.7
REACTOR UTILIZATION REQUEST (RUR) SUMMARY SHEET

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.

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.

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.

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).

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.

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 Specifications limitation.

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.

All active RURs will be reviewed annually by the Reactor Manager. The Reactor Manager will use the annual review to ensure that the experiment descriptions, activities, isotopes, handling procedures, license considerations, and safety analyses are valid for the current range of experiments.

10.4.3 Review Process

The review and approval process of all experiments conducted at the MURR is outlined in the next seven paragraphs.

1. The initial review of the experiment is conducted by MURR staff with technical expertise while assisting the principle experimenter to prepare the RUR. The intent is to raise safety questions and analyze them during this review step.

2. The RUR is then sent to the Reactor Health Physics Manager who reviews the proposal to ensure that all necessary radiological control measures will be performed during the experiment and that the experimenter possesses the experience and equipment to deal with the expected radiation level. He also scrutinizes the applicability and the adequacy of the by-product license(s) under which the experiment is to be conducted. However, his review is not limited to these areas. He may recommend limitations or additional analyses in other areas. If he approves the experiment, he will indicate the additional limitations (if any) recommended and sign the RUR in the space provided.

3. The Assistant Reactor Manager-Physics will critically review the proposed experiment to ascertain the reactivity effect, heat generation considerations, and the possibility of sample decomposition. However, his review is not limited to these areas. He may recommend limitations or additional analyses in other areas. If he approves the experiment, he will indicate the additional limitations (if any) recommended and sign the RUR in the space provided.

4. The Reactor Manager will critically review the proposed experiment and determine if the experiment represents a new class of experiment or a change to an existing experiment, which has a safety significance. If either of these conditions apply, he will submit the RUR to the Reactor Safety Subcommittee (RSSC) for its review. The RSSC conducts the reviews of all new experiments for the Reactor Advisory Committee (RAC). Its review is primarily to determine if the experiment involves a question pursuant to 10 CFR 50.59. If the RSSC determines that the experiment does not involve a potential safety hazard and recommends approval, the RUR is completed. The RSSC may, however, refer the experiment to the RAC for its review. This may be done because of unusual hazards, special conditions involved, or because the RSSC feels that a potential safety hazard does or may exist. If the Reactor Manager determines the experiment <u>does not</u> represent a new class of experiment or a change to an existing experiment, he may approve the RUR and submit it to the RSSC for review.

5. The RAC will review the experiment if said has been referred to it by the RSSC. If the RAC determines that the experiment <u>does not</u> involve a potential safety hazard, then the review process is complete.

6. If the RSSC and/or the RAC feel that the proposed experiment <u>does</u> introduce a potential safety hazard, the experiment proposal must be submitted to the NRC for final review. The MURR staff will generally prepare the necessary documents for submittal.

7. After all of the reviews have been completed, the Reactor Manager will indicate on the RUR any additional limitations required beyond those listed and will then sign the RUR as being approved. Copies of the approved RUR will be distributed to the following personnel:

   a) Document Control (Original);
   b) The Reactor Safety Subcommittee; and
   c) The Reactor Advisory Committee (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 Provost, 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 two individuals will posses 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 (RSSC) shall act in 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 RSSC 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 RSSC 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 RSSC of the course of action selected, or he shall submit a new proposal for review.

The Reactor Safety Subcommittee and its responsibilities are described in greater detail in Chapter 12, Conduct of Operations.

10.5 Limitations on Experiments

Limitations on experiments conducted can be placed into two general classifications:

(1) Limiting Conditions of Operation and

(2) Reactivity Worth.

The limiting conditions for operation of the experimental facilities at the MURR are stated in the Technical Specifications. The objectives of these limitations are 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. In addition to the experiment limitations in the Technical Specifications, the Operations Procedures also list controls on the use or exclusion of corrosive, flammable, and toxic materials in the reactor containment building.

The reactivity worth limits of the experimental facilities are also stated in the Technical Specifications. The objectives of these limitations are to ensure that the reactor can be shut down at all times, and to ensure that the reactor core safety limits will not be exceeded.

CHAPTER 11

RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

REFERENCES

11.1 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," 1990.

11.2 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.11, "Radiation Protection at Research Reactor Facilities," 1993.

11.3 Title 10, "Energy," Code of Federal Regulations, Part 20, "Standards for Protection Against Radiation," Washington, D.C.

11.0 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

This chapter discusses and analyzes all 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 the MURR, the general public and the environment.

11.1 Radiation Protection

The Missouri University Research Reactor (MURR) Radiation Protection Program (RPP) has been established to protect the health and safety of all individuals present at the 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 the MURR, and is sufficient to ensure compliance with the regulations in 10 CFR 20. 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.

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 Health Physics Staff has the authority to interdict or terminate the use of radioactive materials or radiation sources if adequate health physics 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 the MURR: the Broad Scope Material 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 the MURR. Coverage provided by the other license is supplementary and is used to support the research and development mission of the MURR which may not currently be covered under the facility operating license.

Radioactive materials within the reactor facility, whether licensed under the Broad Scope Material 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 Material License differs from that of the facility operating license. The Reactor and Health Physics Managers 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 and Health Physics Managers 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 Material 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 - Airborne sources consist mainly of argon-41 (^{41}Ar, half-life 1.8 h), which accounts for greater than 99% of the radioactivity released through the facility ventilation exhaust stack. ^{41}Ar is produced when the argon-40 in air (~1.0%) is activated by thermal neutrons. The principle production areas within the reactor facility include the pneumatic tube system, the thermal column, and the beamports. Other than ^{41}Ar, no other significant source of airborne radioactivity is produced at the reactor facility as part of its normal operation;

- Liquid - 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 - Solid sources are a bit more diverse, but for the most part are very 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 the MURR, ^{41}Ar is the principal source of airborne radioactivity (>99%) released through the facility ventilation exhaust stack. The assumptions and calculations used to assess the radiological impact of ^{41}Ar during normal operation are described in detail in Appendix B. Therefore, that information will only be summarized in this section.

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. However, the Maximum Hypothetical Accident (MHA) for the MURR, which assumes the melting of four fuel plates, is analyzed and discussed in detail in Chapter 13, Accident Analyses. This analysis does assume a release of fission products from the primary coolant system into the reactor containment building.

11.1.1.1.1 Argon-41 from the Pneumatic Tube System

As stated above, the principle production areas of ^{41}Ar within the MURR includes the pneumatic tube (p-tube) system, and to a much lesser extent, the thermal column and the beamports. It is estimated that approximately 98% of the ^{41}Ar produced at the 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 ^{41}Ar is exhausted from the system through a HEPA filter to the facility ventilation exhaust stack. ^{41}Ar produced in the thermal column and beamports is ducted to the 16-inch hot exhaust line which also exhausts to the facility ventilation exhaust stack.

Table 11-1 provides the measured flow rates and typical ^{41}Ar 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 ^{41}Ar 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 an average ^{41}Ar

concentration of 2.8×10^{-8} µCi/ml, which is well below the NRC regulatory limit stated in 10 CFR 20. When compared to the ^{41}Ar Derived Air Concentration (DAC) limit of 3×10^{-6} µCi/ml, as listed in Appendix B, 10 CFR 20, facility workers are exposed to less than 1% of the DAC value. Normally, there are four reactor operators, four operations management staff, one administrative assistant, and three researchers within the containment building at any one time. At current levels of exposure, an average staff of twelve would receive less than 0.6 person-rem per year due to ^{41}Ar exposure. This is in reality a conservative number, as most staff members do not stay in the containment building 100% of the time.

11.1.1.1.3 Argon-41 Release to the Unrestricted Area

^{41}Ar produced in the p-tube system, thermal column, and beamports is discharged from the 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 ^{41}Ar considerably before the exhaust plume returns to ground-level locations which could be occupied by personnel. Utilization of this dilution credit is allowed by the NRC. The detailed calculations relating to the dispersion of ^{41}Ar released from the exhaust stack are contained in Appendix B. It is important to note that only a small amount of dilution is required to reduce the ^{41}Ar 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 ^{41}Ar 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).

Results of the dispersion calculations for the discharge of ^{41}Ar 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. Additional information is provided in Appendix B.

The maximum annual dose to an individual from exposure to ^{41}Ar 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 50 METERS NORTH FROM THE MURR EXHAUST STACK

Atmospheric Stability		Effective Stack Height	^{41}Ar Concentration[a]
Class	Condition	(meters)	(μCi/ml)
A	Extremely Unstable	35	2.96×10^{-9}
B	Moderately Unstable	27	2.80×10^{-9}
C	Slightly Unstable	23	2.04×10^{-9}
D	Neutral	20	5.71×10^{-10}
E	Slightly Stable	23	2.25×10^{-12}
F	Moderately Stable	30	2.38×10^{-26}

[a]Concentrations are based on a maximum projected ^{41}Ar release concentration of 3.5×10^{-6} μCi/ml.

TABLE 11-3
MAXIMUM Ar-41 CONCENTRATIONS AT 760 METERS NORTH FROM THE MURR EXHAUST STACK

Atmospheric Stability		Effective Stack Height	^{41}Ar Concentration[a]
Class	Condition	(meters)	(μCi/ml)
A	Extremely Unstable	16	1.43×10^{-10}
B	Moderately Unstable	8	5.34×10^{-10}
C	Slightly Unstable	4	9.33×10^{-10}
D	Neutral	1	2.10×10^{-9}
E	Slightly Stable	4	5.26×10^{-9}
F	Moderately Stable	11	1.12×10^{-8}

[a]Concentrations are based on a maximum projected ^{41}Ar release concentration of 3.5×10^{-6} μCi/ml.

Results of the dispersion calculations for the discharge of ^{41}Ar through the facility ventilation exhaust stack at a normal operational value of 2.42 x 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 ^{41}Ar released, the dose estimates for normal operating conditions are easily calculated using the ratios of the stack release rates (given ^{41}Ar 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	^{41}Ar Concentration[a]
Class	Condition	(meters)	(µCi/ml)
A	Extremely Unstable	35	2.07x 10^{-9}
B	Moderately Unstable	27	1.96 x 10^{-9}
C	Slightly Unstable	23	1.43 x 10^{-9}
D	Neutral	20	4.00 x 10^{-10}
E	Slightly Stable	23	1.57 x 10^{-12}
F	Moderately Stable	30	1.67 x 10^{-26}

[a]Concentrations are based on a normal operational ^{41}Ar release concentration of 2.42 x 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 normally accounts for about 81% of the total activity released each year.

TABLE 11-5
NORMAL OPERATIONAL Ar-41 CONCENTRATIONS AT 760 METERS NORTH FROM THE MURR EXHAUST STACK

Atmospheric Stability		Effective Stack Height	^{41}Ar 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 ^{41}Ar release concentration of 2.42×10^{-6} µCi/ml.

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 the 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 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	Activity (Ci)[a]	% of Total Released[b]
Tritium (^{3}H)	12.3 years	1.79×10^{-1}	81.1%
Sulfur-35 (^{35}S)	87.2 days	3.56×10^{-3}	2.3%
Calcium-45 (^{45}Ca)	162.7 days	4.04×10^{-4}	0.3%
Cobalt-60 (^{60}Co)	5.27 years	1.53×10^{-3}	1.0%
Zinc-65 (^{65}Zn)	243.8 days	2.41×10^{-4}	0.2%
Others	N/A	N/A	15.1%

[a]Average amount of activity released into the sanitary sewerage system each year during the reporting period 2001 to 2005.
[b]Average percent of total released each year during the reporting period 2001 to 2005.

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 THE MURR PRIMARY COOLANT
AND THEIR MEASURED CONCENTRATIONS AT 10 MW

Radionuclide	Half-Life	Typical Measured Concentration[a]
Tritium (^{3}H)	12.3 years	7.28×10^{-2} µCi/ml
Magnesium-27 (^{27}Mg)	9.45 minutes	8.78×10^{-2} µCi/ml
Sodium-24 (^{24}Na)	14.96 hours	2.74×10^{-2} µCi/ml
Manganese-54 (^{54}Mn)	312.2 days	3.64×10^{-5} µCi/ml
Manganese-56 (^{56}Mn)	2.58 hours	1.83×10^{-4} µCi/ml
Xenon-135 (^{135}Xe)	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 2 to 3 days after reactor startup.

Although tritium is the major contributor to the total amount of activity released, liquid radioactive waste, as previously mentioned, 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. Furthermore, occupational exposure from primary coolant is also limited because there are few operations which 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 would essentially be gone after 48 hours, with sodium-24 reduced by about a factor of 10. Also, experience at the 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 THE MURR POOL COOLANT AND THEIR MEASURED CONCENTRATIONS AT 10 MW

Radionuclide	Half-Life	Typical Measured Concentration[a]
Tritium (^{3}H)	12.3 years	8.29×10^{-2} μCi/ml
Magnesium-27 (^{27}Mg)	9.45 minutes	1.10×10^{-2} μCi/ml
Sodium-24 (^{24}Na)	14.96 hours	4.64×10^{-3} μCi/ml
Manganese-56 (^{56}Mn)	2.58 hours	2.54×10^{-3} μCi/ml
Technetium-101 (^{101}Tc)	14.2 minutes	4.70×10^{-5} μCi/ml
Technetium-99m (^{99m}Tc)	6.01 hours	9.73×10^{-6} μCi/ml
Antimony-122 (^{122}Sb)	2.70 days	1.01×10^{-5} μCi/ml
Xenon-135 (^{135}Xe)	9.10 hours	1.22×10^{-5} μCi/ml
Silver -110m (^{110m}Ag)	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 2 to 3 days after reactor startup.

11.1.1.2.3 Nitrogen-16 from the Primary and Pool Coolant Systems

Nitrogen-16 (^{16}N, 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 ^{16}N resulting from neutron interactions with oxygen in air and air entrained in the coolant can therefore be neglected.

As described in Section 5.7, radiation levels due to ^{16}N activity 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 ^{16}N, 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. Therefore, occupational exposure from ^{16}N activity is insignificant.

11.1.1.3 Solid Radioactive Sources

The solid radioactive sources associated with normal operation of the 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.

TABLE 11-9
REPRESENTATIVE RADIOACTIVE SOURCES AT MURR

[redacted]

[redacted]

[redacted]

Although solid waste is included in the preceding table, additional information on waste classification, storage, packaging, and shipment is included in Section 11.2. To further elaborate on the waste entry in Table 11-9, 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 1996 to 2005 and its radioactive content, in millicuries.

11.1.2 Radiation Protection Program

The Health Physics Branch is located within the organization for the management and operation of the MURR as shown in Figure 12.1 and discussed in Chapter 2, Conduct of Operations. The Health Physics Branch includes the Reactor Health Physics Manager and the Health Physics Staff under his direction.

The Reactor Health Physics Manager reports to the Reactor Facility Director through the Associate Director, Regulatory Affairs Group. However, there is a communications/consultation line from the Reactor Health Physics Manager to the Office of the Provost. This line of communications/ consultation allows the Reactor Health Physics Manager access to upper University management if Reactor Facility Management does not address radiation protection concerns to the satisfaction of the Health Physics Manager.

TABLE 11-10
MURR WASTE SHIPMENTS AND INVENTORY

Shipment Year	Waste Container Type	Volume (ft^3)	Activity (mCi)
2005	LLW Barrels (72) B-25 Containers (2) (Total)	540.0 190.0 (730.0)	829
2004	LLW Barrels (86) B-25 Containers (1) (Total)	645.0 95.0 (740.0)	698
2003[a]	LLW Barrels (51) B-25 Containers (1) (Total)	383.0 95.0 (478.0)	465
2002[b]	Shielded Containers (2) Self Contained (1) LLW Barrels (39) B-25 Containers (2) Boxes (2) (Total)	41.5 95.3 293.0 190.0 15.0 (634.8)	315,121
2001[c]	Shielded Containers (1) LLW Barrels (104) B-25 Containers (3) (Total)	29.4 780.0 285.0 (1094.4)	5,584
2000[d]	B-25 Containers (2) LLW Barrels (43) (Total)	200.0 322.5 (522.5)	2491
1999	B-25 Containers (1) LLW Barrels (62) (Total)	100.0 465.0 (565.0)	281
1998[e]	B-25 Containers (4) LLW Barrels (28) (Total)	400.0 210.0 (610.0)	53
1997	LLW Barrels (56)	420.0	404
1996	LLW Barrels (45)	337.5	1,409

[a]Year 2003 shipment included 52 pounds of depleted uranium.
[b]Year 2002 shipment included the following: 1 shielded container of reflector elements; 1 shielded container of aluminum; 2 boxes of contaminated lead; and 1 activated neutron scattering instrument.
[c]Year 2001 shipment included aluminum ingots.
[d]Year 2000 shipment included two Surface Contaminated Objects (SCO) - Water Storage Tanks - 685 ft^3 total volume.
[e]Year 1998 shipment included two Surface Contaminated Objects (SCO) - Pool Coolant System Heat Exchangers - 300 ft^3 total volume.

11.1.2.1 Health Physics Branch

The Health Physics Branch is the organization which administers the Radiation Protection Program for the reactor facility. This branch can consist of the following personnel: the Reactor Health Physics Manager, the Assistant Reactor Health Physics Manager, Health Physicists, and Health Physics Technicians.

The qualifications for the positions within the Health Physics Branch are as follows:

- Manager, Reactor Health Physics - The Reactor Health Physics Manager shall have a minimum of four to five years of experience in operational health physics and management. The individual shall have a recognized master's degree in health physics or an equivalent combination of education and experience from which comparable knowledge and abilities can be acquired;

- Assistant Manager, Reactor Health Physics - The Assistant Reactor Health Physics Manager shall have a minimum of three to four years of experience in operational health physics. The individual shall have a recognized master's degree in health physics or an equivalent combination of education and experience from which comparable knowledge and abilities can be acquired;

- Health Physicist - The Health Physicist shall have a minimum of two to three years of experience in applied health physics. The individual shall have a bachelor's degree in health physics or an equivalent combination of education and experience from which comparable knowledge and abilities can be acquired; and

- Health Physics Technician - The Health Physics Technician shall have a minimum of two to three years of experience in operational health physics. The individual shall have an associate's degree in health physics or related field or an equivalent combination of education and experience from which comparable knowledge and abilities can be acquired.

The positions of authority and responsibility within the Health Physics Branch are discussed in Chapter 12, Conduct of Operations.

The working relationship of the Health Physics Branch 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) shall act as an advisory group to the Reactor Advisory Committee (RAC) in regard to matters relating to the custody and use of radiation and radioisotopes within the MURR. The RAC and the IUS are described in detail in Chapter 12, Conduct of Operations.

11.1.2.3 Radiation Safety Committee

The Radiation Safety Committee (RSC) is responsible for establishing the policies relating to the management of programs utilizing radioactive material and radiation sources that are covered by the Broad Scope Material License. The RSC reports to the Chancellor through the Vice Chancellor for Research on all matters pertaining to the safe use of radiation in these programs. The Vice Chancellor for Research is responsible for appointing members to the RSC and for assigning a Chairman. The RSC's primary duties include:

- Reviewing and approval of the use of radioactive materials covered by the Broad Scope Material License;
- Ensuring that the use of licensed material is consistent with the ALARA philosophy and program; and
- Reviewing the performance of the Radiation Safety Officer (RSO) and the Health Physics Branch to assure adequate control of radiation risks with respect to the Broad Scope Material License.

Meetings of the Radiation Safety Committee are conducted in accordance with a written charter.

11.1.2.4 Radiation Safety Officer

The Radiation Safety Officer (RSO) is responsible for the implementation of the policies established by the Radiation Safety Committee (RSC). The RSO is appointed by the Chancellor upon recommendation of the Vice Chancellor for Research. The RSO reports to the Reactor Facility Director. The RSO's primary duties, with the assistance of the Health Physics Branch, 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 Material 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 Material 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 Title 10, Chapter I of the Code of Federal Regulations, Part 19.12 (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 unescorted access to the reactor facility 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 the MURR. The initial training program also includes the completion of documentation for the level of unescorted access, assignment of personnel dosimetry, and a tour of the reactor facility conducted by a member of the Health Physics Branch to reinforce the information provided in the instructional visual presentation. Radiation safety training topics shall cover the following areas in sufficient depth for the work being done:

    a. Biological effects of ionizing radiation;

    b. Principles and practices of radiation safety (ALARA);

    c. Radioactivity measurements and monitoring techniques;

    d. Applicable regulations and license requirements;

    e. Areas where radioactive materials are used and stored;

    f. Appropriate radiation protection procedures and practices;

g. Individual's responsibility to report unsafe conditions (potential regulatory and license violations) to the Health Physics Branch and/or applicable authorities;

h. Appropriate response to emergencies or unsafe conditions;

i. Worker's right to be informed of occupation radiation exposure and bioassay results;

j. Worker's rights as described in 10 CFR 50.7; and

k. 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 having 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 reading the training booklet or reviewing the initial training visual 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 NRC-approved 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 unescorted access are allowed to work with radioactive materials greater than exempt quantities only under the direct supervision of a qualified radiation worker, an authorized supervisor, or a member of the Health Physics Branch.

- 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:

    a. Radioactive decay, and radiation units and quantities;

    b. Specific radiation protection techniques, including external protection;

    c. Radiation instrumentation, including the use of air monitoring and special personnel dosimetry;

    d. Contamination checks and surveys;

    e. Calculations basic to the use and measurement of radioactivity;

f. Decontamination techniques;

g. Radioactive materials transfer/release requirements;

h. Bioassay considerations; and

i. Radioactive waste disposal.

Additional training time ranges from a minimum of ten to 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 general, an individual who has a Class II level of training is approved to work with radioactive materials without direct supervision, or 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 licensed under the MURR Broad Scope Material License. In addition to the requirements for the Class II level, training for Class I will focus on subjects specific to administrative controls, such as:

    a. Requirements for the application of project authorization;

    b. Safety evaluation criteria;

    c. Regulatory requirements and license conditions;

    d. Personnel training and approval; and

    e. Documentation requirements.

This additional training shall include a minimum of twenty hours of combined formal and on-the-job training. Prior documented training and experience may be used to fulfill part of the training requirements.

Records of formal training are maintained by the MURR Training Organization. These records will normally include the following information: course/lecture/visual outline and/or material used; date(s) of training; names of trainees and signatures indicating participation and understanding; and duration and name of instructor/lecturer, if applicable. Formal training will be provided by the Health Physics Branch or authorized supervisors or other individuals who, by their training and experience, are approved by the Reactor Health Physics Manager or Radiation Safety Officer to provide given elements of formal radiation protection training.

11.1.2.6 Health Physics Procedures

The Health Physics Operating Procedures provide the methods and guidelines for the implementation and the maintenance of the MURR Radiation Protection Program. Changes to these procedures are made by the Reactor Health Physics Manager, or his authorized delegate, and are subsequently reviewed by the Reactor Procedures Review Subcommittee (RPRS). The procedures are also audited on an annual basis by the Reactor Health Physics Manager or his authorized delegate.

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 health physics staff:

a. 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;

b. Calibration of area radiation monitors, facility air monitors, laboratory radiation detection systems, personal radiation monitoring devices, and portable radiation monitoring instruments;

c. Administrative guidelines for the facility personnel indoctrination training program;

d. Receiving and opening packages of radioactive materials and any subsequent transfer within the facility;

e. Monitoring of radioactivity in the environment surrounding the facility;

f. Leak testing of sealed sources containing radioactive materials;

g. Shipment of radioactive materials;

h. Radioactive analysis of the primary and pool coolant;

i. Radiation Work Permit procedures;*

j. Controlled special exposures (ALARA considerations);

k. Unplanned personnel radiation exposure investigation procedures;

l. Secondary coolant and facility sump water analyses; and

m. Monitoring a beamport area during a reactor startup.

*Radiation Work Permits (RWPs) provide instructions to workers prior to performing radiological operations. An RWP is used for work not controlled by an operating procedure which has a 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.

11.1.2.7 Health Physics Audits

Periodic audits are performed by the 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:

a. Radiation Protection Program - 10 CFR 20.1101;

b. As Low As Reasonably Achievable (ALARA) Program - Regulatory Guide 8.10.C.1.b;

c. Health Physics Operating Procedures - MURR Technical Specification No. 6.1;

d. Type B Radioactive Materials Shipping Program - 10 CFR 71.137; and

e. Broad Scope Material License - License Application Section 10.2.

Note: Included is the source document that either requires or recommends that the audit be performed.

11.1.2.8 Health Physics Records

Records that document compliance with regulations regarding radiation protection are maintained by the Health Physics Branch. 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 in the Health Physics Office. Other records are retained in long-term storage (remote record retention facilities). Radiation protection records are reviewed by the Reactor Health Physics Manager or his 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:

  a. Provisions of the Radiation Protection Program;

  b. Surveys conducted for dose assessments;

  c. Measurements and calculations used to determine intakes of radioactive material;

  d. Results of air samples, surveys, and bioassays;

  e. NRC Form 4 records for each individual;

  f. Records of planned special exposures;

  g. Records of doses received during work, accidents, emergency conditions and doses received by an embryo/fetus of a declared pregnant woman;

  h. Declaration of pregnancy;

  i. Records to demonstrate compliance with the dose limit for individual members of the public;

  j. Records of the disposal of licensed materials; and

  k. 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.

- Three Year Records - The following records are to be retained for a period of at least three years:

  a. Audits of the Radiation Protection Program;

  b. Surveys conducted to assess radiation and contamination levels;

  c. Surveys conducted for the receipt of radioactive material;

  d. Records of instrument calibrations;

  e. Records used to prepare NRC Form 4; and

  f. Records of receipt, transfer, and disposal of byproduct material; receipt records for as long as the material is possessed, three years following transfer or disposal.

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 Radiation Work Permits (RWPs) or to evaluate the effectiveness of decontamination efforts or temporary shielding placement.

11.1.3 ALARA Program

The ALARA Program for the MURR is dedicated to the fundamental principle of maintaining individual exposures and radioactive effluents as low as is reasonably achievable (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 Health Physics Manager.

The MURR ALARA program, which is a component of the Radiation Protection Program, has been established in accordance with 10 CFR 20.1101 and is audited annually as specified in Regulatory Guide 8.10.C.1.b.

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:

a. Allowing radiation source(s) to decay;

b. Installing portable or temporary shielding;

c. Installing portable or temporary ventilation systems, or temporary enclosures and covering, or both;

d. 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;

e. Conducting pre-operational briefings with those assigned to perform tasks in high radiation areas;

f. Insuring that unnecessary personnel are kept out of areas where radiation exposure may occur;

g. Insuring that sufficient radiation monitoring instruments are available to permit accurate measurements and rapid evaluations of the radiation and contamination levels encountered; and

h. Relocating component(s) to be worked on to a lower radiation area.

The MURR ALARA program also contains the following elements which enhance the effectiveness and reliability of the overall program.

- Commitment - The MURR management is committed to the ALARA program for keeping individual and collective radiation exposures and radioactive effluents as low as is reasonably achievable. 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 and Health Physics Managers. These approvals are reviewed by the Reactor Advisory Committee (RAC). Work with radioactive materials licensed under the Broad Scope Materials License must be approved by the Radiation Safety Committee (RSC) prior to starting such work.

- Responsibilities - Operational responsibilities of the following groups or individuals for adherence to the ALARA program are:

  1. Health Physics Branch:

     a. Inform personnel of ALARA program efforts;
     b. Ensure that personnel who work with radioactive materials are instructed in the ALARA philosophy;
     c. Establish programs for routine radiation surveys in areas where radioactive materials and/or sealed sources of radiation are used, handled or stored;
     d. Procure, distribute, calibrate, and maintain the necessary radiation safety instrumentation;
     e. Quantify all radioactive effluents from the reactor facility;
     f. Trend selected effluents to ensure the adequacy of engineered controls in minimizing radioactive effluents;
     g. Provide advice and assistance regarding disposal of liquid effluents and potential releases of gaseous effluents;
     h. Provide advice and assistance for decontamination and for special surveys or inspections of the facility;
     i. Monitor requisition, receipt, and delivery of all incoming shipments of radioactive materials;
     j. Monitor proper radioactive material transfer, waste handling, and disposal in accordance with ALARA principles;
     k. Perform safety (ALARA) evaluations of facilities, equipment, and procedures employed in areas where radioactive materials are used, handled, or stored; and
     l. Evaluate the personnel dose and effluent release investigation levels on an annual basis to ensure that they are still appropriate.

2. Project Leaders and Authorized Supervisors:

   a. To all individuals under their supervision, explain the ALARA concept and their commitment to it;
   b. Ensure that personnel under their supervision adhere to established protocols and good health physics practices during the performance of occupational duties;
   c. Consult with the Health Physics Branch and obtain the approval of the Reactor and Health Physics Managers for work under the facility operating license, or approval of the RSC before using radioactive materials under the Broad Scope Material License; and
   d. Evaluate all applicable practices and procedures before using radioactive materials to ensure that exposures and releases adhere to the ALARA principle.

3. Radiation Workers and Approved Workers:

   a. Assume responsibility for their own participation in the ALARA program, for adhering to project procedures/protocols, and for employing good health physics practices in daily work;
   b. Know what recourse is available if ALARA is not being promoted on the job;
   c. Take the opportunity to participate in the formulation or modification of the procedures that they are required to follow; and
   d. Make any suggestions, as expected, to their supervisors and/or the Health Physics Branch about improving health physics practices for working with radioactive materials or radiation sources.

- Investigation Levels - Investigation levels for occupational radiation exposures and effluent concentrations are established which, when exceeded, initiate a review or investigation by the Health Physics Branch. This investigation or review is focused on determining the cause of the exposure so that appropriate ALARA actions, if any, can be applied. The investigation levels which apply to the exposure of individual workers are listed in Table 11-11. The investigation levels that apply to monthly averages of gaseous effluents are included in Table 11-12 and the investigation levels which apply to batch releases of liquid effluents are included in Table 11-13.

11.1.4 Radiation Monitoring and Surveying

The radiation monitoring program for the 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. Surveys conducted to assess radiation and contamination levels are retained for a period of at least three years.

TABLE 11-11
INVESTIGATION LEVELS FOR PERSONNEL EXPOSURES

Exposure	Investigation Levels (millirem per month)		
	Level I	Level II	Monthly Average[a]
Whole body; head and trunk; active blood-forming organs; lens of eyes; or gonads (deep dose)	>30 mrem and 133% above average monthly dose[a]	>50 mrem and 150% above average monthly dose[a]	>150 mrem >180 mrem (Health Physics & Operations)
Hands and forearms (shallow dose)	>500 mrem	>1,000 mrem	>1,000 mrem
Skin of whole body[b] (shallow dose)	>300 mrem	>600 mrem	>400 mrem
Fetal	>10 mrem	>20 mrem	

[a]average monthly dose based upon previous four quarters
[b]applicable for significant quantities of beta emitters

TABLE 11-12
INVESTIGATION LEVELS FOR GASEOUS RADIOACTIVE EFFLUENTS

Radionuclide	Investigation Level[a]
Radioactive effluents with the exception of those listed below	1.0%
Radioiodines	1.0%
Tritium (^{3}H)	7.5×10^{-8} μCi/ml
Argon-41 (^{41}Ar)	3.0×10^{-6} μCi/ml

[a]in percent of the air effluent concentrations specified in Appendix B, Table 2, Column 1 of 10 CFR 20.

TABLE 11-13
INVESTIGATION LEVELS FOR RADIOACTIVE RELEASES TO SANITARY SEWER[a]

Radionuclide	Investigation Level[b]
Radioactive effluents with the exception of those listed below	5.0%
Cobalt-60 (^{60}Co)	30.0%
Sulfur-35 (^{35}S)	10.0%
Calcium-45 (^{45}Ca)	10.0%
Tritium (^{3}H)	5.0%
Zinc-65 (^{65}Zn)	10.0%

[a]based upon batch release analysis (no dilution)
[b]in percent of the release to sewer limits specified in Appendix B, Table 3 of 10 CFR 20

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 the MURR which would require a radiation survey:

a. Changes to a beamport or other reactor experiment which could lead to significant alterations in area radiation levels;

b. Entry into the mechanical equipment room (Room 114) following a reactor shutdown;

c. Lowering water level in the reactor pool below the normal operating level;

d. Removal of a Nuclear Instrumentation (NI) detector assembly from a drywell;

e. Opening the thermal column door;

f. Receipt and opening of a package containing radioactive material(s) from another facility or organization;

g. Removal of an activated or contaminated component or irradiated sample from the reactor pool;

h. As determined by a Radiation Work Permit (RWP);

i. Periodically in accessible radiation areas and high radiation areas, and in all other occupied areas in the facility;

j. Upon initial opening of the primary or pool coolant system for inspection, maintenance, or repair;

k. Prior to inspecting an irradiated fuel element;

l. Receipt of new, unirradiated reactor fuel; and

m. 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 the MURR which would require a contamination survey:

a. Receipt and opening of a package containing radioactive material(s) from another facility or organization;

b. Release of equipment, supplies, or materials from the reactor facility for unrestricted use;

c. Periodically in accessible contaminated areas and occupied areas surrounding contaminated areas;

d. Periodically in occupied non-contaminated areas of the facility;

e. Upon initial entry into potentially contaminated ventilation ducting for its inspection, repair, or filter replacement;

f. Decontamination of equipment;

g. Receipt of new, unirradiated reactor fuel;

h. Shipment of irradiated reactor fuel;

i. As determined by a Radiation Work Permit (RWP); and

j. 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 argon-41 to ensure that levels remain below the regulatory limits.

11.1.4.2 Radiation Monitoring Equipment

Radiation monitoring equipment used at the MURR is summarized in Table 11-14. 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 11-14 is available to both the Health Physics Branch and to the MURR radiation workers. Because periodic updating and replacement of instrumentation are performed to take advantage of technological improvements, Table 11-14 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 the 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 N323-1978, "American National Standard Radiation Protection Instrumentation Test and Calibration." Other radiation protection equipment is calibrated in accordance with the manufacturers' suggested procedures as applicable to the use of the instrument at the 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.

TABLE 11-14
RADIATION MONITORING AND RELATED EQUIPMENT

Equipment (Quantity)	Function	Typical Location(s)
Portable Contamination Meters (27)	Measure alpha/beta/gamma contamination levels	Laboratories Health Physics Spaces
Portable Alpha Survey Meters (2)	Measure alpha contamination levels only	Health Physics Spaces
Portable G-M Survey Meters (7) a. Underwater Detector (1) b. High Range 20K R/hr (1) c. High Range 200K R/hr (1) d. Telescoping Detectors (2) e. Standard Hand Held Unit (2)	Measure beta/gamma radiation dose rates	Laboratories Health Physics Spaces Control Room
Portable Ionization Chamber Survey Meters (20)	Measure beta/gamma radiation dose rates	Laboratories Health Physics Spaces
Portable Neutron Survey Meters (3)	Measure neutron radiation dose rates	Beamport Area Health Physics Spaces
Gamma Spectroscopy Systems (27) a. HPGe System (25) b. NaI System (2)	Perform gamma-ray spectroscopy	Laboratories
Dose Calibrators (16)	Measure beta/gamma activities	Laboratories
Alpha/Beta Planchet Counters (4)	Measure alpha/beta/gamma contamination on swipes	Counting Rooms
Liquid Scintillation Counters (2)	Measure beta contamination in liquid	Health Physics Spaces
Thermoluminescent Dosimeters (TLDs)	Measure environmental gamma radiation doses	Various On-Site and Off-Site Locations
Direct Reading Dosimeters (87)	Measure personnel gamma dose	Health Physics Spaces
Hand and Foot Monitors (1)	Measure potential contamination on hands and feet	Exit from Potentially Contaminated Areas
Portal Monitors (3)	Measure potential contamination on the whole body prior to exiting the facility	Various Locations
Air Monitors (9) a. Off-Gas Radiation Monitor (2) b. Continuous Air Monitors (6) c. Portable Air Sampler (1)	 Measure radioactivity in stack effluent Measure area airborne radioactivity Collect grab air samples	 West Tower Various Locations Health Physics Spaces
Area Radiation Monitors (ARMs) (10)	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

TABLE 11-14
RADIATION MONITORING AND RELATED EQUIPMENT (cont)

Equipment (Quantity)	Function	Typical Location(s)
Air Flow Measurement Instruments (2)	Measure facility ventilation flow rates	Health Physics Spaces
Gas Content Meter	Measure gas concentrations in air	MURR Safety Associate Office

The following instrumentation is normally calibrated at the MURR by individuals trained to perform primary instrument calibration:

- Portable Contamination Meters ("friskers");
- Gamma Spectroscopy Counting Systems;
- Dose Calibrators;
- Alpha/Beta Planchet Counters;
- Direct-Reading Pocket Dosimeters;
- Portal Monitors;
- Portable Air Samplers;
- Continuous Air Monitors; and
- Off-Gas Radiation Monitor.

The following instrumentation is normally calibrated at the Research Park Development Building with the assistance of Environmental Health and Safety (EH&S) personnel:

- Portable Alpha Survey Meters;
- 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 Neutron Survey Meters;
- Air Flow Measurement Instruments; and
- Gas Content Meter.

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 Health Physics Branch 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) is attached to all radiation survey meters.

11.1.5 Radiation Exposure Control and Dosimetry

Radiation exposure control at the 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 have been discussed previously in Section 11.1.2 and Section 11.1.3, respectively. Therefore, this section of this chapter will focus on 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. With nearly 40 years of operational history, the installed radiation shielding has performed more than adequately as analyzed and designed.

The following dose rate schedule is used in the analysis and design of the primary reactor shielding:

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

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

(c) 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 millirem/h;

(d) 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 millirem/h;

(e) At one foot (0.3 m) from the demineralizer cell's walls and ceiling, the radiation level shall not exceed 2.5 millirem/h;

(f) 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 millirem/h; and

(g) At any location at the top surface of the reactor pool, the radiation level shall not exceed 20 millirem/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 10^5-second (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 millirem/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 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 millirem per hour 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 millirem/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:

1. 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;*

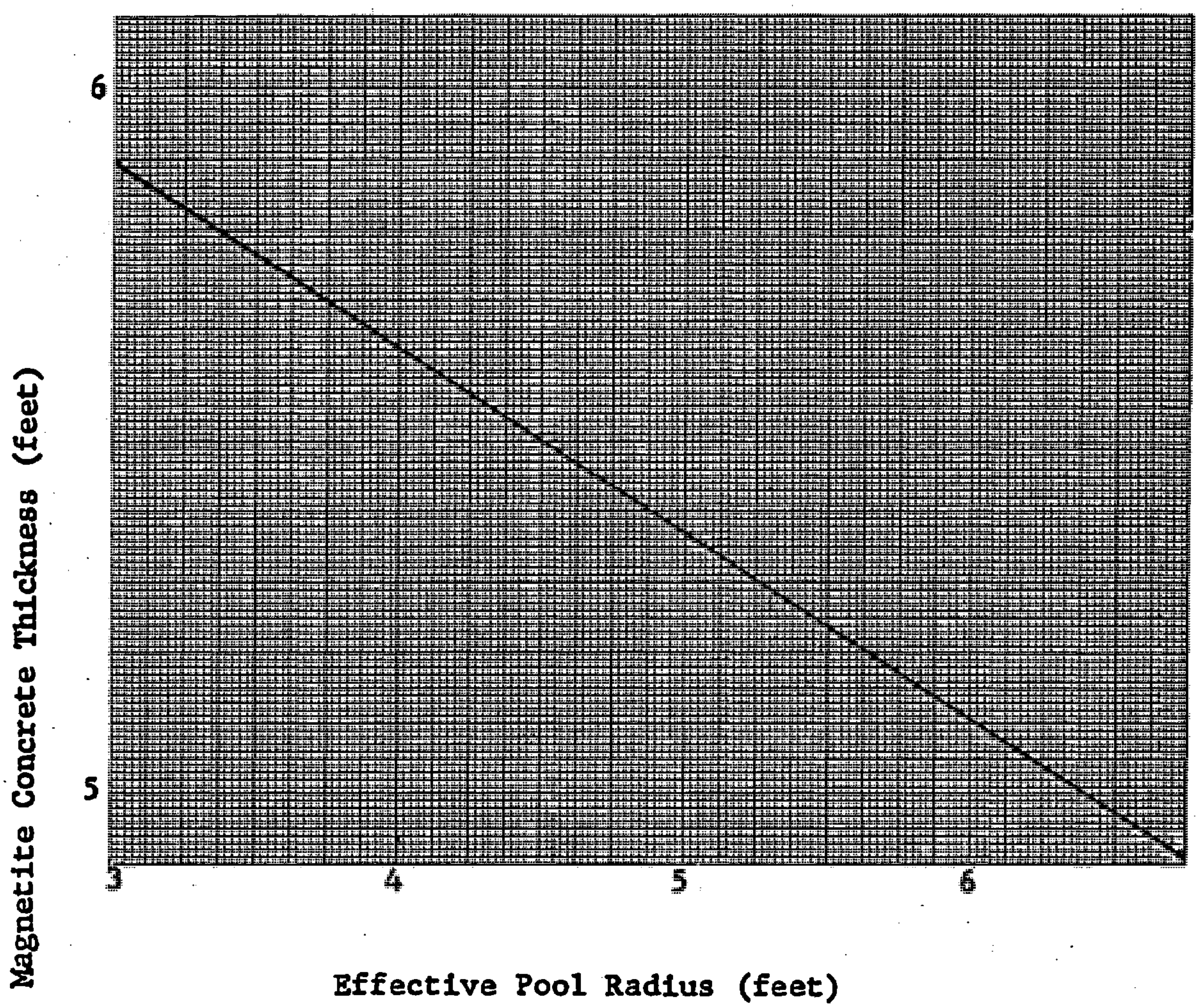

FIGURE 11.1
BIOLOGICAL SHIELD MAGNETITE CONCRETE
VS. EFFECTIVE POOL RADIUM

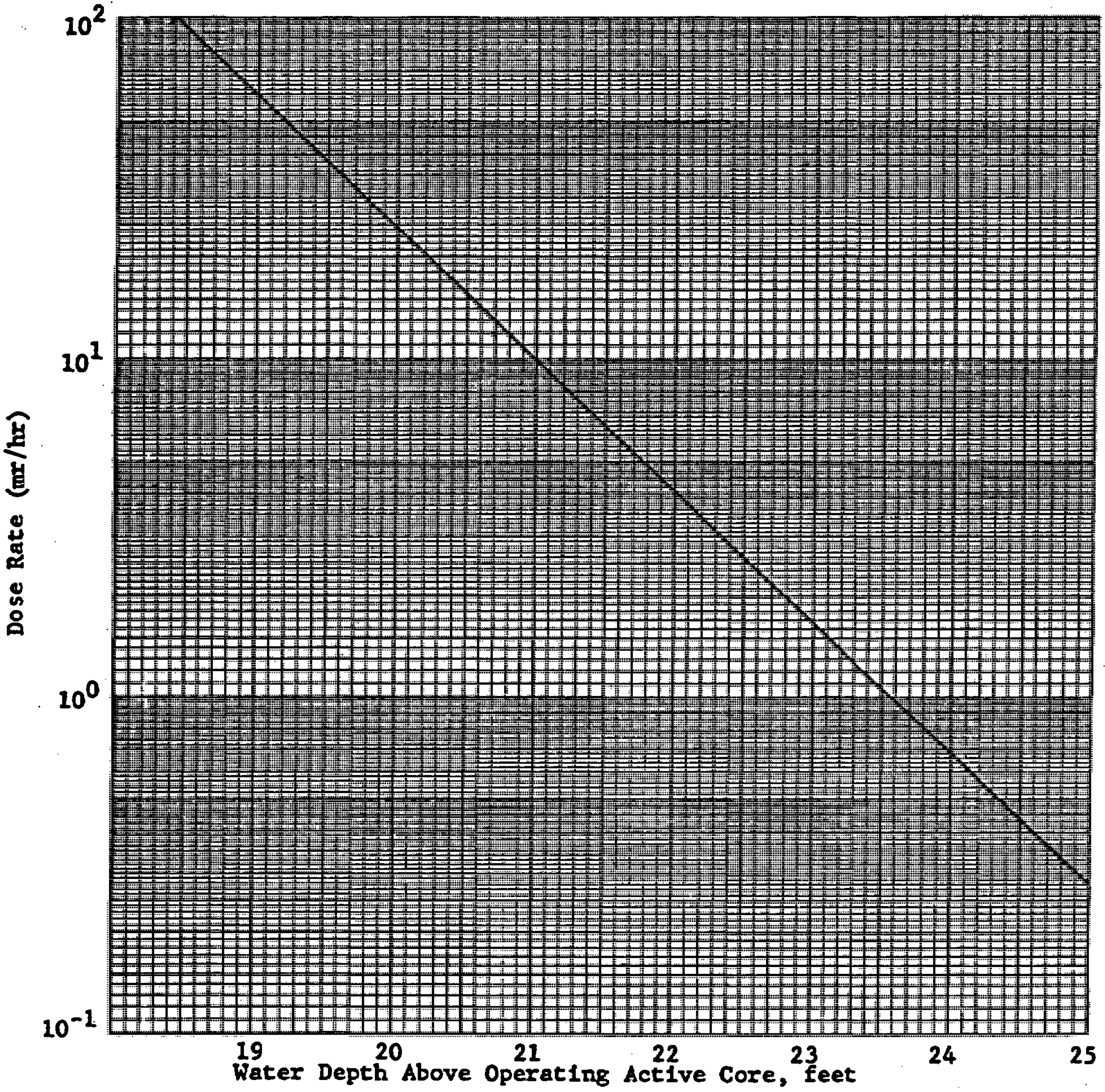

FIGURE 11.2
DIRECT PENETRATION GAMMA-RAY DOSE RATES AS A FUNCTION
OF WATER DEPTH ABOVE THE ACTIVE CORE REGION

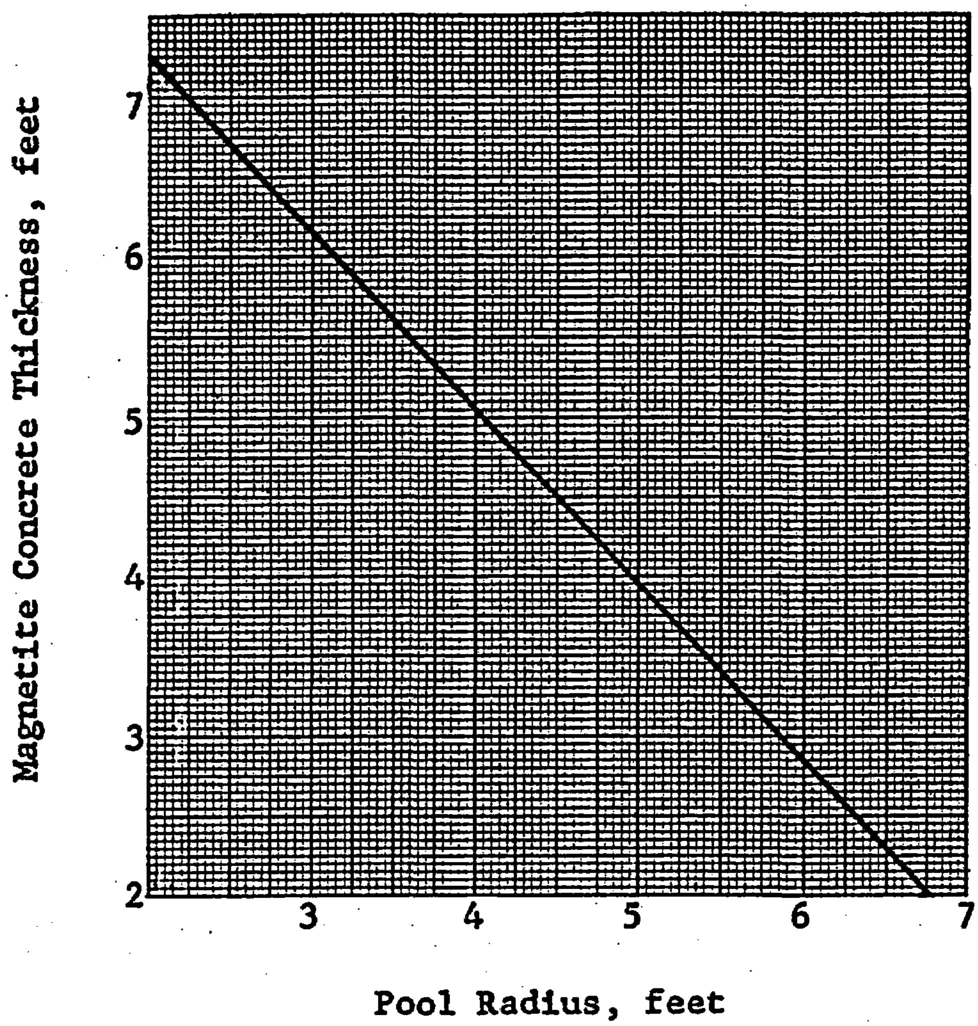

FIGURE 11.3
MAGNETITE CONCRETE BIOLOGICAL SHIELDING REQUIREMENTS
VS. EFFECTIVE POOL RADIUS TO ACHIEVE ONE-TENTH
OF DOSE RATE CRITERIA FROM FAST NEUTRONS

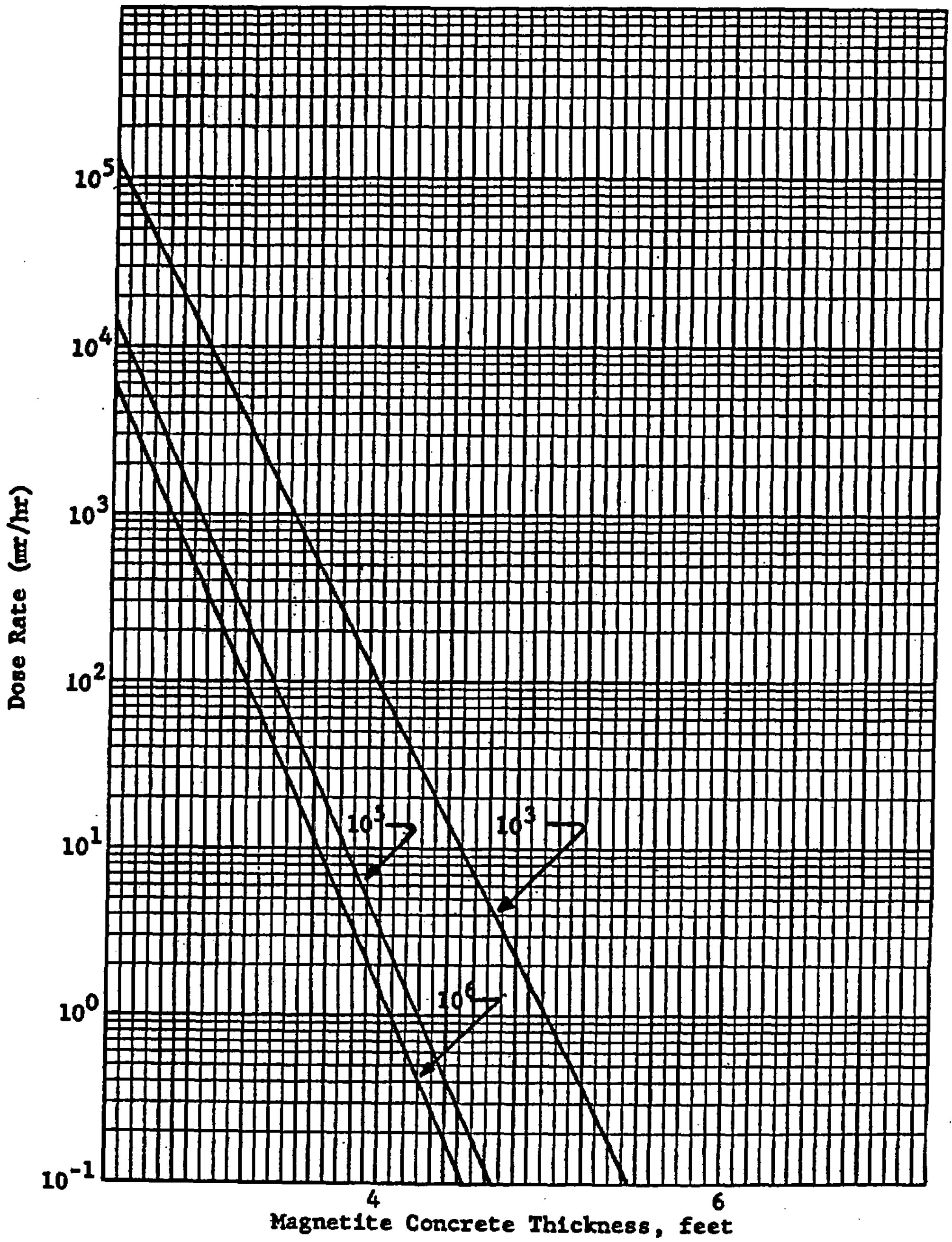

FIGURE 11.4
SPENT FUEL ELEMENT DOSE RATES THROUGH MAGNETITE SHIELDING
FOR A SINGLE ELEMENT
AS A FUNCTION OF FISSION PRODUCT DECAY TIME

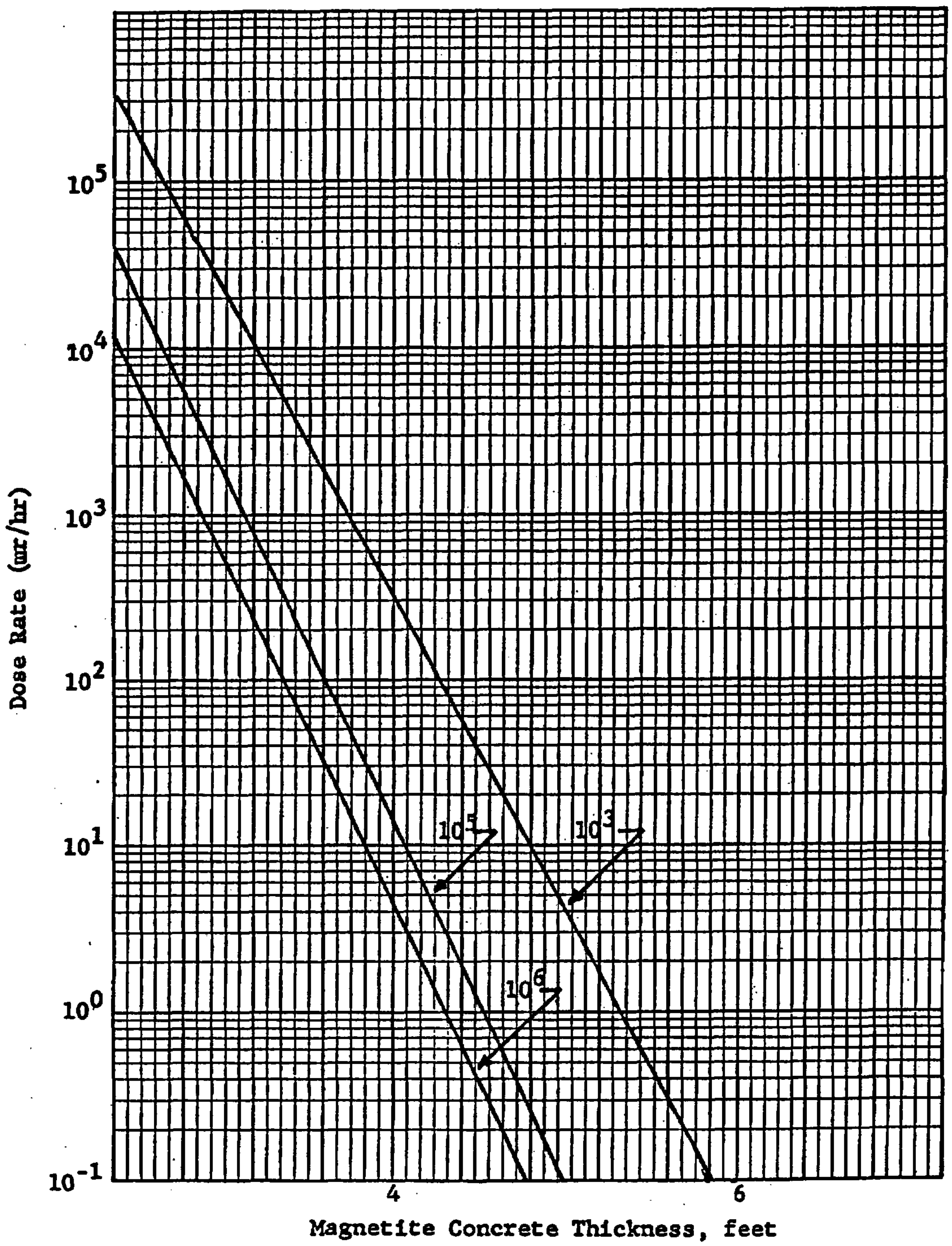

FIGURE 11.5
SPENT FUEL ELEMENT DOSE RATES THROUGH MAGNETITE SHIELDING
FOR FOUR ELEMENTS
AS A FUNCTION OF FISSION PRODUCT DECAY TIME

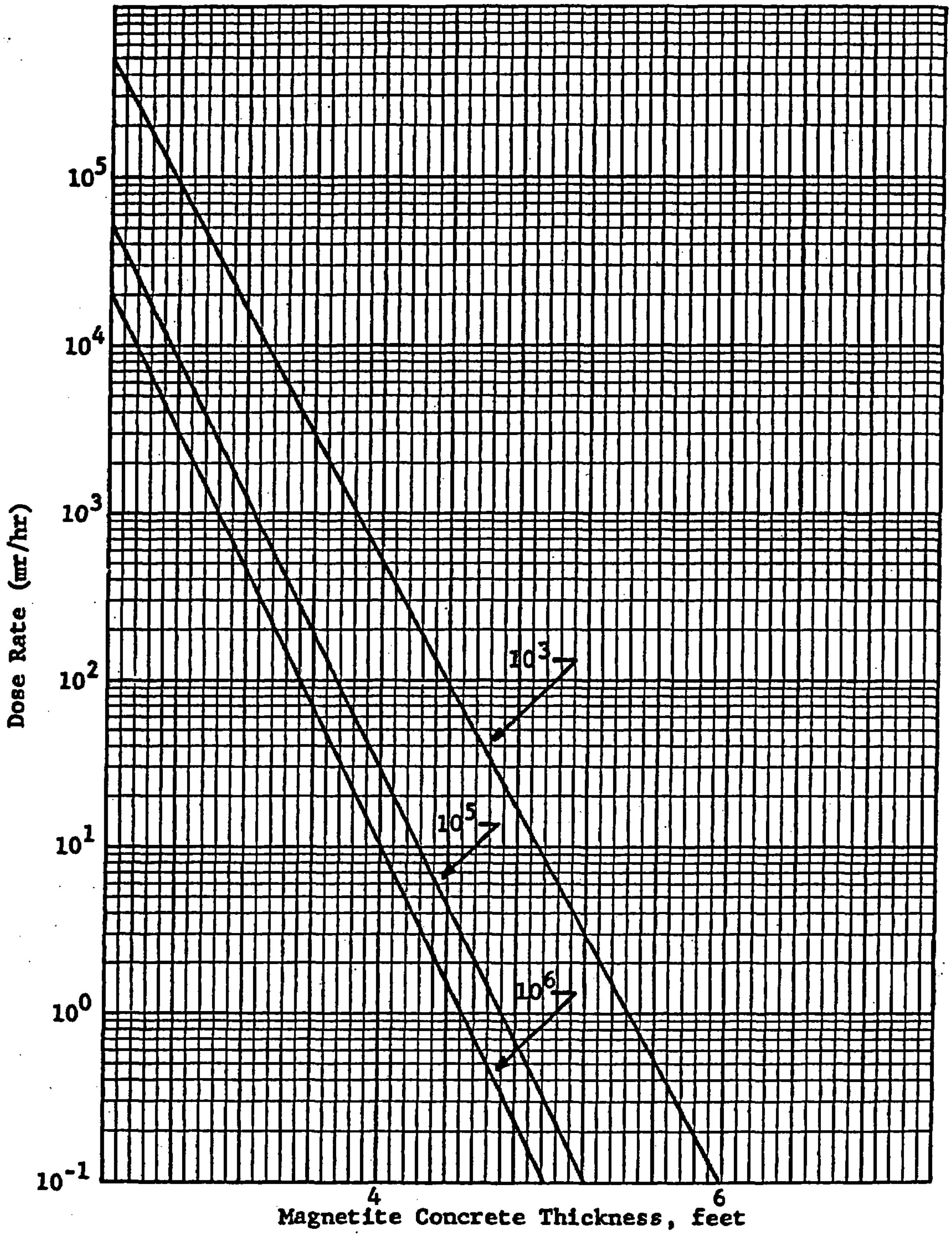

FIGURE 11.6
SPENT FUEL ELEMENT DOSE RATES THROUGH MAGNETITE SHIELDING
FOR EIGHT ELEMENTS
AS A FUNCTION OF FISSION PRODUCT DECAY TIME

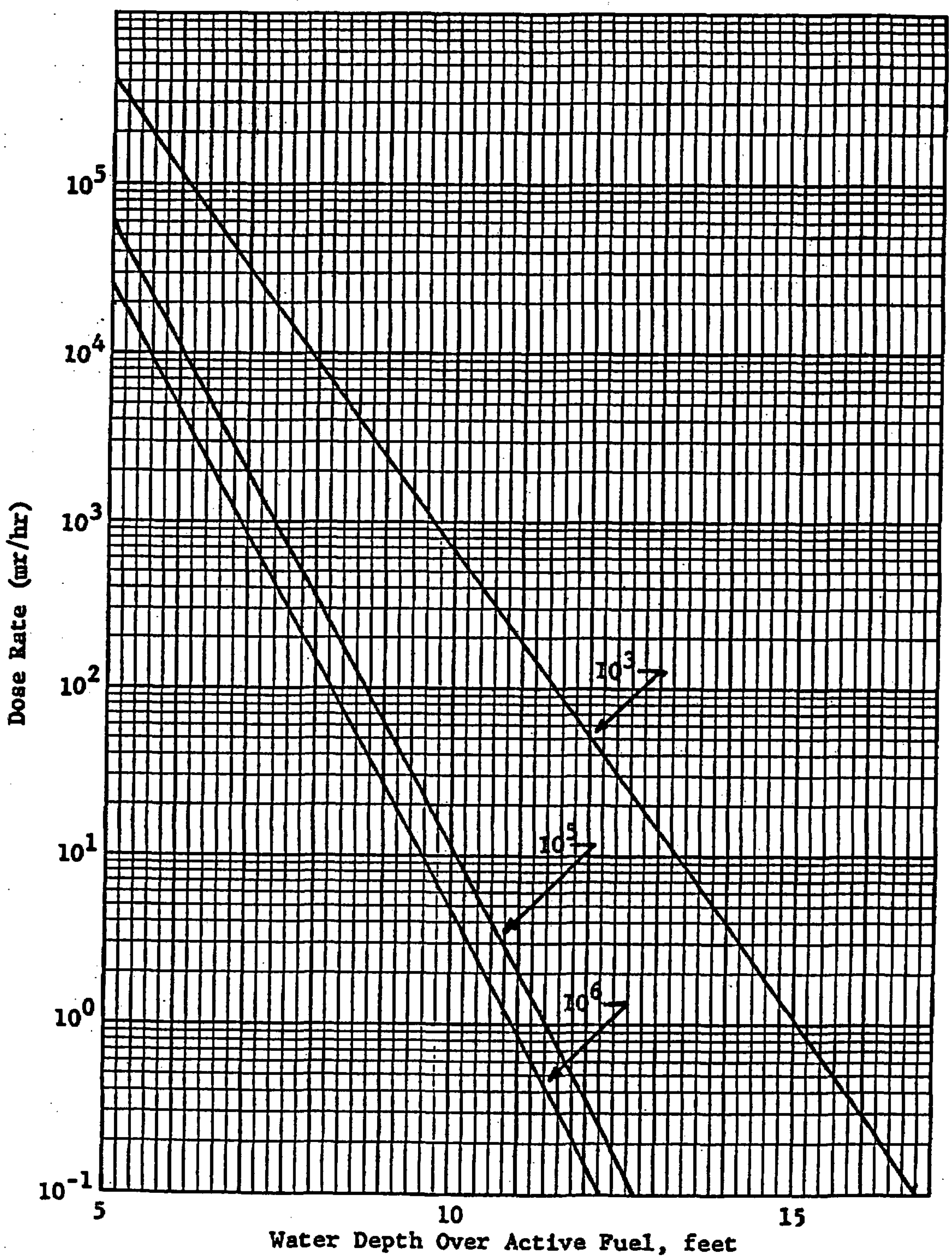

FIGURE 11.7
SPENT FUEL ELEMENT DOSE RATES THROUGH WATER SHIELDING
AS A FUNCTION OF
FISSION PRODUCT DECAY TIME - HORIZONTAL POSITION

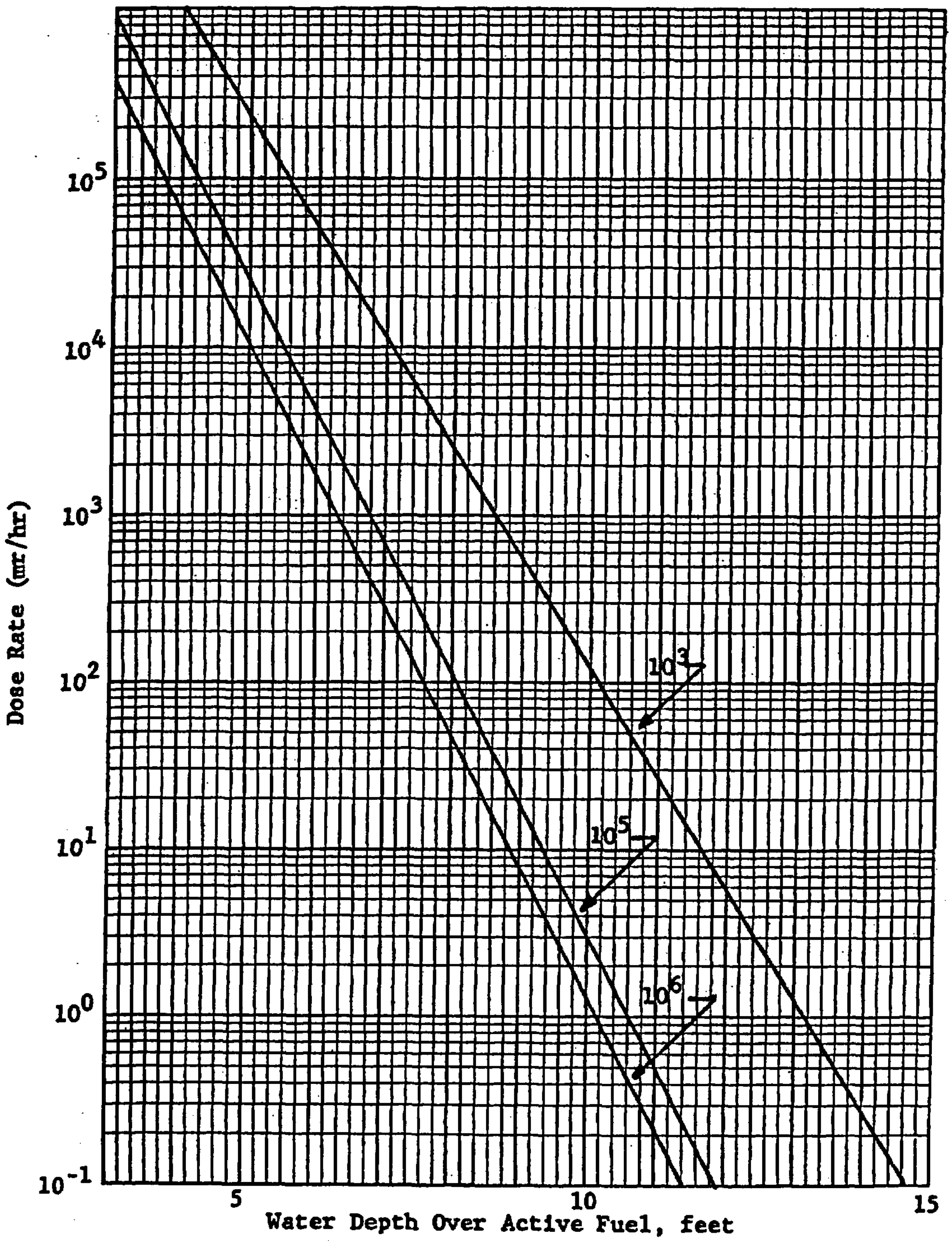

FIGURE 11.8
SPENT FUEL ELEMENT DOSE RATES THROUGH WATER SHIELDING
AS A FUNCTION OF
FISSION PRODUCT DECAY TIME - VERTICAL POSITION

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

3. 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.

*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, Methods of Storage and Transfer.

The functional shielding requirements can then be defined as follows:

1. The minimum concrete thicknesses around stored fuel;

2. 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;

3. 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

4. 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:

1. Storage of Fuel Elements Adjacent to the Primary Reactor Shield:

   a. 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 millirem/h at 1 foot (0.3 m) from the shield surface' requires 52 inches (1.3 m) of magnetite concrete; and

   b. 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 millirem/h for a worker in the deep pool' requires that the dividing wall have a magnetite concrete thickness of 47 inches (1.2 m);

2. Transfer Operations at a Lowered Water Level in the Reactor Pool:

   a. 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 millirem/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

   b. 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 millirem/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 millirem/h' requires 12½ 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];

3. Transfer Operations at Normal Operating Level in the Reactor Pool:

   a. 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 millirem/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

4. Reactor Pool Water Level Lowered to the Pool Weir:

   a. 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 millirem/h at the water surface' is 10 feet (3 m) over the baskets.

11.1.5.1.3 Experimental Facilities

This section describes the 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 millirem/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:

  a. Annuli (water or air filled[1]) between the primary reactor shield casing (fixed beamport liner), the beam tube (removable beamport liner), and the beamport shield plug;[2]

  b. 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^3;

  c. The beamport vestibule at the primary reactor shield surface; and

  d. The coolant activation in the beam tube and/or experiment can and the subsequent passage to the outer portion of the reactor shield.[3]

  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:

  a. An air or water annulus of varying thickness followed by lead and concrete shielding or vice versa;

  b. A homogenized annulus of aluminum and air or water of varying thicknesses followed by lead and concrete shielding or vice versa;

  c. An air or water annulus of varying thickness traversing the entire primary reactor shield; and

[1]Presently, the volume in front of the shield plug is filled with either helium or demineralized water.

[2]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).

[3]Experiment cans are no longer utilized in the beamport assemblies.

d. An 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:

a. 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

b. A minimum step of 2 inches (5.1 cm) is required with the inclusion of the above lead shielding.

Note: 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 and to the mixing of the activity in the beam tube. Further consideration of shielding was given to 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..

- <u>Thermal Column Shielding</u> - 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 11-15 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.

TABLE 11-15
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	68.75 (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.

The thermal column case is constructed of two square boxes. The in-pool portion immediately behind the lead shield is a 3 foot 1½-inch (95.3-cm) square by 12¼-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 ¼-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.

- 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 10½ 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 x 4 x 10 inches (10.2 x 10.2 x 25.4 cm) and 4 x 4 x 10½ inches (10.2 x 10.2 x 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 54½ 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 millirem/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 nitrogen-16 (^{16}N) activity in the two coolant systems. ^{16}N, a high-energy beta and gamma emitter with a half-life of seven seconds, is produced when oxygen in the primary and pool coolant is irradiated with neutrons of sufficient energy. 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 ^{16}N 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 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 primary coolant piping, the 6-inch 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:

1. Invert Loop:

   (a) The invert loop is located in the reactor pool adjacent to the primary reactor shield;
   (b) The radiation level criterion is '2.5 millirem/h at one foot (0.3 m) from the shield surface;' and
   (c) The shielding requirement is '3½ feet (1.1 m) of magnetite concrete or its equivalent in all directions from the pipe section.'

2. Mechanical Equipment Room (Room 114) Pipe Tunnel:

   (a) 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;
   (b) The radiation level criterion is '2.0 millirem/h at one foot (0.3 m) from the shield surface;' and
   (c) The shielding requirement is '5 feet (1.5 m) of ordinary concrete or its equivalent.'

3. Mechanical Equipment Room:

   (a) 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;
   (b) The radiation level criterion is '2.5 millirem/h at one foot (0.3 m) from the shield surface;' and
   (c) The shielding requirement is '5 feet (1.5 m) of ordinary concrete or its equivalent.'

- <u>Demineralizer System</u> - 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, thus avoiding the need to shield ^{16}N 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 (^{19}O) activity which influences the shielding requirements for these lines.

Separate demineralizer tanks are used for the primary and pool coolant systems. However, these tanks, plus the spare unit, are interchangeable, hence the shielding requirements are identical for all three tanks. 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:

1. Demineralizer System Piping:

   (a) The piping extends from the primary and pool coolant systems in Room 114 to the demineralizer tanks;
   (b) The radiation level criterion is '2.5 millirem/h at one foot (0.3 m) from the shield surface;' and
   (c) The shielding requirement is '1 foot (0.3 m) of ordinary concrete or its equivalent.'

2. Demineralizer Tanks:

   (a) 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;
   (b) The radiation level criterion is '2.5 millirem/h at one foot (0.3 m) from the shield surface;' and
   (c) 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 ^{16}N and ^{19}O activities to decay.

3. Regeneration Piping:

   (a) Location of the regeneration piping is in the valve tunnel adjacent to and in front of the demineralizer cells. The regeneration lines are used to "sluice" the resin beds from the tanks to the regeneration unit;
   (b) The radiation level criterion is '2.5 millirem/h at one foot (0.3 m) from the shield surface;' and
   (c) The shielding requirement is '2 feet (0.6 m) of ordinary concrete or its equivalent.'

4. Radioactive Liquid Waste Tanks:

   (a) 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;
   (b) The radiation level criterion is '2.5 millirem/h at one foot (0.3 m) from the shield surface;' and
   (c) 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, thus 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 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 high efficiency particulate air (HEPA) filters.

- The mechanical equipment room's (Room 114) exhaust system contains a pre-filter, two HEPA filters, and two activated charcoal filters.

- The pneumatic tube (p-tube) system exhausts through a HEPA filter. The system is also designed to minimize the instantaneous release of argon-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. Also, a time delay circuit starts the second p-tube blower approximately 15 seconds after the first p-tube blower, thus minimizing an air surge 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 the Reactor Operations' 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 the 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 based on the fact that it is a high radiation area during operation of the reactor. This is due primarily to the nitrogen-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 based on the fact that these rooms are high radiation areas during operation of the reactor. This is due primarily to the oxygen-19 activity in the primary and pool coolant lines supplying the demineralizer tanks and the buildup of activation products (mainly sodium-24 and cobalt-60) in the resin beds. This location is also used as a storage area for radioactive material, thus potentially creating 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 based on the fact that, 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 Radiation Protection Program are summarized in Table 11-16. 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.

TABLE 11-16
PROTECTIVE EQUIPMENT FOR RADIATION PROTECTION PROGRAM

Equipment	Use[a]
Tyvek Coveralls	R & E
Lab Coats	R & E
Rubber Gloves	R & E
Latex Examination Gloves	R & E
Safety Glasses	R & E
Coveralls	R & E
Hoods/Caps	R & E
Nylon Shoe Covers	R & E
Plastic Shoe Covers	R & E
Rubber Overshoes	R & E
Decontamination Sink	R & E
Decontamination Showers	E

[a]Use: R = Routine; E = Emergency.

The Health Physics Branch 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 Research Park Development Building. In addition to some of the protective equipment listed in Table 11-16, 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.

11.1.5.5.1 <u>Respiratory Protection Equipment</u>

Other than argon-41 (^{41}Ar), 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 the MURR. Should the situation change and respiratory protection become necessary to meet the ALARA objectives, the facility will implement and maintain a respiratory protection program in accordance with Title 10, Chapter I of the Code of Federal Regulations, Part 20 (10 CFR 20), Subpart H, "Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas."

11.1.5.5.2 <u>Personnel Dosimetry Devices</u>

Personnel dosimetry devices have been selected to provide monitoring of all radiation types likely to be encountered. Table 11-17 provides a summary of the dosimetry devices typically used at the MURR.

TABLE 11-17
PERSONNEL DOSIMETRY DEVICES

Device	Radiation Measured	Dose
Optically Stimulated Luminescence (OSL) Dosimeter (with track-etch)	X-Ray, Beta, Gamma, and Neutron	Deep-Dose Equivalent Eye Dose Equivalent Shallow-Dose Equivalent
Ring Badge (TLD)	X-Ray, Beta, and Gamma	Shallow-Dose Equivalent, Extremity
Wrist Badge (TLD)	X-Ray, Beta, and Gamma	Shallow-Dose Equivalent, Extremity
Direct Reading Pocket Dosimeters (Electronic or Pencil type)	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. With the exception of the direct reading pocket dosimeters, dosimetry devices are exchanged at regular time intervals. The dosimetry devices are processed by a contractor and the results are returned to the Health Physics Branch 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). The direct reading pocket dosimeters are typically read and recorded daily when used. Occupational exposure records are maintained by the Health Physics Branch and are retained for the life of the facility. In addition, Radiation Work Permits (RWPs) which document radiation exposures received during a radiological operation are also maintained by the Health Physics Branch 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 the 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 Health Physics Branch. 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 previously 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.

Based upon the ALARA dose trend analysis charts maintained by the Health Physics Branch, there has been a relatively downward trend in total person-rem over the past five years. The average annual occupational whole body exposure (deep dose equivalent) for a reactor operations and a health physics staff member is approximately 815 mrem/y and 500 mrem/y, 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. With respect to the MURR, the "restricted" area is considered to be all locations within the operations boundary (the outer walls of the laboratory and reactor containment buildings). 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 the MURR operates 24 hours a day, 6½ days per week, 52 weeks per year, it is assumed that an individual working at the MURR will be in the facility only one shift per day (40 hours per week). Furthermore, it is assumed that an occupationally-exposed individual will spend only a limited amount of time in areas where there is a potential for significant radiation levels (within the demineralizer cells, the waste tank room, or in the mechanical equipment room). Therefore, the 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 Health Physics Branch and Reactor Operations Staff. Table 11-18 shows the film badge recorded annual dose measurements as well as the monthly average at certain locations in the containment building for the period 2001 to 2005. Using the location which has the highest monthly average as a conservative estimate, the calculated average radiation level on the beamport floor with the reactor operating at 10 MW is approximately 0.6 millirem/h (431 millirem/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 or no recorded dose each month.

TABLE 11-18
DOSE MEASUREMENTS BY FILM BADGE AT SELECTED LOCATIONS WITHIN THE REACTOR CONTAINMENT BUILDING (in millirem)

	Total for Year					Monthly
Location	2001	2002	2003	2004	2005	Average
North Beamport Floor	3,330	3,197	3,038	3,770	3,971	288
South Beamport Floor	1,640	1,704	1,440	1,286	1,191	121
East Beamport Floor	1,470	1,587	1,117	964	775	98
West Beamport Floor	5,100	5,420	5,276	5,004	5,009	430
Upper Bridge Area	6,460	7,134	6,929	6,942	6,754	570
Reactor Control Room	580	718	720	623	518	53

Reactor operators spend a large portion of their time in the reactor control room and on the upper bridge where film badges placed at these locations routinely record radiation doses on the order of 53 millirem/month and 570 millirem/month, respectively. In addition, operators perform a large number of 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 10 millirem/h and 20 millirem/h, respectively. Entry into Room 114 is normally not allowed while the reactor is operating. Therefore, it is estimated that personnel exposures from this type of activity will be insignificant.

With the exception of loading a new, or dumping a depleted, resin bed from a demineralizer tank, 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 storage area for radioactive material, potentially creating radiation levels in excess of 5 rem/h. 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 millirem/h). In addition, personnel doses from work performed in the demineralizer cell area are usually documented on a Radiation Work Permit (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 millirem/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 millirem/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 millirem/h.

While the above predictions provide some indication of what the annual Total Effective Dose Equivalent (TEDE) might be for occupationally-exposed individuals at the MURR, the facility has measured staff personnel radiation doses for many years. Using these data to predict annual doses provides even more assurance that the personnel doses will remain low. Table 11-19 shows the total person-rem by group from 2001 to 2005. Table 11-20 shows the average annual dose for an individual in each group during the same reporting period.

TABLE 11-19
PERSON-REM BY GROUP AT THE MURR

	Year				
Group Name	2001	2002	2003	2004	2005
Hot Cell	3.73	3.09	1.91	1.88	2.78
Shipping	2.63	0.95	0.70	0.93	1.70
Health Physics	6.26	4.77	1.92	1.94	2.37
Isotope Production	3.36	1.86	1.93	1.03	0.71
Facility Support	2.01	1.39	1.76	1.31	2.17
Reactor Operations	20.12	17.76	15.27	16.20	18.19

TABLE 11-20
AVERAGE ANNUAL RADIATION DOSE BY GROUP AT THE MURR
(in millirem/yr)

Group Name	Year				
	2001	2002	2003	2004	2005
Hot Cell	932	773	478	470	695
Shipping	526	190	14	186	340
Health Physics	626	477	192	194	237
Isotope Production	336	186	193	172	118
Facility Support	126	87	110	82	135
Reactor Operations	805	710	610	648	724

Note: The groups represented in Tables 11-19 and 11-20 are the top six groups, in terms of total dose, of individuals badged at MURR and collectively represent over 85% of the total facility dose received at MURR.

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 the MURR declares her pregnancy, in most cases she may continue her 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 the 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 1 0% of any limit stated in 10 CFR 20. A Controlled Special Exposure Authorization Form is prepared by the Health Physics Branch which documents authorization of the exposure by the MURR management and the approved dose limit. A Radiation Work Permit (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 argon-41 production during normal operation of the MURR reactor is contained in Section 11.1.1.1.3 and in Appendix B. Using the nearest occupied facilities and most probable wind direction, the annual dose values for the unrestricted area indicate a maximum TEDE, primarily from argon-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, but may actually decrease as the facility continues to explore new techniques and their possible implementation as practical measures to reduce argon-41 releases.

11.1.6 Contamination Control

Radioactive contamination is controlled at the 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 has been previously 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 Radiation Work Permit (RWP) shall provide the necessary instructions for contamination control. Each individual performing work under an 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. Additionally, portal monitoring shall be performed when deemed necessary by Health Physics 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 an RWP. Anti-C clothing is typically laundered after each use.

- 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. A portal monitor located in the containment building lobby is 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 Health Physics Branch 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 the Reactor Operations' 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 the techniques for avoiding, limiting, and controlling contamination.

11.1.7 <u>Environmental Monitoring</u>

The MURR Environmental Monitoring Program has continued for the period of time from initial operation in October 1966 to the present 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. 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 Health Physics 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 for gamma emitters ~1.30 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 for gamma emitters ~3.00 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 for gamma emitters ~215 pCi/liter; and

- Integrated gamma dose measurements using thermoluminescent dosimeters (TLDs) which are exchanged quarterly at up to forty-five (45) locations (this part of the program was instituted in 1991) with typical sensitivity ~1.0 millirem/quarter.

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. TLDs are processed by a contractor and the results are returned to the Health Physics Branch for review and compilation. The TLDs 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.

The procedures for carrying out the environmental monitoring program are contained in the MURR Regulatory Assurance Procedures Manual. The procedures are focused on ensuring a comprehensive 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

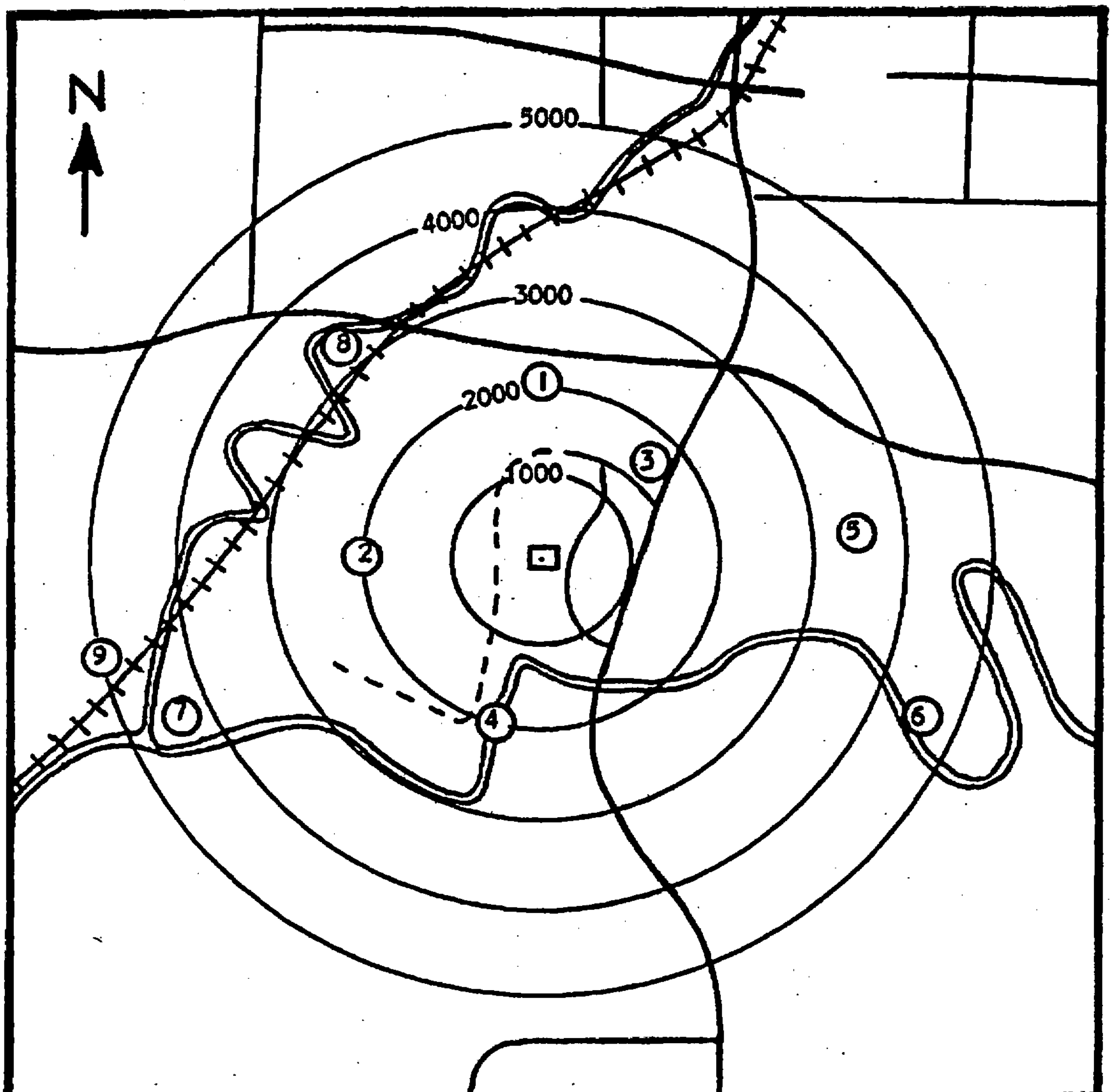

FIGURE 11.9
ENVIRONMENTAL SAMPLING STATIONS

sensitivity and reported in the Reactor Operations Annual Report. Document control measures for these procedures are described in Section 11.1.2.3.

The procedures for carrying out the environmental monitoring program are contained in the MURR Regulatory Assurance Procedures Manual. The procedures are focused on ensuring a comprehensive 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 11.1.2.3.

Environmental monitoring locations and the types of measurements made or the samples collected are summarized in Tables 11-21 and 11-22. Environmental samples are no longer collected at Stations 8 and 9. In September 1983, the City of Columbia's sewage treatment facilities at these locations were closed. A new wastewater treatment plant was constructed 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 11-21
ENVIRONMENTAL MONITORING PROGRAM

Sample Station	Location	Direction	Type
1	University Hall	N	V, S
2	A. L. Gustin Golf Course	WNW	V, S
3	Research Park Development Building	NE	V, S
4	Hinkson Creek	SSW	V, S, W
5	Providence Point	E	V, S
6	Hinkson Creek Nature Preserve	ESE	V, S, W
7	A. L. Gustin Golf Course	WSW	V, S
10	Regional Wastewater Treatment Plant	WSW	V, S, W

V = Vegetation Sample; S = Soil Sample; W = Water Sample

TABLE 11-22
ENVIRONMENTAL THERMOLUMINESCENT DOSIMETERS

Badge No.	Location	Direction	Distance[a]
1	Unassigned	N/A	N/A
2	Unassigned	N/A	N/A
3	Reactor Facility Site	N/A	N/A
4	Reactor Facility Site	SW	37
5	Reactor Facility Site	SW	42
6	Reactor Facility Site	N	34
7	Reactor Facility Site	NE	57
8	Reactor Facility Site	SW	27
9	Reactor Facility Site	S	27
10	USDA Research Laboratory	NNE	149
11	USDA Research Laboratory	N	149
12	Dalton Research Center	NE	301
13	Laboratory Animal Center	NNE	316
14	Science Instrument Shop	S	156
15	Reactor Facility Cooling Tower	S	65
16	Reactor Facility Site	SE	107
17	Red Cross Building	E	293
18	Research Park Development Building	NE	476
19	Tom N. Taylor Sports Complex	NE	606
20	Corner of Stadium Blvd & Providence Rd	NE	907
21	Research Park - Picnic Shelter	SSE	236
22	Research Park - Reactor Field	ESE	168
23	Reactor Facility Site	WNW	110
24	Research Park	SSW	328
25	Agronomy Research Drive	SSW	480
26	A. L. Gustin Golf Course - 13th Tee	SW	301
27	Southwest Well	SW	141
28	A. L. Gustin Golf Course - 15th Tee	WNW	210
29	A. L. Gustin Golf Course - 12th Tee	NW	255
30	A. L. Gustin Golf Course - Ground Crew Bldg	NNW	328
31	University Hall	NNW	671
32	Old Alumni Center	NNW	724
33	Practice Field	E	671
34	Memorial Stadium Parking Lot	ENE	587
35	Hinkson Creek	SSE	499
36	Hinkson Creek	SE	419
37	Faurot Field Parking Lot	NE	690
38	A. L. Gustin Golf Course - 9th Tee	NW	556
39	A. L. Gustin Golf Course - 17th Tee	W	491
40	University Hall Parking Lot	N	514
41	USDA Research Laboratory	NE	137
42	Reactor Facility Laboratory Building	N/A	N/A
43	Reactor Facility Laboratory Building	N/A	N/A
44	MURR Site (Pad)	SW	110
45	Reactor Facility Cooling Tower	S	65

[a]Distance is measured in meters from the reactor facility ventilation exhaust stack.

11.2 Radioactive Waste Management

The 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 reactor 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.

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 the MURR to ensure contaminated and non-contaminated material is properly segregated, contained, and labeled.

The Health Physics Branch is responsible for administering the Radioactive Waste Management Program. The organizational structure, the authorities and responsibilities, and the position qualifications for the Health Physics Branch are discussed in Section 11.1.2.1 and Chapter 12, Conduct of Operations. The Radiation Safety Officer (RSO), with the assistance of the Health Physics Branch, is responsible for the safe disposal of radioactive waste from materials licensed under the MURR Broad Scope Material 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 Radiation Protection Program and other radiation safety programs (e.g., ALARA program). The audit is performed by the 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 Health Physics Branch. 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 Health Physics 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 Health Physics Branch 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 the experimental facilities. Section 11.1.1.3 summarizes the sources of solid waste at the MURR.

Receptacles lined with polyethylene 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 on the below-grade level of the laboratory building 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 the 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 Health Physics Branch 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.

As previously stated, it is the policy of the reactor facility to keep the volume of waste materials generated to the absolute minimum. Although there are no numerical volume goals set at the facility, the Reactor and Reactor Health Physics 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

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. The liquid waste retention and disposal system consists of four tanks, three with a capacity of approximately 5,000 gallons (18,927 l) and a fourth with a capacity of 550 gallons (2,082 l), three transfer pumps, three filter banks, and associated piping and valves.

Liquid waste is retained or chemically treated until an assay indicates 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 which tends to precipitate-out the radionuclides or may consist of 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 coolant is, 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 argon-41 (^{41}Ar) 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 previously described. Releases through the MURR stack are usually classified as an 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 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.

A complete description of ^{41}Ar production and its subsequent discharge into the unrestricted environment is contained in Section 11.1.1.1 and Appendix B. Furthermore, 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 argon-41) and particulate activity through the facility ventilation exhaust stack has been previously discussed in Section 11.1.1.1 and Appendix B. 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. It is the policy of the MURR to use 5% of each isotope's total limit as an administrative limit; although there are a few isotopes that have a higher administrative limit. These limits ensure that the liquid waste is retained as long as practical to allow the activity to decay.

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 Title 49, Chapter I of the Code of Federal Regulations, Part 172, Subpart H.

CHAPTER 12

CONDUCT OF OPERATIONS

TABLE OF CONTENTS

LIST OF FIGURES

REFERENCES

12.1 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," 1990.

12.2 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.4, "Selection and Training of Personnel for Research Reactors," 1988.

12.3 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.11, "Radiation Protection at Research Reactor Facilities," 1993.

12.4 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.16, "Emergency Planning for Research Reactors," 1982.

12.5 Office of Nuclear Reactor Regulation, NUREG-1478, "Non-Power Reactor Operator Licensing Standards," 1994.

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 (Ref. 12.1).

12.1 Organization

The Missouri University Research Reactor (MURR) falls within the organizational structure of the University of Missouri at Columbia 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 main categories: academic departments, each of which is supervised by a dean; non-academic services, which include the budget, construction, auditing, etc.; and academic research and service organizations, which include research facilities such as the MURR.

The reactor and laboratory facilities of the MURR are available to faculty members or graduate and undergraduate students interested in pursing 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 the 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-member Board of Curators appointed by the Governor and confirmed by the State Senate to serve a six-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 to the MURR Director's Office. The Director's Office consists of the Reactor Facility Director and the Chief Operating Officer. The Reactor Facility Director has overall responsibility for the direction and operation of the MURR. He delegates the internal direction of the Reactor Operations Branch to the Chief Operating Officer and the Health Physics Branch to the Associate Director, Regulatory Assurance Group. The Reactor Facility Director reports to the Office of the Provost.

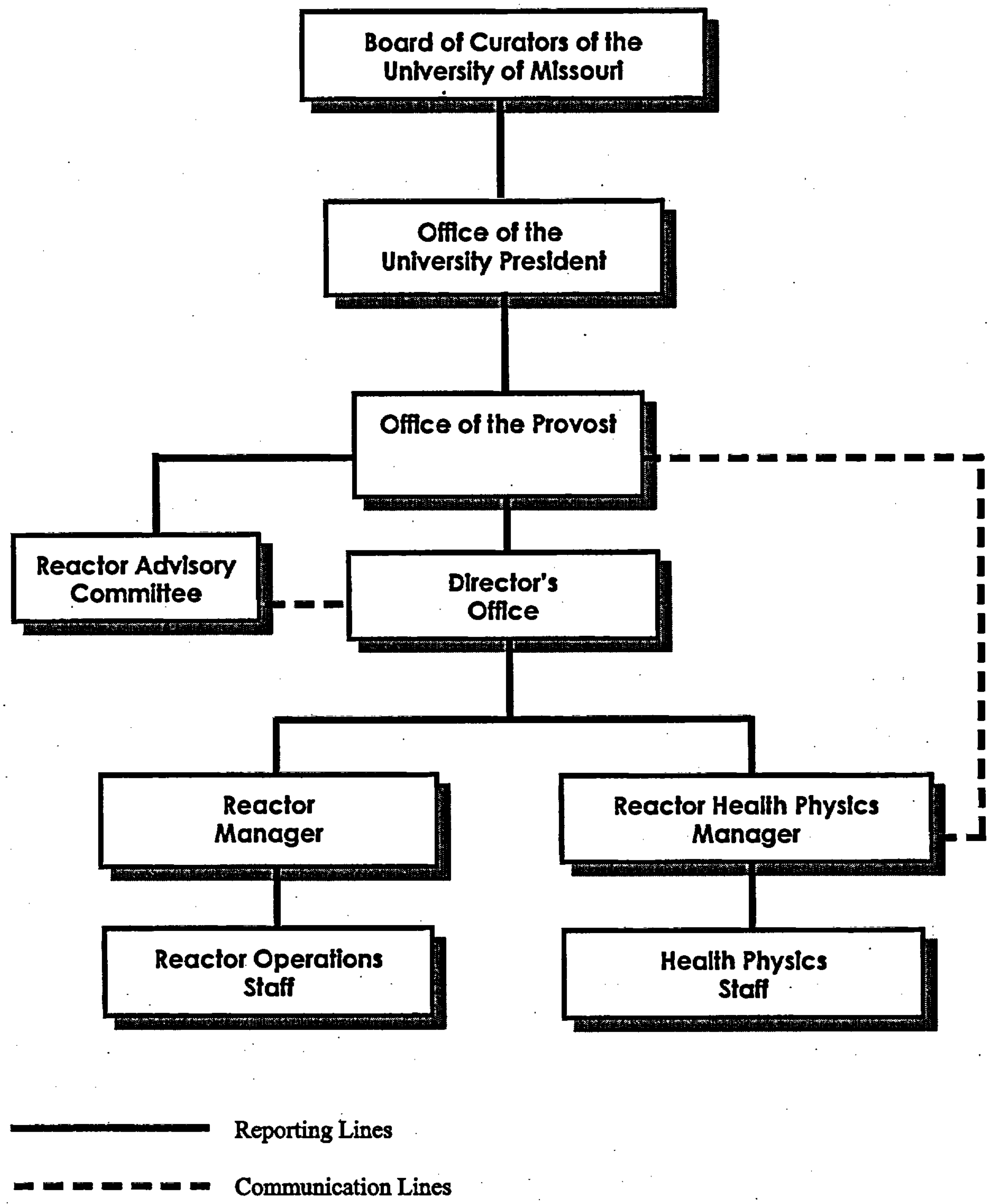

FIGURE 12.1
MISSOURI UNIVERSITY RESEARCH REACTOR (MURR)
ORGANIZATION

The Reactor Operations Branch includes the Reactor Manager and the Reactor Operations Staff under his direction. The Reactor Operations Staff consists of the following personnel: the Assistant Reactor Manager-Operations, the Assistant Reactor Manager-Physics, the Assistant Reactor Manager-Engineering, Senior Reactor Operators, and Reactor Operators.

The Health Physics Branch includes the Reactor Health Physics Manager and the Health Physics Staff under his direction. The Health Physics Staff consists of the following personnel: the Assistant Reactor Health Physics Manager, Health Physicists, and Health Physics 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 reports to and is appointed by the Office of the Provost.

There is a communications/consultation line from the Reactor Health Physics Manager to the Office of the Provost. This line of communications/consultation allows the Reactor Health Physics Manager access to upper Missouri University (MU) management if Reactor Facility Management does not address radiation protection concerns to the satisfaction of the Reactor Health Physics Manager.

12.1.2 Responsibility

12.1.2.1 Reactor Facility Director

The Reactor Facility Director has overall responsibility for the reactor and all its associated laboratories. The Director's primary duties include:

- Supervising, through the subordinate managers and their staffs, the operation and utilization of the reactor facility;
- Preparing and administering the facility budget;
- Negotiating with and employing the necessary staff for operation of the facility; and
- Being available to faculty members for consultation on research proposals for the utilization of the reactor or associated laboratory facilities.

The Reactor Facility Director is assisted in the performance of these duties by the Chief Operating Officer and the reactor facility staff.

12.1.2.2 Chief Operating Officer

The Chief Operating Officer (COO) is responsible for the reactor and all of its associated laboratories. The COO reports to the Reactor Facility Director. The COO's primary duties include:

- Supervising, through the subordinate managers and their staffs, the operation and utilization of the reactor facility;
- Negotiating with, and employing the necessary staff for operation of the facility;
- Serving as Acting Director in the absence of the Reactor Facility Director; and
- Taking the lead position in communications with the U.S. Nuclear Regulatory Commission (NRC).

The COO is assisted in the performance of these duties by the Reactor Manager and his staff.

12.1.2.3 Reactor Manager

The Reactor Manager is responsible for operating the reactor in a safe, efficient, and reliable manner. The Reactor Manager reports to the COO. The Reactor Manager's primary duties include:

- Ensuring that the fullest possible use of the experimental facilities is made available to the experimenters;
- Establishing procedures for the accommodation of experiments, reviewing and processing Reactor Utilization Requests (RURs), and providing advice regarding the safety aspects of proposed experiments;
- Reviewing, evaluating, and approving all facility modifications prior to implementation;
- Instructing reactor operations personnel in their duties; issuing the operating schedule; supervising non-routine reactor evolutions;
- Supervising, preparing, and submitting reports, correspondence, and other communications to the NRC as required;
- Reviewing and approving all reactor operating, safety, emergency, and security procedures;
- Supervising and maintaining documentation necessary to demonstrate continuous compliance with the facility operating license; and

- Developing and requesting a budget for reactor operations.

The Reactor Manager is assisted in the performance of these duties by the Reactor Operations Staff. The Reactor Manager shall be certified at the Senior Reactor Operator level 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 Health Physics Manager is responsible for the development and implementation of the Radiation Protection Program. The Reactor Health Physics Manager reports to the Associate Director, Regulatory Assurance Group. The Reactor Health Physics Manager's primary duties include:

- Acting as a consultant to the Director's Office;
- Administration of the ALARA Program;
- Serving on the Reactor Advisory Committee;
- Providing radiation safety supervision for research personnel;
- Monitoring operations involving accessing a neutron beam from the beamports or thermal column;
- Conducting training and orientation programs in radiation safety and applied health physics for all personnel; and
- Maintaining records of personnel exposure, radiation and contamination control, radioactive waste control, on- and off-site environmental control, and off-site transfers of radioactive materials.

The Reactor Health Physics Manager is assisted in the performance of these duties by the Health Physics Staff.

12.1.2.5 Reactor Operations Staff

The Reactor Operations Staff assists the Reactor Manager in the performance of his duties. The Reactor Operations Staff includes the Assistant Reactor Manager-Operations, the Assistant Reactor Manager-Physics, the Assistant Reactor Manager-Engineering, Senior Reactor Operators (SROs), and the Reactor Operators (ROs).

12.1.2.5.1 Assistant Reactor Manager-Operations

The Assistant Reactor Manager-Operations is responsible for the safe operation of the reactor in conformance with NRC regulations. The Assistant Reactor Manager-Operations reports to the Reactor Manager. The Assistant Reactor Manager-Operations' primary duties include:

- Planning, assigning, and reviewing the work of the shift operating crews, including the daily inspection of all logs and records maintained by the crews;
- Coordinating the shift operating schedule;
- Interviewing and recommending the employment of reactor operators and support staff;
- Conducting training programs in the areas of reactor operator licensing and reactor safety;
- Regularly conducting safety inspections of the facility and submitting reports of findings as required; and
- Supervising reactor operations through the Lead Senior Reactor Operators.

The Assistant Reactor Manager-Operations shall be certified at the Senior Reactor Operator level pursuant to 10 CFR 55.

12.1.2.5.2 Assistant Reactor Manager-Physics

The Assistant Reactor Manager-Physics is responsible for the storage, use, and transfer of Special Nuclear Material (SNM). The Assistant Reactor Manager-Physics reports to the Reactor Manager. The Assistant Reactor Manager-Physics' primary duties include:

- Maintaining a fuel inventory, preparing fuel requisition orders, and overseeing the shipment of spent fuel for reprocessing or storage;
- Performing computer code analysis in support of reactor utilization and license amendments;
- Determining the reactivity worth of sample materials to ensure that the reactor's reactivity limits are not exceeded;
- Developing, writing, and implementing procedures, in compliance with applicable regulations, for the use of SNM in experiment work, for its accountability, and for facility security;

- Developing and maintaining procedures for a Special Nuclear Material Control and Accounting Program in order to properly account for all SNM; and

- Providing or approving an Estimated Critical Position (ECP) for any reactor startup following a shutdown in which fuel handling has taken place.

The Assistant Reactor Manager-Physics shall be or is expected to be certified at the Reactor Operator level pursuant to 10 CFR 55.

12.1.2.5.3 Assistant Reactor Manager-Engineering

The Assistant Reactor Manager-Engineering is responsible for compliance with the facility operating license through documentation, reviews, and audits. The Assistant Reactor Manager-Engineering reports to the Reactor Manager. The Assistant Reactor Manager-Engineering's primary duties include:

- Maintaining and overseeing the Facility Modification Program; ensuring that all Title 10, Chapter 1 of the Code of Federal Regulations, Part 50.59 (10 CFR 50.59) screens and evaluations are performed, as required;

- Managing the Compliance Program and maintaining the documentation necessary to demonstrate continuous compliance with the facility operating license as well as periodically monitoring compliance testing to assure tests are adequate, and performed and documented properly;

- Preparing or assisting in the preparation of license proposals, reports, correspondence, and other communications for submission to the NRC; and

- Evaluating maintenance activities to ensure that adequate post-maintenance testing is performed and verifying the operability of components or systems.

The Assistant Reactor Manager-Engineering shall be certified at the Senior Reactor Operator level pursuant to 10 CFR 55.

12.1.2.5.4 Senior Reactor Operators

Senior Reactor Operators (SROs) are responsible for directing the licensed and non-licensed activities of the Reactor Operators. SROs report to the Lead Senior Reactor Operator[1] of their assigned shift. An SRO's primary duties include:

- Planning, assigning, and reviewing the work of reactor operations support staff in the control of the research reactor;
- Monitoring and reviewing the maintenance of various records such as the console log, nuclear and process data, and routine patrol logs;
- Conducting, or supervising the conduction of facility inspections to ensure building security and the proper maintenance and operation of the reactor and related equipment, as well as to ensure compliance with the applicable government and MU regulations;
- Instructing and advising the reactor support staff on operating methods and procedures and assisting in the development of these methods and procedures;
- Reviewing all maintenance performed on reactor and license-related systems and equipment to ensure operability of these systems prior to reactor operation;
- Maintaining a working knowledge of the facility operating license, the Technical Specifications, the Operations Procedures, and the applicable sections of the CFR; and
- Assuming the duties and responsibilities of an RO (Section 12.1.2.5.5) when required.

The SROs shall be certified at the Senior Reactor Operator level pursuant to 10 CFR 55.

12.1.2.5.5 Reactor Operators

Reactor Operators (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

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[1]The importance of one supervisor to coordinate the licensed and non-licensed activities is recognized for safety and effective control of reactor operation. The Lead Senior Reactor Operator is responsible for the supervision of the staff which supports the operation of the reactor during his assigned shift. The Lead Senior Reactor Operator reports to the Assistant Reactor Manager-Operations.

the operating and maintaining of reactor-related equipment. ROs report to the Lead Senior Reactor Operator of their assigned shift. An RO's primary duties include:

- Starting-up and shutting down of the reactor and associated systems under the supervision of an SRO;
- Monitoring of the reactor plant during steady-state operation; and
- Handling of the reactor fuel from the time of arrival at the facility till the time the fuel is prepared for shipment.

The ROs shall be certified at the Reactor Operator level pursuant to 10 CFR 55.

12.1.2.6 <u>Health Physics Staff</u>

The Health Physics Staff assists the Reactor Health Physics Manager in the performance of his duties. The Health Physics Staff includes the Assistant Reactor Health Physics Manager, Health Physicists, and the Health Physics Technicians.

12.1.2.6.1 <u>Assistant Reactor Health Physics Manager</u>

The Assistant Reactor Health Physics Manager is responsible for conducting training and for monitoring programs so as to protect personnel from radiation hazards and to assure compliance with federal, state, and MU regulations. The Assistant Reactor Health Physics Manager reports to the Reactor Health Physics Manager. The Assistant Reactor Health Physics Manager's primary duties include:

- Assisting the Reactor Health Physics Manager in ensuring that the reactor facility's needs for health physics training, surveying, and operational support are met;
- Serving as Acting Manager in the absence of the Reactor Health Physics Manager;
- Preparing regulatory-required written reports, operating procedures, and program audits;
- Instructing and advising the staff and other users of the facility on methods and procedures to comply with the ALARA principle;
- Supervising and performing radiation and contamination surveys on all facility laboratories in which radioactive materials are used; leading test surveys on sealed sources of radioactive materials and environmental sampling; and

- Maintaining a working knowledge of regulations concerning the use, transfer, and storage of radioactive materials; interpreting applicable regulations and developing procedures to ensure adherence to them.

12.1.2.6.2 Health Physicists

Health Physicists 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. Health Physicists report to the Assistant Reactor Health Physics Manager. A Health Physicist's primary duties include:

- Conducting research to develop inspection standards, radiation exposure limits for personnel, safe work methods and decontamination procedures; testing surrounding areas to ensure that radiation is not in excess of permissible standards;

- Consulting with radiation users regarding their experiment or general use of radioactive material and assisting them in defining and alleviating potential hazards;

- Assisting in developing standards of permissible concentrations of radioisotopes in liquids and gases;

- Instructing and advising technical support staff in principles and regulations relating to radiation hazards; and

- Conducting lectures and demonstrations on the use and handling of radioactive material, protective equipment, and detection and measuring devices.

12.1.2.6.3 Health Physics Technicians

Health Physics Technicians are responsible for ensuring, through measurements and observations, that users of radioactive materials and radiation sources are following the procedures and methods established by federal, state, and MU regulations. Health Physics Technicians report to the Assistant Reactor Health Physics Manager. A Health Physics Technician's primary duties include:

- Processing radioactive material for shipment, storage, and disposal;

- Performing periodic testing of radiation detection equipment and ensuring that such equipment is periodically calibrated and properly maintained;

- Performing radioactive contamination surveys of those areas where radiation sources are being utilized, measuring exposure rates, and inspecting isotope records;

- Conducting indoctrination and information programs that demonstrate proper techniques of handling and using radioactive materials and related instrumentation and equipment;
- Assisting in various experiments and maintenance activities associated with the radiation/reactor facilities;
- Assisting in the processing of radioactive material shipments;
- Leak-testing of sealed sources containing radioactive materials; and
- Collecting and analyzing environmental samples for radioactivity.

12.1.3 Staffing

a. 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:

   (1) Director's Office personnel;

   (2) Reactor Operations personnel;

   (3) Health Physics (Radiation Safety) personnel;

   (4) Facility Support Operations personnel;

   (5) Additional personnel (based on their expertise); and

   (6) Emergency Support Organizations (e. g., MU Police, MU Hospital and Clinics, City of Columbia Fire Department, etc.).

b. A call list of key personnel employed at the reactor facility is available in the reactor control room for use by the reactor operators, if required.

c. The following staffing requirements shall be satisfied as part of reactor start-up, steady-state operation, and shutdown:

   (1) As 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 must be knowledgeable of the facility; and

(2) As 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 shut down, but not secured, the reactor control room boundary may be expanded to include the operating (upper) bridge during refueling or maintenance activities.

d. A Health Physics Technician, Electronic Technician, and an Engineering (machinery shop) Technician will normally be readily available for emergencies (i.e., capable of arriving at the reactor facility within one hour of notification).

12.1.4 <u>Selection and Training of Personnel</u>

In order to develop and maintain an organization qualified for operation and maintenance of the 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 certified 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 certification is received from the NRC.

12.1.4.1 <u>Initial Training and Certification</u>

Initial training of personnel (trainees) to be certified 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. Certification of a candidate is achieved after extensive training followed by the successful completion of an examination administered by the NRC.

12.1.4.2 <u>Requalification and Recertification</u>

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. Certified individuals who have successfully completed the requalification program may be recertified by the NRC when required. The MURR Requalification Program is described in greater detail in Section 12.9, Operator Training and Requalification.

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 Health Physics Staff has the authority to interdict or terminate the use of radioactive materials or radiation sources if adequate health physics 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

To ensure that the operation of the reactor facility does not jeopardize the health and safety of the general public, a committee has been chartered to provide objective and independent reviews, evaluations, and recommendations on matters affecting reactor safety. The Reactor Advisory Committee (RAC) is a committee of the University of Missouri, appointed by the Office of the Provost, to satisfy the requirements of the facility operating license. The University of Missouri and the NRC expect the RAC to provide independent oversight on experimental and operational activities at the facility.

12.2.1 Composition and Qualification

Members of the RAC, its Chairman, and its Vice Chairman are 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, its Chairman, and its Vice Chairman are appointed annually during the fall semester to serve one year. Members may be reappointed indefinitely.

The RAC, through its Chairman, may appoint subcommittees (See Section 12.2.5) consisting of 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. Subcommittees may be authorized to act on behalf of the RAC.

12.2.2 Charter and Rules

The RAC shall meet at least once during each calendar quarter. It shall maintain minutes of its meetings which will include the items considered, actions taken, and the recommendations made.

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, must be present 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, and any resolution coming before the RAC for action shall require approval by the majority of the voting members present. A meeting of a subcommittee shall not be deemed to satisfy the requirement for the RAC to meet at least once during each calendar quarter.

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 Provost acting on the recommendation of the Reactor Facility Director.

12.2.3 <u>Review Function</u>

Responsibilities of the RAC shall include but are not limited to the following:

(1) Reviewing and making recommendations concerning proposed changes to reactor equipment or procedures when such changes have a safety significance, involve an amendment to the operating license including a change in the Technical Specifications, or any questions pursuant to 10 CFR 50.59;

(2) Reviewing and making recommendations concerning proposed tests or experiments which are significantly different from any previously reviewed or which involve a question pursuant to 10 CFR 50.59; and

(3) Reviewing the circumstances of all abnormal occurrences and violations of the Technical Specifications and the measures taken, or to be taken, to prevent recurrence.

The RAC shall act in an advisory capacity to the Reactor Facility Director in matters pertaining to the safe operation of the reactor and with regard to planned research activities and use of the facility building and equipment. It 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 <u>Audit Function</u>

The charter for the RAC does not require it to perform periodic audit activities. However, annual audits are performed by the Reactor and Reactor Health Physics 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. These annual audits shall include, but are not limited to the following:

a. Quality Assurance Program;

b. Physical Security Plan and Security Procedures;

c. Emergency Plan and Emergency Plan Implementing Procedures;

d. Operator Requalification Program;

e. Operating Procedures (Reactor Operations and Health Physics);

f. Radiation Protection Program; and

g. As Low As Reasonably Achievable (ALARA) Program.

12.2.5 Subcommittees

Presently, the following five subcommittees have been appointed by the Reactor Advisory Committee: Reactor Safety, Reactor Action, Reactor Procedures Review, Isotope Use, and Reactor Service.

12.2.5.1 Reactor Safety Subcommittee

The Reactor Safety Subcommittee (RSSC) shall act in behalf of the RAC in performing reviews of the following:

a. Proposed changes to reactor equipment when such changes have safety significance, involve an amendment to the operating license (including a change to the Technical Specifications) or involve a question pursuant to 10 CFR 50.59;

b. Proposed tests or experiments significantly different from any previously reviewed or which involve a question pursuant to 10 CFR 50.59; and

c. Circumstances of all abnormal occurrences and violations of the Technical Specifications and the remedial measures taken, or to be taken, to prevent recurrence.

Upon completion of the review, the RSSC shall make a recommendation concerning the proposed change, experiment, or remedial measure and report this recommendation to the Chairman of the RAC and to the Reactor Manager. When the review results in a negative recommendation, the RSSC shall recommend alternatives for the proposed change, experiment, or remedial measure 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 RSSC of the course of action selected, or he shall submit a new proposal for review.

A subset of the RSSC will act in behalf of the RAC as the Reactor Action Subcommittee (See Section 12.2.5.2).

Members of the RSSC and its Chairman will be appointed by the Chairman of the RAC with the concurrence of the RAC committee members. The term of appointment will be coincident with the term of appointment of the RAC. The Chairman of the RSSC will be a member of the RAC. Other members will be faculty, staff, or students of the University. There will be no less than six members and no more than one student member. Meetings of the RSSC are conducted in accordance with a written charter.

12.2.5.2 Reactor Action Subcommittee

The Reactor Action Subcommittee shall 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 Reactor Action Subcommittee is a subset of the RSSC. Its Chairman will be appointed by the Chairman of the RAC with the concurrence of the RAC committee members. The term of appointment will be coincident with the term of appointment of the RAC. The Chairman of the Reactor Action Subcommittee will be a member of the RAC. Other members will be any three remaining members of the RSSC. At least two of the four members will be non-MURR staff. Meetings of the Reactor Action Subcommittee are conducted in accordance with a written charter.

12.2.5.3 Reactor Procedures Review Subcommittee

The Reactor Procedures Review Subcommittee (RPRS) shall act as an advisory group to the Reactor Manager and to the RAC in matters relating to the review of proposed changes to reactor procedures when such changes have safety significance or involve an amendment to the facility operating license, a change to the Technical Specifications incorporated in the license, or a question pursuant to 10 CFR 50.59.

The RPRS Chairman will be appointed by the Chairman of the RAC. The remaining members of the RPRS are specifically designated by a written charter. Meetings of the RPRS are also conducted in accordance with this charter.

12.2.5.4 Isotope Use Subcommittee

The Isotope Use Subcommittee (IUS) shall act as an advisory group to the RAC in regard to matters relating to the custody and use of radiation and radioisotopes within the MURR. The IUS provides assistance in reviewing radiation safety for project applications for the use of radiation sources and radioactive materials which are covered by the Amended Facility License (No. R-103) and not covered under the other U.S. Nuclear Regulatory Commission (NRC) licenses granted to The Curators of the University of Missouri governing work at the MURR.

Members of the IUS and its Chairman shall be appointed by the Chairman of the RAC with the concurrence of the rest of the RAC. The term of appointment shall be coincident with the term of appointment of the RAC. Members shall be selected who possess experience and training in the safe use of radioactive materials in research and development applications. Meetings of the IUS are conducted in accordance with a written charter.

12.2.5.5 Reactor Service Subcommittee

The Reactor Service Subcommittee shall act as an advisory group to the RAC in regard to matters relating to service performed by or service that might be performed by the reactor staff. The Reactor Service Subcommittee evaluates and advises that such service meets the mission and goals of the reactor facility and the University.

Members of the Reactor Service Subcommittee will be nominated by the Reactor Facility Director and approved by the RAC. Meetings of the Reactor Service Subcommittee are conducted in accordance with a written charter.

12.3 Procedures

Written procedures are established for the Reactor Operations and Health Physics 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.

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 Reactor Procedures Review Subcommittee (RPRS) 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 RPRS. The following is a list of evolutions or programs which typically require written procedures for the reactor operations staff:

a. Start-up, steady-state operation, and shutdown of the reactor;

b. Fuel loading, unloading, and movement in the reactor core and/or pool;

c. Removal and installation of a control blade offset mechanism;

d. Pre-start-up operational checks of the reactor control and process instrumentation systems;

e. Start-up and shutdown of the primary and pool coolant systems and the associated auxiliary systems;

f. Administrative control of the experimental facilities which could affect reactor safety and core reactivity;

g. Emergencies requiring immediate actions by reactor operations staff to place the reactor in a safe condition; and

h. Implementation of required plans for the facility such as the emergency or physical security plans.

The Reactor Manager shall annually review and approve the Operations Procedures and the Emergency Plan Implementing Procedures.

12.3.2 Health Physics

Health physics operating procedures provide methods and guidelines for the implementation and the maintenance of the Radiation Protection Program. This program has been established to protect the health and safety of the MURR staff, research associates, students, and the general public. Changes to these procedures are made by the Reactor Health Physics Manager, or his authorized delegate, and are subsequently reviewed by the RPRS. The following is a list of evolutions or programs which typically require written procedures for the health physics staff:

a. 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;

b. Calibration of area radiation monitors, facility air monitors, laboratory radiation detection systems, personal radiation monitoring devices, and portable radiation monitoring instruments;

c. Administrative guidelines for the facility personnel indoctrination training program;

d. Receiving and opening packages of radioactive materials and their subsequent transfer within the facility;

e. Monitoring of radioactivity in the environment surrounding the facility;

f. Leak-testing of sealed sources containing radioactive materials;

g. Shipment of radioactive materials;

h. Radioactive analysis of the primary and pool coolant; and

i. Preparation for shipping and the shipping of byproduct material.

The Reactor Health Physics Manager shall annually review and approve the Health Physics Standard Operating Procedures and the procedures for preparation for shipping and the shipping of byproduct material.

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 (Ref. 12.1). Occurrences which are considered reportable events also require certain actions prior to returning the reactor to its normal condition. These actions are outlined below.

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. A prompt report of the safety limit violation to the NRC with a subsequent detailed follow-up report (Licensee Event Report) is required. The Licensee Event Report (LER) 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. Prompt reporting of the violation shall be made to the NRC Project Manager for MURR no later than the following working day. The LER will be submitted to the NRC Document Control Desk, with a copy to the NRC Project Manager, within fourteen days.

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. If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed until authorized by the Reactor Manager. The NRC Project Manager for MURR shall be notified no later than the following working day. The LER will be submitted to the NRC Document Control Desk, with a copy to the NRC Project Manager, within fourteen days.

12.4.3 Other Reportable Occurrences

Other occurrences that are considered reportable events are listed below. The NRC Project Manager for MURR shall be notified no later than the following working day. The LER will be submitted to the NRC Document Control Desk, with a copy to the NRC Project Manager, within fourteen days. A return to normal reactor operation will not be allowed until authorized by the Reactor Manager. (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.) Those "other reportable occurrences" are:

a. Operation with actual safety system settings for required systems less conservative than the Limiting Safety System Settings (LSSSs) specified in the Technical Specifications;

b. Operation in violation of limiting conditions for operation established in the Technical Specifications;

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

d. An unanticipated or uncontrolled change in reactivity greater than 0.006 Δk. Reactor trips resulting from a known cause are excluded;

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

f. An observed inadequacy in the implementation of administrative or procedural controls such that this inadequacy causes or could have caused the existence or the development of an unsafe condition involving the operation of the reactor.

12.4.4 Other Reports

A written report shall be submitted to the NRC Document Control Desk within 30 days of:

a. Any significant change(s) in the transient or accident analyses as described in the SAR; and

b. Permanent changes in the facility organization involving the Office of the Provost or the Director's Office.

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 60 days following each calendar year. Each annual report shall include the following information:

a. A brief narrative summary including:

1. Operating experience (including operations designed to measure reactor characteristics);

2. Changes in the reactor facility design, performance characteristics, and operating procedures related to reactor safety during the reporting period; and

3. Results of surveillance tests and inspections;

b. A tabulation showing the energy generated by the reactor (in megawatt-days);

c. The number of emergency shutdowns and inadvertent scrams (unscheduled shutdowns);

d. Discussion of the major maintenance operations performed during the reporting period, including the effects, if any, on the safe operation of the reactor;

e. A summary of each change to the reactor facility, operating procedures, tests, and experiments carried out under the conditions of 10 CFR 50.59;

f. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee at or prior to the point of such release or discharge;

g. A description of any environmental surveys performed outside the reactor facility; and

h. A summary of radiation exposures received by facility personnel 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 (Ref. 12.1). 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.

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.)

a. Gaseous and liquid radioactive effluents released to the environs;

b. Off-site environmental-monitoring surveys required by the Technical Specifications;

c. Radiation exposure for all monitored personnel; and

d. Updated drawings of the reactor facility.

12.5.2 Five Year Records

The following records are to be maintained for a period of at least five years or for the life of the component involved, whichever is shorter:

a. 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);

b. Principal maintenance operations;

c. Reportable occurrences;

d. Surveillance activities required by the Technical Specifications;

e. Reactor facility radiation and contamination surveys required by applicable regulations;

f. Experiments performed with the reactor;

g. Fuel inventories, receipts, and shipments;

h. Approved changes to operating procedures; and

i. Records of meetings and audit reports of the review and audit group.

12.5.3 Operator Requalification Records

Records of retraining and requalification of ROs and SROs who are licensed pursuant to 10 CFR 55 shall be maintained at all times while the individual is employed or until certification 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.

Emergency preparedness for the reactor facility is the responsibility of the Reactor Manager and is maintained by the following four programs:

1. 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;

2. 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;

3. 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

4. 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 Reference 12.4 to conform with 10 CFR 50, Appendix E.

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 the 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. The Physical Security Plan and Security Procedures shall be annually reviewed and revised as necessary.

12.8 Quality Assurance

The MURR Quality Assurance (QA) Program 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 Associate Director of the MURR Regulatory Assurance Group is responsible for the QA Program, which 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 four main components:

a. Written Examinations;

b. On-The-Job Training;

c. Operating Tests; and

d. 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:

a. Reactor Theory, Thermodynamics, and Facility Operating Characteristics;

b. Normal and Emergency Procedures, and Radiological Controls; and

c. 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 his competence in manipulating the plant controls and in operating all apparatuses and mechanisms required by his license, and (2) that he has 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 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 refuelings, reactor start-ups, or time spent as a console operator. For SROs, these functions include directing refuelings, 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) has been prepared by the 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, the MURR staff used the guidance provided by the following documents:

1. "Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses," Regulatory Guide 4.2, Supplement 1, September 2000;

2. "Standard Review Plans for Environmental Reviews for Nuclear Power Plants," NUREG-1555, Supplement 1, September 2000; and

3. "Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants," NUREG-1437.

The ER supports the application for a twenty-year renewal of the Class 104c Amended Facility License No. R-103 (NRC Docket No. 50-186).

CHAPTER 13

ACCIDENT ANALYSES

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

REFERENCES

13.1 Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Standard Format and Content, NUREG-1537, Part 1, February 1996.

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13.0 ACCIDENT ANALYSES

This chapter demonstrates that various facility design features, safety limits, 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 extremely 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 past forty years of safe operation. Although the inherent safety of the 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 assure that reactor safety limits are 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" (Ref. 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 the 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, hence it 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 a release of fission products to the primary coolant system. However, one accident scenario, termed the Maximum Hypothetical Accident (MHA), assumes the melting of four fuel plates in the reactor core. 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 therefore, are not analyzed. The MHA for the MURR is consistent with the assumed MHA at other similar research reactor facilities.

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 the 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. In addition, the reactor is housed within a containment building, thus providing 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 the MURR, it is assumed that an accident condition has caused the melting of the number-1 fuel plate in four separate fuel elements (Ref. 13.11). It is further assumed that the four number-1 fuel plates are in the peak power region of the core.

Because this postulated accident is considered worse than any credible accident, the conditions that lead to this event are immaterial to the analysis. While one might postulate that the MHA 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 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 (Ref. 13.2).

13.2.1.2 Accident Analysis and Consequences

The MHA postulates partial fuel melting with an associated release of fission products into the primary coolant system. The MHA 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. [redacted]

The potential energy release from the melting of four number-1 fuel plates could occur as a possible metal-water reaction (Ref. 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 number-1 fuel plates melting is not predictable as the amount is dependent 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 reactor 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:

$$Al + nH_2O \rightarrow AlO_n + nH_2 + \text{heat.}$$

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 core. Additionally, any steam that would form in the vicinity of the molten area would also assist in dissipating the heat. Since the MHA 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 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 the MHA is calculated below.

The four number-1 fuel plates in the peak power region of the core contain 78.58 grams of uranium-235 (^{235}U). Considering a total core mass of 6.2 Kg of ^{235}U, 1.27% of the core melts. For purposes of calculation, a 1.3% meltdown shall be assumed, but as shown below, this increases to approximately 2.1% when a power peaking factor is incorporated. The release of radioisotopes of krypton, xenon and iodine are the major sources of radiation exposure to personnel in the containment building and will, therefore, serve as the basis for the source term for this dose calculation. For operation at 10 MW for 1,200 MWD in twelve 10-day cycles over a 300-day period with 6.2 Kg of ^{235}U (normal operating cycle is 6.5 days with a total of less than 700 MWD on the core), the following radioiodine, krypton and xenon activities will conservatively be present in the core (Ref. 13.39).

<u>Radioiodine and Noble Gas Activities in the Core</u>
(in curies)

^{131}I - 1.7×10^5 Ci	^{85}Kr - 4.7×10^2 Ci	^{133}Xe - 4.2×10^5 Ci
^{132}I - 3.3×10^5 Ci	$^{85}Kr_m$ - 1.1×10^5 Ci	^{135}Xe - 9.6×10^4 Ci
^{133}I - 5.1×10^5 Ci	^{87}Kr - 2.1×10^5 Ci	$^{135}Xe_m$ - 9.4×10^4 Ci
^{134}I - 6.3×10^5 Ci	^{88}Kr - 3.0×10^5 Ci	^{137}Xe - 4.9×10^5 Ci
^{135}I - 5.2×10^5 Ci	^{89}Kr - 3.8×10^5 Ci	^{138}Xe - 5.2×10^5 Ci
	^{90}Kr - 3.8×10^5 Ci	^{139}Xe - 4.2×10^5 Ci

A power peaking factor of 1.6 is taken into consideration by increasing the fuel meltdown from 1.3 to 2.08%, and 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. Therefore, 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:

(Note: It would take approximately 5 minutes for Operations personnel to secure the primary coolant system and verify that the containment building has been evacuated following a containment building isolation. For the purpose of the MHA calculations, a conservative assumption of 10 minutes is used.)

Example calculation of ^{131}I released into the reactor pool:

= ^{131}I in fuel x 0.0208 x 1/2,000 gal x (7.9 x10^{-3} gpm) x 10 min x 1,000 mCi/Ci
= (1.7 x 10^5 Ci) x (8.22 x 10^{-4} mCi/Ci)
= 1.40 x 10^2 mCi

Note: Same calculation is used for the other isotopes listed below.

Radioiodine and Noble Gas Activities Released Into the Pool After 10 Minutes
(in millicuries)

^{131}I - 1.40 x 10^2 mCi	^{85}Kr - 3.86 x 10^{-1} mCi	^{133}Xe - 3.45 x 10^2 mCi
^{132}I - 2.71 x 10^2 mCi	^{85}Kr$_m$ - 9.04 x 10^1 mCi	^{135}Xe - 7.89 x 10^1 mCi
^{133}I - 4.19 x 10^2 mCi	^{87}Kr - 1.73 x 10^2 mCi	^{135}Xe$_m$ - 7.72 x 10^1 mCi
^{134}I - 5.18 x 10^2 mCi	^{88}Kr - 2.46 x 10^2 mCi	^{137}Xe - 4.03 x 10^2 mCi
^{135}I - 4.27 x 10^2 mCi	^{89}Kr - 3.12 x 10^2 mCi	^{138}Xe - 4.27 x 10^2 mCi
	^{90}Kr - 3.12 x 10^2 mCi	^{139}Xe - 3.45 x 10^2 mCi

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 iodine isotopes. The krypton and xenon noble gases released into the reactor pool over this same time period 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.

Radioiodine Concentrations in the Pool Water
(in microcuries per gallon)

^{131}I - 7.0 µCi/gal	^{133}I - 21.0 µCi/gal	^{135}I - 21.4 µCi/gal
^{132}I - 13.6 µCi/gal	^{134}I - 25.9 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 (303 l) of water per day. However, for the purposes of the MHA, the assumption is that a total of 40 gallons (151 l) of pool water containing the previously listed radioiodine concentrations evaporates over 10 minutes into the isolated containment building. This assumption results in about seventy times more radioiodine activity in the containment building air than would be present at the end of the 10 minutes of evaporation. In addition, considering that containment air at a temperature of 75 °F

(25 °C) and 100% relative humidity contains H_2O vapor equal to 40 gallons (151 l) of water, and recognizing that the air in containment is normally at about 50% relative humidity, the assumption that 40 gallons (151 l) of pool water evaporating overestimates by a factor of two how much pool water could actually evaporate into the isolated structure after initiation of the MHA. It is also conservatively assumed that all of the iodine activity in the 40 gallons (151 l) of pool water, which was assumed to evaporate over 10 minutes, is released into containment and 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 air volume:

Example calculation of ^{131}I released into containment air:

= ^{131}I concentration in pool water x 40 gal x 1/225,000 ft^3 x 35.3147 ft^3/m^3
= 7.0 μCi/gal x (6.28 x 10^{-3} gal/m^3)
= 4.4 x 10^{-2} μCi/m^3

$$(4.4 \times 10^{-2}\ \mu Ci/m^3) \times (1\ m^3/10^6\ ml) = 4.4 \times 10^{-8}\ \mu Ci/ml$$

Note: Same calculation is used for the other isotopes listed below.

Radioiodine Concentrations in the Containment Building Air After 10 Minutes
(in microcuries per milliliter)

^{131}I - 4.4 x 10^{-8} μCi/ml	^{134}I - 1.62 x 10^{-7} μCi/ml
^{132}I - 8.5 x 10^{-8} μCi/ml	^{135}I - 1.34 x 10^{-7} μCi/ml
^{133}I - 1.32 x 10^{-7} 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 the MHA (Note: the primary coolant system is shut down and secured, and the leakage driving force is stopped within 10 minutes), 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. Based on the 225,000-ft^3 volume of containment building air and the previously listed millicurie quantities of these gases released into the reactor pool, the maximum noble gas concentrations in the containment building at the end of 10 minutes would be as follows:

Example calculation of ^{85}Kr released into containment air:

= ^{85}Kr activity in pool water x 1/225,000 ft^3 x 35.3147 ft^3/m^3 x 1,000 μCi/mCi
= (3.86 x 10^{-1} mCi) x (1.57 x 10^{-1} μCi/mCi-m^3)
= 6.10 x 10^{-2} μCi/m^3

$$(6.10 \times 10^{-2}\ \mu Ci/m^3) \times (1\ m^3/10^6\ ml) = 6.06 \times 10^{-8}\ \mu Ci/ml$$

Note: Same calculation is used for the other isotopes listed below.

Noble Gas Concentrations in the Containment Building Air after 10 Minutes
(in microcuries per milliliter)

^{85}Kr - 6.06 x 10^{-8} µCi/ml	^{133}Xe - 5.42 x 10^{-5} µCi/ml
^{85}Kr$_m$ - 1.42 x 10^{-5} µCi/ml	^{135}Xe - 1.24 x 10^{-5} µCi/ml
^{87}Kr - 2.71 x 10^{-5} µCi/ml	^{135}Xe$_m$ - 1.21 x 10^{-5} µCi/ml
^{88}Kr - 3.87 x 10^{-5} µCi/ml	^{137}Xe - 6.32 x 10^{-5} µCi/ml
^{89}Kr - 4.90 x 10^{-5} µCi/ml	^{138}Xe - 6.71 x 10^{-5} µCi/ml
^{90}Kr - 4.90 x 10^{-5} µCi/ml	^{139}Xe - 5.42 x 10^{-5} µCi/ml

The objective of this calculation is to present a worst-case dose assessment for a person who remains in the containment building for 10 minutes following the MHA. Therefore, as noted previously, the radioactivity in the evaporated pool water is assumed to be instantaneously and uniformly distributed into the building once released into the air. Although evacuation of the containment building would occur within about 2 minutes for research staff, and about 5 minutes for Operations personnel, the 10-minute interval, as previously noted, allows more than sufficient time for necessary Operations personnel to secure the primary coolant system, which would then stop the primary coolant system leakage into the reactor pool.

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. As previously noted, the exposure time for this dose assessment is 10 minutes. However, since leakage from the primary coolant system into the reactor pool will occur at a uniform rate over this time, the buildup of radioiodines and noble gases will be approximately linear over a 10-minute period and the maximum concentrations shown above will not occur until the end of the 10-minute interval. Therefore, assuming no decay for the iodines or noble gases, personnel in the containment building for 10 minutes after the MHA begins will be exposed to an average concentration of radioiodines and noble gases that is best represented by the concentration existing at 5 minutes after the onset of the MHA (i.e., one-half of the previously listed maximum concentrations for containment air after 10 minutes). These values are given below for the radioiodines along with the applicable radioiodine dose conversion factors used to calculate the dose.

Average Radioiodine Concentrations in the Containment Building Air
During the 10-Minute Period Following the MHA
(in microcuries per milliliter)

^{131}I - 2.19×10^{-8} μCi/ml	^{134}I - 8.12×10^{-8} μCi/ml
^{132}I - 4.26×10^{-8} μCi/ml	^{135}I - 6.71×10^{-8} μCi/ml
^{133}I - 6.58×10^{-8} μCi/ml	

Committed Dose Equivalent Per Unit Intake to the Thyroid (Ref. 13.36)
(in Sieverts per Becquerel)

^{131}I - 2.92×10^{-7} Sv/Bq	^{134}I - 2.88×10^{-10} Sv/Bq
^{132}I - 1.74×10^{-9} Sv/Bq	^{135}I - 8.46×10^{-9} Sv/Bq
^{133}I - 4.86×10^{-8} Sv/Bq	

Air Submersion Dose Equivalent H_T to the Whole Body (Ref. 13.37)
(in Sieverts per Becquerel-m^3)

^{131}I - 1.82×10^{-14} Sv/Bq-m^3	^{134}I - 1.30×10^{-13} Sv/Bq-m^3
^{132}I - 1.12×10^{-13} Sv/Bq-m^3	^{135}I - 7.98×10^{-14} Sv/Bq-m^3
^{133}I - 2.94×10^{-14} Sv/Bq-m^3	

Sv	=	Sieverts,
Bq	=	Becquerel,
Curie (Ci)	=	3.7×10^{10} Bq,
Microcurie (μCi)	=	3.7×10^{4} Bq, and
Breathing Rate	=	3.3×10^{-4} m^3/sec (Ref. 13.38)

Since the airborne radioiodine source is composed of five different iodine isotopes, it will be necessary to determine the dose contribution from each individual isotope and to then sum the results. The calculation of the doses from inhalation and submersion for ^{131}I is shown below and is the same calculation performed for the other iodine isotopes. The results of these calculations are then summed to show the total iodine dose to an individual who remains in the containment building for 10 minutes after the MHA occurs.

Example calculation of thyroid and whole body doses from ^{131}I:

from inhalation:

$$\left(2.19 \times 10^{-2}\,\frac{\mu Ci}{m^3}\right) \times \left(3.7 \times 10^4\,\frac{Bq}{\mu Ci}\right) \times 10\,\text{min} \times \frac{60\,\text{sec}}{1\,\text{min}} \times \left(3.3 \times 10^{-4}\,\frac{m^3}{\text{sec}}\right) = 1.6 \times 10^2\,Bq$$

Thyroid Dose - $$\left(1.6 \times 10^2\,Bq\right) \times \left(2.92 \times 10^{-7}\,\frac{Sv}{Bq}\right) = 4.67 \times 10^{-5}\,Sv$$

from submersion:

$$\left(2.19 \times 10^{-2}\,\frac{\mu Ci}{m^3}\right) \times \left(3.7 \times 10^4\,\frac{Bq}{\mu Ci}\right) \times 10\,\text{min} \times \frac{60\,\text{sec}}{1\,\text{min}} = 4.86 \times 10^5\,\frac{Bq-\text{sec}}{m^3}$$

Whole Body Dose - $$\left(4.86 \times 10^5\,\frac{Bq-\text{sec}}{m^3}\right) \times \left(1.82 \times 10^{-14}\,\frac{Sv}{Bq-\text{sec}-m^3}\right) = 8.85 \times 10^{-9}\,Sv$$

A tabulation of the results of the calculations for the five different iodine isotopes along with a total dose for the thyroid and whole body resulting from inhalation and submersion is shown below. These totals are then expressed as current dose quantities where:

CDE	=	Committed Dose Equivalent
CEDE	=	Committed Effective Dose Equivalent
DDE	=	Deep Dose Equivalent
TEDE	=	Total Effective Dose Equivalent

<u>Dose to the Thyroid from Inhalation</u>
(in Sieverts)

^{131}I - 4.67 x 10^{-5} Sv	^{134}I - 1.75 x 10^{-7} Sv
^{132}I - 5.52 x 10^{-7} Sv	^{135}I - 4.21 x 10^{-6} Sv
^{133}I - 2.37 x 10^{-5} Sv	Total Thyroid Dose - 7.53 x 10^{-5} Sv

Dose to the Whole Body from Submersion
(in Sieverts)

^{131}I - 8.85×10^{-9} Sv	^{134}I - 2.39×10^{-7} Sv
^{132}I - 1.08×10^{-7} Sv	^{135}I - 1.20×10^{-7} Sv
^{133}I - 4.35×10^{-8} Sv	Total Whole Body Dose - 5.19×10^{-7} Sv

By converting these totals to current dose quantities, where rem = 10^{-2} Sv and millirem = 10^{-5} Sv, the following values can be derived and will represent the dose from radioiodine to an individual remaining inside the MURR containment building for 10 minutes after the initiation of the MHA.

10-Minute Dose from Radioiodines in Containment

CDE (thyroid)	=	7.5 mrem
CEDE (thyroid)	=	0.23 mrem
DDE (whole body)	=	0.052 mrem
TEDE (whole body)	=	0.28 mrem

Dose from the kryptons and xenons that are present in the containment building is assessed in much the same manner as the iodines, and the dose contribution from each individual radionuclide is calculated and then added together to arrive at the final noble gas dose. Since the dose from the noble gases is only an external dose due to submersion, and since the Derived Air Concentrations (DACs) for these radionuclides are based on this type of exposure, the individual noble gas doses for 10 minutes in containment are based on their average concentration in the containment air and the corresponding DAC value in Appendix B of 10 CFR 20. However, doses derived in this manner were selectively verified by using dose conversion factors similar to those used for the radioiodines. The average noble gas concentrations in the containment building during the 10-minute period following the MHA and the corresponding doses for a 10-minute occupancy are given below.

Average noble gas concentrations in the containment building air during the 10-minute period following the MHA	Noble gas doses for a 10-minute containment occupancy following the MHA (mrem)
^{85}Kr - 3.0×10^{-8} µCi/ml	0
$^{85}Kr_m$ - 7.1×10^{-6} µCi/ml	0
^{87}Kr - 1.4×10^{-5} µCi/ml	1
^{88}Kr - 1.9×10^{-5} µCi/ml	4
^{89}Kr - 2.5×10^{-5} µCi/ml	5
^{90}Kr - 2.5×10^{-5} µCi/ml	3
^{133}Xe - 2.7×10^{-5} µCi/ml	0
^{135}Xe - 6.2×10^{-6} µCi/ml	0
$^{135}Xe_m$ - 6.1×10^{-6} µCi/ml	1
^{137}Xe - 3.2×10^{-5} µCi/ml	1
^{138}Xe - 3.4×10^{-5} µCi/ml	4
^{139}Xe - 2.7×10^{-5} µCi/ml	113
Total Deep Dose Equivalent (DDE) Whole Body for 10-minute exposure to Noble Gases in the containment building following the MHA	132 mrem

To finalize the occupational dose in terms of TEDE for a 10-minute exposure in the containment building after the MHA, the doses from the radioiodines and noble gases are added together, and result in the following values:

<u>10-Minute Dose from Radioiodines and Noble Gases in Containment</u>

CDE (thyroid)	=	7.5 mrem
CEDE (thyroid)	=	0.23 mrem
DDE (radioiodines)	=	0.052 mrem
DDE (noble gases)	=	132 mrem
TEDE (whole body)	=	132.28 mrem

It is also worth noting that individuals exposed in the containment building for only 2 minutes after the MHA (the expected evacuation time for most occupants of the building) would receive doses about 25 times lower than those shown above and would receive a TEDE of only about 5 millirem.

However, comparison of the maximum TEDE and CDE for those occupationally-exposed during the MHA to applicable NRC dose limits in 10 CFR 20 shows that the final values are well within the published regulatory limit and, in fact, less than 10% of any occupational limit.

As noted earlier in this analysis, with the onset of the MHA, the containment building ventilation system will shut down and the building itself will be isolated from the surrounding areas. The MHA will not cause an increase in pressure inside the reactor containment structure and so 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 change had occurred in conjunction with the onset of the MHA. A reasonable 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 structure. Making the conservative assumption that the containment building will leak at the Technical Specification leakage rate limit [10% of the contained volume over a 24-hour period from an initial overpressure of 2 psig (13.8 kPa above atmosphere)], the air leakage from the containment structure in standard cubic feet per minute (scfm) as a function of containment pressure can be expressed by the following equation:

$$LR = 17.85 \times (CP-14.7)^{1/2};$$

where:

LR = leakage rate from containment (scfm); and
CP = containment pressure (psia).

The leakage rate is proportional to the square root of the pressure differential between the containment building and outside atmosphere; therefore, the initial leakage rate out of the containment structure would be approximately 10.3 scfm and it would take about 16.5 hours for the leak rate to go to zero after an initial pressure differential of 0.33 psig (2.28 kPa above atmosphere). The average leakage rate over the 16.5-hour period would be about 5.2 scfm.

Several factors exist that will mitigate the radiological impact of any air leakage from the containment building following the MHA. First of all, most leakage pathways from containment discharge into the reactor facility laboratory building, which surrounds the containment structure. Since the laboratory building ventilation system continues to operate during the MHA, 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 cfm. Mixing of containment air leakage with the laboratory building ventilation flow, followed by discharge out the exhaust stack and subsequent atmospheric dispersion according to the model developed in Appendix B of this SAR, results in extremely low radionuclide concentrations and very small radiation doses in the unrestricted area. A tabulation of these concentrations and doses is given below.

A second factor which helps to reduce 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). From these experiments, it was shown that up to 75% of the iodine released will be deposited in the containment vessel. If, due to this 75% iodine deposition in the containment building, each cubic meter of air released from the containment has a radioiodine concentration that is 25% of each cubic meter within containment building air, then the radioiodine leaking from the containment building into the laboratory building, in microcuries per milliliter, will be:

<u>Radioiodine Concentration in Air Leaking from Containment</u>

^{131}I - 1.10×10^{-8} μCi/ml

^{132}I - 2.13×10^{-8} μCi/ml

^{133}I - 3.29×10^{-8} μCi/ml

^{134}I - 4.06×10^{-8} μCi/ml

^{135}I - 3.35×10^{-8} μCi/ml

Assuming, as stated earlier, that (1) the average leakage rate from the containment building is 5.2 scfm, (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, (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 312 (See SAR Appendix B), and (5) there is no decay of any radioiodines or noble gases, then the following average concentrations of radioiodines and noble gases with their corresponding radiation doses will 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 further assuming that the most conservative (worst-case) meteorological conditions exist for the entire 16.5-hour period of containment leakage following the MHA. 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.

Average Noble Gas Concentrations at the Point of Maximum Concentration in the Unrestricted Area and Corresponding Radiation Doses (16.5-Hour Containment Leak Following the MHA)

Noble Gas	Average Concentration		Radiation Dose
^{85}Kr	7.5×10^{-14} μCi/ml		3.2×10^{-8} mrem
$^{85}Kr_m$	1.7×10^{-11} μCi/ml		3.6×10^{-5} mrem
^{87}Kr	3.3×10^{-11} μCi/ml		2.8×10^{-4} mrem
^{88}Kr	4.8×10^{-11} μCi/ml		9.8×10^{-4} mrem
^{89}Kr	6.0×10^{-11} μCi/ml		1.2×10^{-3} mrem
^{90}Kr	6.0×10^{-11} μCi/ml		8.3×10^{-4} mrem
^{133}Xe	6.7×10^{-11} μCi/ml		2.8×10^{-5} mrem
^{135}Xe	1.5×10^{-11} μCi/ml		6.3×10^{-5} mrem
$^{135}Xe_m$	1.5×10^{-11} μCi/ml		6.8×10^{-5} mrem
^{137}Xe	7.8×10^{-11} μCi/ml		1.6×10^{-4} mrem
^{138}Xe	8.2×10^{-11} μCi/ml		8.5×10^{-4} mrem
^{139}Xe	6.7×10^{-11} μCi/ml		2.8×10^{-2} mrem
		Total	3.2×10^{-2} mrem

Therefore, the (DDE - Noble Gas) = 0.03 mrem.

Average Radioiodine Concentrations at the Point of Maximum Concentration in the Unrestricted Area (16.5-Hour Containment Leak Following the MHA)

^{131}I - 1.36×10^{-14} μCi/ml

^{132}I - 2.62×10^{-14} μCi/ml

^{133}I - 4.04×10^{-14} μCi/ml

^{134}I - 4.99×10^{-14} μCi/ml

^{135}I - 4.12×10^{-14} μCi/ml

Doses in the Unrestricted Area Due to Radioiodine
(16.5-Hour Containment Leak Following the MHA)

CDE (thyroid)	=	0 mrem
CEDE (thyroid)	=	0 mrem
DDE (radioiodine)	=	0 mrem
TEDE (radioiodine)	=	0 mrem

Summing the doses from the noble gases and the radioiodines simply substantiates earlier statements regarding the very low levels in the unrestricted area should an MHA occur, and should the containment building leak following such an event. Because the dose values are so low, the dose from the noble gases becomes the dominant value, but the overall TEDE is still only 0.03 mrem, a value far below the applicable 10 CFR 20 regulatory limit for the unrestricted area. Additionally, leakage in mechanical equipment room 114 from such items as valve packing, flange gaskets, pump mechanical seals, etc. was also considered in the MHA analysis. A realistic leakage rate of 60 milliliters within the 10-minute time interval was used - after 10 minutes the primary coolant system would be shutdown, isolated and depressurized as part of the control room operator's actions. 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.

13.2.1.3 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 MHA. 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 (Ref. 13.2). A considerable margin of safety has been designed into the system in this regard. The selection of the melting of four fuel plates in the reactor as the MHA thus represents a condition worse than any credible postulated accident, but it is not expected that there will be any occupational radiation doses or doses in the unrestricted area that exceed any regulatory limits, and it is not expected that any fission products will reach the unrestricted environment. 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.

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 the 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

Previous studies (Refs. 13.12, 13.13) have extensively evaluated the expected results of a sudden step insertion of positive reactivity in the MURR core and concluded that the MURR could withstand a positive reactivity step insertion of 0.008 $\Delta k/k$ without fuel damage. This study was based on an initial power level of 10 MW, with nominal flow, pressure, and reactor temperature conditions, and on the calculated primary temperature and void coefficients of -7.0×10^{-5} $\Delta k/k/°F$ and -2.0×10^{-3} $\Delta k/k/\%$ void, respectively.

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 the MURR's 6.2-Kg uranium-aluminide core, the temperature and void coefficients were carefully re-measured and found to be very close to the original calculated values. The values observed were -7.0×10^{-5} $\Delta k/k/°F$ and -2.51×10^{-3} $\Delta k/k/\%$ void, respectively (Ref. 13.14).

As part of the safety evaluation for the power upgrade to 10 MW, another study was undertaken to determine the maximum reactivity step insertion that the MURR could withstand with no core damage. The MURR was modeled with Chic-Kin (Ref. 13.15), a reactivity transient analysis code developed by the Bettis Atomic Power Laboratory. This code combined hydraulic and heat transfer analyses with reactor kinetics to predict power, temperature, and pressure changes during reactor transients for either pin- or plate-type fuel. Rather than considering transients from nominal conditions, the reactor was modeled for this study with all critical parameters set to their scram set points, i.e., the worst possible conditions for full power operation (Ref. 13.16).

The Chic-Kin analyses at 10-MW operation concluded that the MURR could withstand a positive step insertion of up to 0.006 $\Delta k/k$ without any core damage. Consistent with previous studies (Refs. 13.12, 13.13), for this study it was also assumed that the MURR core could withstand the prompt power burst, since it is of such short time duration, and that fuel failure will occur at the hot spot only if the reactor continues to operate with a sustained normalized power increase factor of greater than or equal to 2.52. From previous work (Ref. 13.17), the most conservative steady-state power level at which burnout would occur was determined to be 25.23 MW.

Analyses presented in Addendum 5 (Ref. 13.32) to the Hazards Summary Report used the computer code PARET, which is a newer version of the Chic-Kin code, to extend the reactivity step insertion analyses to a wider range of reactivity steps. To consolidate and confirm the reactivity insertion analyses presented in the various addenda to the Hazards Summary Report, the most recent version of the reactivity transient analysis code PARET-ANL was obtained from Argonne National Laboratory (ANL). PARET-ANL has been extensively tested and used by the Reduced Enrichment for Research and Test Reactors (RERTR) program.

With the use of PARET-ANL, the response of the MURR core to various step insertions of positive reactivity was analyzed. All applicable reactor parameters in the PARET input were set to conservative values, but not to their nominal or scram set point values. Primary temperature and void coefficients of -6.0×10^{-5} $\Delta k/k/°F$ and -2.0×10^{-3} $\Delta k/k/\%$ void, respectively, were used as input values. Experimental results (Ref. 13.14) have shown that the MURR 6.2-Kg core has temperature and void coefficients more negative than those cited. All transients were initiated from a nominal steady-state power level of 10 MW. For comparison, other key reactor parameters used for this analysis and their nominal values are given below.

Parameter	Assumed Value	Nominal Value
Reactor Power	10 MW	10 MW
Primary Coolant Flow Rate	3,600 gpm (13,627 lpm)	3,800 gpm (14,385 lpm)
Reactor Core Inlet Pressure	75 psia (517 kPa)	75 psia (517 kPa)
Reactor Core Inlet Temperature	130 °F (54.4 °C)	120 °F (48.9 °C)

Figure 13.1 displays the results from the PARET-ANL analyses for positive reactivity step insertions of 0.004, 0.005, 0.006 and 0.007 $\Delta k/k$. Based on these results, it is evident that no core damage will occur even for a positive step insertion of 0.007 $\Delta k/k$. Post burst reactor power level remains below the burnout value of 25.23 MW. Experimental data indicate that either one of two short period or one of three high power trips in the MURR safety system will initiate a reactor scram within 150 milliseconds after the scram set point is reached, and sufficient redundancy in instrumentation certainly exists to ensure that a post burst scram will occur. Experimentally-measured control blade worth and drop time data enabled the modeling of a reactor scram after the step insertion by PARET-ANL. Figure 13.2 shows reactor power behavior after the initiation of a scram 150 milliseconds after reactor power has exceeded 12.5 MW for the 0.007 $\Delta k/k$ step insertion case. It clearly demonstrates that such a scram will safely shut down the reactor with no fuel damage occurring.

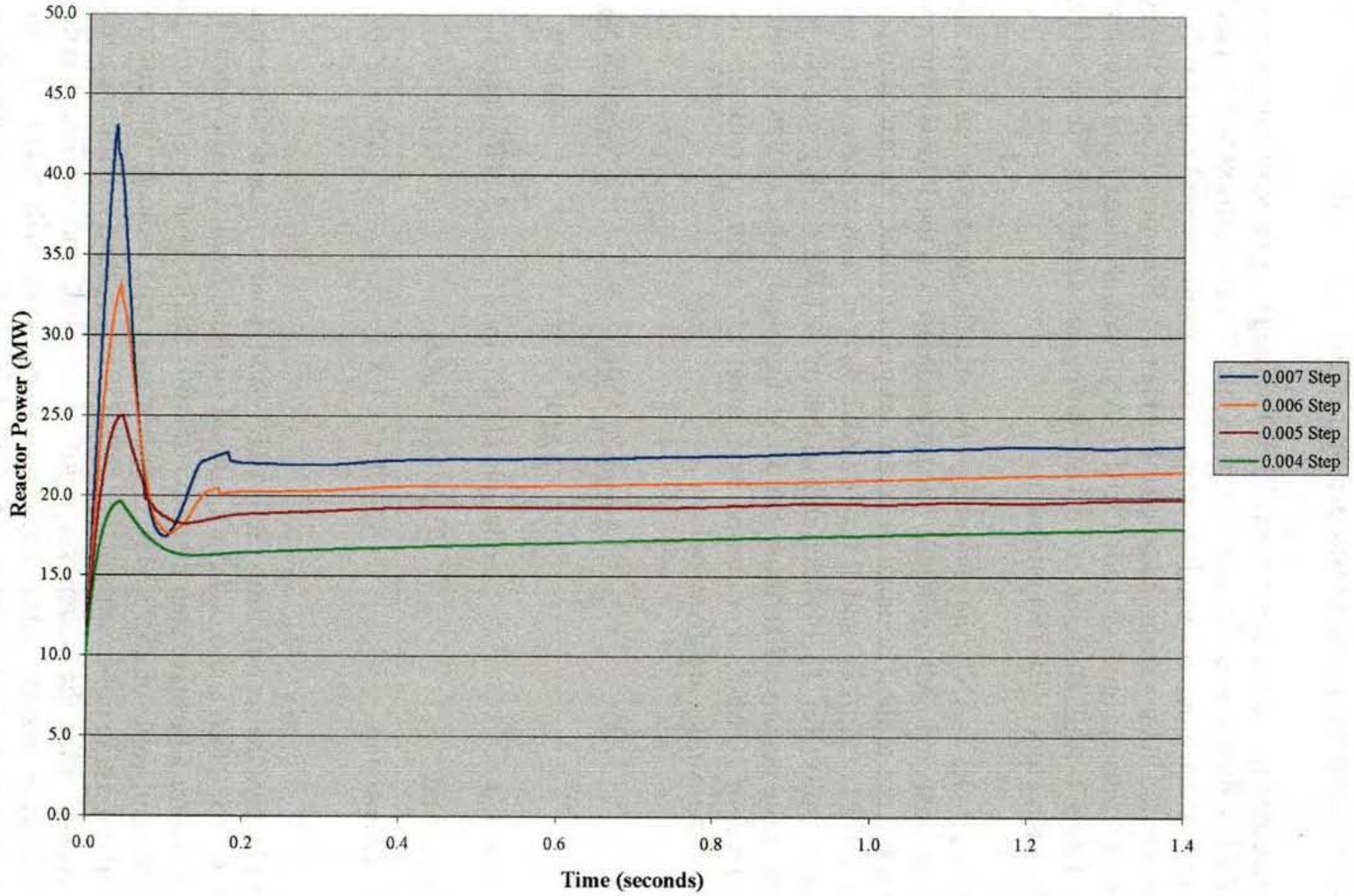

FIGURE 13.1
REACTOR POWER VERSUS TIME FOR POSITIVE REACTIVITY STEP INSERTIONS
WITHOUT AN ACCOMPANYING SCRAM

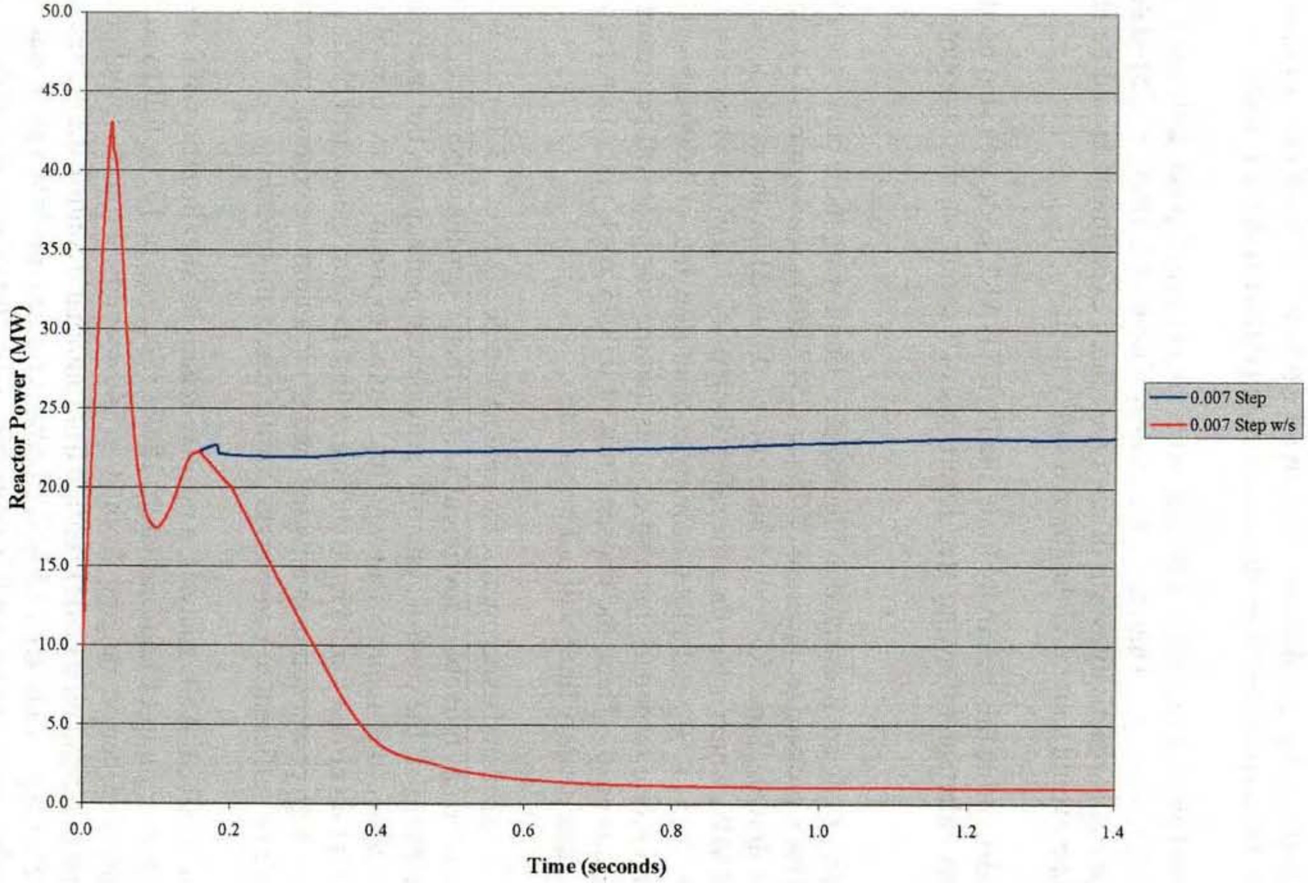

FIGURE 13.2
REACTOR POWER VERSUS TIME FOR A POSITIVE REACTIVITY STEP INSERTION OF 0.007 ΔK/K
WITH AN ACCOMPANYING SCRAM

C-06

In summary, under the worst possible conditions, the MURR can withstand a positive reactivity step insertion of 0.007 Δk/k without core damage. This value is then used to establish the more restrictive 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. Therefore, Regulatory Guide 2.2 (Ref. 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 determine 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 13.3. Employing these curves and the MURR safety limit curves for a pressurizer pressure of 75 psia (517 kPa) (See Figure 13.4), 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 safety 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 (Ref. 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 13.3), 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.

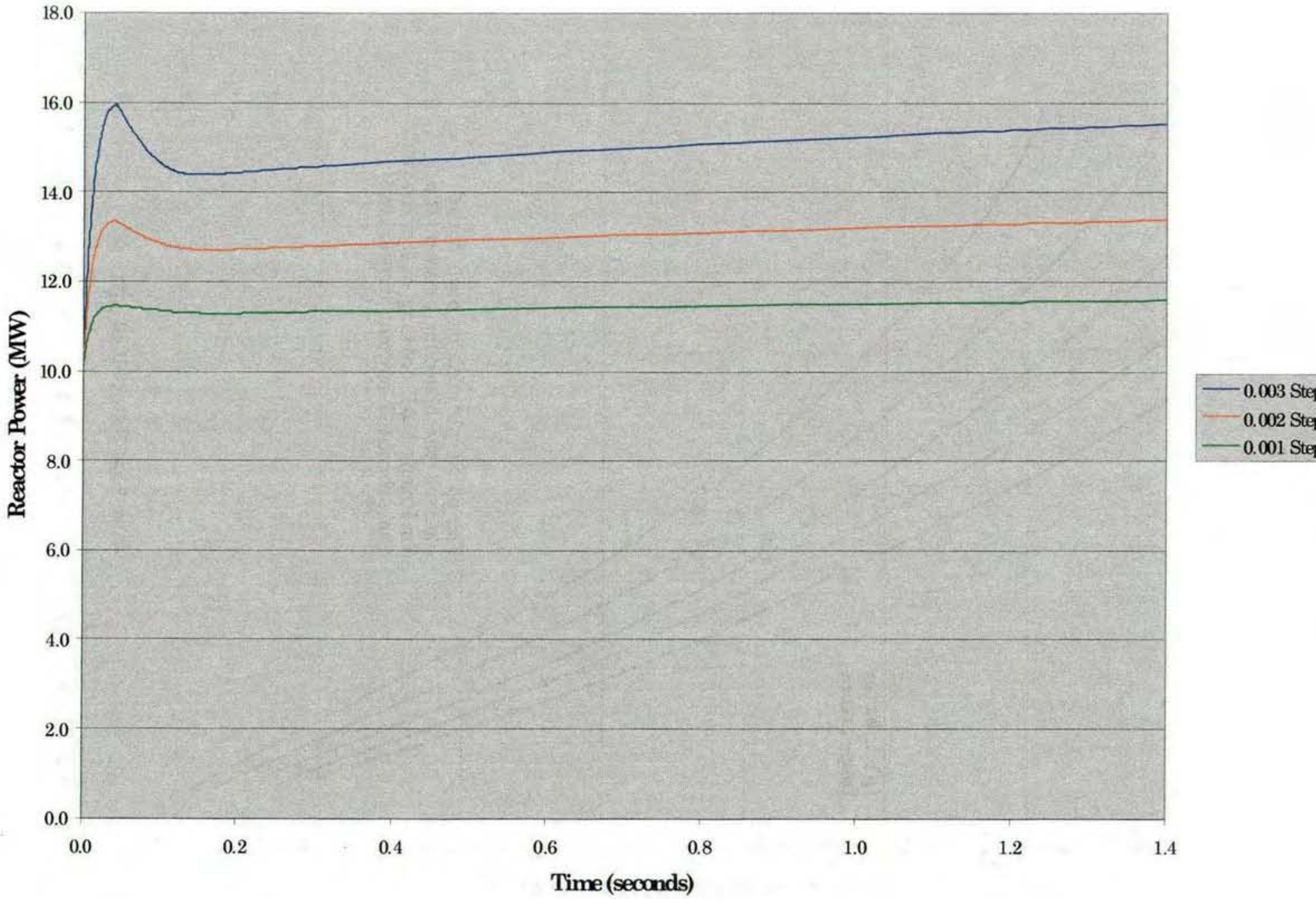

FIGURE 13.3
REACTOR POWER VERSUS TIME FOR VARIOUS SMALLER
POSITIVE REACTIVITY STEP INSERTIONS

C-07

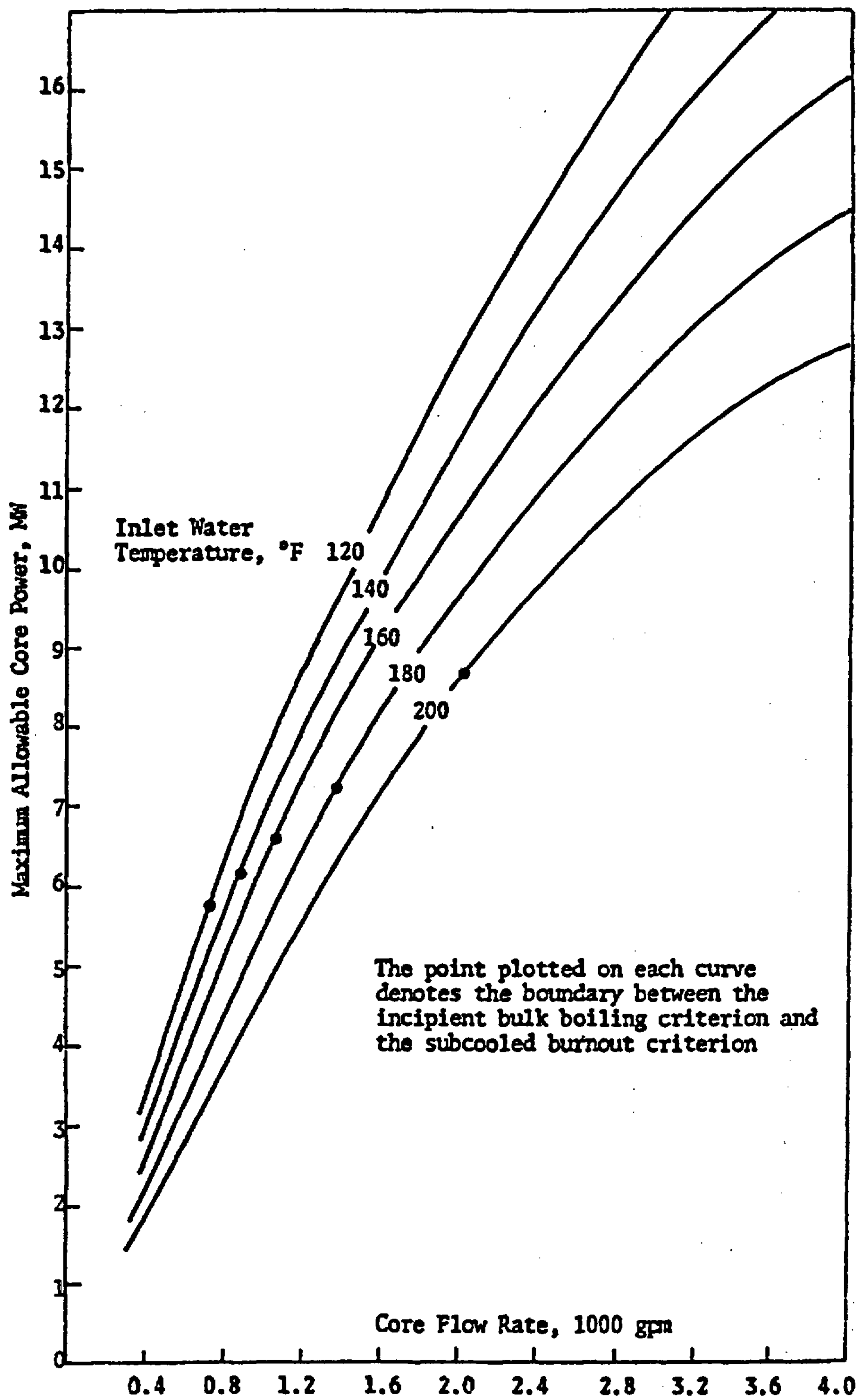

FIGURE 13.4
MURR SAFETY LIMIT CURVES (PRESSURIZER PRESSURE AT 75 PSIA)

In automatic control with the regulating blade near the top (15.6 inches-withdrawn) 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; therefore, 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) 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 inch-withdrawn 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 either in 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), 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, therefore, 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 extremely low during the entire transient, as shown in Figure 13.5, no major reactivity feedback mechanisms exist to effect reactor behavior. The entire reactivity effect is due to the withdrawal of the control blades.

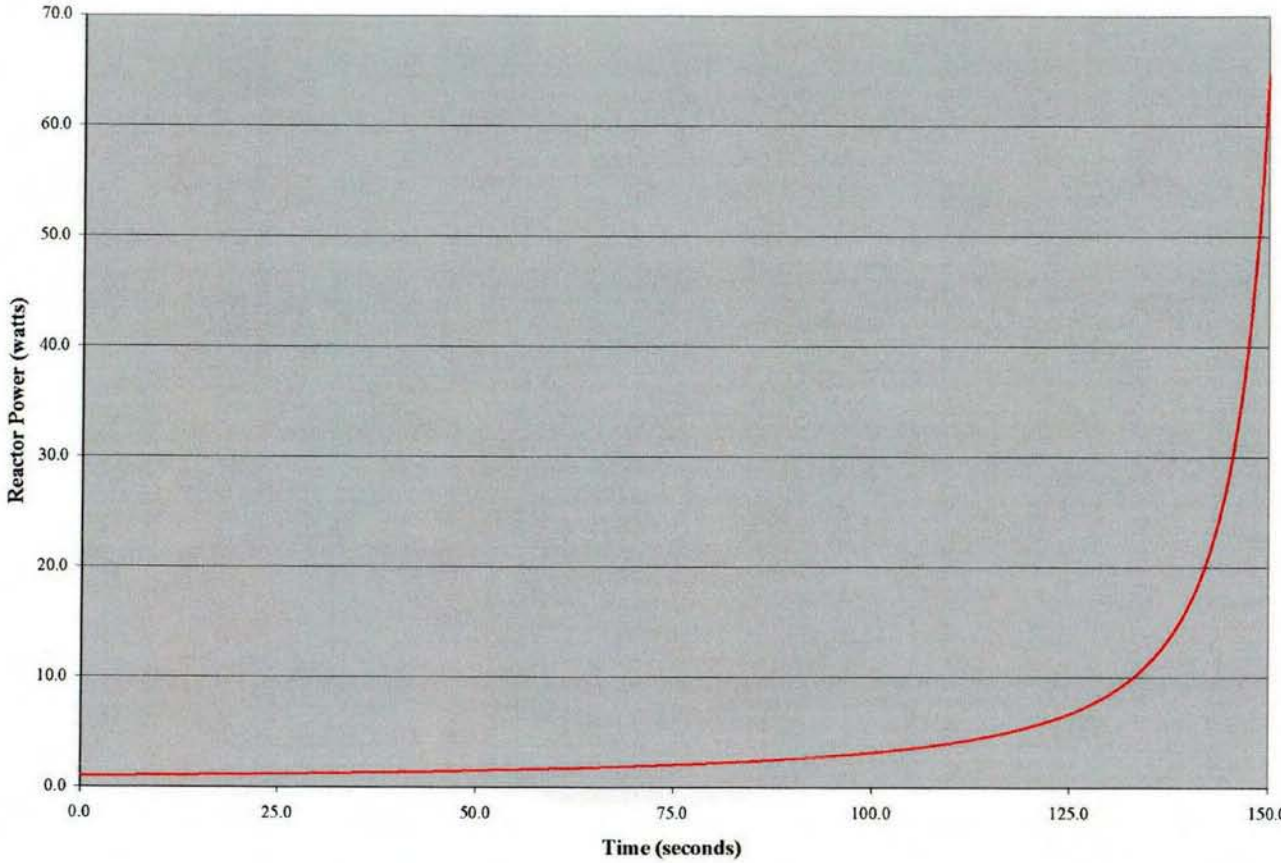

FIGURE 13.5
REACTOR POWER VERSUS TIME FOR
CONTINUOUS CONTROL BLADE WITHDRAWAL

Approximately 140.0 seconds into the transient, the reactor passes through the critical state. At this time, 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 very 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 fact means that the transient can be detected earlier by the nuclear instrumentation and the accident would be terminated by the reactor safety system. It is further noted that, 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.

It can be concluded that 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 (Ref. 13.31). Thus, upon reaching the short period scram set point the reactor will shut down. It is further noted that no credit was taken for the short period rod run-in which will, in actuality, 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 (Ref. 13.31) criteria, and thus, its failure is also not considered credible.

Additionally, a reactor startup is performed in a very controlled manner by the reactor operator. Control blade withdrawals are paused at 5.0-inch 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 below the estimated critical position (ECP), only individual blade withdrawals are permitted. All of these administrative controls further prevent and/or 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 $\Delta k/k$/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. Again, it is concluded that 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 (Ref. 13.31).

13.2.2.2 Conclusions

The insertion of excess reactivity at the 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 the MURR can, under the worst possible conditions, withstand a positive reactivity step insertion of 0.007 $\Delta k/k$ without core damage. This value was then used to establish the more restrictive Technical Specification limit of 0.006 $\Delta k/k$ 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 $\Delta k/k$ was established as a safe value, while each movable experiment shall be limited to 0.001 $\Delta k/k$.

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.

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 13-1 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 13-1
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]	62 - 66 psig (427 - 455 kPa)[1]
Anti-Siphon Pressure	26 psig (179 kPa)[1]	36 psig (248 kPa)[1]

[1]Pressure above atmosphere.

13.2.3.2 Accident Analysis and Consequences

Figure 13.6 is a schematic of the in-pool portion of the primary coolant system. A rupture in a section of the 12-inch 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), the following actions to occur:

1. Actions 1 and 3 listed above;
2. In-pool heat exchanger isolation valves V546A and V546B will open; and
3. 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 sensors DPS 929, DPS 928A and DPS 928B. It should be noted that 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 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.

[redacted]

FIGURE 13.6
SCHEMATIC OF IN-POOL PORTION OF PRIMARY COOLANT SYSTEM

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 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 13-1, 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. 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. The more severe cold leg break analysis is described below in detail.

NOTE: 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 13.7).

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 13.8, 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 13.9). 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 13.10 through 13.13). Table 13-2 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 13-2
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 x 10^{-9}, then 20 is added to the mode. Thus Mode 23 would be subcooled nucleate boiling with non-condensable quality greater than 1 x 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 13-3 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 13-3
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 13.9, and 13.14 through 13.18, a lot of 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 13.8 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 six-inch 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 13-4 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 13-4
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 13-5 lists the heat transfer modes in coolant channels 3, 4 and 5 that occur at the 4.0-second point. 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 13-5
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 13-6 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 13-6
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 affect 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 13.8 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 13.19.

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 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 voidings 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 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 occur 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 occasionally occurs in the top volume during this same period. The five inches 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 six 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 at 17 feet of static head) 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).

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 13.20). 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 13.21, 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 are 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 <u>Conclusions</u>

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. It should also be noted that, because of the large heat sink that is created by the reactor pool, any steam postulated from the accident boil-off will very quickly condense in the empty piping and in-pool heat exchanger and drain back into the reactor pressure vessel.

Therefore, the 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.

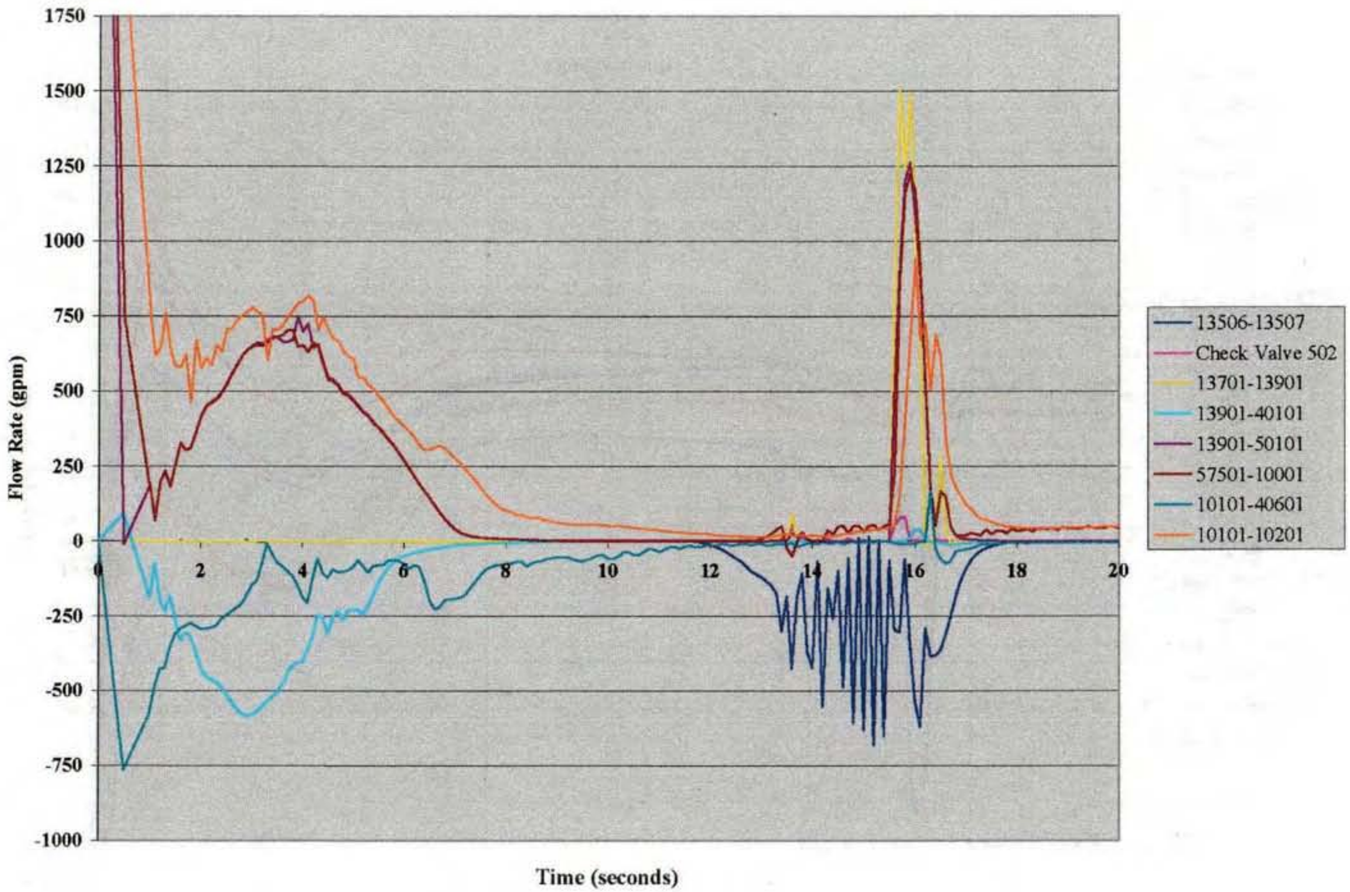

FIGURE 13.7
JUNCTION FLOW RATES DURING
THE FIRST 20 SECONDS OF THE COLD LEG LOCA

C-08

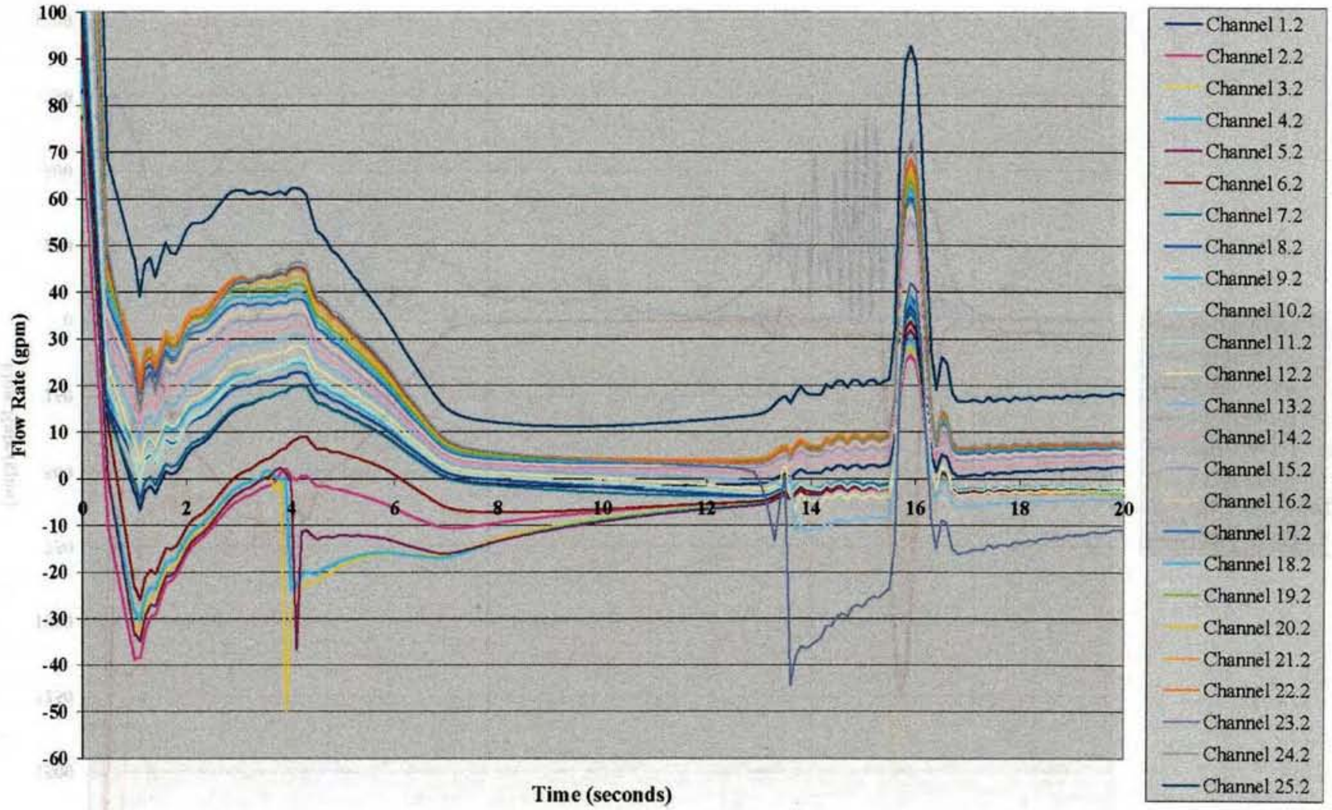

FIGURE 13.8
COOLANT FLOW THROUGH THE 25 INDIVIDUAL CHANNELS
DURING THE FIRST 20 SECONDS OF THE COLD LEG LOCA

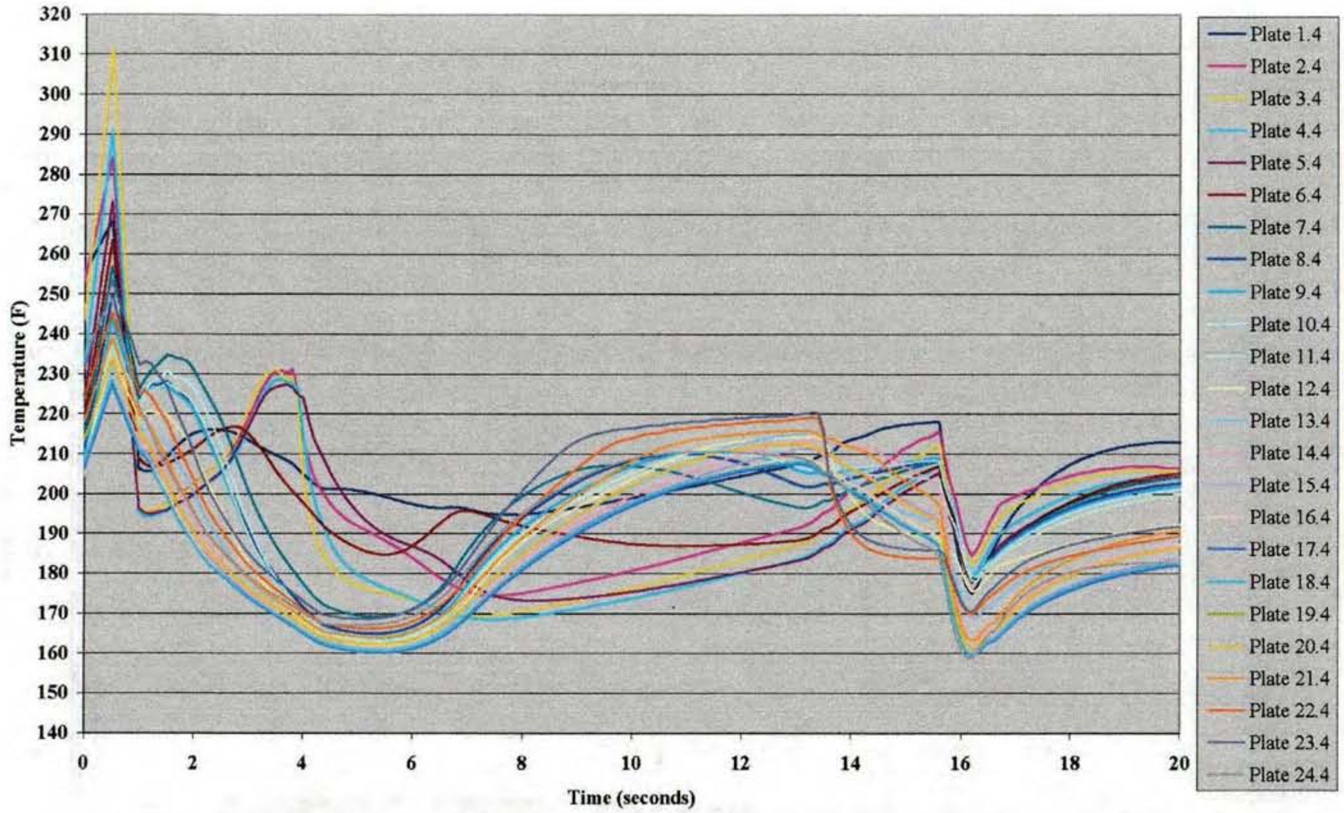

FIGURE 13.9
CENTERLINE TEMPERATURE OF THE 24 FUEL PLATES (SECTION 4)
DURING THE FIRST 20 SECONDS OF THE COLD LEG LOCA

C-10

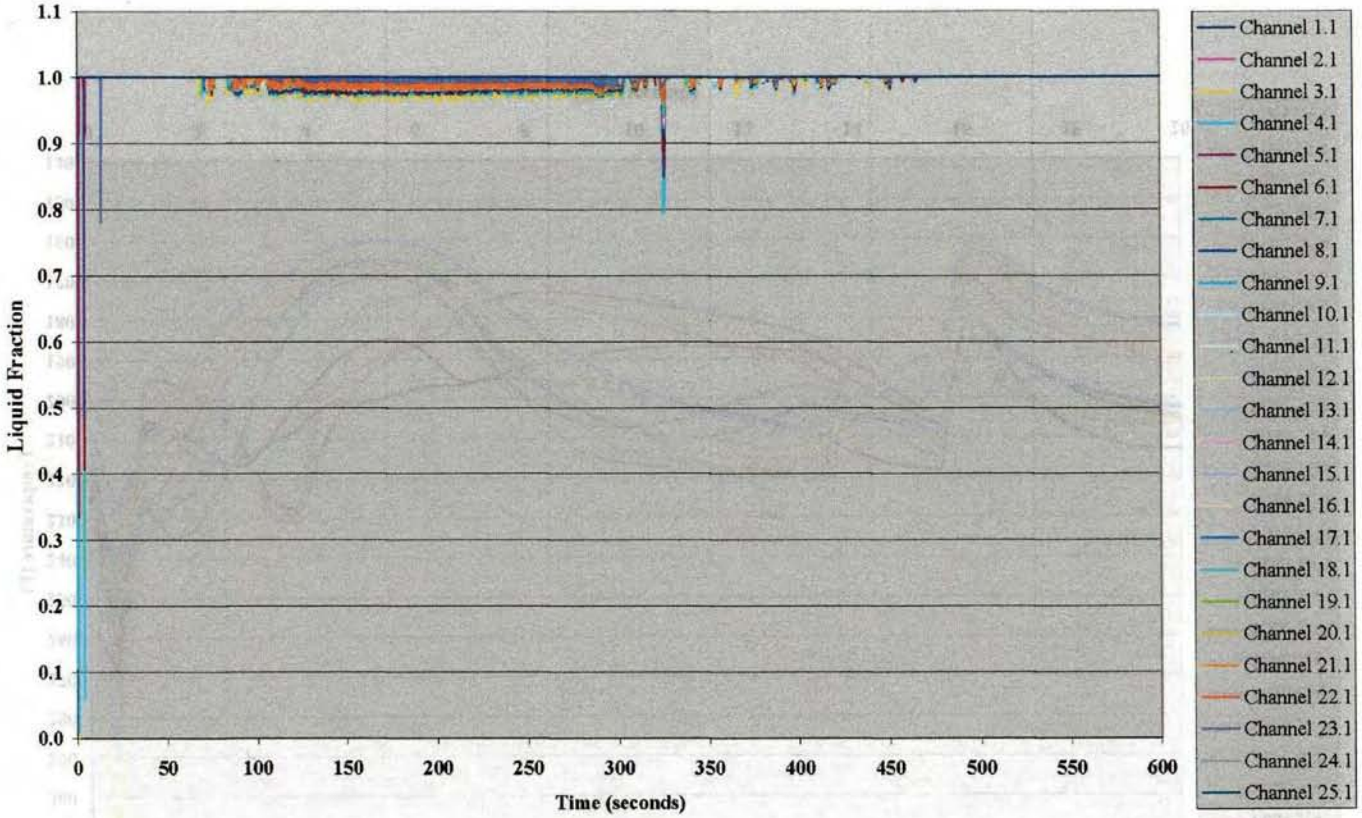

FIGURE 13.10
LIQUID FRACTION OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 1)
DURING THE FIRST 600 SECONDS OF THE COLD LEG LOCA

C-11

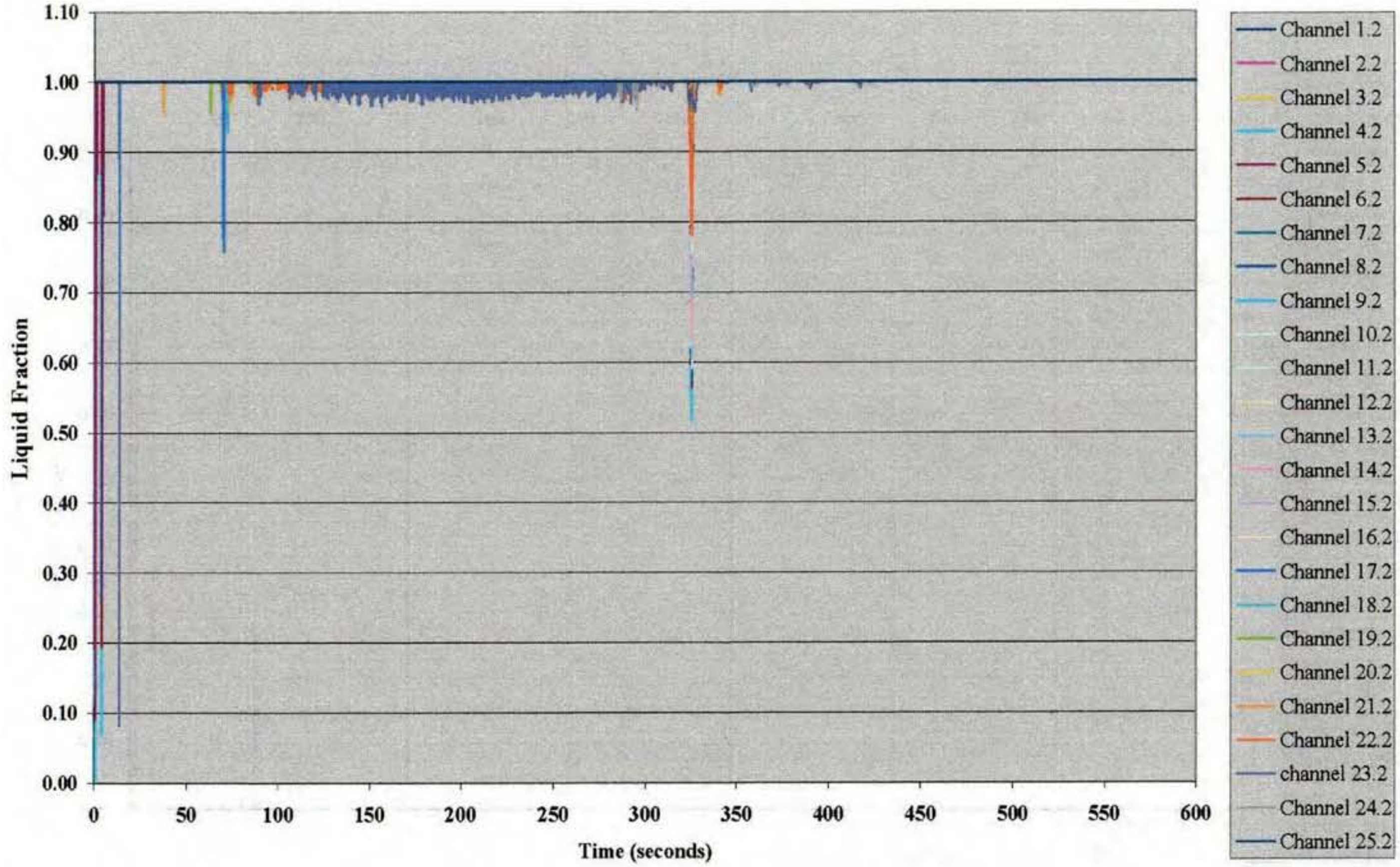

FIGURE 13.11
LIQUID FRACTION OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 2)
DURING THE FIRST 600 SECONDS OF THE COLD LEG LOCA

C-12

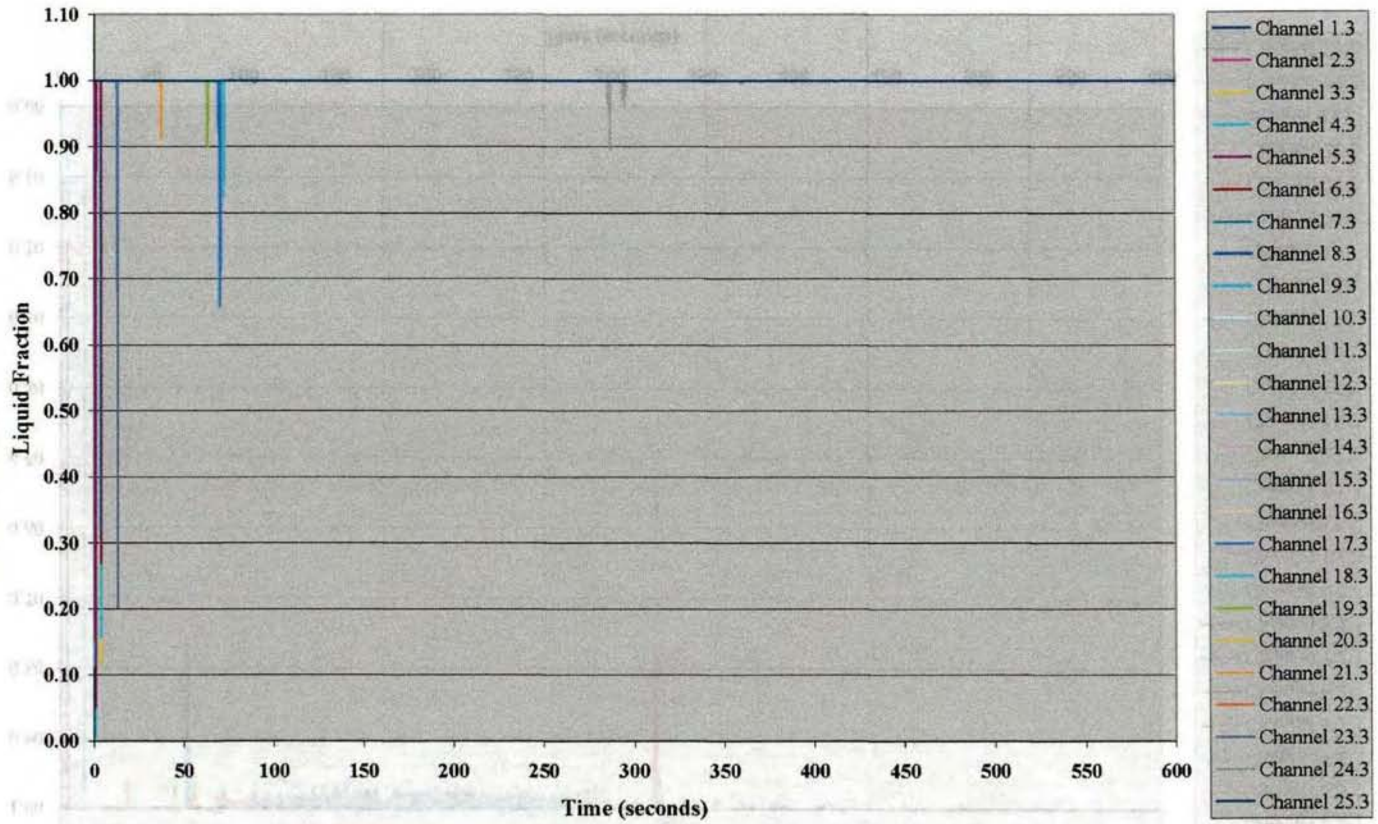

FIGURE 13.12
LIQUID FRACTION OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 3)
DURING THE FIRST 600 SECONDS OF THE COLD LEG LOCA

C.13

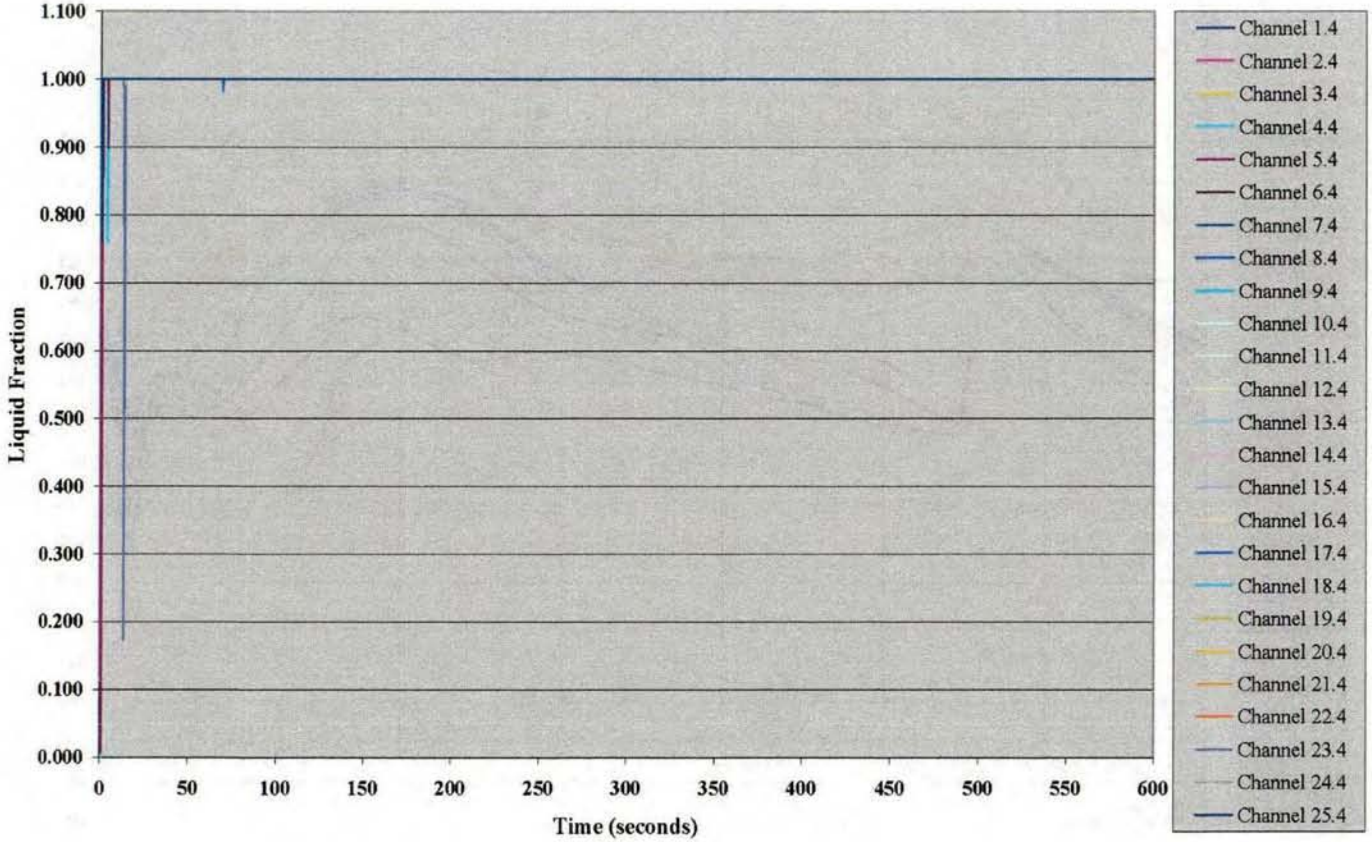

FIGURE 13.13
LIQUID FRACTION OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 4)
DURING THE FIRST 600 SECONDS OF THE COLD LEG LOCA

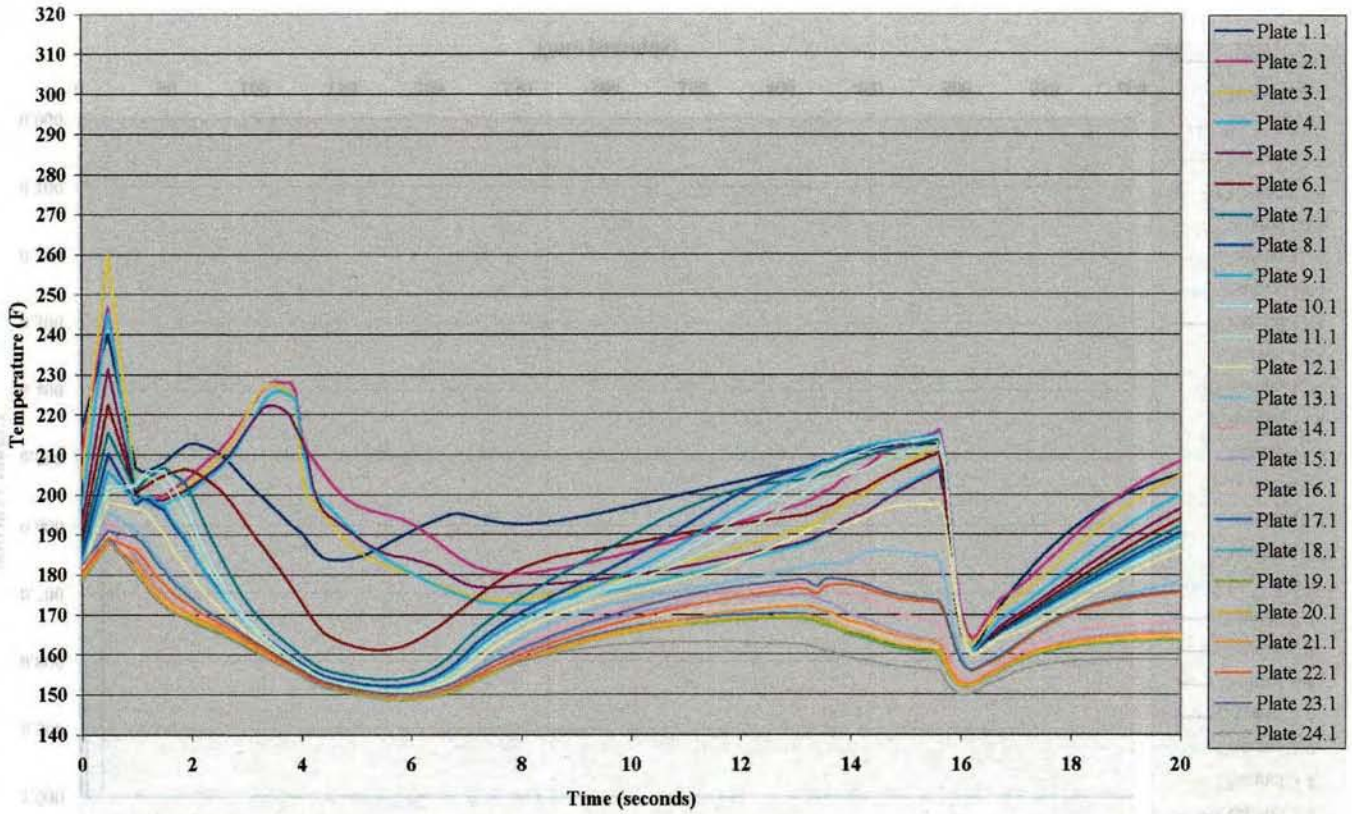

FIGURE 13.14
CENTERLINE TEMPERATURE OF THE 24 FUEL PLATES (SECTION 1)
DURING THE FIRST 20 SECONDS OF THE COLD LEG LOCA

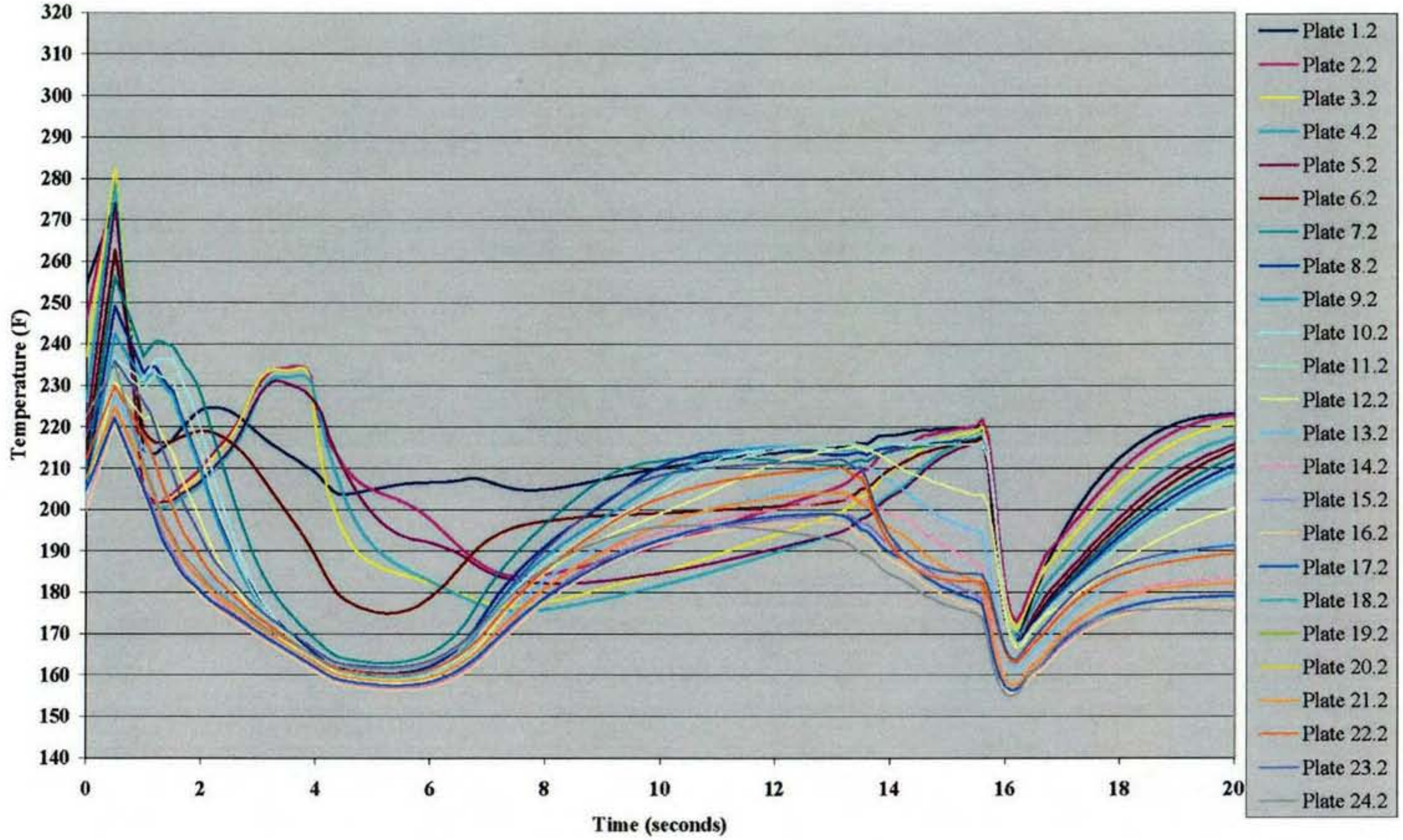

FIGURE 13.15
CENTERLINE TEMPERATURE OF THE 24 FUEL PLATES (SECTION 2)
DURING THE FIRST 20 SECONDS OF THE COLD LEG LOCA

C-16

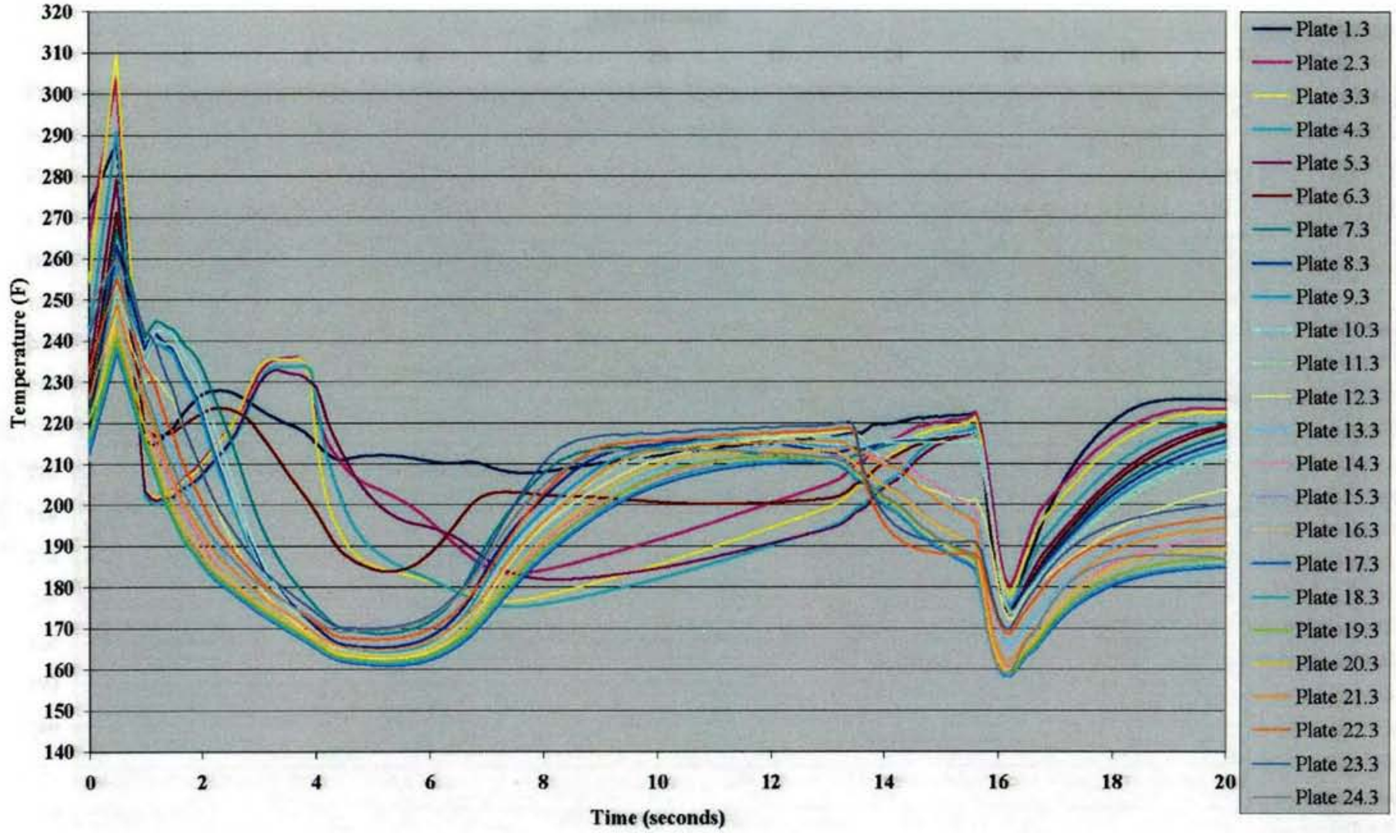

FIGURE 13.16
CENTERLINE TEMPERATURE OF THE 24 FUEL PLATES (SECTION 3)
DURING THE FIRST 20 SECONDS OF THE COLD LEG LOCA

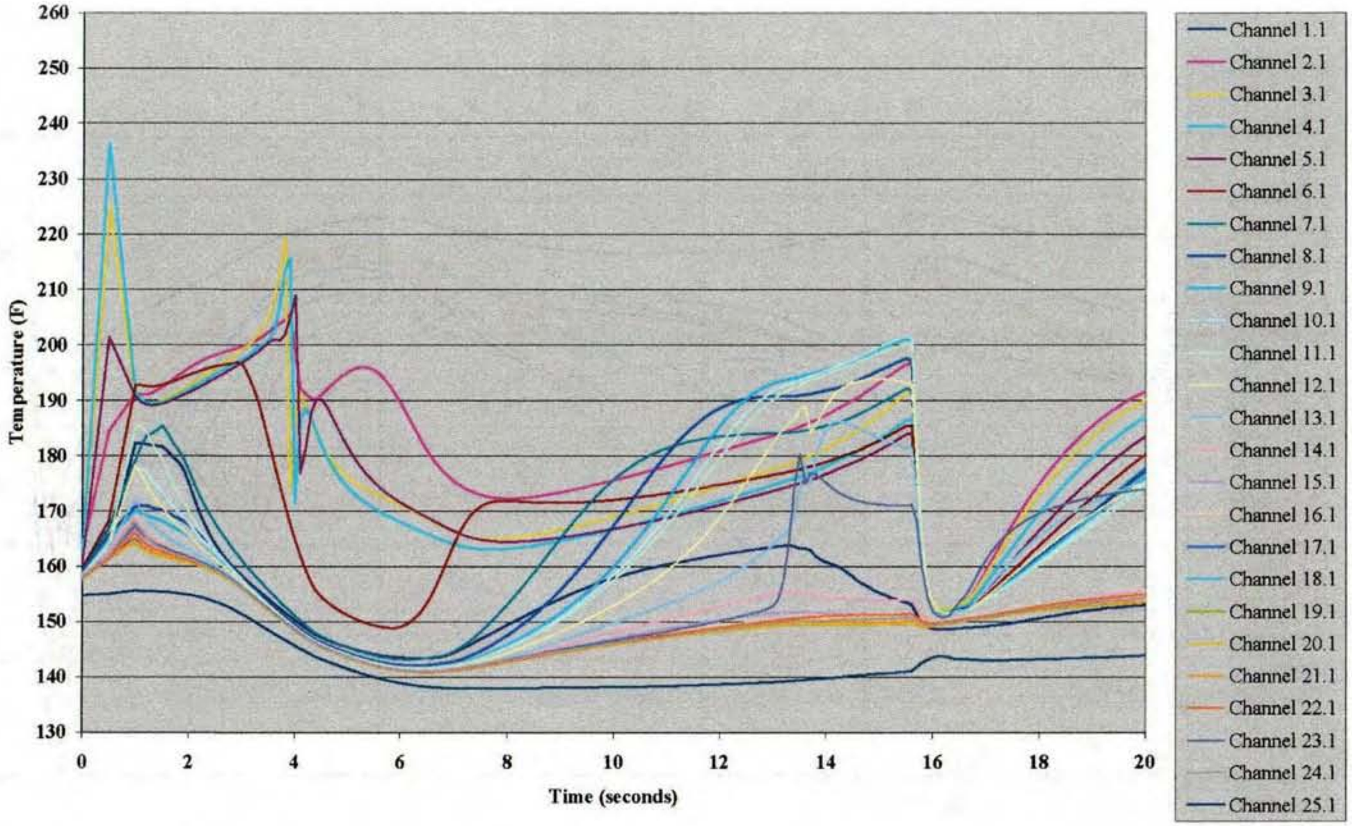

FIGURE 13.17
TEMPERATURE OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 1)
DURING THE FIRST 20 SECONDS OF THE COLD LEG LOCA

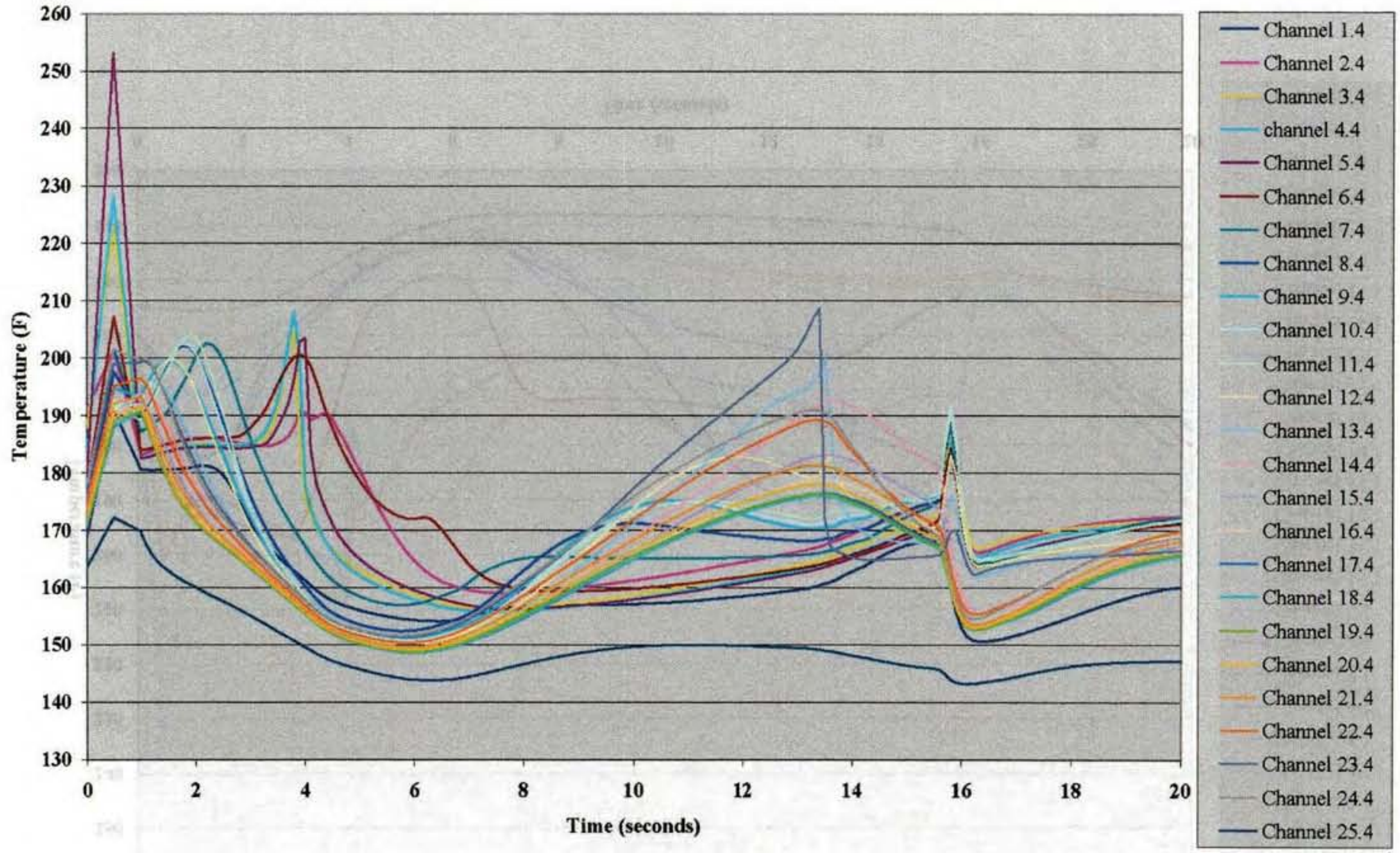

FIGURE 13.18
TEMPERATURE OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 4)
DURING THE FIRST 20 SECONDS OF THE COLD LEG LOCA

C-19

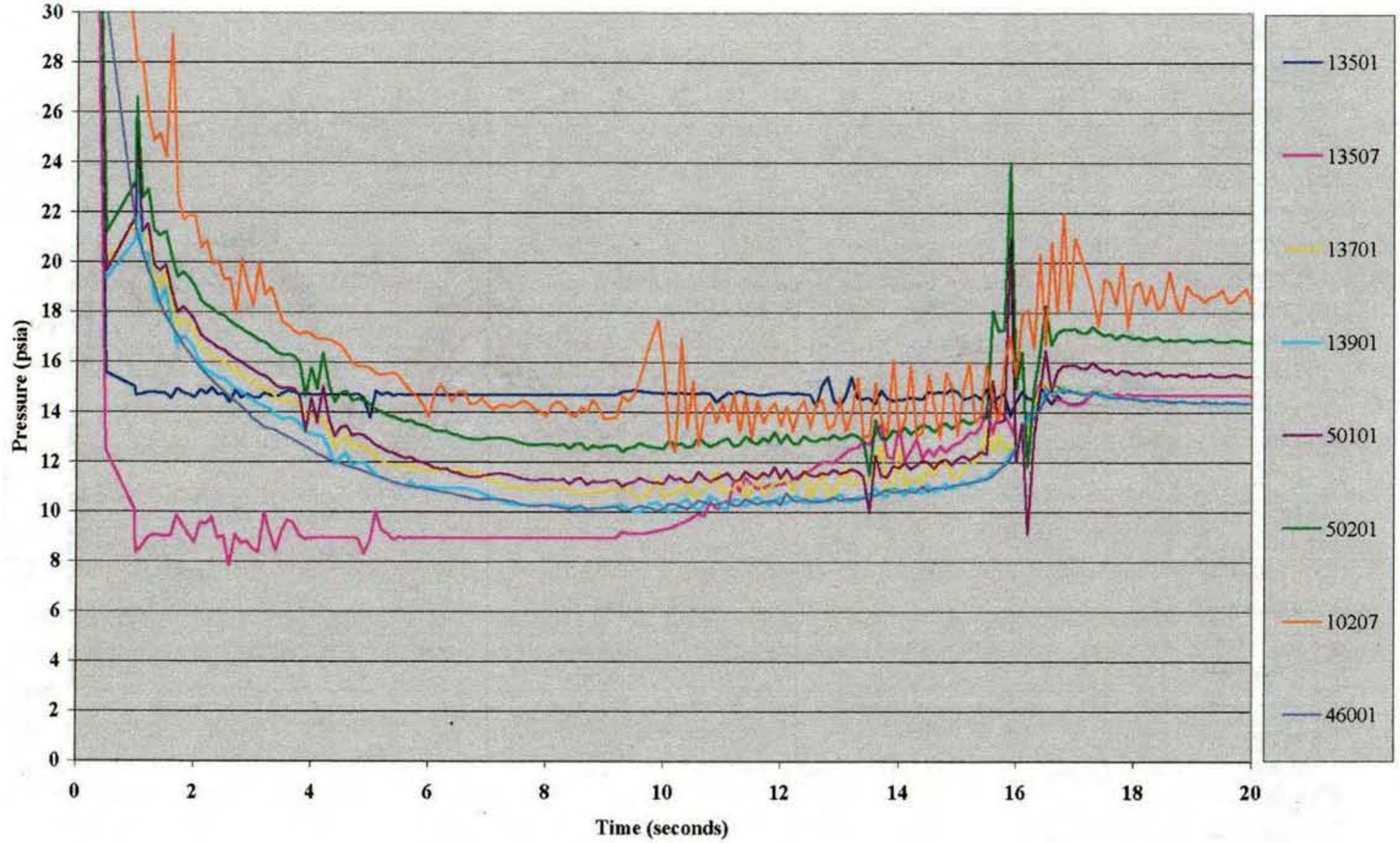

FIGURE 13.19
VOLUME PRESSURES
DURING THE FIRST 20 SECONDS OF THE COLD LEG LOCA

C-20

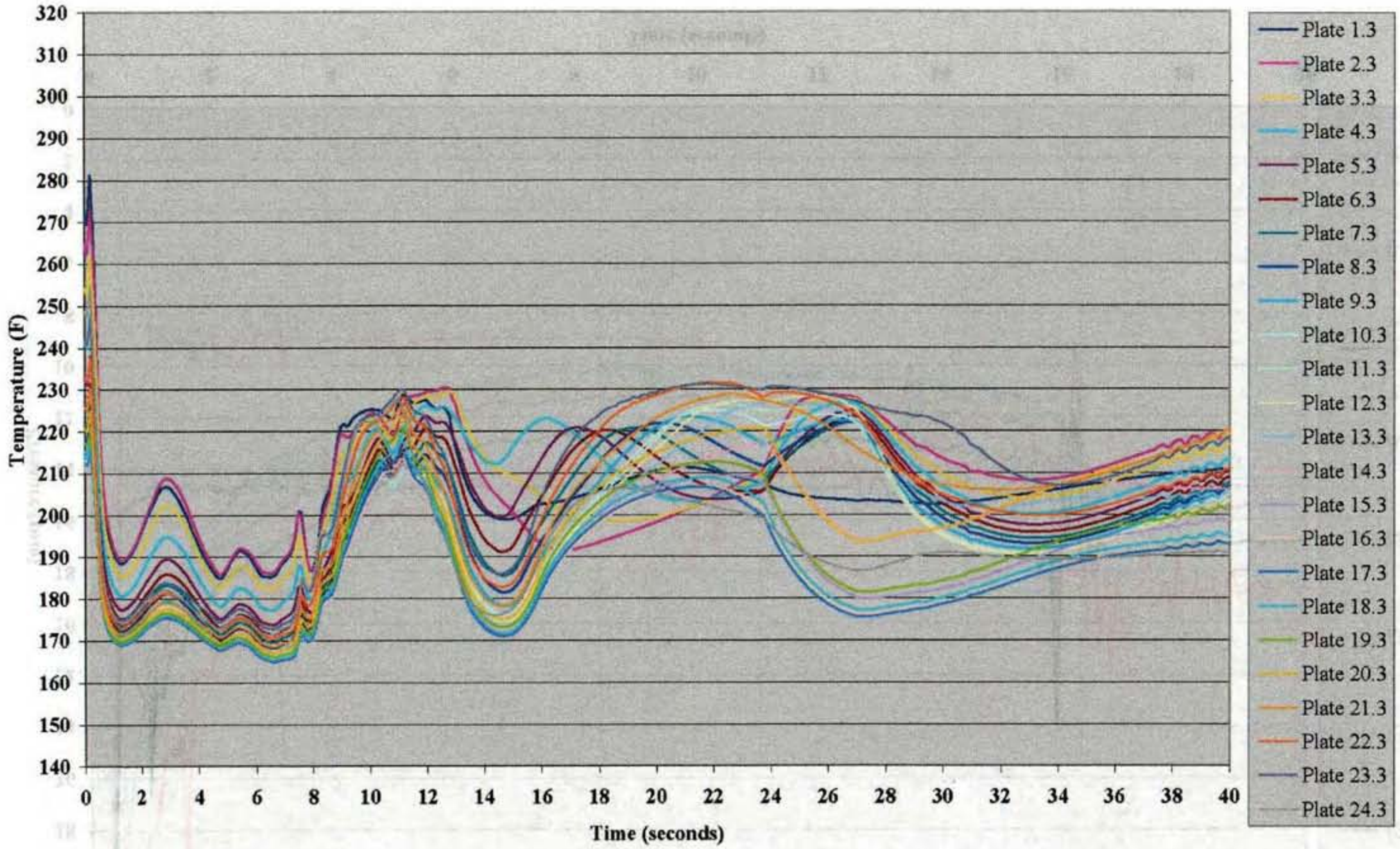

FIGURE 13.20
CENTERLINE TEMPERATURE OF THE 24 FUEL PLATES (SECTION 3)
DURING THE FIRST 40 SECONDS OF THE HOT LEG LOCA

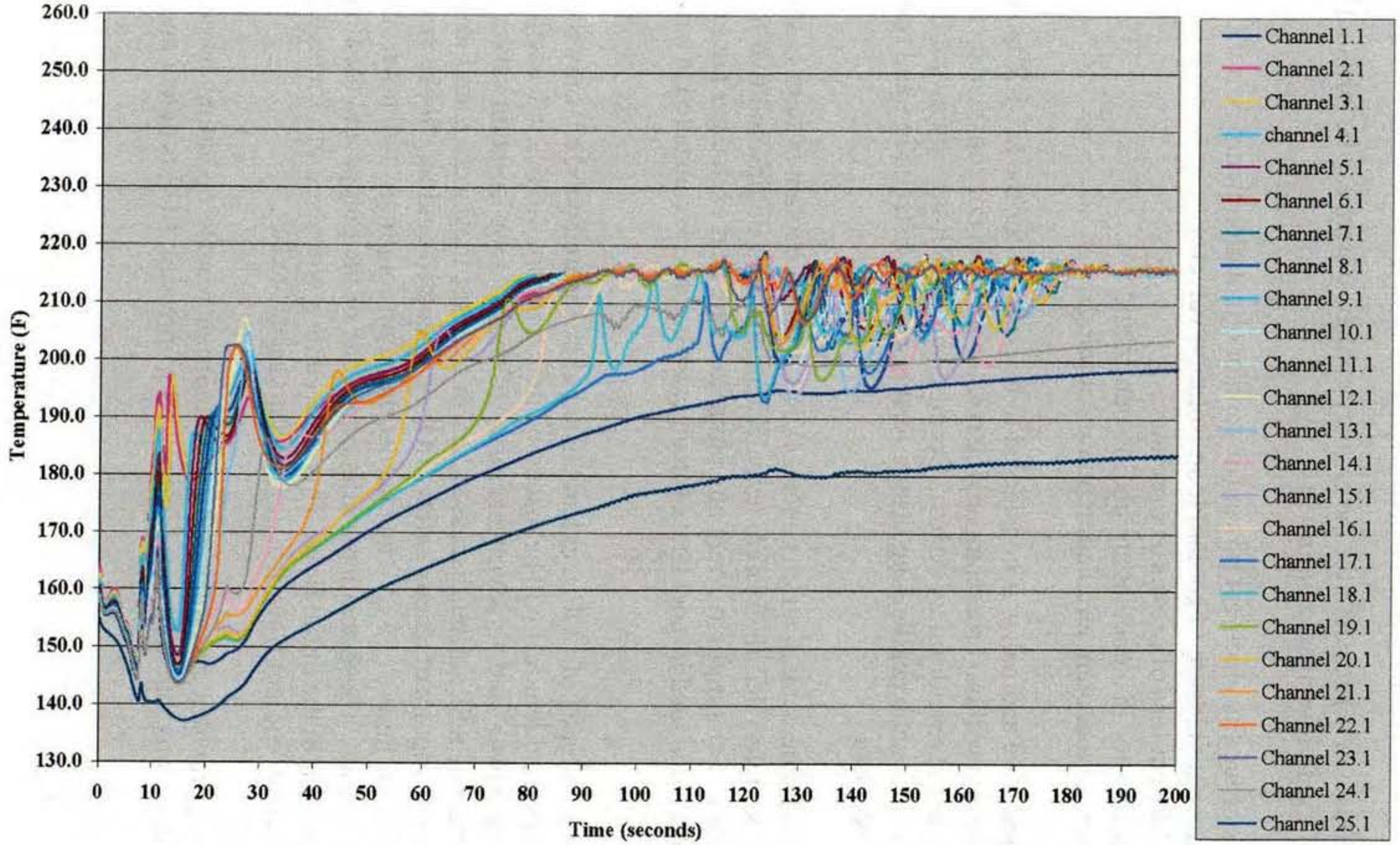

FIGURE 13.21
TEMPERATURE OF THE 25 INDIVIDUAL COOLANT CHANNELS (VOLUME 1)
DURING THE FIRST 200 SECONDS OF THE HOT LEG LOCA

C-99

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 the 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 13.22. It should be noted that the system actually contains two pumps and two heat exchangers operating in parallel. For simplicity, only one pump and one 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 will not be 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 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.

Table 13-7 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.

[redacted]

FIGURE 13.22
SIMPLIFIED SCHEMATIC OF THE PRIMARY COOLANT SYSTEM

TABLE 13-7
NORMAL REACTOR OPERATING CONDITIONS AND CONSERVATIVE ASSUMPTIONS WHEN THE LOF ACCIDENT 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]	62 - 66 psig (427 - 455 kPa)[1]
Anti-Siphon Pressure	26 psig (179 kPa)[1]	36 psig (248 kPa)[1]

[1]Pressure above atmosphere.

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:

1. Actions 1 and 3 listed above;
2. In-pool heat exchanger isolation valves V546A and V546B will open; and
3. 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. It should be noted that 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 13.23 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 95% 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, thus 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 13.24. 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).

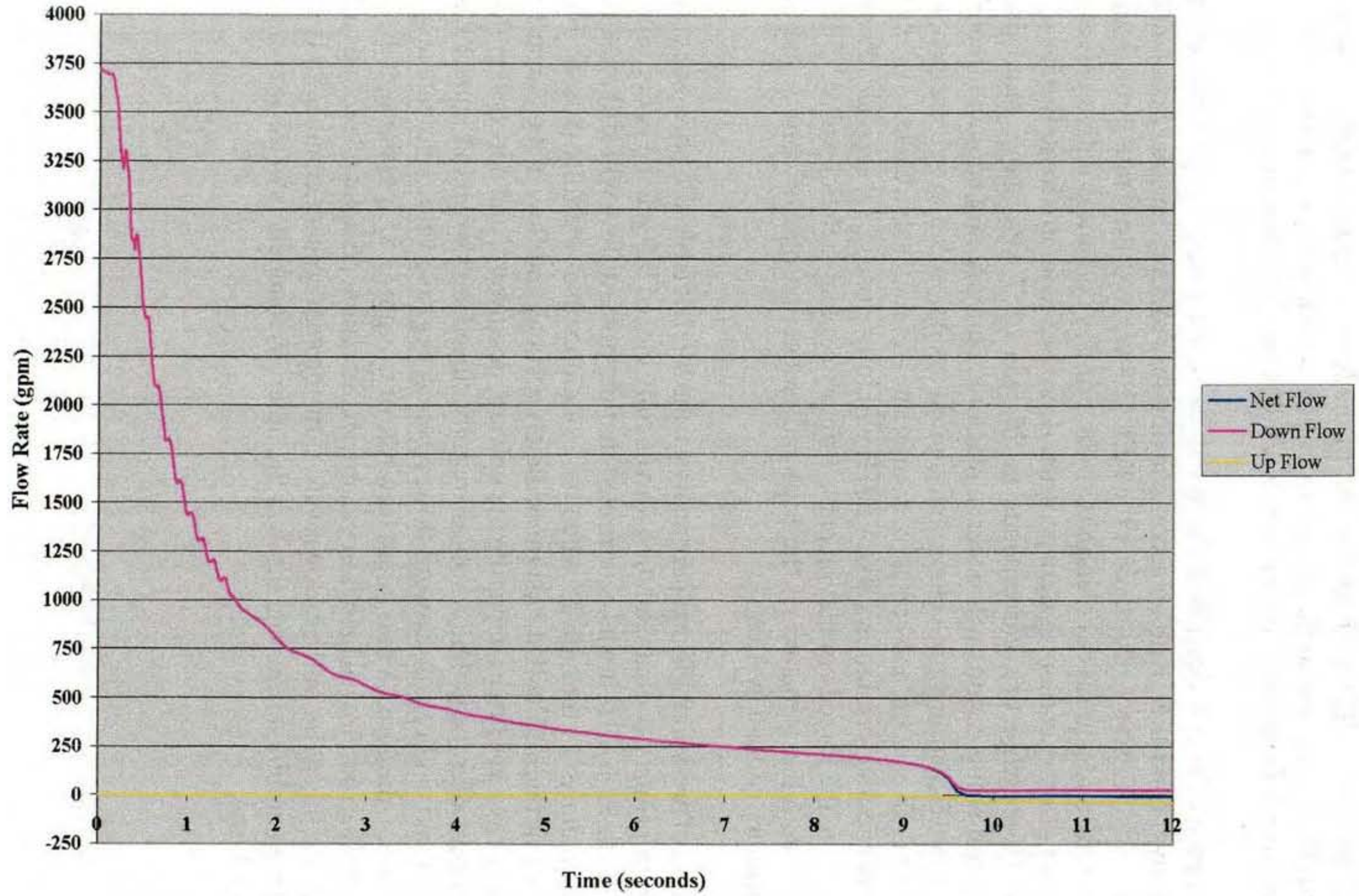

FIGURE 13.23
DOWNWARD, UPWARD, AND NET COOLANT FLOW THROUGH THE CORE
DURING THE FIRST 12 SECONDS OF THE LOF ACCIDENT

C-23

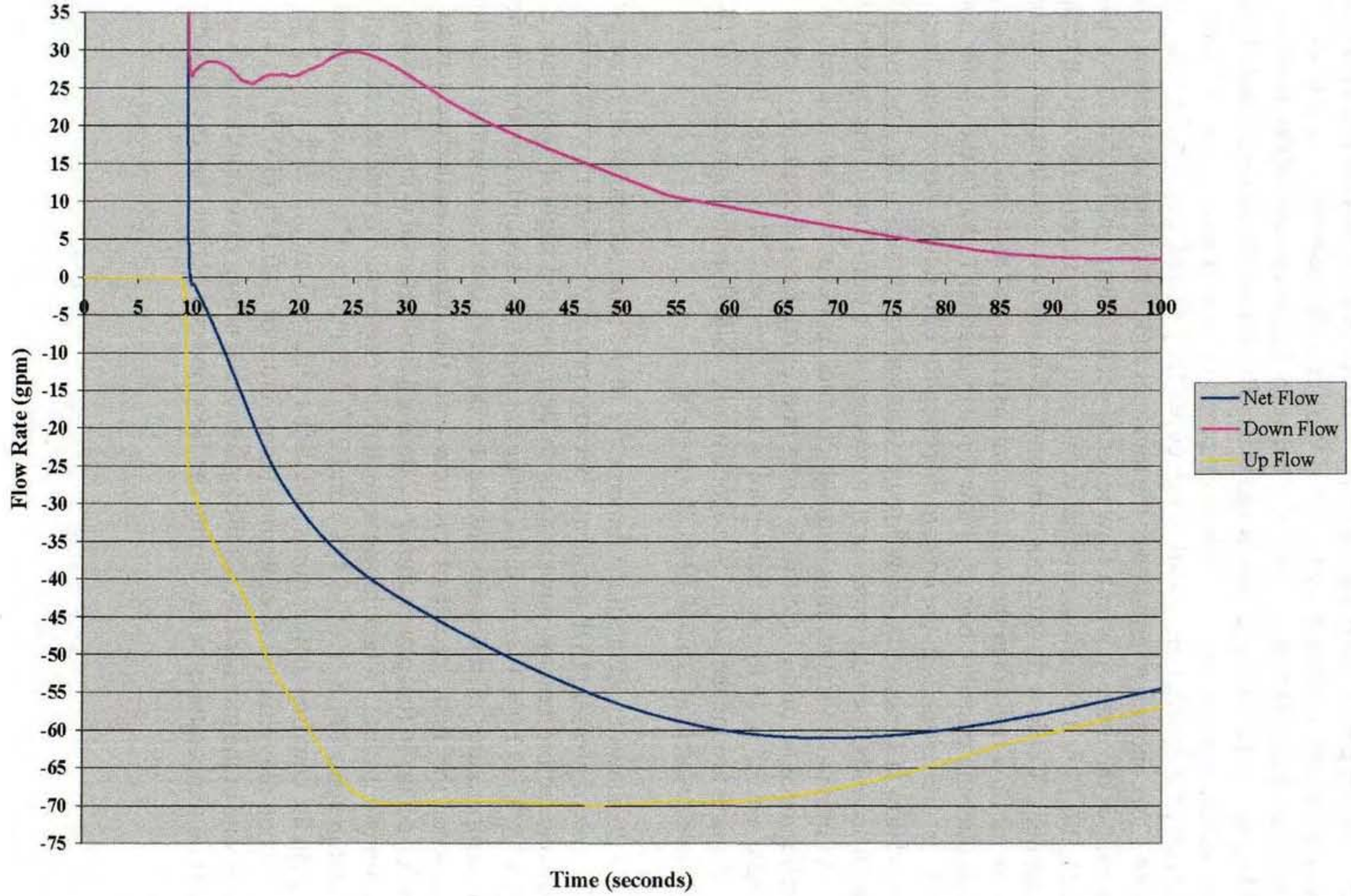

FIGURE 13.24
DOWNWARD, UPWARD, AND NET COOLANT FLOW THROUGH THE CORE
DURING THE FIRST 100 SECONDS OF THE LOF ACCIDENT

C-24

The individual flow rates through all twenty-five coolant channels are shown in Figure 13.25. 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 13.26 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.

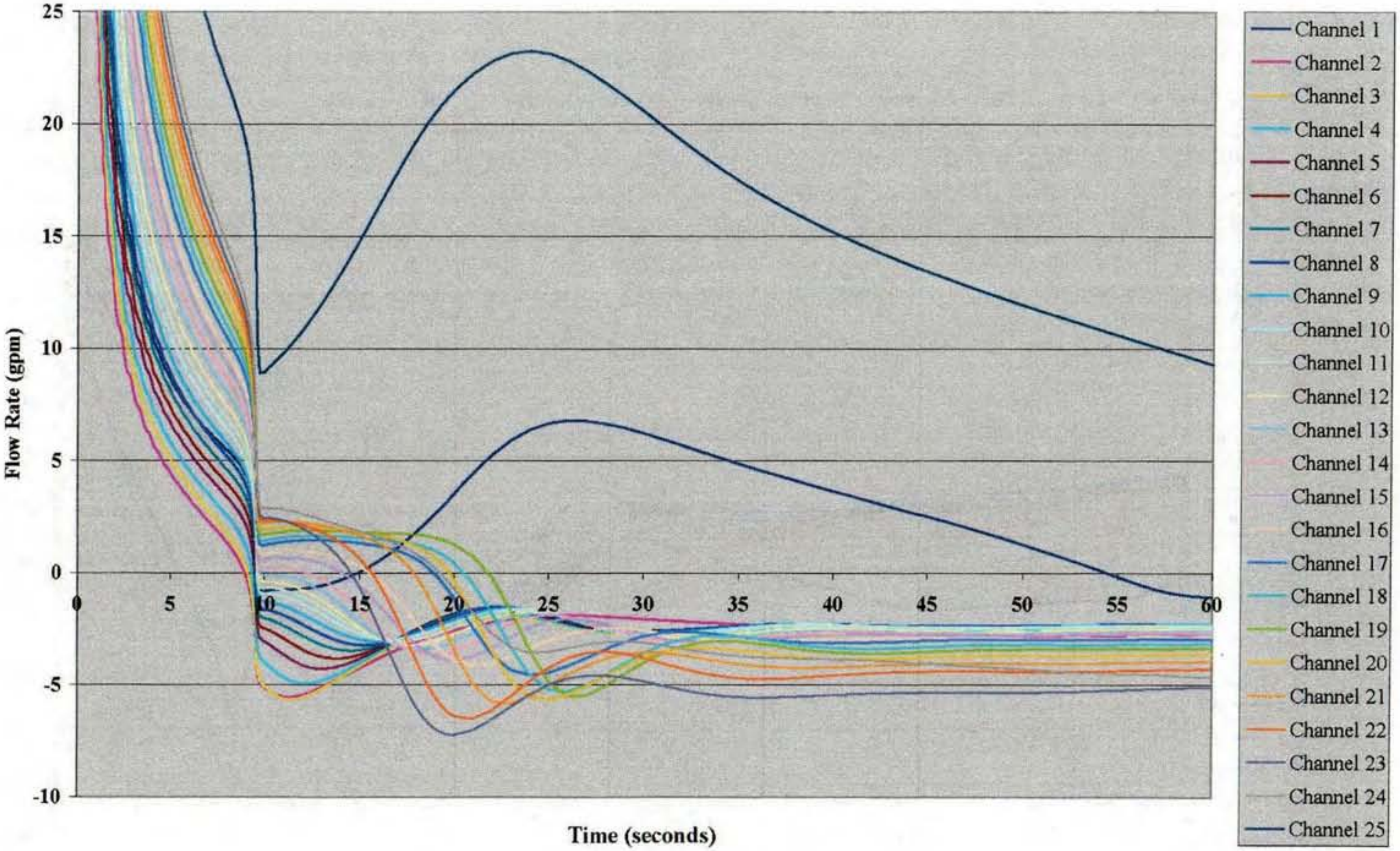

FIGURE 13.25
COOLANT FLOW THROUGH THE 25 INDIVIDUAL COOLANT CHANNELS
DURING THE FIRST 60 SECONDS OF THE LOF ACCIDENT

C-25

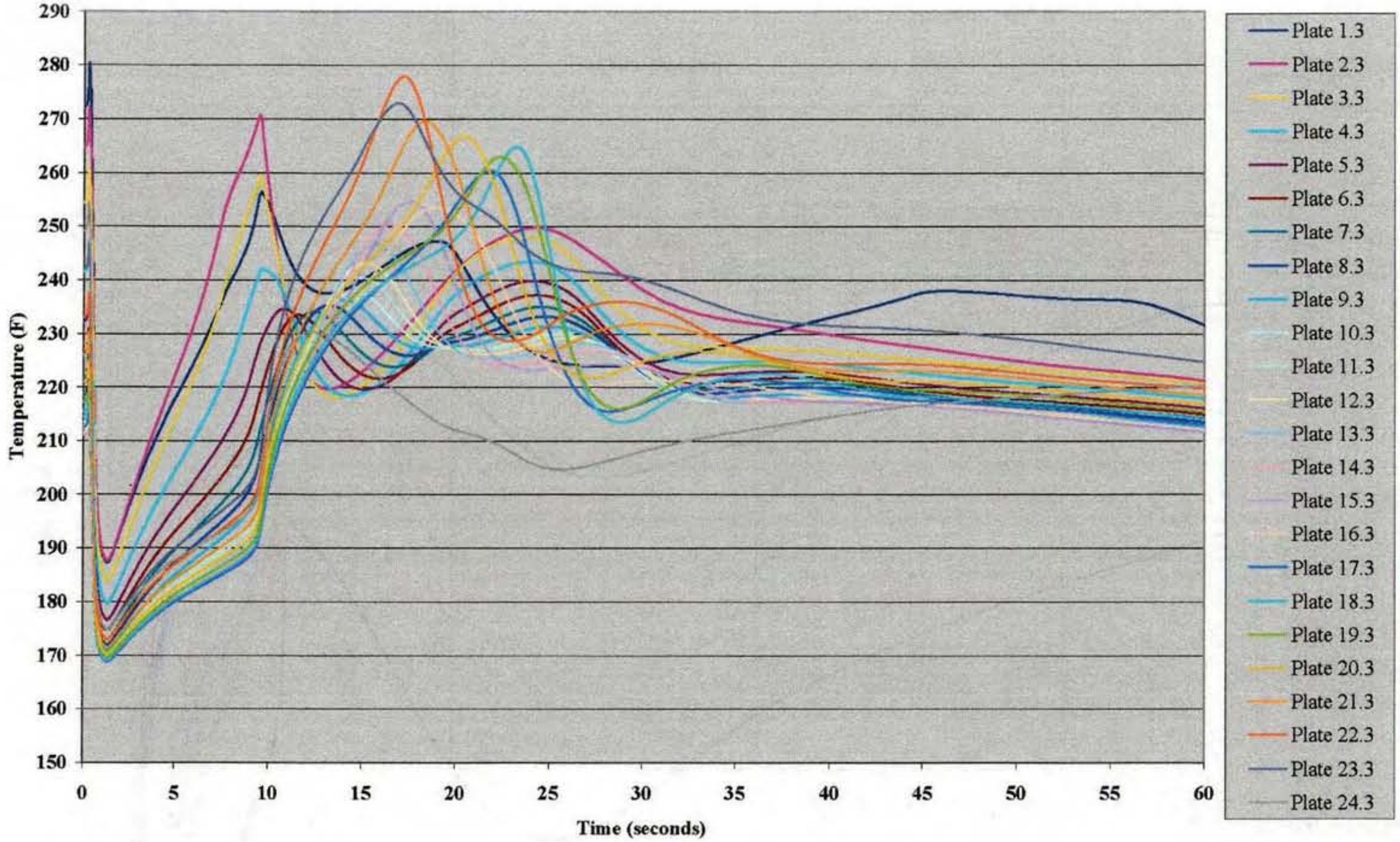

FIGURE 13.26
CENTERLINE TEMPERATURE OF THE 24 FUEL PLATES
DURING THE FIRST 60 SECONDS OF THE LOF ACCIDENT

C-26

Figure 13.27 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). Therefore, 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.

13.2.4.3 Conclusions

The accident discussed in this section represents the worst-case LOF accident that could be realized at the 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 very conservative conditions, it is concluded that reactor safety is not jeopardized by any type of LOF accident. Finally, as discussed in Section 13.2, any fuel element damage that may occur from the partial flow blockage of a fuel element channel is bounded within the Maximum Hypothetical Accident (MHA) analysis.

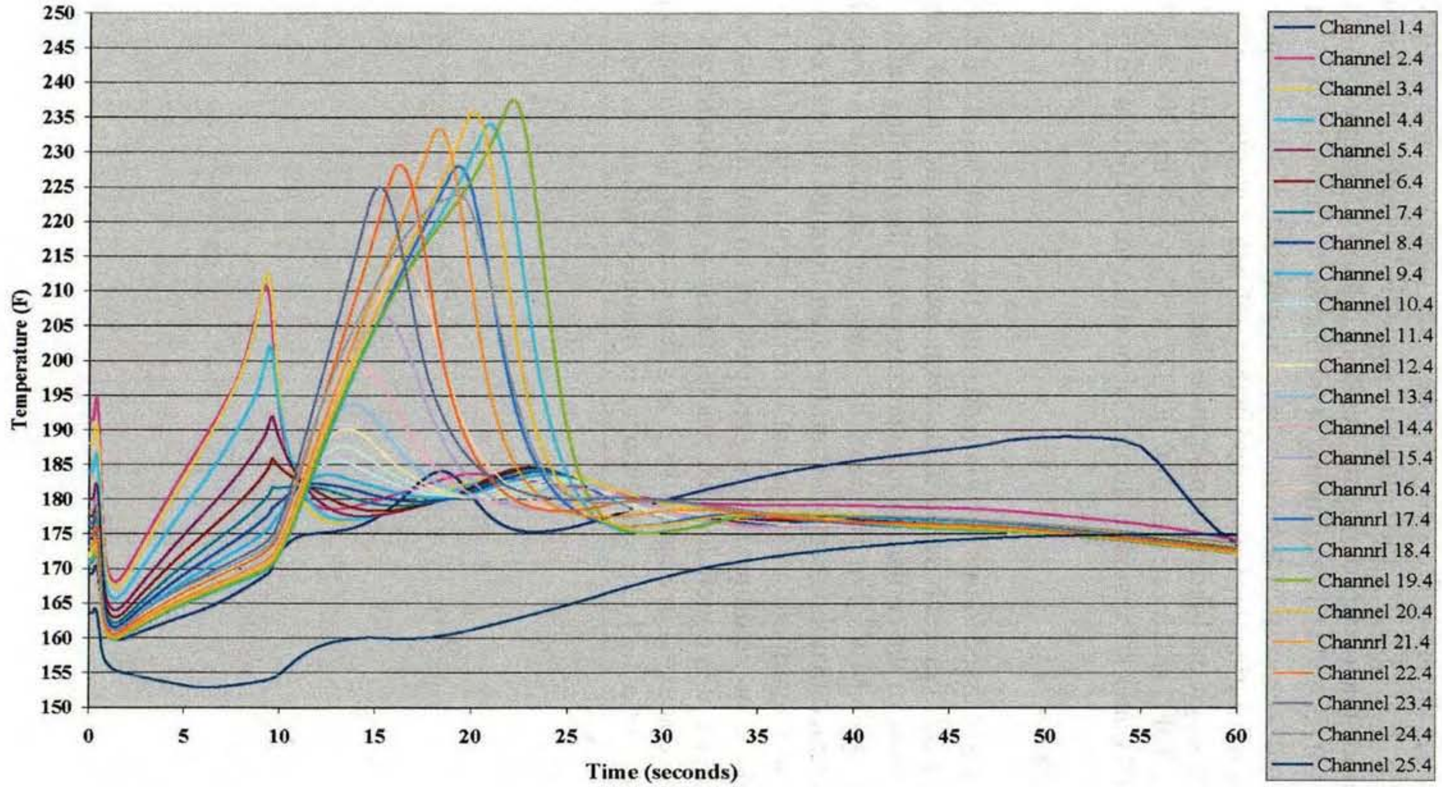

FIGURE 13.27
TEMPERATURE OF THE 25 INDIVIDUAL COOLANT CHANNELS
DURING THE FIRST 60 SECONDS OF THE LOF ACCIDENT

C-27

13.2.5 Mishandling or Malfunction of Fuel

13.2.5.1 Accident-Initiating Events and Scenarios

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.

13.2.5.2 Accident Analysis and Consequences

13.2.5.2.1 Damage to a Fuel Element Due to Mishandling

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 normal 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. [redacted]. These storage locations are designed to ensure a geometry such that the calculated K_{eff} is less than 0.9 under all conditions of moderation, thus allowing sufficient convection cooling and providing sufficient radiation shielding.

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 the in-pool storage locations to the cask is done manually with the cask underwater and resting on the shelf behind the weir wall. The 15-ton capacity overhead rectilinear crane is used to move the cask from the reactor pool. A spent fuel element is not loaded into a shipping cask for shipment until a predetermined cooling period has elapsed from the time 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 cooling time ensures that a fuel element has decayed to a level where air cooling, in the horizontal position, is adequate to maintain fuel temperature below the design limits. Thus, in the event of a dropped loaded cask or a loss of coolant water from the cask, the fuel element would not release fission products by a meltdown. The handling and storage of this fuel is discussed in greater detail in Section 9.2.

As described above, 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 capacity crane, is rigorously maintained, including preventive maintenance and magnetic particle testing, as appropriate. A dye penetrant inspection is also performed on the shipping cask. Therefore, no specific accidents regarding the handling of fuel have been identified for the MURR. The probability of dropping a fuel element while underwater and damaging it severely enough to breach the fuel cladding was considered. However, only the inner and outer most fuel plates are exposed and could potentially become damaged. The MHA assumes the melting of four fuel plates at the peak flux position. Therefore, this accident poses a much smaller risk than the MHA.

13.2.5.2.2 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 ^{235}U loading of 1,270 grams per assembly (Extended Life Aluminide Fuel or ELAF). The ELAF Program, conducted by 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. Studies of deformation mechanisms fo[redacted]- and 1,270-gram 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 (Ref. 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. The MURR has experienced no fuel element failures due to pitting (Ref. 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 ^{131}I level in the primary coolant (Ref. 13.40). Pitting corrosion is not catastrophic in nature and can be detected by conventional monitoring techniques in place at the MURR as demonstrated in this example.

To place an upper limit on a pitting event, a worst-case pit release scenario was analyzed. The MHA addresses a release of fission products by assuming the melting of four number-1 fuel plates, which results in the melting of 2.08% of the fuel and a theoretical release of 2.08% of the fission products if a 100% fuel element fission product release fraction is assumed (which is ultra conservative).

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 thick fuel plate cladding would be 0.090 inches in diameter. To be conservative, the analysis assumes a pit diameter of 0.20 inches, 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). This volume contains 0.015 grams of ^{235}U compared to the 78.58 grams contained in the number-1 plate of four fuel elements. Based upon these assumptions, the release would be equivalent to 1.91×10^{-4} times the release assumed in the MHA. Assuming instantaneous dilution in the 2,000 gallons (7,571 l) of primary coolant and an equilibrium ^{131}I activity of 1.7×10^{5} curies in the core at 10-MW operation, the primary coolant ^{131}I concentration would be calculated as follows (Ref. 13.11):

^{131}I activity in the primary coolant:

$$\frac{(1.7\times10^{11}\,\mu Ci)\times(1.91\times10^{-4})\times0.0208}{2{,}000gal\times3{,}785\,{}^{\mu Ci}/_{gal}}=8.92\times10^{-2}\,{}^{\mu Ci}/_{ml}$$

where:

1.7×10^{11} µCi	=	^{131}I activity in the core;
1.91×10^{-4}	=	0.015 gm/78.58 gm, the ratio of the amount of fission product activity assumed released by "the pit" compared to the MHA release; and
0.0208	=	the fraction of the total fuel melted (including peaking factor) and the total fission product activity released in the MHA assuming a 100% fission product release fraction.

Aluminum film corrosion studies (Ref. 13.28) report that the oxide film rate of formation is dependent on coolant pH, operating time, surface temperature and the magnitude of the heat flux. The MURR has also been able to measure and predict oxide thickness (Ref. 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 predicated to be 0.000854 inches. The 300-MWD power history corresponds to a peak burnup of slightly greater than 2.3 x 10^{21} fissions per cm^3. The three standard deviations value for the oxide thickness at 300 MWD is 1.75 mils. The probability of having an oxide thickness greater than 1.75 mils is less than 0.14% (Ref. 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 MHA is still the most significant accident and brackets any potential risks from fuel corrosion or cladding deformation.

13.2.5.2.3 Overheating of Fuel with Subsequent Potential Cladding Failure

Overheating of the MURR fuel with subsequent potential cladding failure has been analyzed in detail in Section 13.2.3, Loss of Primary Coolant, and Section 13.2.4, Loss of Primary Coolant Flow. Both analyses concluded that the operational and environmental impacts of these accidents did not include fuel melting or fission product release and are easily bounded by the MHA.

13.2.5.3 Conclusions

Three scenarios were proposed for this section. In each case, the analysis of the accident conditions and consequences led to a conclusion that the risks were measurably less than, and well bounded by, the previously analyzed MHA.

13.2.6 Experiment Malfunction

13.2.6.1 Accident-Initiating Events and Scenarios

Experiments conducted at the 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 cannister 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 (Ref. 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 the 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.

MURR 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 is of the utmost importance in ensuring the safety of reactor experiments and has been successfully used for many years at other research reactors and for nearly forty years at the MURR. Therefore, this approach is expected to continue as an effective measure in assuring experiment safety at the MURR.

Limiting the generation of certain fission products in a fueled experiment helps assure 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 ^{131}I through ^{135}I for each fueled experiment is orders of magnitude less than the approximately 4,500 curies of ^{131}I through ^{135}I which are present in the four fuel plates in the peak flux position evaluated in the MHA. In the case of the MHA, 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 significantly less than doses in the MHA:

- 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 ^{131}I through ^{135}I greater than 1.5 curies or strontium-90 (^{90}Sr) 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.

Therefore, limiting fueled experiments to 150 curies of radioiodine will result in a projected dose well within the limits of 10 CFR 20. Similarly, the generation of ^{90}Sr in a fueled experiment is limited to 300 millicuries, which is far below the 78 curies present in the four fuel plates mentioned above. Since no dose limits in the unrestricted area will be exceeded by the MHA, doses from fueled experiments where the ^{90}Sr inventory is limited to 300 millicuries will be safely within the limits of 10 CFR 20.

The amount of explosive materials which can be irradiated, or which is allowed to generate in any experiment, has been limited to 25 milligrams of TNT-equivalent explosives in order to reduce the likelihood of damage to the reactor or pool should the explosive material detonate. 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.

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 $\Delta k/k$. The 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 to the point where any identified risks are far less than those from 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. 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) would provide 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 (Ref. 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, hence they would fail to close (or open) in response to any electrical signal. Also, the control system which inflates the gaskets is actuated only when the isolation doors are in the closed position, 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 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. [redacted] 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 affect 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. In addition, 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 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 the 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 run-off from 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. Boone County has experienced 32 reported tornadoes within the recording period 1950 to 2005. However, structural damage has generally been limited to frame/lumber and mobile home residential units. Thunderstorms are observed during every month of the year. During the summer they are most frequent, and may occur weekly. The MURR is designed to withstand the extreme wind speeds associated with any thunderstorm activity. Section 2.3.1 gives more detail on the overall meteorology of the area, while Sections 2.4.1 and 2.4.2 describe the situation with respect to surface water and water drainage. Review of these sections does not provide a basis for significant meteorologically-related accident scenarios for the MURR.

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 very low probability. In addition, there are no nearby industrial, transportation, or military facilities with the potential of causing a credible accident thereby preventing a safe reactor shutdown or resulting in a release of radioactive material from the reactor facility that would exceed 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 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). [redacted]. 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:

1. In the section of piping between the reactor pool and return line isolation valve V509;
2. In the section of piping between the reactor pool and supply line check valve V519A;
3. In a section of the system not specifically mentioned as one of the two previous locations; and
4. 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, Shearing of a Beamport.

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 unisolatable portion of the pool coolant system, is discussed below.

13.2.9.1.2 Accident Analysis and Consequences

The primary consideration when discussing an unisolatable 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 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 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 supply line, drainage would be through the pool diffuser. The diffuser is a vertical section of pipe with 252 ⅝-inch 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 very 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 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 unisolatable 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 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 the 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.

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. Nevertheless, 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 insures that the radiation level above the pool surface from direct core radiation will remain less than 2.5 millirem/h. 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 beamport with no impediments in the port. The emergency pool fill system is actuated by the opening of a 4-inch 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 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 main to provide a portion of the campus water supply. The 8-inch wet fire line that provides the flow path for the emergency pool fill system is connected to this 10-inch main. To ensure a continuous supply of water to the 10-inch 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 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 fire main and the 4-inch 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 probability of shearing a beamtube due to a dropped object is basically 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 extremely difficult, if not impossible, to suspend a heavy object above a beamtube. In addition, as previously stated, it is standard practice at the MURR to have the removable collimator liner installed in the beamtube, thereby preventing a loss of pool water should a beamtube be sheared. Nevertheless, if a sheared beamtube caused 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 well 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 (Ref. 13.31). Therefore, the failure of both 546 isolation valves to open when required is not considered credible. Nevertheless, 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. Although a detailed discussion of this analysis is not included in this SAR, 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 far 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 a 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 primary coolant system pressure would be caused by any one of the following three scenarios:

1. The addition of nitrogen gas through nitrogen addition valve V526;
2. A reactor plant heat-up from 70 to 160 °F (21 to 71 °C); or
3. 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) (Ref. 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 (Ref. 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, the MURR is adequately protected from postulated high pressure transients. A system high pressure alarm alerts the operator to an abnormal high pressure condition, thus 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), thus providing a sufficient margin to assure 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. Therefore, 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 would be detected during the weekly pool water analysis. The detection limits for antimony-122 (^{122}Sb) and antimony-124 (^{124}Sb) 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 1.06×10^8 ml (28,000 gals). 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.

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 (Ref. 13.33) points out that beryllium has been irradiated to neutron fluences of 1.8 x 10^{20} nvt with no dimensional changes. To achieve this nvt, 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 ^{124}Sb (the license limit). Furthermore, there are no direct gaseous products from antimony's decay, and therefore its expansion should be no greater than that of the beryllium. Consequently, a pressure buildup due to a volumetric expansion is also negligible.

Strict administrative controls prevent the inadvertent removal of the neutron source from the reactor pool. [redacted]. 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 millirem/h (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. Also, administrative controls limiting the irradiation time of the neutron source prevent the activity level from exceeding the facility license limit.

CHAPTER 14

TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

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.

14.0 TECHNICAL SPECIFICATIONS

This chapter discusses the development, format, and contents of the reactor facility Technical Specifications.

14.1 Introduction

The Technical Specifications (TS) 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 TS. Procedures or actions employed to meet the requirements of these TS are not included in the TS. Normal operation of the reactor within the limits of the TS 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 Missouri University Research Reactor (MURR) TS that are being 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" (Ref. 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 (Ref. 14.2).

To ensure that all items that may be relevant for inclusion in the TS have been considered, the TS are divided into the following six (6) sections:

Section 1 - Definitions;
Section 2 - Safety Limits and Limiting Safety System Settings;
Section 3 - Limiting Conditions for Operation;
Section 4 - Surveillance Requirements;
Section 5 - Design Features; and
Section 6 - Administrative Controls.

Specifications in Sections 2, 3, and 4 provide related information in the following format:

- Applicability - This indicates which components are involved;
- Objective - This indicates the purpose of the specification(s);
- Specification(s) - This provides specific data, conditions, or limitations that bound a system or operation. This is the most important statement in the TS agreement; and
- Bases - This provides the background or reasoning for the choice of specification(s), or references a particular section of the MURR SAR that does.

It is important to note that, although the applicability, objective, and bases provide important information, only the "specification(s)" statement is governing. Section 5, Design Features, and 6, Administrative Controls, simply state the applicable specification(s).

14.2.1 Definitions

The definitions listed in this section provide a uniform interpretation of terms and phrases used in Reference 14.1 and other associated standards. The definitions listed in Reference 14.1, that are applicable to MURR, are typically stated verbatim. Modifications and additional definitions presented in Reference 14.2 have been used to help clarify the meaning of terms used in ANSI/ANS 15.1. Definitions specific to the MURR are included to clarify terms referred to in the TS.

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 TS. 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. For the MURR, the measurable parameters include reactor power, primary coolant flow, reactor inlet water temperature, and pressurizer pressure. The primary barrier for heterogeneous-core, non-power reactors is the aluminum cladding of the fuel plates. Cladding integrity could be lost by softening, melting, blistering, or yielding to excessive internal pressure, all of which are dependent on temperature and operating history.

The MURR safety limit analysis, as presented in Section 4.6.3 of the SAR, provides 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 the departure from nucleate boiling (DNB).

14.2.2.2 <u>Limiting Safety System Settings</u>

For each measurable parameter on which a safety limit has been established in the SAR, 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. 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 of the SAR provides the bases for the LSSSs for the MURR.

14.2.3 <u>Limiting Conditions for Operation</u>

Limiting Conditions for Operation (LCO) 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.

For the MURR, the following ten (10) systems or operational characteristics have LCOs placed on them:

1. Reactivity Limits;
2. Control Blades;
3. Reactor Safety System;
4. Reactor Instrumentation;
5. Reactor Containment Building;
6. Experiments;
7. Facility Airborne Effluents;
8. Reactor Fuel;
9. Reactor Coolant Systems; and
10. Auxiliary Systems.

14.2.4 <u>Surveillance Requirements</u>

Typically, a specific system from a Section 3 specification will establish the lowest functional capability or the minimum performance level, and a companion Section 4 surveillance specification requirement will prescribe the frequency and scope of surveillance to demonstrate such functional capability or performance.

In general, three types of surveillance requirements are specified: operability checks, calibrations, and system inspections. For the MURR, surveillance requirements exist for the following six (6) systems:

1. Containment System;
2. Reactor Coolant Systems;
3. Control Blades;
4. Reactor Instrumentation;
5. Reactor Fuel; and
6. Auxiliary Systems.

Maximum allowable surveillance intervals, as specified in the TS, provide operational flexibility and are not used to reduce frequency. Established frequencies are maintained over the long term. Generally, 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 of the TS 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 the 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.

For the MURR, the following specific areas or systems have been addressed in Section 5:

1. Site Description;
2. Reactor Containment Building;
3. Reactor Coolant Systems;
4. Reactor Core and Fuel; and
5. Emergency Electrical Power System.

14.2.6 Administrative Controls

The information and controls on staffing and operations of the reactor facility, as specified in this section of the TS, will ensure that facility management and staff are acceptably knowledgeable and aware of the technical requirements to operate a safe facility and that all applicable regulations and license conditions are complied with.

Chapter 12, Conduct of Operations, provides the bases for this section of the TS. The following sections are included in Administrative Controls:

1. Organization - The organizational structure of the University of Missouri as it relates to the MURR and the minimum staffing requirements to operate the reactor are provided by this section;

2. Review and Audit - This section discusses the Reactor Advisory Committee (RAC): a committee, appointed by the Office of the Provost, University of Missouri at Columbia, to provide objective and independent reviews, evaluations, and recommendations on matters affecting reactor safety;

3. Procedures - The written procedures that have been established for the reactor operations and health physics groups are described in this section of the TS. These procedures provide detailed guidance in the operation and utilization of the reactor and the laboratory facilities and are 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;

4. Records - This section of the TS lists the records that shall be maintained by the facility in addition to those otherwise required under the facility operating license and applicable regulations. 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; and

5. Reportable Events and Required Actions - This section of the TS lists the reports that shall be made to the NRC by the facility and the required actions following certain incidents and conditions relating to the operation of the reactor in addition to those otherwise required by Title 10, Code of Federal Regulations.

14.3 Changes to the MURR Technical Specifications

The MURR TS, as revised through Amendment No. 33, are officially Appendix A to the facility operating license [License No. R-103 (NRC Docket No. 50-186)]. As part of the application for license renewal, an updated version of the MURR TS are being submitted as Appendix A to both the facility operating license and the SAR. All key sections of the TS remain the same as described above in Section 14.2, however sections 4 and 5, Design Features and Surveillance Requirements, respectively, have been swapped to conform with the format provided by Reference 14.1.

The MURR, however, as part of the analyses which supported the preparation of the SAR, has updated many of the bases in the TS. Some of the various analyses described in the SAR, particularly in Chapters 4, 6 and 13, have resulted in improved and more accurate analytical conclusions. These improved conclusions have been incorporated into the TS bases where appropriate. Other changes include the addition of eight (8) new definitions, the removal of the applicability, objective and basis information from Section 5, and changes to Section 6 that are consistent with the Administrative Controls section of Reference 14.1.

New definitions have been added to help clarify terms referred to in the TS, and to remove the potential for any ambiguity. Information removed from Section 5 is consistent with Reference 14.1, Section 1.2.2, which states, "Section 5, Design Features, and 6, Administrative Controls, should state the specifications without the related information." Changes to Section 6 are consistent with the guidance provided by Reference 14.1. Additionally, some reformatting, correction of typos and the insertion of the correct names for contacting the NRC in Section 6 were also performed.

CHAPTER 15

FINANCIAL QUALIFICATIONS

TABLE OF CONTENTS

LIST OF TABLES

15.0 FINANCIAL QUALIFICATIONS

The University of Missouri-Columbia (MU) is financially qualified to own, operate, and decommission the Missouri University Research Reactor (MURR). This chapter describes the University's financial ability to safely operate and decommission the facility.

15.1 Financial Ability to Construct a Non-Power Reactor and Related Fuel Cycle Costs

The MURR is an existing facility, therefore the issue of construction is not relevant.

As stated in Section 1.6 of this report, The Nuclear Waste Policy Act of 1982, Section 302(b)(1)(B) states that the Nuclear Regulatory Commission (NRC) may require, as a precondition to renewing an operating license for a research reactor under Section 104 of the Atomic Energy Act of 1954, as amended (the Act), that the applicant shall have entered into an agreement with the Department of Energy (DOE) for the disposal of high-level radioactive wastes and spent nuclear fuel. The DOE provides fuel assistance to 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 states that the DOE retains title to the fuel and is obligated to take the spent fuel for storage or reprocessing. The Curators of the University of Missouri have entered into such a contract with the DOE, hence the applicable requirements of the Nuclear Waste Policy Act of 1982 are satisfied.

The DOE funding applicable to the MURR's fuel cycle includes the cost associated with fuel fabrication, transport of new fuel to the facility, and transport of spent fuel from the facility. The cost to the MURR is $4,000 per fuel assembly. This equates to an approximate annual expenditure of $100,000 for new fuel. In addition, there are minor shipping charges for spent fuel that the MURR incurs of approximately $10,000 per year. These costs are included in the "Overhead" category of Tables 15-1 and 15-2.

15.2 Financial Ability to Operate a Non-Power Reactor

MU has the financial ability to operate the MURR. Table 15-1, "Costs of Operation of MU Research Reactor: FY 05" lists the actual operating expenses for fiscal year 2005. This table identifies that the cost of conducting the facility's commercial activities is less than 50% of the cost of owning and operating the facility. Therefore, the MURR continues to be classified as a Class 104 licensed facility.

TABLE 15-1

COSTS OF OPERATION OF MU RESEARCH REACTOR: FY 05

Research Direct; Commercial Direct; Mixed (Overhead); and Total Expenditures

	Research Direct	Commercial Direct	Overhead	Total
Funds Source	FY 05 actual	FY 05 actual	FY 05 actual	FY 05 actual
Total Operating Funds	5,351,059	2,695,274	4,091,528	12,137,861
Overhead Allocation	2,720,992	1,370,536	(4,091,528)	-
Grand Total	$ 8,072,051	$ 4,065,810	$ -	$ 12,137,861
Research Percentage	67%			
Commercial Percentage		33%		

NOTES:

Overhead is allocated to Research Direct and Commercial Direct according to each one's respective share of total direct expenditures in a fiscal year.

Included above are only those expenditures made directly from MU Research Reactor accounts, and does not include all University contributions in support of the reactor. The University, in support of the reactor's research mission, provides significant, in-kind "institutional support" and about a third of the utilities.

TABLE 15-2
MURR 2006-2011 BUDGET

(in thousands of dollars)	FY06 Budget	FY07 Projected	FY08 Projected	FY09 Projected	FY10 Projected	FY11 Projected
REVENUES						
General Operating (Campus Allocation)	$ 2,640	$ 2,746	$ 2,855	$ 2,970	$ 3,088	$ 3,212
Grant Funds	1,500	500	520	541	562	585
Service Operations	8,600	8,944	9,302	9,674	10,061	10,463
TOTAL REVENUES	$12,740	$12,190	$12,677	$13,184	$13,712	$14,260
EXPENDITURES						
Administrative Services						
Salary/Wages	$ 784	$ 815	$ 848	$ 882	$ 917	$ 954
Benefits	220	229	238	247	257	268
Supplies	31	32	34	35	36	38
Subtotal	$ 1,035	$ 1,076	$ 1,119	$ 1,164	$ 1,211	$ 1,259
Reactor Operations						
Salary/Wages	$ 1,230	$ 1,279	$ 1,330	$ 1,384	$ 1,439	$ 1,496
Benefits	344	358	372	387	402	419
Supplies	140	146	151	157	164	170
Subtotal	$ 1,714	$ 1,783	$ 1,854	$ 1,928	$ 2,005	$ 2,085
Technical Support Services						
Salary/Wages	$ 1,241	$ 1,291	$ 1,342	$ 1,396	$ 1,452	$ 1,510
Benefits	347	361	375	390	406	422
Supplies	113	118	122	127	132	137
Subtotal	$ 1,701	$ 1,769	$ 1,840	$ 1,913	$ 1,990	$ 2,070
Product and Service Operations						
Salary/Wages	$ 1,265	$ 1,316	$ 1,368	$ 1,423	$ 1,480	$ 1,539
Benefits	354	368	383	398	414	431
Supplies	282	293	305	317	330	343
Subtotal	$ 1,901	$ 1,977	$ 2,056	$ 2,138	$ 2,224	$ 2,313
Research & Development						
Salary/Wages	$ 1,776	$ 1,645	$ 1,711	$ 1,780	$ 1,851	$ 1,925
Benefits	416	433	450	468	487	506
Supplies	297	309	321	334	347	361
Subtotal	$ 2,489	$ 2,387	$ 2,482	$ 2,582	$ 2,685	$ 2,792
Health Physics						
Salary/Wages	$ 443	$ 461	$ 479	$ 498	$ 518	$ 539
Benefits	124	129	134	139	145	151
Supplies	85	88	92	96	99	103
Subtotal	$ 652	$ 678	$ 705	$ 733	$ 763	$ 793
Overhead	$ 1,702	$ 1,415	$ 1,472	$ 1,531	$ 1,592	$ 1,655
Plant/Equipment	$ 1,060	$ 599	$ 623	$ 648	$ 674	$ 701
Incentive Compensation	$ -	$ -	$ -	$ -	$ -	$ -
Other	$ 290	$ 302	$ 314	$ 326	$ 339	$ 353
TOTAL EXPENDITURES	$12,544	$11,986	$12,465	$12,964	$13,482	$14,022
TOTAL REVENUES LESS TOTAL EXPENDITURES	$ 196	$ 204	$ 212	$ 220	$ 229	$ 238

Footnote: The University of Missouri fiscal year (FYXX) begins July 1 and ends June 30.

Table 15-2, "MURR 2006-2011 Budget" provides the projected budget for fiscal years 2006 through 2011. The table identifies both revenue and expenses.

The revenue category of this table, titled "General Operating (Campus Allocation)," represents an annual allocation of the State of Missouri funds for MU. This allocation has been steadily increased on an annual basis to cover the cost-of-living adjustments. MU is in sound financial condition as evidenced by its recent Financial Report (2005). The revenue category titled "Grant Funds" is a revenue source received from non-MU sources. Grant funding has been steady throughout the last several years at the MURR, and it is expected to continue. The reduction in grant funding between FY 06 and FY 07 reflects the end of a multi-year grant provided by DOE for the renewal of MURR facilities and equipment in preparation for relicensing. The "Product and Service Operations" category is primarily based on the sale of irradiation, processing and analytical services. Revenue is subject to market fluctuations, however many of the MURR's major customers have been clients for several years and the MURR continues to develop significant new customers each year. If service revenues are jeopardized in the short-term, the MURR could implement cost reducing measures, request additional funding from Campus Allocated Funding, or seek short-term loans from MU.

15.3 Financial Ability to Decommission the Facility

Title 10, Code of Federal Regulations 50.33 (k)(2) states that "on or before July 26, 1990, each holder of an operating license for a production or utilization facility in effect on July 27, 1990, shall submit information in the form of a report as described in §50.75 of this part, indicating how reasonable assurance will be provided that funds will be available to decommission the facility."

The Curators of the University of Missouri, a state government entity and holder of License R-103 for the MURR, complied with this regulation by submitting a statement of intent as the mechanism that provides reasonable assurance that funds will be available to decommission the MURR when necessary. This statement of intent was in conformance with 10 CFR 50.75(e)(2)(iv) and was submitted in a letter to the NRC, dated June 29, 1990 and assures that the University of Missouri will request appropriation of funds for decommissioning sufficiently in advance of decommissioning to prevent delay of required activities.

As was required by 10 CFR 50.75(e)(2)(iv), a cost estimate for decommissioning the MURR was enclosed with the June 29, 1990 letter. This estimate was developed using NUREG/CR-1756, Technology, Safety and Costs of Decommissioning Reference Research and Test Reactors. The original reported estimate was $9 million, and has been adjusted periodically over the life of the facility as required by 10 CFR 50.75(d). The MURR has used the inflation formula provided in 10 CFR 50.75(c)(2) to adjust the decommissioning cost estimate at five year intervals. The most recent cost estimate prepared in August 2005 is approximately $40 million in year-2005 dollars. A more detailed explanation of how this cost estimate was derived is included in Chapter 17, Decommissioning.

CHAPTER 16

OTHER LICENSE CONSIDERATIONS

TABLE OF CONTENTS

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 Yahr, G.T., "Fatigue Design Curves for 6061 T-6 Aluminum," ORNL, 1993.

16.5 Sargent and Lundy, LLC., "Condition Evaluation of Aluminum Reactor Pool Liner," July 2001.

16.6 Sargent and Lundy, LLC., "Containment Structure Condition Assessment," July 2001.

16.7 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 2001.

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 Missouri University 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

The 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 proposed licensing period.

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. The MURR has used over 700 fuel elements since 1971 with no failures, 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 ^{131}I level in the primary coolant (Ref. 16.1). This slight increase was easily detected by the online fission product monitor and subsequently validated by radio-chemical 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 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 replacement performed in January 2006, 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 (Ref.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 60-year 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 independent primary coolant low pressure scrams and each one 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 (0.689 MPa above atmosphere). The limiting safety system setting (LSSS) for primary coolant system pressure is 75 psia (0.517 MPa) 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 (1.034 MPa 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.

$$
\begin{aligned}
\text{max stress on outer PV} &= \text{internal pressure x radius/thickness} \\
&= 0.689\text{ MPa x } 5.963\text{ inches}/0.3125\text{ inches} \\
&= 13.1\text{ MPa } (1{,}900\text{ psi})
\end{aligned}
$$

$$
\begin{aligned}
\text{max stress on inner PV} &= \text{external pressure x OD/thickness} \\
&= 0.689\text{ MPa x } 5.06\text{ inches}/0.280\text{ inches} \\
&= 12.5\text{ MPa } (1{,}813\text{ psi})
\end{aligned}
$$

The original design calculations gave 8,500 psi (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 vessels' 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.

To date, the reactor pressure vessels have been in service approximately 105,500 MWD of operation. This has resulted in the following:

1. A peak fast fluence [>0.1 Mev] of 2.31×10^{27} n/m^2 to the inner pressure vessel;
2. A peak thermal fluence of 1.79×10^{27} n/m^2 to the inner pressure vessel;
3. A peak fast fluence of 1.82×10^{27} n/m^2 to the outer pressure vessel; and
4. A peak thermal fluence of 1.59×10^{27} n/m^2 to the outer pressure vessel.

A report on the effect of the radiation environment on material properties of aluminum 6061-T6 (Ref. 16.3) describes the variation of strength and elongation as a function of thermal neutron fluence. 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 through October 2026, the peak thermal fluence on the pressure vessels will be 2.90×10^{27} n/m^2, within the measured envelop for aluminum 6061-T6. Therefore the high neutron fluence does not put the pressure vessels at risk of failure due to the stress during the proposed licensing period.

Thermal neutron flux has generated approximately 3.5 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% swelling during the entire service life of the pressure vessels. Therefore swelling will not cause a significant increase in stress during the timeframe of the proposed license period.

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 psi) (Ref. 16.4). 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 psi (0.41 MPa) causing a stress of approximately 8 MPa (1,160 psi), 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 12,000 cycles over a 60-year operating period. Therefore, the accumulated fatigue stress does not limit the service life of the inner and outer pressure vessels.

16.1.3 In-Pool Components Receiving High Neutron Fluence

The 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 was discussed in detail in Section 16.1.2. The remaining four 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] (Ref. 16.5). 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 coolant 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 licensing 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] (Ref. 16.6). 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 annual 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.

6.1.6 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 effect 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 (Ref. 16.7). 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 license 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 license period.

16.2 Medical Use of Non-Power Reactors

The MURR has not been utilized for medical purposes and there are no immediate future plans for such use.

CHAPTER 17

DECOMMISSIONING

TABLE OF CONTENTS

LIST OF TABLES

REFERENCES

17.1 NUREG-0586, Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities, August 1988.

17.2 NUREG-1307, Report on Waste Burial Charges, Rev. 11, June 2005.

17.3 U.S. Department of Labor - Bureau of Labor Statistics, Employment Cost Index - Midwest Region [Series ID: ecu13302i].

17.4 U.S. Department of Labor - Bureau of Labor Statistics, Producer Price Index - Commodities [Series ID: wpu0543].

17.5 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 Missouri University'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 Missouri University Research Reactor (MURR). As required by 10 CFR 50.75(e)(2)(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. In the June 29, 1990 letter, MURR committed to using the adjustment factor provided in 10 CFR 50.75(c)(2) to adjust the decommissioning cost estimate at five-year intervals. The most recent cost adjustment that was performed in August 2005 estimated a total decommissioning cost of $39.95 million.

17.2 Decommissioning Alternatives

Three decommissioning alternatives are defined in NUREG-0586, Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities (Ref. 17.1):

- 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	$121,000.00[a]

[a]The 30-year total cost estimate is 3.6 million dollars.

TABLE 17-2
SUMMARY OF COST (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 (1989 dollars) and the fractional escalation of these three categories of cost over the time period of interest. That equation is:

Estimated Cost (2005):
= [1989 $ Cost]*[A L_x + B E_x + C B_x];
= estimated decommissioning costs in 2005 dollars;

where:

[1989 $ Cost]
= estimated decommissioning costs in 1989 dollars;

A = fraction of the [1989 $ Cost] attributable to labor, materials, and services;

B = fraction of the [1989 $ Cost] attributable to energy and transportation;

C = fraction of the [1989 $ Cost] attributable to waste burial;

L_x = labor, materials, and services cost adjustment, Jan. 1989 to Jan. 2005;

E_x = energy and waste transportation cost adjustment, Jan.1989 to Jan. 2005; and

B_x = LLW burial/disposition cost adjustment, Jan. 1989 to Jan. 2005.

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 (2005)
= [1989 $ 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. 11, June 2005 (Ref 17.2), and by the U.S. Department of Labor - Bureau of Labor Statistics (Refs. 17.3, 17.4, 17.5).

A. Labor Adjustment Factors
L_x is calculated for each region by multiplying the 4th Quarter 2004 value (Ref. 17.3) by the scaling factor and then dividing by the reference value. For the Midwest region:

L_x = $(177.9)_{\text{Base 1989}}$ (ecu13302i - Qtr 4, 2004),
* $(1.409)_{\text{Base 1981/Base 1989}}$ (column 4),
÷ $(125.0)_{\text{Base 1981}}$ (column 2),
= 2.005.

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 (Refs. 17.4, 17.5) divided by the corresponding indexes for January 1989.

P_x = 146.2 (wpu0543 - Dec. 2004),
÷ 112.0 (wpu0543 - Jan. 1989),
= 1.310.

F_x = 133.8 (wpu0573 - Dec. 2004),
÷ 54.9 (wpu0573 - Jan. 1989),
= 2.437.

Therefore:

$E_x(PWR)$ = (0.58*1.310) + (0.42*2.437),
= 1.783; and

$E_x(BWR)$ = (0.54*1.310) + (0.46*2.437),
= 1.828.

E_x for MURR is calculated as an average of $E_x(PWR)$ and $E_x(BWR)$, therefore:

$E_x(AVE)$ = (1.783 + 1.828)/2,
= 1.806.

C. Waste Burial Adjustment Factors
The adjustment factor for waste burial/disposition, Bx, is taken directly from data on the appropriate LLW burial location as given in Table 2.1 of Reference 17.2 for the year 2004.

B_x for MURR is calculated as an average of the PWR and BWR, Non-Atlantic Compact, "Direct Disposal with Vendors" cost for the South Carolina Site (Barnwell), therefore:

$$B_x(\text{AVE}) = (7.934 + 8.863)/2, \\ = 8.399.$$

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 17.1 indicates 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:	South Carolina (Non-Atlantic Compact)

Decommissioning Cost (2005 $)

= [1989 $ Cost]*[A L_x + B E_x + C B_x]
= (11.8 Million) [(0.65)*(2.005) + (0.13)*(1.806) + (0.22)*(8.399)]
= (11.8 Million) [1.303 + 0.235 + 1.848]
= (11.8 Million) [3.386]
= $39.95 Million

CHAPTER 18

HIGHLY-ENRICHED TO LOW-ENRICHED URANIUM CONVERSIONS

TABLE OF CONTENTS

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 1986; and supplemented by Letter Response to US NRC Request for Additional Information Supporting the Unique Purpose Exemption Request, February 1987.

18.0 HIGHLY-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 highly-enriched to low-enriched uranium fuel and the current status of the Missouri University Research Reactor (MURR) within this conversion process.

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 (Ref. 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 highly-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 Current Status for 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 the MURR to operate. Additionally, in 1986, the University of Missouri requested that a determination be made by the U.S. Nuclear Regulatory Commission (NRC) that the MURR has a *Unique Purpose*, as defined by 10 CFR 50.2, and is, therefore, exempt from the conversion from HEU to LEU fuel (Ref. 18.2).

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.

APPENDIX A

TECHNICAL SPECIFICATIONS

FOR

THE MISSOURI UNIVERSITY RESEARCH REACTOR

FACILITY OPERATING LICENSE R-103
DOCKET 50-86

Introduction

The Technical Specifications 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 Title 10, Chapter I of the Code of Federal Regulations, Part 50.36 (10 CFR 50.36).

This document is divided into the following six sections:

Section 1 - Definitions
Section 2 - Safety Limits (SL) and Limiting Safety System Settings (LSSS)
Section 3 - Limiting Conditions for Operation (LCO)
Section 4 - Surveillance Requirements
Section 5 - Design Features
Section 6 - Administrative Controls

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 the Safety Analysis Report (SAR), 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 Technical Specifications. Procedures or actions employed to meet the requirements of these Technical Specifications are not included in the Technical Specifications. Normal operation of the reactor within the limits of the Technical Specifications will not result in off-site radiation exposure in excess of Title 10, Chapter I of the Code of Federal Regulations, Part 20 (10 CFR 20) guidelines.

Specifications in Sections 2, 3, and 4 provide related information in the following format shown:

- **Applicability** - This indicates which components are involved;
- **Objective** - This indicates the purpose of the specification(s);
- **Specification(s)** - This provides specific data, conditions, or limitations that bound a system or operation. This is the most important statement in the Technical Specifications agreement; and
- **Bases** - This provides the background or reasoning for the choice of specification(s), or references a particular section of the SAR that does.

It is important to note that although the applicability, objective, and bases provide important information, only the "specification(s)" statement is governing. Section 5, Design Features, and 6, Administrative Controls, simply state the applicable specification(s).

TABLE OF CONTENTS

1.0 DEFINITIONS

1.1 **Abnormal Occurrences** - An abnormal occurrence is any of the following which occurs during reactor operation:

a. Operation with actual safety system settings for required systems less conservative than specified in Section 2.2, Limiting Safety System Settings;

b. Operation in violation of Limiting Conditions for Operation established in Section 3.0;

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

d. An unanticipated or uncontrolled change in reactivity in excess of 0.006 Δk. Reactor trips resulting from a known cause are excluded;

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

f. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition involving operation of the reactor.

1.0 **DEFINITIONS** - Continued

1.2 **Calibration or Testing Interval** - A calibration or testing interval is that period of time between normal checks for accuracy or operability of a system or component. To allow for some margin of time for proper scheduling and yet reasonably assure reliability, the calibration or testing interval shall be interpreted as follows:

Interval	Maximum Period Between Checks
Weekly:	9 days
Monthly:	6 weeks
Quarterly:	4 months
Semi-annually or greater:	Interval plus 2 months.

1.3 **Center Test Hole** - The center test hole is that volume in the flux trap occupied by the removable experiment sample canister.

1.4 **Control Blade (Rod)** - A control blade (rod) is either a shim blade (rod) or the regulating blade (rod). The words blade and rod can be used interchangeably.

1.5 **Excess Reactivity** - Excess reactivity is that amount of reactivity that would exist if all of the shim blades were moved to the fully withdrawn position from the point where the reactor is exactly critical (K_{eff}= 1).

1.6 **Exclusion Area** - The exclusion area is that area bounded by the outer perimeter of the reactor laboratory building.

1.7 **Experiment** - An experiment, as used herein, is either of the following:

a. Any device or material which is exposed to significant radiation from the reactor and is not a normal part of the reactor.

b. Any operation designed to measure or monitor reactor characteristics or parameters.

1.0 **DEFINITIONS** - Continued

1.8 **Flux Trap** - The flux trap is that portion of the reactor through the center of the core bounded by the 4.5-inch inside diameter tube and 15 inches above and below the reactor core horizontal center line.

1.9 **Instrument Channel** - An instrument channel is an arrangement of sensors, components, and modules as required to provide a single trip or other output signal relating to a reactor or system operating parameter.

1.10 **Instrument Channel Test** - An instrument channel test is the introduction of a simulated input signal to an instrument channel and the observation of proper channel response. When applicable, the test shall include verification of proper safety trip operation.

1.11 **Irradiated Fuel** - Irradiated fuel is any fuel element which has been used to an integrated power of greater than 1 megawatt-day.

1.12 **Limiting Safety System Settings** - Limiting Safety System Settings (LSSS) are settings for automatic protection devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting shall be so chosen that automatic protective action will correct the most severe abnormal situation anticipated before a safety limit is exceeded.

1.13 **Movable Experiment** - A movable experiment is one which is designed with the intent that it may be moved into and out of the reactor while the reactor is operating.

1.14 **Operable** - Operable means a system or component is capable of performing its intended function in a normal manner.

1.0 **DEFINITIONS** - Continued

1.15 **Operational Modes** - The reactor may be operated in any of three operating modes, depending upon the configuration of the reactor coolant systems and the protective system set points.

a. Operational Mode I - Reactor can be operated safely at a thermal power level of ten megawatts or less.

b. Operational Mode II - Reactor can be operated safely at a thermal power level of five megawatts or less.

c. Operational Mode III - Reactor can be operated safely at a thermal power level of fifty kilowatts or less.

1.16 **Reactor Containment Building** - The reactor containment building is a reinforced concrete structure within the facility site which houses the reactor core, pool, and irradiated fuel storage facilities.

1.17 **Reactor Containment Integrity** - For reactor containment integrity to exist, the following conditions must be satisfied:

a. The truck entry door is closed and sealed;

b. The utility entry seal trench is filled with water to a depth required to maintain a minimum water seal of 4.25 feet;

c. All of the reactor containment building ventilation system's automatically-closing doors and automatically-closing valves are operable or placed in the closed position;

d. The reactor mechanical equipment room ventilation exhaust system, including the particulate and halogen filters, is operable;

e. The personnel airlock is operable (one door shut and sealed); and

f. The most recent reactor containment building leakage rate test was satisfactory.

1.18 **Reactor Core** - The reactor core shall be considered to be that volume inside the reactor pressure vessels occupied by eight or less fuel elements.

1.0 **DEFINITIONS** - Continued

1.19 **Reactor Operator** - A reactor operator is an individual who is certified to manipulate the controls of a reactor.

1.20 **Reactor in Operation** - The reactor shall be considered in operation unless it is either shutdown or secured.

1.21 **Reactor Safety System** - The reactor safety system is that combination of sensing devices, electronic circuits and equipment, signal conditioning equipment, and electro-mechanical devices that serves to either effect a reactor scram, or activate the engineered safety features.

1.22 **Reactor Scram** - A reactor scram is the insertion of all four shim rods by gravitational force as a result of removing the holding current from the shim rod drive mechanism electromagnets.

1.23 **Reactor Secured** - The reactor shall be considered secured when:

(1) There is insufficient fuel in the reactor core to attain criticality with all four shim rods removed,

OR

(2) Whenever all of the following conditions are met:
  a. All four shim rods are fully inserted;
  b. One of the two following conditions exits:
    1. The Master Control Switch is in the "OFF" position with the key locked in the key box or in custody of a licensed operator,

    OR

    2. The dummy load test connectors are installed on the shim rods and a licensed operator is present in the reactor control room;
  c. No work is in progress involving the transfer of fuel in or out of the reactor core;
  d. No work is in progress involving the shim rods or shim rod drive mechanisms with the exception of installing or removing the dummy load test connectors; and
  e. The reactor pressure vessel cover is secured in position and no work is in progress on the reactor core assembly support structure.

1.0 **DEFINITIONS** - Continued

1.24 **Reactor Shutdown** - The reactor shall be considered shutdown when all four of the shim rods are fully inserted and power is unavailable to the shim rod drive mechanism electromagnets.

1.25 **Regulating Blade (Rod)** - The regulating blade (rod) is a low worth control blade (rod) used for very fine adjustments in the neutron density in order to maintain the reactor at the desired power level. The regulating blade (rod) may be controlled by the operator with a manual switch or push button, or by an automatic controller.

1.26 **Removable Experiment** - A removable experiment is any experiment which can reasonably be anticipated to be moved during the life of the reactor.

1.27 **Safety Limits** - Safety Limits (SL) are limits placed upon important process variables which are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity.

1.28 **Secured Experiment** - 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 to.

1.29 **Senior Reactor Operator** - A senior reactor operator is an individual who is certified to direct the activities of reactor operators and manipulate the controls of a reactor.

1.30 **Shim Blade (Rod)** - A shim blade (rod) is a high worth control blade (rod) used for coarse adjustments in the neutron density and to compensate for routine reactivity losses. The shim blade (rod) is magnetically coupled to its drive mechanism allowing it to perform its safety function when the electromagnet is de-energized.

1.31 **Shutdown Margin** - Shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition and with the most reactive shim blade and the regulating blade in the fully withdrawn positions, and that the reactor will remain subcritical without further operator action.

1.32 **True Value** - The true value is the actual value of a parameter.

1.0 **DEFINITIONS** - Continued

1.33 **Unsecured Experiment** - An unsecured experiment is any experiment which is not secured as defined by 1.28, or the moving parts of secured experiments when they are in motion.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

Applicability:
This specification applies to the interrelated variables associated with reactor core thermal and hydraulic performance. These measurable operating or process variables include reactor power level, core flow rate, reactor inlet water temperature, and pressurizer pressure.

Objective:
The objective of this specification is to define a four-dimensional safety limit envelope such that operation within this envelope will assure that the integrity of the fuel element cladding is maintained.

Specification:
Reactor power level, core flow rate, reactor inlet water temperature, and pressurizer pressure shall not exceed the following limits during reactor operation:

a. Mode I and II Operation (Core Flow Rate ≥ 400 gpm)
The combination of the true values of reactor power level, reactor core flow rate, and reactor inlet water temperature shall not exceed the limits plotted on Figures 2.0, 2.1, and 2.2. The limits are considered exceeded if, for core flow rates greater than or equal to 400 gpm, the point defined by reactor power level and core flow rate is at any time above the curve corresponding to the true values of reactor inlet water temperature and pressurizer pressure. To define values of the safety limits for temperatures and/or pressures not shown in Figures 2.0, 2.1, and 2.2, interpolation or extrapolation of the data on the curves shall be used. For pressurizer pressures greater than 85 psia, the 85 psia curves (Figure 2.2) shall be used and no pressure extrapolation shall be permitted.

b. Mode I and II Operation (Core Flow Rate < 400 gpm)
Steady-state power operation in Modes I and II is not authorized for a core flow rate less than 400 gpm. Reactor operation with a core flow rate below 400 gpm will occur only after a normal reactor shutdown when the primary coolant circulation pumps are secured or following a loss of flow transient. Under the above conditions, the maximum fuel element cladding temperature shall not approach a temperature that would challenge the integrity of the fuel element cladding.

c. Mode III Operation
Reactor power is limited to a maximum of 150 kilowatts.

2.1 **Safety Limits - Continued**

Bases:

a. A complete safety limit analysis for the MURR is presented in Section 4.6.3 of the SAR. A family of curves is presented which relate reactor inlet water temperature and core flow rate to the reactor power level corresponding to a departure from nucleate boiling ratio (DNBR) of 1.2. This is based on the burnout heat flux data experimentally verified for Advanced Test Reactor (ATR) type fuel elements. Curves are presented for pressurizer pressures of 60, 75, and 85 psia. The safety limits were chosen from the results of this analysis for Mode I and II operation, i.e., forced convection operation with greater than 400 gpm flow.

b. Steady-state reactor operation is prohibited for core flow rates less than 400 gpm by the low flow scram settings in the reactor safety system. The region below 400 gpm will only be entered following a reactor shutdown when the primary coolant circulation pumps are secured or during a loss of flow transient where the reactor scrams, the flow coasts to zero, reverses, and natural convective cooling is established through the decay heat removal system. The analysis of a loss of flow transient presented in Section 13.2.4 of the SAR, from the ultraconservative conditions of 11 MW of power, a core flow rate of 3,800 gpm, and a reactor inlet water temperature of 155 °F, indicated a maximum fuel plate centerline temperature of 280.3 °F and a maximum coolant channel temperature of 237.5 °F, which is well below the saturation temperature of 277 °F.

c. Analysis of natural convection cooling of the core (Mode III Operation) is presented in Section 4.6.1 of the Safety Analysis Report.

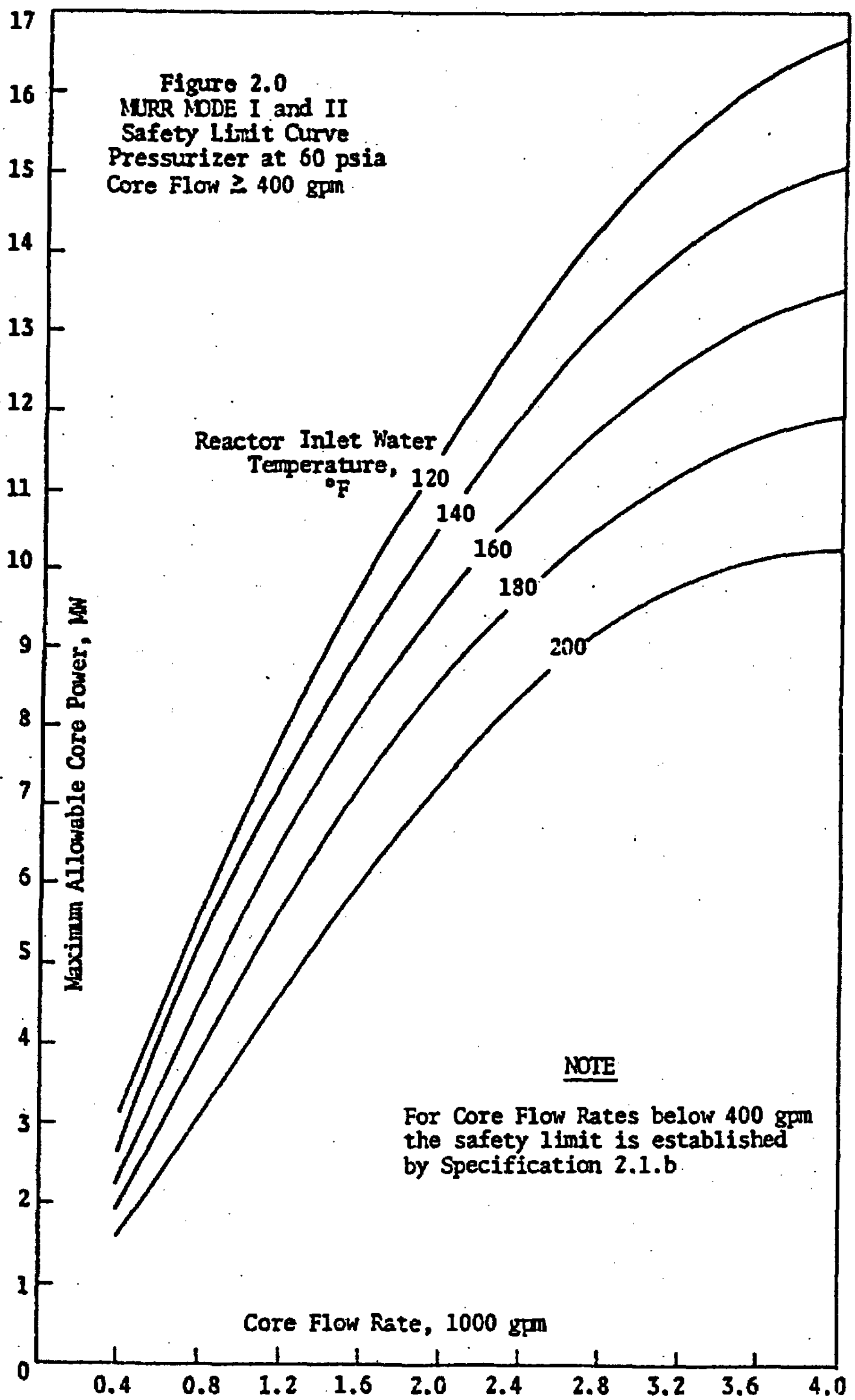
Figure 2.0
MURR MODE I and II
Safety Limit Curve
Pressurizer at 60 psia
Core Flow ≥ 400 gpm
Reactor Inlet Water Temperature, °F
120
140
160
180
200
Maximum Allowable Core Power, MW
Core Flow Rate, 1000 gpm
NOTE
For Core Flow Rates below 400 gpm the safety limit is established by Specification 2.1.b
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
0.4
0.8
1.2
1.6
2.0
2.4
2.8
3.2
3.6
4.0

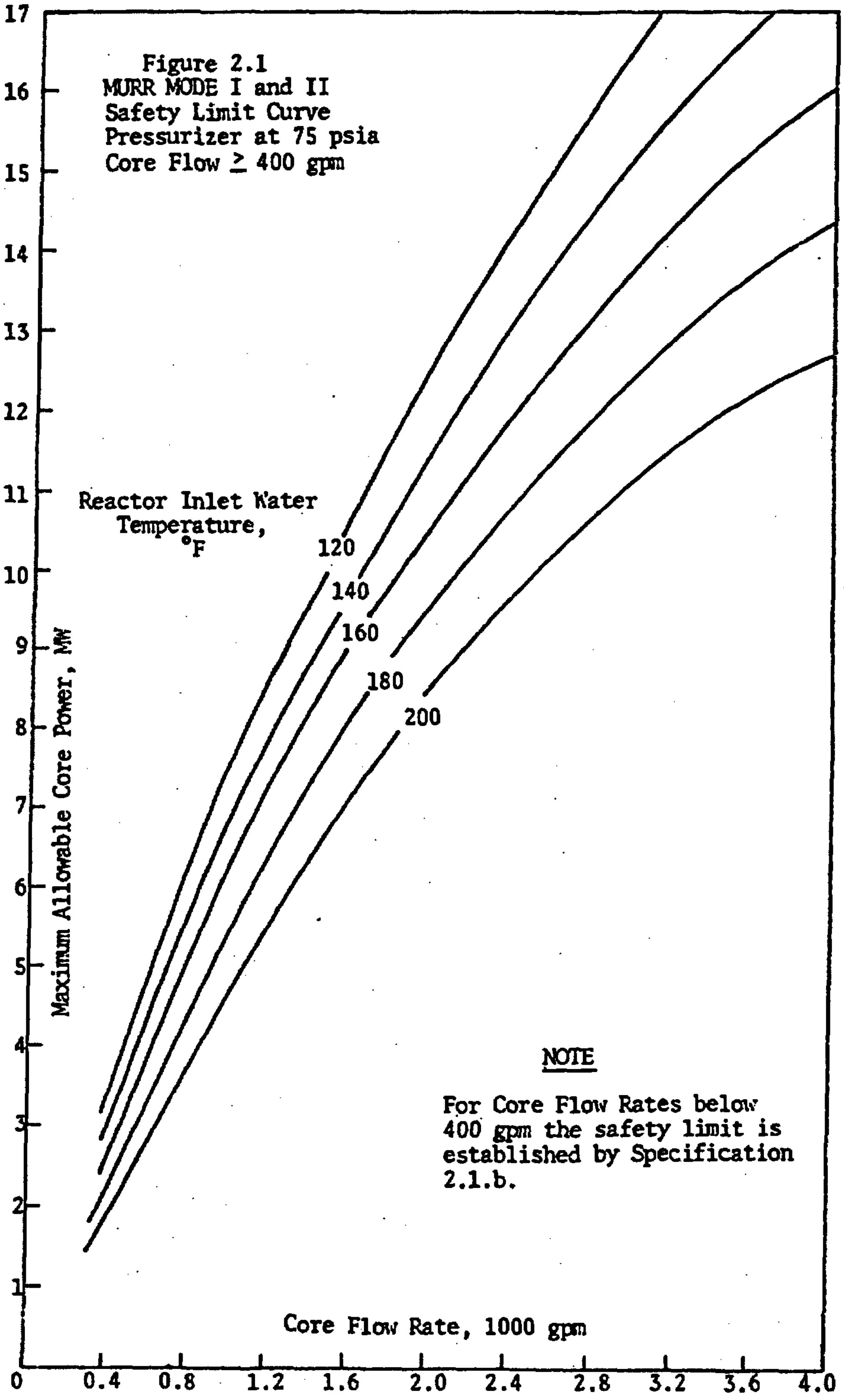
Figure 2.1
MURR MODE I and II
Safety Limit Curve
Pressurizer at 75 psia
Core Flow ≥ 400 gpm
Reactor Inlet Water Temperature, °F
120
140
160
180
200
Maximum Allowable Core Power, MW
Core Flow Rate, 1000 gpm
0
0.4
0.8
1.2
1.6
2.0
2.4
2.8
3.2
3.6
4.0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
NOTE
For Core Flow Rates below 400 gpm the safety limit is established by Specification 2.1.b.

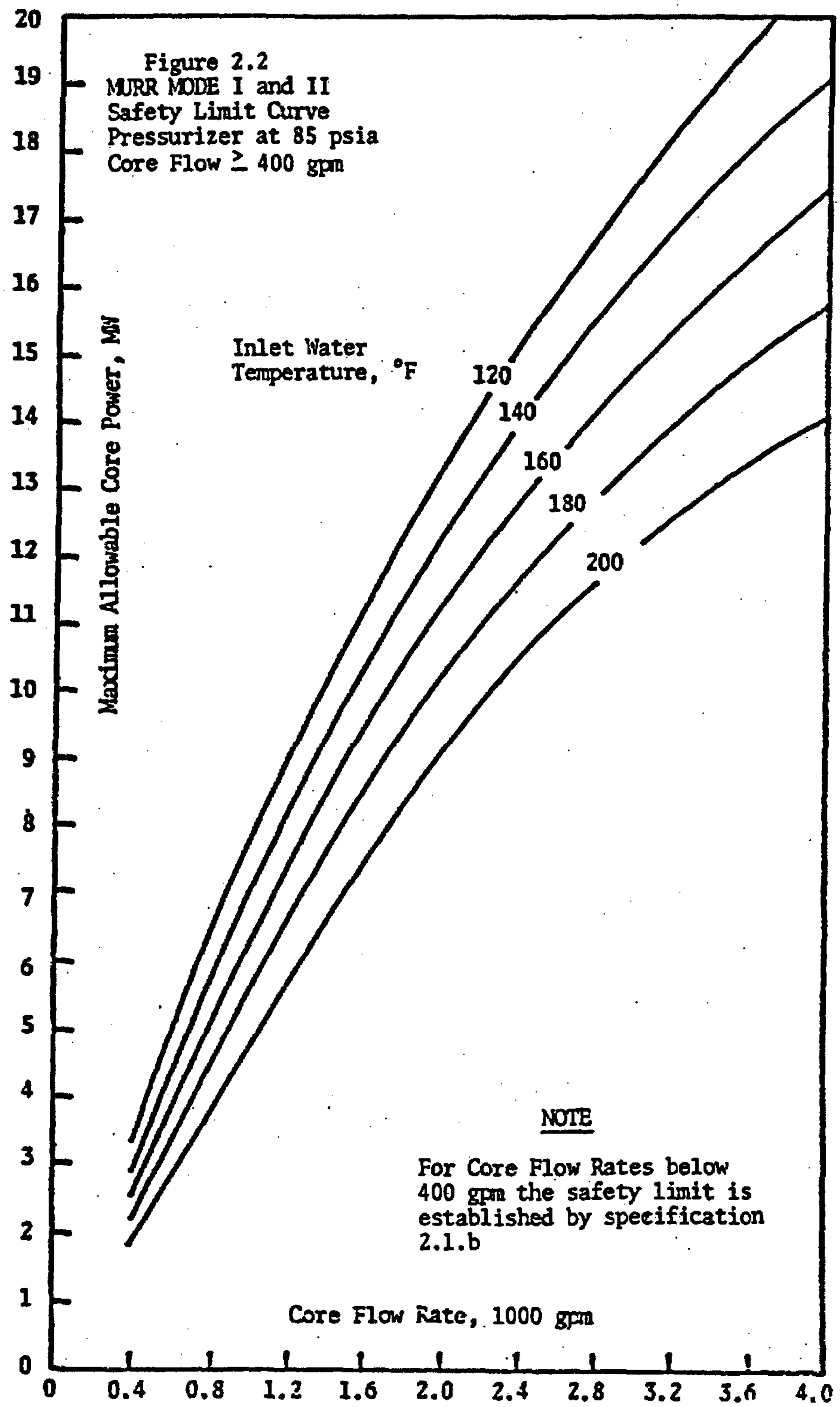
Figure 2.2
MURR MODE I and II
Safety Limit Curve
Pressurizer at 85 psia
Core Flow ≥ 400 gpm
Inlet Water
Temperature, °F
120
140
160
180
200
Maximum Allowable Core Power, MW
Core Flow Rate, 1000 gpm
NOTE
For Core Flow Rates below
400 gpm the safety limit is
established by specification
2.1.b
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
0
0.4
0.8
1.2
1.6
2.0
2.4
2.8
3.2
3.6
4.0

2.2 Limiting Safety System Settings

Applicability:
This specification applies to the set points for the reactor safety channels monitoring reactor power level, primary coolant flow, reactor coolant inlet water temperature, and pressurizer pressure.

Objective:
The objective of this specification is to assure that automatic protective action is initiated to prevent a safety limit from being exceeded.

Specification:

a. Mode I Operation

Reactor Power Level (10 MW)	125% of full power (Maximum)
Primary Coolant Flow	1,625 gpm either loop (Minimum)
Reactor Inlet Water Temperature	155 °F (Maximum)
Pressurizer Pressure	75 Psia (Minimum)

b. Mode II Operation

Reactor Power Level (5 MW)	125% of full power (Maximum)
Primary Coolant Flow	1,625 gpm (Minimum)
Reactor Inlet Water Temperature	155 °F (Maximum)
Pressurizer Pressure	75 Psia (Minimum)

c. Mode III Operation

Reactor Power Level (50 kW)	125% of full power (Maximum)

Bases:

a. - b. The limiting safety system settings (LSSS) are set points which, if exceeded, will cause the reactor safety system to initiate a reactor scram. The LSSS were chosen such that the true value of any of the four safety-related variables, i.e., reactor power level, core flow rate, reactor inlet water temperature, and pressurizer pressure will not exceed a safety limit under the most severe anticipated transient. Section 4.6.4 of

2.2 Limiting Safety System Settings - Continued

the SAR presents analyses to show that the LSSS for Mode I and II operation meet this criterion.

c. For Mode III operation, the high power scram set point 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.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactivity Limitations

Applicability:
This specification applies to the reactivity of the reactor core and the reactivity worths of the control blades and experiments.

Objective:
The objective of this specification is to assure that the reactor can be controlled and shutdown at all times and that the safety limits will not be exceeded.

Specification:

a. The average reactor core temperature coefficient of reactivity shall be more negative than -6.0×10^{-5} Δk/k/°F.

b. The average reactor core void coefficient of reactivity shall be more negative than -2.0×10^{-3} Δk/k/% void.

c. The regulating blade total reactivity worth shall be a maximum of 6.0×10^{-3} Δk/k and the maximum rate of reactivity insertion shall be 2.5×10^{-4} Δk/k/sec.

d. The maximum rate of reactivity insertion for the four shim blades operating simultaneously shall not exceed 3.0×10^{-4} Δk/k/sec.

e. The reactor shall be subcritical by a margin of at least 0.02 Δk/k with the most reactive shim blade and the regulating blade in the fully withdrawn positions.

f. The reactor core excess reactivity above cold, clean, critical shall not exceed 0.098 Δk/k. Core excess reactivity shall be verified after any changes are made in the reactor core.

g. The reactivity worth of each secured removable experiment shall be limited to 0.006 Δk/k.

h. The absolute value of the reactivity worth of all experiments in the center test hole shall not exceed 0.006 Δk/k.

i. Each movable experiment or the movable parts of any individual experiment shall have a maximum absolute reactivity worth of 0.001 Δk/k.

3.1 Reactivity Limitations - Continued

j. The magnitude of the reactivity worth of each unsecured experiment shall not exceed 0.0025 $\Delta k/k$.

k. The sum of the magnitudes of the reactivity worths of all unsecured experiments which are in the reactor shall not exceed 0.006 $\Delta k/k$.

Bases:

a. Specification 3.1.a limits one of the parameters which assures that core damage will not occur following any credible step reactivity insertion as analyzed in Section 13.2.2 of the SAR.

b. The average core void coefficient of reactivity also limits the step reactivity insertion accident as analyzed in Section 13.2.2 of the SAR.

c. The regulating blade total reactivity worth is limited by Specification 3.1.c such that any condition resulting in the step insertion of the maximum worth of 6×10^{-3} $\Delta k/k$ will not result in fuel plate damage. The limit on the rate of reactivity addition provides for reasonable response from operator control.

d. Specification 3.1.d assures that power increases caused by control rod motion will be safely terminated by the reactor safety system. The continuous control rod withdrawal accident is analyzed in Section 13.2.2 of the SAR.

e. Specification 3.1.e assures that a shutdown margin, as defined by Definition 1.31, is maintained.

f. Specification 3.1.f provides additional assurance that Specification 3.1.e is satisfied.

g. Specification 3.1.g 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 is analyzed in Section 13.2.2 of the SAR.

h. The reactivity worth of experiments in the center test hole is limited by Specification 3.1.h such that the introduction of the maximum reactivity worth of all experiments would not result in damage to the fuel plates as analyzed in Section 13.2.2 of the SAR.

3.1 **Reactivity Limitations** - Continued

i. Specification 3.1.i 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 as analyzed in Section 13.2.2 of the SAR.

j. Specification 3.1.j prevents the installation of an unsecured experiment which could introduce, as a positive step change, sufficient reactivity to place the reactor on a transient that would cause a violation of a safety limit as analyzed in Section 13.2.2 of the SAR..

k. Specification 3.1.k assures that the reactivity worth of all unsecured experiments shall not exceed the maximum value authorized for a single secured removable experiment.

3.2 Control Blades

Applicability:
This specification applies to the operation of the reactor control blades.

Objective:
The objective of this specification is 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.

Specification:

a. All control blades, including the regulating blade, shall be operable during reactor operation.

b. Above 100 kilowatts, the reactor shall be operated so that the maximum distance between the highest and lowest shim blade shall not exceed one inch.

c. The shim blades shall be capable of insertion to the 20% withdrawn position in less than 0.7 seconds.

Bases:

a. Specification 3.2.a ensures that the normal method of reactivity control is used during reactor operation.

b. Specification 3.2.b provides a restriction on the maximum neutron flux tilting that can occur in the core to ensure the validity of the power peaking factors described in Section 4.5 of the SAR.

c. Specification 3.2.c assures prompt shutdown of the reactor in the event a scram signal is received as analyzed in Section 13.2.2 of the SAR. The 20% level is defined as 20% of the shim blade full travel as measured from the fully inserted position. Below the 20% level, the fall of the shim blade is cushioned by a dashpot assembly. Approximately 91% of the shim blade total worth is inserted at the 20% level.

3.3 Reactor Safety System

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

Objective:
The objective of this specification is to specify the minimum number of reactor safety system instrument channels that must be operable for safe reactor operation.

Specification:

a. The reactor safety system and the number (N) of associated instrument channels necessary to provide the following scrams shall be operable whenever the reactor is in operation. Each of the safety system functions shall have 1/N logic where N is the number of instrument channels required for the corresponding mode of operation.

Safety System or Measuring Channel	Number Required (N) Mode I	Mode II	Mode III	Trip Set Point
High Power Level	3	3	3	125% of full power (Max)
Reactor Period	2	2	2	8 Seconds (Min)
Primary Coolant Flow	4	2	2(1)	1,625 gpm(2) (Min)
Differential Pressure Across the Core	1	0	0	3,200 gpm(3) (Min)
Differential Pressure Across the Core	0	1	1(1)	1,600 gpm(3) (Min)
Primary Coolant Low Pressure	4	4	4(1)	75 psia(5) (Min)
Reactor Inlet Water Temperature	2	1	1(1)	155 °F (Max)
Reactor Outlet Water Temperature	1	1	1(1)	175 °F (Max)
Pool Coolant Flow	2	2	0	850 gpm(4) (Min)

3.3 Reactor Safety System - Continued

Safety System or Measuring Channel	Number Required (N) Mode I	Mode II	Mode III	Trip Set Point
Differential Pressure Across the Reflector	1	0	0	2.52 psi (Min) 8.00 psi (Max)
Differential Pressure Across the Reflector	0	1	0	0.63 psi (Min) 2.00 psi (Max)
Pressurizer High Pressure	1	1	1[1]	95 psia (Max)
Pressurizer Low Water Level	1	1	1[1]	16 inches below centerline (Min)
Pool Low Water Level	0	0	1	23 feet (Min)
Primary Coolant Isolation Valves 507A/B Off Open Position	1	1	1[1]	Either valve off open position
Pool Coolant Isolation Valve 509 Off Open Position	1	1	0	Valve 509 off open position
Power Level Interlock	1	1	1	Scram as a result of incorrect selection of operating mode
Facility Evacuation	1	1	1	Scram as a result of actuating the facility evacuation system
Reactor Isolation	1	1	1	Scram as a result of actuating the reactor isolation system
Manual Scram	1	1	1	Push button on Control Console

3.3 **Reactor Safety System - Continued**

(1) Not required below 50-kW operation if the natural convection flange and pressure vessel cover are removed or in operation with the reactor subcritical by a margin of at least 0.015 $\Delta k/k$.
(2) Flow orifice or heat exchanger ΔP (psi) in each operating heat exchanger leg corresponding to the flow value in the table.
(3) Core ΔP (psi) corresponding to the core flow value in the table.
(4) Flow orifice ΔP (psi) corresponding to the flow value in the table.
(5) Trip pressure is that which corresponds to the pressurizer pressure indicated in the table with normal primary coolant flow.

Bases:

a. The specifications on high power level, primary coolant flow, primary coolant pressure, and reactor inlet water temperature provide for the limiting safety system settings outlined in the Technical Specifications 2.2.a, 2.2.b, and 2.2.c. In Mode I and II operation, the core differential temperature is approximately 17 °F and, therefore, the reactor outlet water temperature scram set point at 175 °F provides a backup to the high reactor inlet water temperature scram. The core differential pressure scram provides a backup to the primary coolant low flow scrams.

The reactor period scram assures protection of the fuel elements from a continuous control blade withdrawal accident as analyzed in. Section 13.2.2 of the SAR.

The pool coolant low flow scram assures the adequate cooling of the reactor pool, reflectors, control rods, and the flux trap. With the reflector plenum natural convection valve V547 in the open position and pool coolant flow rate at 425 gpm, the total flow through the reflectors, control rods, and the flux trap will be 350 gpm (Ref. Section 5.3.5). The reflector high and low differential pressure scram provides a backup to the low pool coolant flow scram.

The pressurizer high pressure scram provides assurance that the reactor will be shut down during a high pressure transient before the relief valve set point or the pressure limit of the primary coolant system is reached as analyzed in Section 13.2.9.4 of the SAR.

The pressurizer low level scram provides assurance that the reactor will be shut down on a loss of coolant accident before the pressurizer level decreases sufficiently to introduce nitrogen gas into the primary coolant system.

3.3 Reactor Safety System - Continued

The pool water low level scram assures that the radiation level above the reactor pool from direct core radiation remains below 2.5 mrem/h (Ref. Section 11.1.5.1 of the SAR).

The reactor scrams caused by the primary and pool coolant isolation valves (507A/B and 509) leaving their full open position provide the first line of protection for a loss of flow accident (in their respective system) initiated by an inadvertent closure of the isolation valve(s).

The power level interlock (PLI) scram provides assurance that the reactor cannot be operated with a power level greater than that authorized for the mode of operation selected on the Power Level Switch. The PLI scram also provides the interlocks to assure that the reactor cannot be operated in Mode I with a primary or pool coolant low flow scram bypassed.

The facility evacuation and reactor isolation scrams provide assurance that the reactor is shut down for any condition which initiates or leads to the initiation of a facility evacuation or an isolation of the reactor containment building.

The manual scram provides assurance that the reactor can be shut down by the operator if an automatic function fails to initiate a reactor scram or if the operator detects an impending unsafe condition prior to the initiation of an automatic scram.

3.4 Reactor Instrumentation

Applicability:
This specification applies to the instruments that provide information which must be available to the operator during reactor operation.

Objective:
The objective of this specification is to ensure that sufficient reliable information is presented to the operator to assure safe operation of the reactor.

Specification:

a. The reactor shall not be operated unless the following instrument channels are operable:

Channel	Minimum Numbers Operable Mode I	Mode II	Mode III
Source Range Nuclear Instrument Channel	1[1]	1[1]	1[1]
Reactor Pool Temperature	1	1	1
Reactor Bridge Radiation Monitor	1[2]	1[2]	1[2]
Reactor Containment Building Exhaust Plenum Radiation Monitor	1	1	1
Off-Gas Radiation Monitor	1[3]	1[3]	1[3]

(1) Required for reactor startup only.
(2) The trip setting may be temporarily set upscale during periods of maintenance and sample handling. During these periods, the radiation monitor indication will be closely observed.
(3) The off-gas radiation monitor may be placed out of service for up to 2 hours for calibration and maintenance. During this out-of-service time, no experimental or maintenance activites will be conducted which could likely result in the release of unknown quantities of airborne radioactivity.

3.4 **Reactor Instrumentation - Continued**

b. Sufficient instrumentation shall be provided to assure that the following limits are not exceeded during steady-state operation:

Parameter	Limit
Primary Coolant System Pressure	110 psig (Max)
Anti-Siphon System Pressure	27 psig[(1)] (Min)
Reactor Pool Temperature	120 °F (Max)

[(1)] Not required for Mode III operation.

c. The reactor shall not be operated unless the following rod run-in functions are operable. Each of the rod run-in functions shall have 1/N logic where N is the number of instrument channels required for the corresponding mode of operation.

Rod Run-In Function	Number Required (N) Mode I	Mode II	Mode III	Trip Set Point
High Power Level	3	3	3	115% of full power (Max)
Reactor Period	2	2	2	10 Seconds (Min)
Pool Low Water Level	1	1	0	27 feet (Min)
Vent Tank Low Level	1	1	0	1 foot below centerline (Min)
Rod Not-In-Contact With Magnet	4	4	4	Magnet disengaged from any rod
Anti-Siphon System High Level	1	1	1[(1)]	6 inches above valves (Max)
Truck Entry	1	1	1	Loss of entry door seal pressure

3.4 **Reactor Instrumentation - Continued**

Rod Run-In Function	Number Required (N) Mode I	Mode II	Mode III	Trip Set Point
Regulating Blade Position	2	2[2]	2[2]	< 10% withdrawn and bottomed
Manual Rod Run-In	1	1	1	Push button on Control Console

[1] Not required below 50-kW operation if the natural convection flange and reactor pressure vessel cover are removed or in operation with the reactor subcritical by a margin of at least 0.015 Δk.
[2] Not required during calibration measurements of the regulating blade.

d. A minimum of one decade of overlap shall exist between adjacent ranges of nuclear instrument channels.

e. The reactor shall not be started up unless:

(1) The Source Range Channel is indicating a neutron count rate of at least 1 count per second and the Wide Range Monitor is indicating a power level greater than 1 watt,

OR

(2) The Source Range Channel is indicating a neutron count rate of at least 2 counts per second and is verified just prior to startup by a neutron test source or movement on the Source Range meter demonstrating that the channel is responding to neutrons.

Bases:

a. The Source Range Nuclear Instrument Channel provides a neutron monitor that is very sensitive to neutrons and thus provides improved indication of the low neutron flux levels present during a startup.

The reactor pool temperature instrument is required to ensure that pool temperature does not increase to a level which would jeopardize the ability to cool in-pool components.

3.4 **Reactor Instrumentation - Continued**

The radiation monitors provide information of an impending or existing danger from radiation so that corrective action can be initiated to prevent the spread of radioactivity to the surroundings and so that there will be sufficient time to evacuate the facility should it be necessary to do so.

b. The maximum primary coolant pressure of 110 psig assures that the system design pressure of 125 psig is not exceeded.

Maintaining the minimum anti-siphon system pressure ensures that the system will adequately perform its intended function (Ref. Section 6.3 of the SAR).

The reactor pool temperature limit provides an operating limit to assure the adequate cooling of pool components during all modes of operation.

c. The specifications on high power level and short reactor period are provided to introduce shim blade insertion on a reactor transient before the reactor safety system trip is actuated.

The low pool level rod run-in provides assurance that the radiation level from direct core radiation above the pool will not exceed 2.5 m/h (Ref. Section 11.1.5.1).

The vent tank low level rod run-in prevents reactor operation with a vent tank level which could result in the introduction of air into the primary coolant system (Ref. Section 9.13 of the SAR).

The anti-siphon system high level rod run-in provides assurance that the introduction of air to the invert loop is sufficiently rapid to prevent a siphoning action following a rupture of the primary coolant piping (Ref. Section 6.3 of the SAR).

The rod not-in-contact with magnet rod run-in assures the reactor cannot be operated in violation of Specification 3.2.b due to a dropped rod.

The specification on the truck entry door prohibits reactor operation without the door's contribution to containment integrity as required by Specification 1.17.a.

The regulating blade rod run-ins ensure termination of a transient which, in automatic control, is causing a rapid insertion of the regulating blade.

3.4 **Reactor Instrumentation** - Continued

d. Specification 3.4.d ensures that, during a startup, the reactor power level is continuously monitored over the entire range.

e. Specification 3.4.e provides for adequate neutron flux level monitoring to ensure that subcritical multiplication and criticality can be observed during a startup.

3.5 Reactor Containment Building

Applicability:
This specification applies to the reactor containment building.

Objective:
The objective of this specification is to assure 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.

Specification:

a. Containment integrity shall be maintained at all times except when:

   (1) The reactor is secured,

   AND

   (2) Irradiated fuel with a decay time of less than sixty (60) days is not being handled.

b. While containment integrity is required, the reactor containment building shall be automatically isolated if the activity in the ventilation exhaust plenum or at the reactor bridge indicates an increase of 10 times above previously established levels at the same operating condition. Exception; The containment isolation set point may temporarily be increased to avoid an inadvertent scram and isolation during controlled evolutions such as experiment transfers or minor maintenance in the reactor pool area. The pool area shall be continuously monitored, and, if necessary, a manual containment isolation actuated, until the automatic set point is reset to its normal value.

Bases:

a. Specification 3.5.a assures that the reactor containment building can be isolated at all times except when plant conditions are such that the probability of a release of radioactivity is negligible.

b. Radiation monitors located at the reactor bridge and in the containment building ventilation exhaust plenum supply input signals to meters located in the reactor control room. A containment isolation will occur when radiation levels in these areas exceed a predetermined value. During operations such as the removal of experiments or equipment from the pool, the radiation level at the level of the reactor bridge or in the exhaust plenum can increase significantly for short periods. To prevent inadvertent containment isolations, it may be necessary to raise the set point on the reactor bridge or exhaust plenum monitor. During periods in which the set point is

3.5 Reactor Containment Building - Continued

raised to more than one decade above the normal reading, the radiation level in the area of the monitor will be continuously monitored. Thus, should the radiation level increase from unknown causes or from material which could be released to the unrestricted environment, the containment building can be quickly isolated by manually actuating the isolation system.

3.6 Experiments

Applicability:
This specification applies to all experiments which directly utilize neutrons or other radiation produced by the reactor. Radioactive sources shall meet the requirements for experiments.

Objective:
The objective of this specification 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.

Specification:

a. Each fueled experiment shall be limited such that the total inventory of iodine-131 through iodine-135 in the experiment is not greater than 150 curies and the maximum strontium-90 inventory is no greater than 300 millicuries.

b. No experiments shall be placed in the reactor pressure vessel or water annulus surrounding the center test hole other than for reactor calibration.

c. Where the possibility exists that the failure of an experiment could release radioactive gases or aerosols into the containment building atmosphere, the experiment shall be limited to that amount of material such that the airborne concentration of radioactivity when averaged over a year will not exceed the limits of 10 CFR 20, Appendix B, Table I. Exception: Fueled experiments (See Specification 3.6.a).

d. Explosive materials shall not be irradiated nor shall they be allowed to generate in any experiment in quantities over 25 milligrams.

e. Only movable experiments in the center test hole shall be removed or installed with the reactor operating. All other experiments in the center test hole shall be removed or installed only with the reactor shutdown. Secured experiments shall be rigidly held in place during reactor operation.

f. Experiments shall 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.

3.6 **Experiments** - Continued

g. Experiments shall be designed such that a failure of an experiment will not lead to a direct failure of another experiment, a failure of a reactor fuel element, or to interfere with the action of the reactor control system or other operating components.

h. Cooling shall be provided to prevent the surface temperature of a submerged irradiated experiment from exceeding the saturation temperature of the cooling medium.

i. Irradiation containers to be used in the reactor, in which a static pressure will exist or in which a pressure buildup is predicted, shall be designed and tested for a pressure exceeding the maximum expected pressure by at least a factor of two (2).

j. Corrosive materials shall be doubly encapsulated in corrosion-resistant containers to prevent interaction with reactor components or pool water.

k. Fluids utilized in loop experiments placed in the beamports shall be of types which will not chemically react in the event of leakage and shall be maintained at pressure and temperature conditions such that the integrity of the beamtube will not be impaired in the event of loop rupture.

l. The normal operating procedures shall include controls on the use or exclusion of corrosive, flammable, and toxic materials in experiments or in the reactor containment building. These procedural controls shall include a current list of those materials which shall not be used and the specific controls and procedures applicable to the use of corrosive, flammable, or toxic materials which are authorized.

m. Cryogenic liquids shall not be used in any experiment within the reactor pool.

n. 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. First-of-a-kind fueled experiments shall be instrumented to measure temperature.

o. Fueled experiments containing inventories of iodine-131 through iodine 135 greater than 1.5 curies or strontium-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.

3.6 **Experiments - Continued**

Bases:

a. Specification 3.6.a restricts the generation of hazardous materials to levels that can be handled safely and easily. Analysis of fueled experiments containing a greater inventory of fission products has not been completed, and therefore their use is not permitted. (Ref. 13.2.6 of the SAR).

b. Specification 3.6.b is intended to reduce the likelihood of accidental voiding in the reactor core or water annulus surrounding the center test hole by restricting materials which could generate or accumulate gases or vapors.

c. The limitation on experiment materials imposed by Specification 3.6.c assures that the limits of 10 CFR 20, Appendix B, are not exceeded in the event of an experiment failure.

d. Specification 3.6.d is intended to reduce the likelihood of damage to reactor or pool components resulting from the detonation of explosive materials (Ref. 13.2.6 of the SAR).

e. Specification 3.6.e is intended to limit the experiments that can be moved in the center test hole while the reactor is operating to those that will not introduce reactivity transients more severe than one that can be controlled without initiating safety system action (Ref. 13.2.2 of the SAR).

f. - g. Specifications 3.6.f and 3.6.g provide guidance for experiment safety analysis to assure that anticipated transients will not result in radioactivity release and that experiments will not jeopardize the safe operation of the reactor.

h. Specification 3.6.h is intended to reduce the likelihood of reactivity transients due to accidental voiding in the reactor or the failure of an experiment from internal or external heat generation.

i. Specification 3.6.i is intended to reduce the likelihood of damage to the reactor and/or radioactivity releases from experiment failure.

j. Specification 3.6.j provides assurance that no chemical reaction will take place to adversely affect the reactor or its components.

3.6 **Experiments - Continued**

k. Specification 3.6.k provides assurance that the integrity of the beamports will be maintained for all loop-type experiments.

l. Specification 3.6.l assures that corrosive materials which are chemically incompatible with reactor components, highly flammable materials, and toxic materials are adequately controlled and that this information is disseminated to all reactor users.

m. The extremely low temperatures of the cryogenic liquids present structural problems that enhance the potential of an experiment failure. Specification 3.6.m provides for the proper review of proposed experiments containing or using cryogenic materials.

n. Specification 3.6.n is intended to reduce the likelihood of damage to the reactor and/or radioactivity releases from experiment failure.

o. Specification 3.6.o restricts the generation of hazardous materials to levels that can be handled safely and easily. Analysis of fueled experiments containing a greater inventory of fission products has not been completed, and therefore their use is not permitted. (Ref. 13.2.6 of the SAR).

3.7 Facility Airborne Effluents

Applicability:
This specification applies to the release of gaseous and particulate activity from the facility ventilation exhaust stack.

Objective:
The objective of this specification is to assure that 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.

Specification:

a. The maximum discharge rate through the ventilation exhaust stack shall not exceed the following:

Type of Radioactivity	Max. Concentration Averaged Over One Year	Max. Controlled Instantaneous Release Concentration
Particulates and halogens with half-lives greater than 8 days	AEC	AEC
All other radioactive isotopes	350 AEC	3,500 AEC

AEC = Air Effluent Concentration as listed in Appendix B, Table II, Column I of 10 CFR 20, "Standards for Protection Against Radiation."

Bases:

a. Dispersion calculations based upon standard reference material and experiment data obtained at the reactor show that argon-41 concentrations under average conditions will be 0.008 of the AEC limits in the unrestricted area surrounding the reactor facility. Dilution factors under conservative conditions are in the range of 5×10^4 under both average and stable conditions at ground level from the facility building.

The normal short burst releases at the facility are five to ten seconds in duration and occur on an average of ten times per day five days per week. The short bursts affect the concentration by less than 1% when averaged over a one-day period.

It is concluded that these concentrations as specified will not constitute a hazard to the health and safety of the public.

3.8 Reactor Fuel

Applicability:
This specification applies to the fuel elements used in the reactor core.

Objective:
The objective of this specification is to assure that the reactor fuel is operated within acceptable design considerations thus ensuring fuel element integrity is maintained.

Specification:

a. The peak burnup for UAL_X dispersion fuel shall not exceed a calculated 2.3×10^{21} fissions per cubic centimeter.

b. The reactor will not be operated using fuel in which anomalies have been detected or in which the dimensional changes of any coolant channel between the fuel plates exceeds ten (10) mils.

c. The reactor core shall consist of eight fuel assemblies.
Exception: The reactor may be operated to 100 watts above shutdown power on less than eight assemblies for the purposes of reactor calibration or multiplication measurement studies.

d. All fuel elements or fueled devices outside the reactor core shall be stored in a geometry such that the calculated K_{eff} is less than 0.9 under all conditions of moderation.

e. 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 its design values.

Bases:

a. Specification 3.8.a restricts the peak fissions per cubic centimeter burnup to values that have been correlated to result in less than 10% swelling of the fuel plates. It has been found that fuel plate swelling of less than 10% has no detrimental effect on fuel plate performance (Ref.: Change No. 4 to Facility License R-103, Change No. 6 to Facility License R-103, and Application dated September 12, 1986 with supplements).

b. Specification 3.8.b assures that fuel elements which have been inspected and found to be defective are no longer used for reactor operation.

3.8 Reactor Fuel - Continued

c. To assure the validity of the safety limit curves (Specification 2.1) and other safety analyses, Specification 3.8.c limits the reactor core to eight fuel elements at any significant power level.

d. - e. The limits imposed by Specifications 3.8.d and 3.8.e are conservative and assure safe fuel storage.

3.9 Reactor Coolant Systems

Applicability:
This specification applies to the reactor coolant systems.

Objective:
The objective of this specification is to protect the integrity of the reactor fuel and to prevent the release of fission product radioisotopes.

Specification:

a. The reactor shall not be operated in Modes I or II unless the following components or systems are operable:

   (1) Anti-Siphon System;
   (2) Primary Coolant Isolation Valves V507A/B; and
   (3) In-Pool Convective Cooling System.

b. The reactor shall not be operated with forced circulation unless:

   (1) a continuous primary coolant system fuel element failure monitor is operable,
   OR
   (2) the primary coolant system is sampled and analyzed at least once every four hours for evidence of fuel element failure.

c. The reactor shall not be operated if a radio-chemical analysis of the primary coolant system indicates an iodine-131 concentration of greater than 5×10^{-3} µCi/ml.

d. The anti-siphon system will be maintained pressurized to a value of 30 to 45 psig. In the event of a system low pressure alarm, immediate action will be taken to add air to obtain the specified pressure. The system pressure will be verified, recorded, and readjusted as required every 4 hours as part of the facility routine patrol. Procedures will be established for manual verification of water level in the anti-siphon system for conditions when system pressure has an unexplained rise of 4 psi or more. If water level is 6 inches or more above the anti-siphon isolation valves or a system leak or other malfunction prevents the maintenance of pressure in the specified range, the reactor will be shutdown until the malfunction can be corrected.

3.9 Reactor Coolant Systems - Continued

Bases:

a. The first line of protection against a loss of core water resulting from a rupture of the primary coolant system is provided by the check valve on the inlet line and by the invert loop and the anti-siphon system on the outlet line. Upon opening, the anti-siphon isolation valves will admit a fixed volume of air to the highest point of the invert loop, thus preventing the reactor core from becoming uncovered by breaking any potential siphon which may have been created by the pipe rupture (Ref. Section 6.3 of the SAR).

The primary coolant isolation valves are located on the inlet and outlet primary coolant lines as close as practicable to the biological shield. Proper operation of these valves is not required for protection of the integrity of the fuel elements, however, their operation provides a means for isolation of the in-pool portions of the primary coolant from the remainder of the system.

The in-pool convective cooling system is not required for core protection (Ref. Section 13.2.9.3 of the SAR), however, its operation is desirable to prevent the formation of steam in the loop and to reduce thermal cycling of the reactor fuel.

b. - c. Specifications 3.9.b and 3.9.c provide for the early detection of a fuel element failure so that corrective action can be taken to prevent the release of fission products. Refer to Specification 4.2.c for surveillance sampling of the primary coolant system.

d. Specification 3.9.d ensures that the anti-siphon system will perform its intended function as designed by imposing certain operational limits on the system (Ref. Section 6.3 of the SAR).

3.10 Auxiliary Systems

Applicability:
This specification applies to the reactor auxiliary systems.

Objective:
The objective of this specification is to provide for the operation of certain auxiliary systems and thus further protect the reactor fuel and personnel.

Specification:

a. The reactor shall not be operated unless the emergency electrical power system is operable.

b. The reactor shall not be operated unless the primary coolant make-up water system is operable and connected to a source of at least 2,000 gallons of primary grade water.

c. The reactor shall not be operated unless the emergency pool fill system is operable.

Bases:

a. On a loss of normal electrical power, the emergency electrical power system will supply power to the containment ventilation isolation doors, personnel entry doors, facility ventilation exhaust fans, emergency lighting panel, and reactor instrumentation and control systems. Therefore, on a loss of normal electrical power, the emergency electrical power system is not required for protection of the integrity of the fuel elements. In the extremely unlikely event of a simultaneous loss of normal electrical power and fuel element failure, the operation of the emergency electrical power system would be required to provide for continuous containment isolation (Ref. Section 13.2.7 of the SAR).

b. Specification 3.10.b provides for an adequate supply of primary grade water for make-up during all modes of operation.

c. The emergency pool fill system is capable of supplying water at approximately 1,000 gpm to the reactor pool. This supply assures that the water level in the pool will remain above the reflector in case a 6-inch beamport or a 6-inch pool coolant line is sheared (Ref. Sections 13.2.9.1 and 13.2.9.2 of the SAR).

4.0 SURVEILLANCE REQUIREMENTS

4.1 Containment System

Applicability:
This specification applies to the surveillance requirements on the containment system.

Objective:
The objective of this specification is to reasonably assure proper operation of the containment system.

Specification:

a. The reactor containment building leakage rate shall be measured annually, plus or minus four (4) months. No special maintenance shall be performed just prior to the test.

b. The containment actuation (reactor isolation) system, including each of its radiation monitors, shall be tested for operability at monthly intervals.

Bases:

a. Annual measurement of the containment building leakage rate has proven adequate to ensure that the leakage rate of the structure will remain within the design limits outlined in Specification 5.2.c. No special maintenance will be performed prior to the test so that the results demonstrate the historic integrity of the containment structure.

b. The reliability of the containment actuation (reactor isolation) system has proven that monthly verification of its proper operation is sufficient to assure operability.

4.2 Reactor Coolant Systems

Applicability:
This specification applies to the surveillance requirements on the reactor coolant systems.

Objective:
The objective of this specification is to reasonably assure proper operation of the reactor coolant systems.

Specification:

a. The primary coolant system relief valves shall be tested for operability at two-year intervals, with at least one of the valves tested on an annual basis.

b. The primary coolant isolation valves and the anti-siphon isolation valves shall be tested for operability at monthly intervals except during extended shutdown periods when the valves shall be tested prior to reactor operation.

c. A primary coolant sample shall be taken during each week of reactor operation and a radio-chemical analysis performed to determine the concentration of iodine-131.

Bases:

a. Satisfactory performance of both relief valves during the testing program over the past 40 years has demonstrated the reliability of the valves and the assurance of operability gained by the testing frequency outlined in Specification 4.2.a.

b. The past 40 years of operation of the primary coolant and anti-siphon isolation valves has shown that monthly testing is adequate to provide assurance of continued operability.

c. The weekly radio-chemical analysis will provide assurance that a fuel element leak will be discovered so that corrective action can be taken to prevent the release of fission products. Specification 4.2.c establishes the frequency of verification of compliance with Specification 3.9.c.

4.3 Control Blades

Applicability:
This specification applies to the surveillance requirements of the reactor control blades.

Objective:
The objective of this specification is to reasonably assure proper operation of the reactor control blades.

Specification:

a. The drop-time of each of the four shim blades shall be measured at quarterly intervals.

b. A different one of the four shim blades shall be inspected each six months so that every blade is inspected every two years. The reactor shall not be operated with a control blade that exhibits abnormal swelling or abnormalities that affect performance.

Bases:

a. Measurement of the drop-time of each of the four shim blades is normally made quarterly to demonstrate that the blades are capable of performing properly. In over 40 years of operation, to date, the shim blades have never failed to meet Specification 3.2.c.

b. Periodic inspection of the shim blades provides detection of singular blade abnormalities and any potential generic blade design deficiencies. Specification 4.3.b further assures that the reactor will not be operated using shim blades with suspected generic design deficiencies.

4.4 Reactor Instrumentation

Applicability:
This specification applies to the surveillance requirements of the reactor instrumentation systems.

Objective:
The objective of this specification is to reasonably assure proper operation of the reactor instrumentation systems.

Specification:

a. All instruments, as required by these specifications, shall be calibrated on a semiannual basis.

b. Radiation monitoring instrumentation, as required by these specifications, shall be checked for operability with a radiation source at monthly intervals.

c. All nuclear instrumentation channels shall be channel-tested before each reactor startup. This test shall not be required prior to a restart within two (2) hours following a normal reactor shutdown or an unplanned scram where the cause of the scram is readily determined not to involve an unsafe condition or a failure of one or more nuclear instrumentation channels.

Bases:

a. Semiannual calibration of the reactor instrument channels will assure that long-term drift of the channels will be corrected.

b. Experience has shown that monthly verification of operability of the radiation monitoring instrumentation in conjunction with the semiannual calibration is adequate assurance of proper operation over a long time period.

c. The nuclear instrumentation channel test will assure that the channels are operable.

4.5 Reactor Fuel

Applicability:
This specification applies to the surveillance requirements of the reactor fuel elements.

Objective:
The objective of this specification is to reasonably assure proper performance of the reactor fuel.

Specification:
a. One out of every eight (8) fuel elements that have reached their end-of-life will be inspected for anomalies.

Bases:
a. The specified fuel element inspections along with the continuous primary coolant system fission product monitoring and the weekly radio-chemical analysis of the primary coolant provide for the detection of anomalies resulting from reactor operation and reduces the possibility of fission product release to the primary coolant system. Inspecting the fuel elements at the end of their life has the added advantage of allowing for the decay of the fuel elements and, hence, reducing exposure to personnel.

4.6 Auxiliary Systems

Applicability:
This specification applies to the surveillance requirements of the reactor auxiliary systems.

Objective:
The objective of this specification is to reasonably assure proper operation of the auxiliary systems.

Specification:

a. The operability of the emergency pool fill system shall be tested on a semiannual basis.

b. The operability of the emergency power generator shall be verified on a weekly basis.

c. The ability of the emergency power generator to assume the emergency electrical loads shall be verified on a semiannual basis.

Bases:

a. The Missouri University water supply system provides a virtually unlimited source of raw water for the emergency pool fill system. Water supply is maintained at a high pressure by automatically-controlled pumping stations. The above check, in light of the reliability of the emergency pool fill system, provides assurance that Specification 3.10.c is satisfied.

b. The emergency power generator tests provide assurance that the generator is operable.

c. The semiannual electrical load test has proven satisfactory in providing reasonable assurance that the emergency power generator electrical control and distribution system will remain operable.

5.0 DESIGN FEATURES

5.1 Site Description

The MURR is situated on a 7.5-acre lot in the central portion of the University Research Park, an 84-acre tract of land approximately one mile southwest of the University of Missouri at Columbia's main campus. This campus is located in the southern portion of Columbia, the county seat and largest city in Boone County, Missouri. The University Research Park consists of low occupancy research buildings.

Approximate distances to the University property lines from the reactor facility are 2,400 feet (732 m) to the north, 4,800 feet (1,463 m) to the east, 2,400 feet (732 m) to the south, and 3,600 feet (1,097 m) to the west.

The operations boundary consists of the outer walls of the laboratory and reactor containment buildings. The area within this boundary is a "restricted access" area where the Reactor Facility Director has direct authority and control over all activities, normal and emergency. There are pre-established evacuation routes and procedures known to personnel frequenting this area. The operations boundary is within the site boundary.

The site boundaries consist of the following: Stadium Boulevard; Providence Road (Route K)[1]; the MU Recreational Trail; and the MKT Nature and Fitness Trail. The area within these boundaries is owned and controlled by MU 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.

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[1]Providence Road crosses MU property separating the University Research Park from another MU-owned tract of land lying to the east. The road runs north and south with the closest point of approach being approximately 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.

5.2 Reactor Containment Building

The reactor containment building is a five-level, poured-concrete structure with 12-inch thick reinforced exterior walls configured to form the shape of a cube, with each side being approximately 60 feet long. Below grade within the containment structure is a space extending to the north that is 15 feet high by 37 feet deep by 40 feet wide. The following design features apply to the MURR reactor containment building:

a. The reactor and fuel storage facilities shall be enclosed in a containment building with a free volume of at least 225,000 cubic feet.

b. Whenever containment integrity, as defined by Technical Specification 1.17, is required, containment building ventilation exhaust shall be discharged at a minimum of 55 feet above containment building grade level.

c. The containment building leakage rate shall not exceed 16.3 cubic feet per minute at 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. The test shall be performed by the make-up flow, pressure decay, or reference volume techniques.

d. The containment building shall have a secured fuel storage room with the key or combination under control of the Reactor Manager.

5.3 Reactor Coolant Systems

The MURR utilizes three reactor coolant systems: primary, pool, and secondary. The following design features apply to these coolant systems:

a. The reactor coolant systems shall consist of not less than a reactor pressure vessel, a primary pressurizer, two primary coolant circulation pumps, two primary coolant heat exchangers, two pool coolant circulation pumps, one pool coolant heat exchanger, and one pool water hold-up tank, plus all associated piping and valves.

b. The secondary coolant system shall be capable of continuous discharge of heat generated at the operating power of the reactor.

c. The circulation pumps and heat exchangers of the primary coolant system shall constitute two parallel systems separately instrumented to permit safe operation at five megawatts on either system or ten megawatts with both systems operating simultaneously.

d. The pool coolant circulation pumps shall be instrumented and connected so as to permit safe operation at five or ten megawatts on either pump or both pumps operating simultaneously.

e. 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.

f. The pool and primary coolant systems shall have a water clean-up system.

g. The pool and primary coolant piping shall have isolation valves between the reactor and mechanical equipment room.

h. The primary coolant system shall have two anti-siphon isolation valves.

i. The reactor shall have a natural convection coolant flow path for Mode III operation except for operation with the reactor subcritical by a margin of at least 0.015 Δk.

j. The reactor shall have a decay heat removal system.

k. The primary coolant system shall contain at least two operable pressure relief valves.

5.3 Reactor Coolant Systems - Continued

Exceptions:

a. The reactor may be operated in Mode II with any component removed from the shutdown leg of the system for emergency repairs.

b. Some materials in off-the-shelf commercial components may be excepted from Specification 5.3.e.

5.4 Reactor Core and Fuel

The following design features apply to the reactor core and fuel:

a. Each reactor fuel element shall contain 24 fuel-bearing plates with a nominal active length of 24 inches and a plate thickness of 0.050 inches. The nominal distance between the fuel plates shall be 0.080 inches. Plate nominal cladding thickness shall be 0.015 inches.

b. The fuel material shall be aluminide dispersion UAl_X fully enriched in the isotope uranium-235.

c. Each fuel element shall have a maximum uranium-235 loading of 775 grams.

d. The reactor fuel shall be contained in an aluminum pressure vessel.

e. The reactor shall have a beryllium and graphite reflector.

f. The reactor shall have five control blades between the pressure vessel and beryllium reflector. Four blades shall be for coarse control (shim blades) and one for fine control (regulating blade) of reactor power.

g. The reactor shall have the following experimental facilities:
   1. Six beam tubes which penetrate the graphite reflector;
   2. A center test hole located in the flux trap;
   3. A portion of the graphite reflector;
   4. A bulk pool consisting of the water region above and outside the graphite reflector; and
   5. A thermal column.

5.5 Emergency Electrical Power System

The following design feature applies to the emergency electrical power system:

a. The MURR shall have an emergency power generator capable of providing emergency electrical power to the emergency lighting system, the facility ventilation exhaust system, reactor instrumentation, and the personnel air lock doors.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

a. The organizational structure of the Missouri University (MU) relating to the Missouri University Research Reactor (MURR) shall be as shown in Figure 6.0.

b. As a minimum during reactor operation, there shall be two facility staff personnel at the facility. One of these individuals shall be a Reactor Operator or a Senior Reactor Operator licensed pursuant to 10 CFR 55. The other individual must be knowledgeable of the facility.

6.2 Review and Audit

a. A Reactor Advisory Committee (RAC) shall 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. The RAC shall review:

(1) Proposed changes to the MURR equipment or procedures when such changes have safety significance, or involve an amendment to the facility operating license, a change in the Technical Specifications incorporated in the license, or a question pursuant to 10 CFR 50.59. Changes to procedures that do not change their original intent may be made without prior RAC review if approved by the Reactor Manager or a designated alternate who is a licensed senior reactor operator. All such changes to the procedures shall be documented and subsequently reviewed by the RAC;

(2) Proposed experiments significantly different from any previously reviewed or which involve a question pursuant to 10 CFR 50.59; and

(3) The circumstances of all abnormal occurrences and violations of the Technical Specifications and the measures taken to prevent a recurrence.

b. The RAC may appoint subcommittees consisting of students, faculty, and staff of MU when it deems it necessary in order to effectively discharge its primary responsibilities. When subcommittees are appointed, these are to consist of no less than three members with no more than one student appointed to each committee. The subcommittees may be authorized to act in behalf of the parent committee.

6.2 Review and Audit - Continued

The RAC and its subcommittees are to maintain minutes of meetings in which the items considered and the committees' recommendations are recorded. Independent actions of the subcommittees are to be reviewed by the parent committee at the next regular meeting. A quorum of the committee or the subcommittees consisting of at least fifty percent of the appointed members must be present at any meeting to conduct the business of the committee or subcommittee. The RAC shall meet at least once during each calendar quarter.

A meeting of a subcommittee shall not be deemed to satisfy the requirement of the parent committee to meet at least once during each calender quarter.

c. Any additions, modifications or maintenance to the systems described in these Specifications shall be made and tested in accordance with the specifications to which the system was originally designed and fabricated or to specifications approved by the U.S. Nuclear Regulatory Commission (NRC).

d. Following a favorable review by the NRC, the RAC, or the Reactor Facility Management, as appropriate, and prior to conducting any experiment, the Reactor Manager shall sign an authorizing form which contains the basis for the favorable review.

6.3 Procedures

a. Written procedures shall be in effect for normal operations of the reactor, emergencies, radiological control, and the preparation for shipping and the shipping of byproduct material produced under the facility operating license.

b. The Reactor Manager shall annually review and approve the Reactor Operating and Emergency Procedures. The Reactor Health Physics Manager shall annually review and approve the radiological control procedures and the procedures for the preparation for shipping and the shipping of byproduct material.

6.4 Records

a. In addition to those otherwise required under this license and applicable regulations, and in no way substituting therefor, the following records will be maintained:

6.4 **Records - Continued**

(1) Reactor operating records, including power levels and periods of operation at each power level;

(2) Records showing radioactivity released or discharged into the air or water beyond the effective control of the licensee as measured at or prior to the point of such release or discharge;

(3) Records of emergency shutdowns and inadvertent scrams, including the reasons for the emergency shutdowns;

(4) Records of maintenance operations involving the substitution or replacement of reactor equipment or components;

(5) Records of experiments installed including description, reactivity worths, locations, exposure time, total irradiation, and any unusual events involved in their performance and in their handling; and

(6) Records of tests and measurements performed pursuant to these Specifications.

6.5 **Reportable Events and Required Actions**

a. Safety Limit Violation - In the event of a safety limit violation, the following actions shall be taken:

(1) The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC pursuant to 10 CFR 50.36(c)(1);

(2) The safety limit violation shall be promptly reported to the NRC. Prompt reporting of the violation shall be made by MU, by telephone or email, to the NRC Project Manager for MURR no later than the following working day;

(3) A detailed follow-up report shall be prepared. The report shall include the following:

a. Applicable circumstances leading to the violation including, when known, the causes and contributing factors;

b. Date and approximate time of the occurrence;

6.5 Reportable Events and Required Actions - Continued

c. Effect of the violation upon reactor and associated systems;

d. Effect of the violation on the health and safety of the facility staff and general public; and

e. Corrective actions to prevent recurrence.

(4) The follow-up report will be submitted within fourteen (14) days to the NRC Document Control Desk.

b. Release of Radioactivity - Should a release of radioactivity greater than the allowable limits occur from the reactor facility boundary, the following actions shall be taken:

(1) Reactor conditions shall be returned to normal or the reactor shall be shut down;

(2) The release of radioactivity shall be promptly reported to the NRC. Prompt reporting of the violation shall be made by MU, by telephone or email, to the NRC Project Manager for MURR no later than the following working day;

(3) If it is necessary to shutdown the reactor to correct the occurrence, operations shall not be resumed until authorized by the Reactor Manager; and

(4) A detailed follow-up report shall be prepared. The follow-up report will be submitted within fourteen (14) days to the NRC Document Control Desk.

c. Other Reportable Occurrences - In the event of an Abnormal Occurrence, as defined by Definition 1.1, the following actions shall be taken:

(Note: Where components or systems are provided in addition to those required by these Technical Specifications, the failure of the extra components or systems is not considered reportable provided that the minimum number or components or systems specified or required perform their intended reactor safety function.)

(1) The abnormal occurrence shall be promptly reported to the NRC. Prompt reporting of the violation shall be made by MU, by telephone or email, to the NRC Project Manager for MURR no later than the following working day;

6.5 Reportable Events and Required Actions - Continued

(2) A detailed follow-up report shall be prepared. The follow-up report will be submitted within fourteen (14) days to the NRC Document Control Desk; and

(3) A return to normal reactor operation will not be allowed until authorized by the Reactor Manager.

d. Other Reports - A written report shall be submitted to the NRC Document Control Desk within thirty (30) days of:

(1) Any significant change(s) in the transient or accident analyses as described in the SAR; and

(2) Permanent changes in the facility organization involving the Office of the Provost or the Director's Office.

e. Annual Report - An annual operating report shall be submitted to the NRC within sixty (60) days following the end of each calender year. The report shall include the following information for the preceding year:

(1) A brief narrative summary of (a) operating experience (including operations designed to measure reactor characteristics), (b) changes in the reactor facility design, performance characteristics, and operating procedures related to reactor safety occurring during the reporting period, and (c) results of surveillance tests and inspections;

(2) A tabulation showing the energy generated by the reactor (in megawatt-days);

(3) The number of emergency shutdowns and inadvertent scrams, including the reasons therefor and corrective action, if any, taken;

(4) Discussion of the major maintenance operations performed during the period, including the effects, if any, on the safe operation of the reactor;

(5) A summary of each modification to the reactor facility or change to the procedures, tests, and experiments carried out under the conditions of 10 CFR 50.59;

6.5 **Reportable Events and Required Actions** - Continued

(6) 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;

(7) A description of any environmental surveys performed outside the reactor facility; and

(8) A summary of radiation exposures received by facility staff, experimenters, and visitors, including the dates and time of significant exposure, and a brief summary of the results of radiation and contamination surveys performed within the facility.

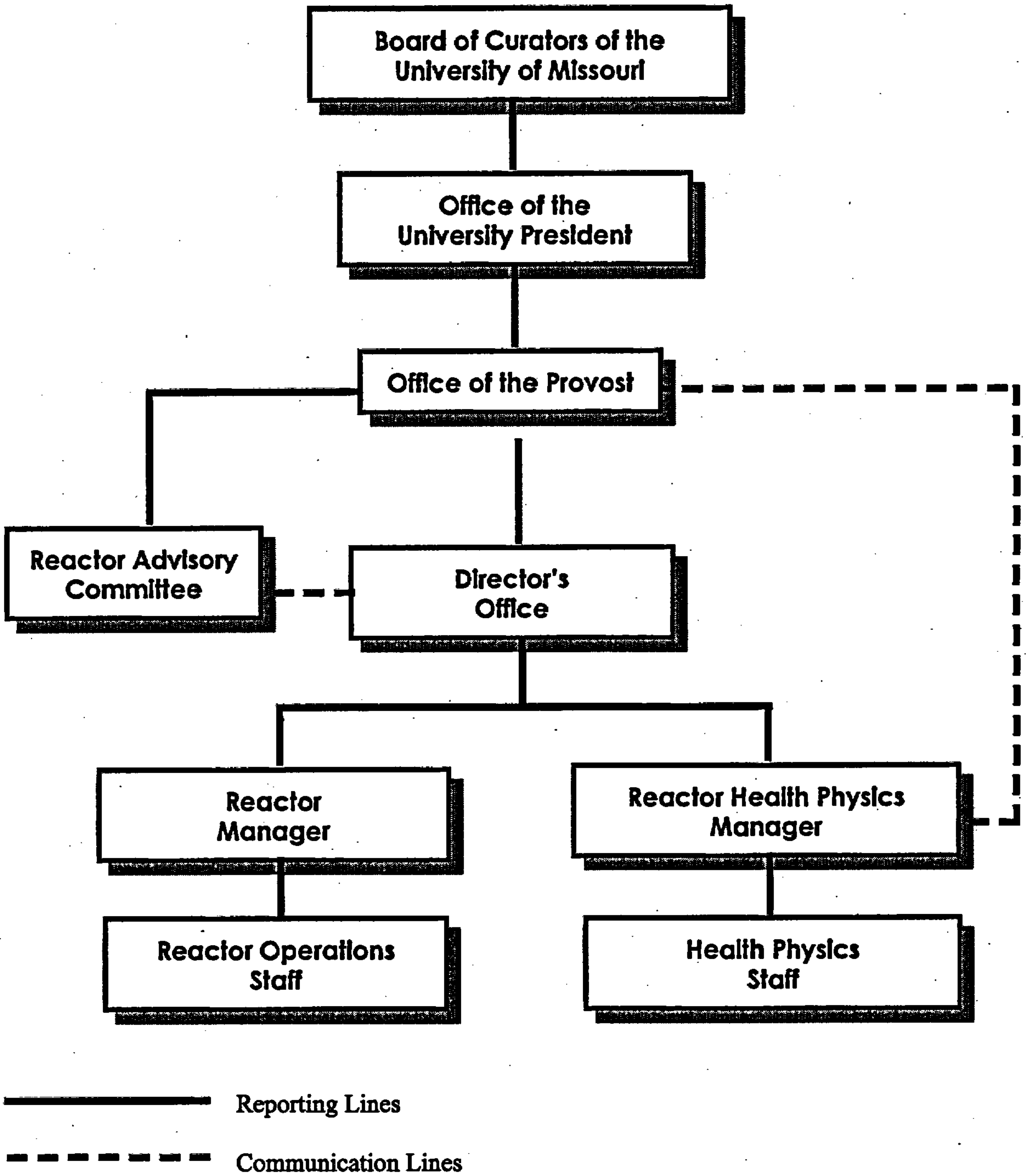

FIGURE 6.0
MISSOURI UNIVERSITY RESEARCH REACTOR (MURR)
ORGANIZATION

APPENDIX B

RADIOLOGICAL IMPACT OF AR-41 DURING NORMAL OPERATIONS

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

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.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 Missouri University Research Reactor (MURR). The source term for normal operation is the noble gas argon-41 (^{41}Ar), 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, the 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 the radioiodine and noble gases released by the melting of four fuel plates.

^{41}Ar is produced when the argon-40 (^{40}Ar) in air (~1.0%) is activated by thermal neutrons. The principle production areas within the reactor facility include the pneumatic tube (p-tube) system, the thermal column, and the beamports. The ^{41}Ar 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 the 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 ^{41}Ar, which accounts for greater than 99% of the radioactivity released through the facility exhaust stack. 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 by sealing the thermal column collimator to prevent the release of gaseous activity. These modifications actually reduced the amount of ^{41}Ar being released during 10-MW operation to levels less than that which were previously released during 5-MW operation. Greater than 98% of the remaining ^{41}Ar 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 when in operation, it would be impractical to use bottled gas not containing ^{40}Ar 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 ^{41}Ar since the system is pressurized.

Several alternatives exist to reduce the production of ^{41}Ar 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 capability of the MURR. Nevertheless, the MURR will continue to explore new techniques and will implement all practical measures to further reduce ^{41}Ar releases.

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). More recently, Committee II reviewed the current state of knowledge and published updated recommendations in Publication 30 (1978/99), which supercedes Publication 2. In Publication 30, the ICRP recommends 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 ^{41}Ar are:

- For accessible areas inside the operations boundary: 3×10^{-6} μCi/ml; and
- 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.

(1) Elevation above sea level	=	687 feet (209 m);
(2) Diameter	=	40 inches (1.02 m);
(3) Maximum Flow Rate	=	30,500 ft^3/min;
(4) 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);
(5) 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 ^{41}Ar release from the MURR is 350 times the Air Effluent Concentration (AEC) listed in Appendix B, Table II, of 10 CFR 20, or:

$$Q = 350 \times \text{AEC} \times \text{Flow Rate};$$

$$= 350 \times (1 \times 10^{-8}\ \mu\text{Ci/ml}) \times (30{,}500\ \text{ft}^3\text{/min}) \times (2.831 \times 10^4\ \text{ml/ft}^3);$$

$$= (3.0 \times 10^3\ \mu\text{Ci/min}) \times (1 \times 10^{-6}\ \text{Ci}/\mu\text{Ci}) \times (1\ \text{min/60 sec});$$

$$= 5.0 \times 10^{-5}\ \text{Ci/sec}.$$

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. These data were collected between May 5, 1973 and May 4, 1975, and were 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 were 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}

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}

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}

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}

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}

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) × P(N/A) = 3.1×10^{-4}.

B.3 Calculations

The following three equations were used to calculate the ^{41}Ar 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 m^3/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 m^3/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,

$$H = h + d\left(\frac{v}{\mu}\right)^{1.4}.$$

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 ^{41}Ar 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 ($\mu Ci/ml$ 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 ^{41}Ar (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);
	=	8.84×10^{-3} (mrem-m^3/ρCi-Y) for ^{41}Ar;
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 ^{41}Ar 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)	Dose with %s (mrem/y)
A	35	35	23	5.92×10^{-5}	2.96×10^{-9}	0.0
B	27	25	15	5.60×10^{-5}	2.80×10^{-9}	0.1
C	23	19	11	4.07×10^{-5}	2.04×10^{-9}	0.3
D	20	12	7	1.14×10^{-5}	5.71×10^{-10}	0.3
E	23	9	5	4.50×10^{-8}	2.25×10^{-12}	0.0
F	30	6.6	3.2	4.76×10^{-22}	2.38×10^{-26}	0.0
Total						0.7

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)	Dose with %s (mrem/y)
A	16	160	330	2.87×10^{-6}	1.43×10^{-10}	0.0
B	8	110	90	1.07×10^{-5}	5.34×10^{-10}	0.0
C	4	81	50	1.87×10^{-5}	9.33×10^{-10}	0.1
D	1	54	25	4.21×10^{-5}	2.10×10^{-9}	1.2
E	4	41	18	1.05×10^{-4}	5.26×10^{-9}	1.4
F	11	30	12	2.23×10^{-4}	1.12×10^{-8}	1.5
Total						4.2

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):

$$H = 209 + 1.02 \times \left(\frac{17.7}{4.0}\right)^{1.4} - 213;$$

$$= 4.2 \text{ meters.}$$

^{41}Ar Concentration:

$$\frac{\chi}{Q} = \frac{1}{\pi \times 41 \times 18 \times 4} \times \exp\left[-\frac{1}{2} \times \left(\frac{4.2}{18}\right)^2\right];$$

$$= (1.08 \times 10^{-4}) \times (0.97);$$

$$= 1.05 \times 10^{-4} \text{ sec/m}^3;$$

$$\chi = (1.05 \times 10^{-4} \text{ sec/m}^3) \times (5.0 \times 10^{-5} \text{ Ci/sec});$$

$$= 5.3 \times 10^{-9} \text{ Ci/m}^3;$$

$$= 5.3 \times 10^{-9} \ \mu\text{Ci/ml.}$$

Class E occurs 17.6% of the time and, of that time, the wind blows from the South 17.6% of the time.

$$\%\text{s comb} = \left(\frac{17.6}{100}\right) \times \left(\frac{17.6}{100}\right);$$

$$= 0.031.$$

$$\text{Dose}_B = \left(5.3 \times 10^{-9} \frac{\text{Ci}}{\text{m}^3}\right) \times (0.031) \times \left(8.84 \times 10^{-3} \frac{\text{mrem - m}^3}{\text{pCi - Y}}\right) \times \left(10^{12} \frac{\text{pCi}}{\text{Ci}}\right);$$

$$= 1.4 \text{ mrem/y.}$$

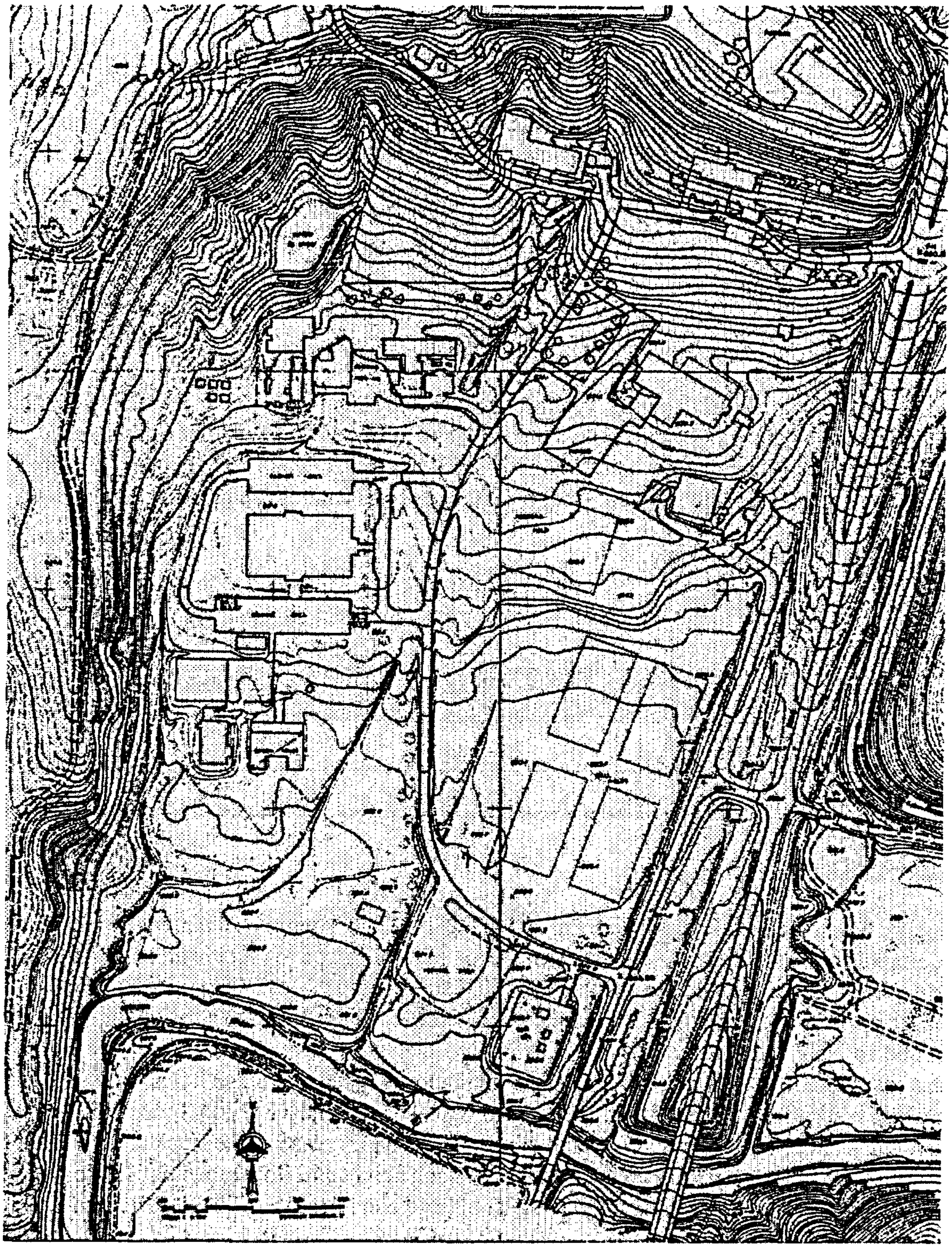

FIGURE B.1
UNIVERSITY OF MISSOURI TOPOGRAPHICAL MAP

B.5 Normal Operational Releases

For the past ten years, the MURR has released approximately 1,100 Ci/y of ^{41}Ar with a stack flow rate of about 30,500 ft^3/ min. Production of ^{41}Ar is expected to remain relatively the same, and so the average ^{41}Ar concentration is predicted to be:

$$(30{,}500\ \text{ft}^3/\text{min}) \times (2.831 \times 10^4\ \text{ml/ft}^3) \times (60\ \text{min/h}) \times (24\ \text{h/day}) \times (365\ \text{day/y}) = 4.54 \times 10^{14}\ \text{ml/y};$$

$$(1{,}100\ \text{Ci/y}) \times (10^6\ \mu\text{Ci/Ci}) \times (1/(4.54 \times 10^{14})\ \text{y/ml}) = 2.42 \times 10^{-6}\ \mu\text{Ci/ml};$$

which is approximately 69% of the Technical Specifications limit. Because the calculated dose estimates are proportional to the total amount of ^{41}Ar released, the dose estimates for actual operating conditions are easily calculated using the ratios of the stack release rates (given ^{41}Ar 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 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 is no significant reduction in safety as the result of the continued operation of the current MURR stack release conditions.

APPENDIX C

THERMAL-HYDRAULIC TRANSIENT ANALYSIS

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

REFERENCES

C.1 "RELAP5/MOD3.3 Code Manual," NUREG/CR-5535/Rev 1, Information Systems Laboratories, Inc., Rockville, MD and Idaho Falls, ID, December 2001.

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 Missouri University 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 (Ref. 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 the 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. (Ref. 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.

C.2.2 Modeling of the 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 the 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.

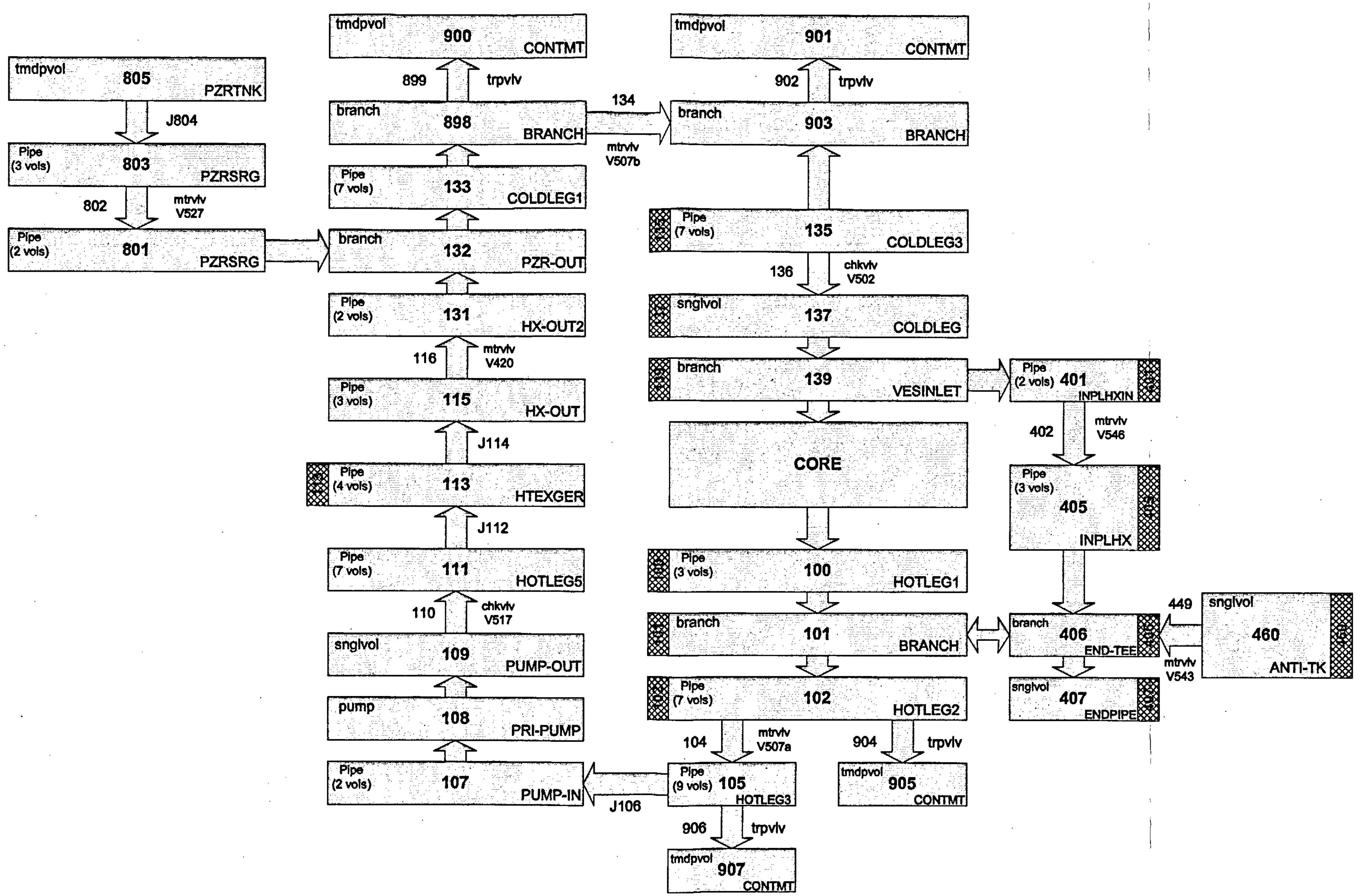

FIGURE C.1
DETAILED RELAP5 MODEL OF THE MURR PRIMARY COOLANT LOOP

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.

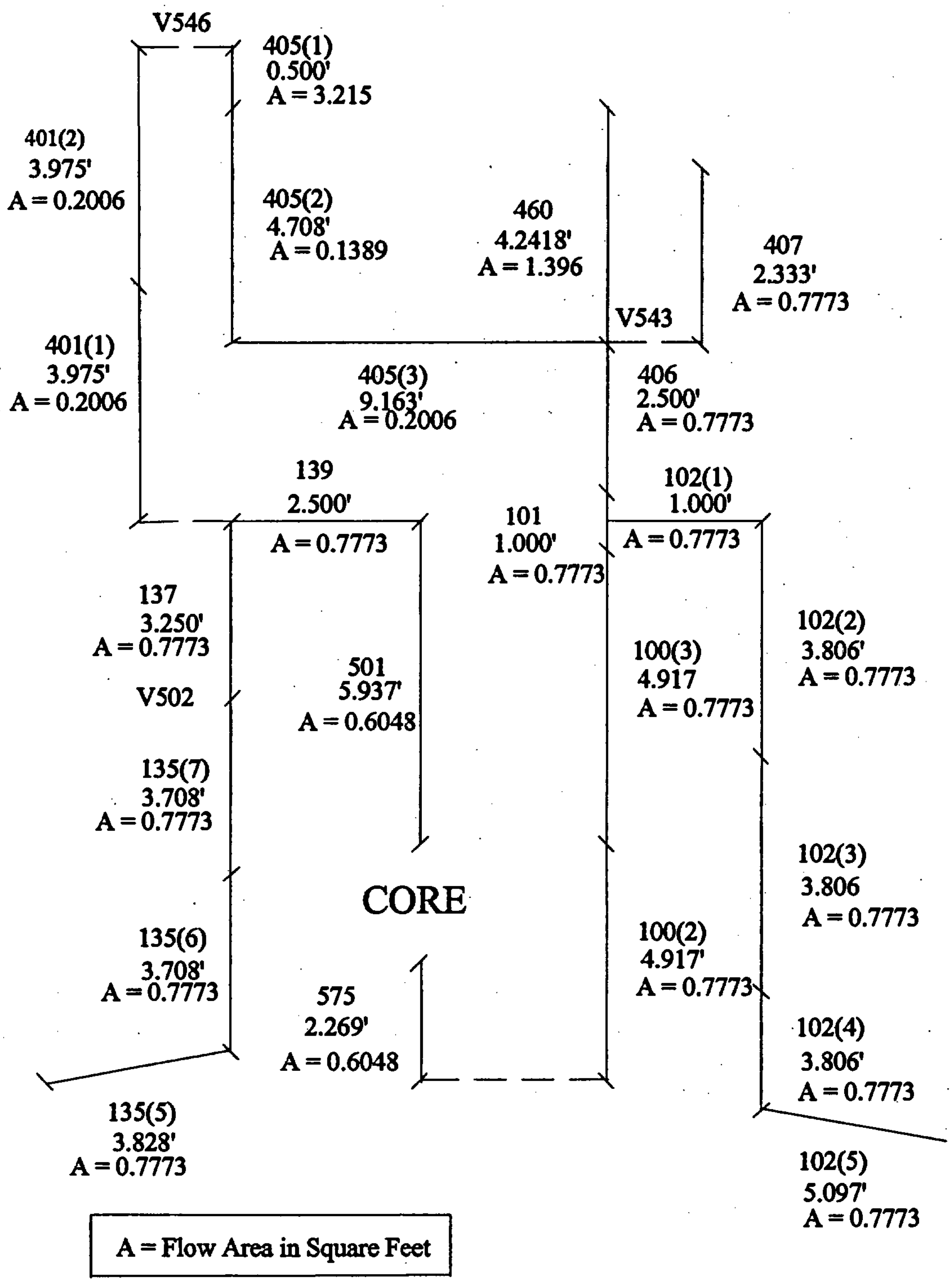

FIGURE C.2
IN-POOL PRIMARY COOLANT LOOP PIPING

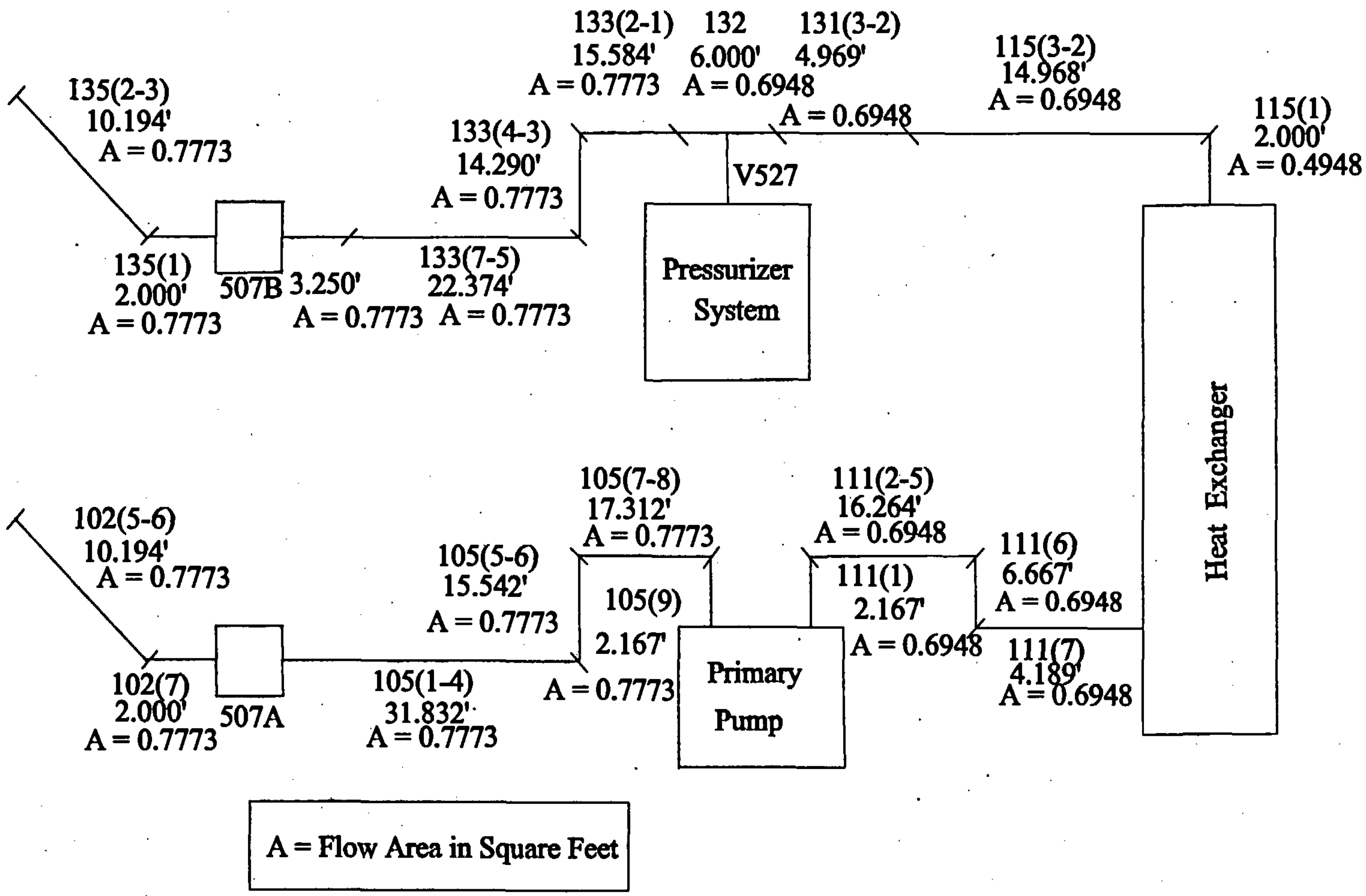

FIGURE C.3
EX-POOL PRIMARY COOLANT LOOP PIPING - HOT AND COLD LEGS

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, thus 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.

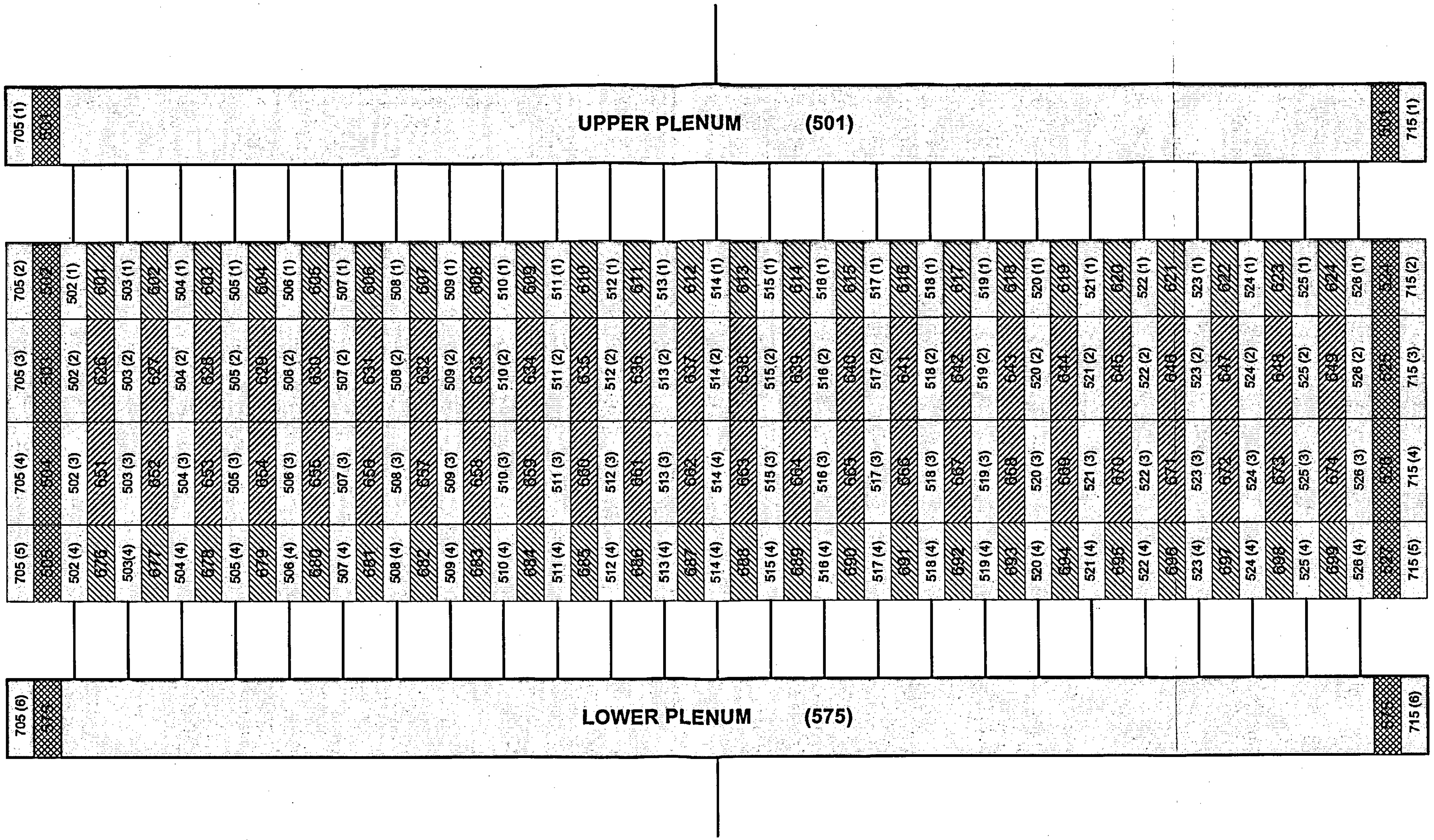

FIGURE C.4
DETAILED RELAP5 MODEL OF THE MURR CORE

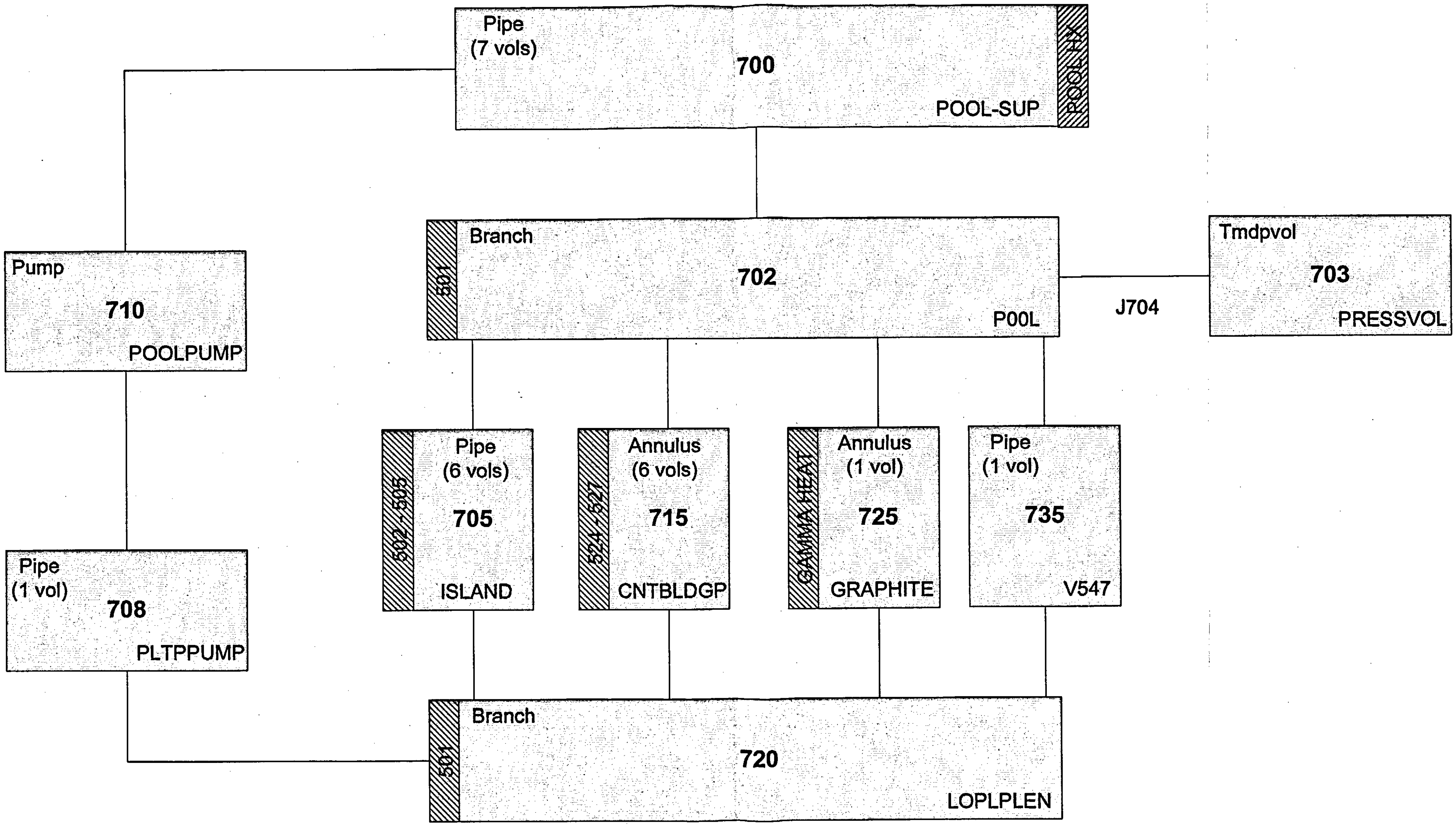

FIGURE C.5
DETAILED RELAP5 MODEL OF THE MURR BULK REACTOR POOL AND POOL COOLANT LOOP

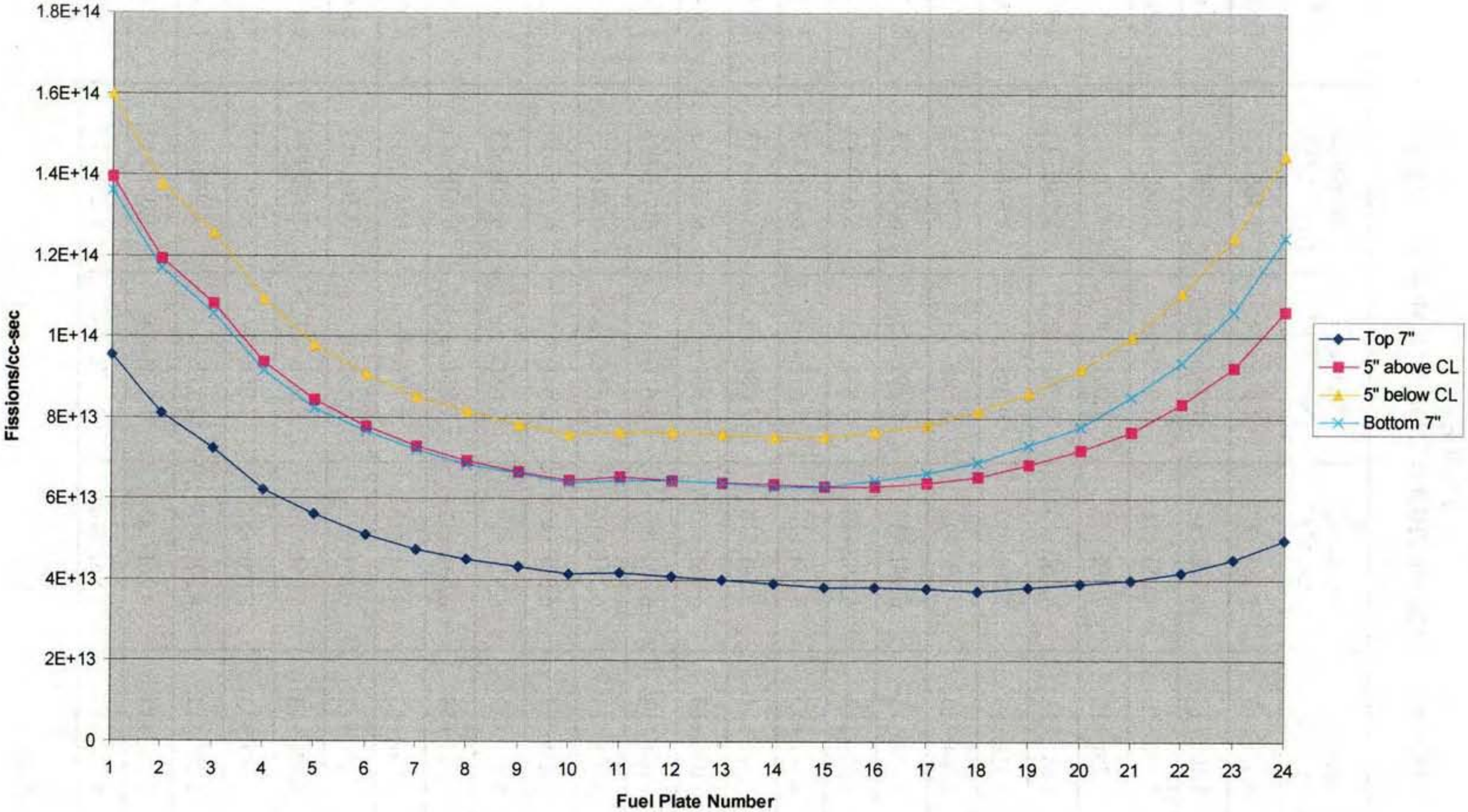

FIGURE C.6
AVERAGE FISSIONS/CC-S IN THE FUEL PLATE SECTIONS

C-28

TABLE C-1
POWER DENSITY PROFILE OF THE MODELED CORE

Plate	Section 1 (Top 7.75") *	Section 2 (Next 5") *	Section 3 (Next 5") *	Section 4 (Bot. 7.75") *	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
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

* fissions per cm^3-sec

TABLE C-2
RELAP5 POWER PROFILE OF THE MODELED CORE

Plate	Section 1 (Top 7.75")	Section 2 (Next 5")	Section 3 (Next 5")	Section 4 (Bottom 7.75")
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%
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 piping is replaced in the RELAP5/MOD3.3 model with a single 12-inch 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 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 the 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. 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.

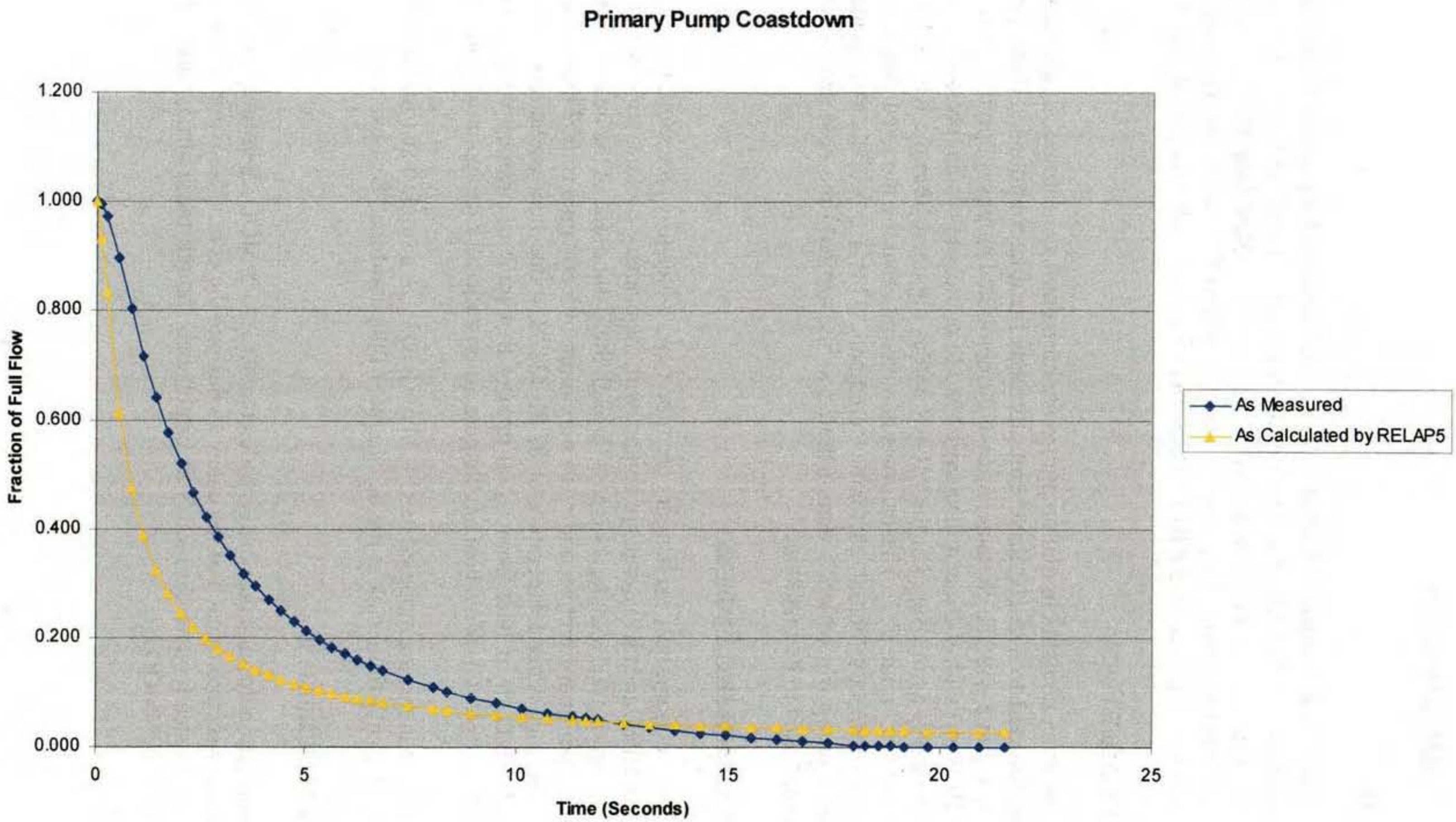

FIGURE C.7
ACTUAL VERSUS MODEL PREDICTED PRIMARY COOLANT PUMP COASTDOWN

C-39

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, thus the need for an additional heat removal system to protect the integrity of the fuel is not required. However, for redundancy, and to provide a greater safety margin, a reactor decay heat removal system is installed to remove decay heat following an emergency shutdown accompanied by reactor loop isolation or in the event of a loss of normal primary coolant flow.

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

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 ensures that the reactor core does not become uncovered during a LOCA. A rupture of the primary coolant system, followed by a loss of pressure, causes the anti-siphon system to admit a fixed volume of air to the high point of the reactor outlet piping, thus breaking any potential siphon which may have been created by the pipe rupture.

The 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.

ATTACHMENT 2
EX-POOL COMPONENTS OF THE PRIMARY COOLANT LOOP THERMAL-HYDRAULIC MODEL

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.

<u>Components V905 & V907 - Establishing Containment Volume for Hot Leg Break Simulation</u>

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