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MCCLELLAN NUCLEAR RESEARCH CENTER
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SAFETY ANALYSIS REPORT
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SUBMITTED JULY 6 2020

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CHAPTER 1

INTRODUCTION AND GENERAL

DESCRIPTION OF PLANT

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1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLAN

1.1 Introduction

This Safety Analysis Report supports an application to the Nuclear Regulatory Commission (NRC) by the University of California - Davis (UCD) for the utilization of a steady-state 1000 kW TRIGA®. The reactor is owned and operated by UCD for neutron radiography and irradiation services for both university and non-university tasks. The facility is known as the University of California - Davis/McClellan Nuclear Research Center (UCD/MNRC).

This document addresses only the safety issues associated with the operation of the UCD/MNRC reactor. Accident scenarios are analyzed in Chapter 13. The industrial safety issues involving the handling of radiographic parts and irradiation experiments are addressed in the UCD/MNRC Operational Safety Hazards Analysis and support documents.

1.1.1 Purpose of Facility

The UCD/MNRC provides a broad range of radiographic and irradiation services to the military and non-military sector. The facility presently provides four radiography bays and consequently four beams of neutrons for radiography purposes. In addition to the radiography bays, the UCD/MNRC reactor core and associated experiment facilities are completely accessible for the irradiation of material. These irradiation services include silicon doping, isotope production, both medical and industrial, and neutron activation analysis (e.g., geological samples). All bays contain the equipment required to position parts for inspection as well as the radiography equipment.

1.1.2 Location of Facility

The reactor is located in the UCD/MNRC Building [redacted] on the former McClellan AFB, an industrial park of 2600 acres located approximately 8 miles northeast of Sacramento, California.

The industrial park is adequately suited for the location of the UCD/MNRC reactor. This is substantiated by this document and by the fact there are dozens of TRIGA® reactors in operation worldwide, including 16 in the United States. Many of these reactors are located on university campuses and in hospitals with surrounding high populated areas.

1.2 General Plant Description

1.2.1 Building

The UCD/MNRC is a three-level 18,000 ft^2 rectangular shaped building that incorporates a TRIGA® reactor, as shown in Figures 1.1 through 1.6. This facility provides space, shielding, and environmental control for the radiography and irradiation services work. Adequate room has been provided to handle the experiments and components in the facility in a safe manner.

The ground-level elements of the UCD/MNRC are constructed of reinforced concrete and concrete unit masonry with minor elements of exposed steel. The exterior walls of the upper portions feature factory-colored metal panels, concrete, and concrete unit masonry walls.

The exterior walls of the radiography bays are made of reinforced concrete and vary in thickness from 2 to 3 feet. The interior walls and the roofs of the radiography bays are constructed of 2-ft thick reinforced concrete.

The reactor room is above the radiography bays. Its walls are constructed of standard-filled reinforced concrete block and it has a typical metal deck built-up roof.

The reactor is located in a cylindrical aluminum walled tank with the core positioned approximately 4.5 ft below grade (i.e., tank bottom is ~6.5 ft below grade) (Figure 1.2). The reactor tank is surrounded by a monolithic block of reinforced concrete. Below ground level, the concrete is approximately [redacted] ft thick. Above ground level, the concrete varies in thickness from approximately [redacted] ft with the smaller dimension at the tank top. The tank is supported by a concrete pad approximately 9.5 ft thick.

The basic purpose of the massive concrete structures is to provide biological shielding for personnel working in and around the UCD/MNRC. However, due to the massiveness of these structures, they provide excellent protection for the reactor core against natural phenomena.

Another irradiation facility had been added to the original UCD/MNRC structural design. This facility is located in the lower level of Bay 4 and is called Bay 5. This facility was created by cutting a cavity into the biological shield so that a fifth neutron beam can be extracted from approximately the core centerline. The cavity extends from the outer surface of the biological shield to the tank wall. The cavity cross section is 10 feet high by 8.5 feet wide until about six inches from the tank wall. The last six inches has a window that exposes about a 3 feet high by 3.5 feet wide rectangular area of the tank wall. For now, the cavity has been filled with concrete block to keep the radiation levels below allowable limits (Section 11.1.1.3.1).

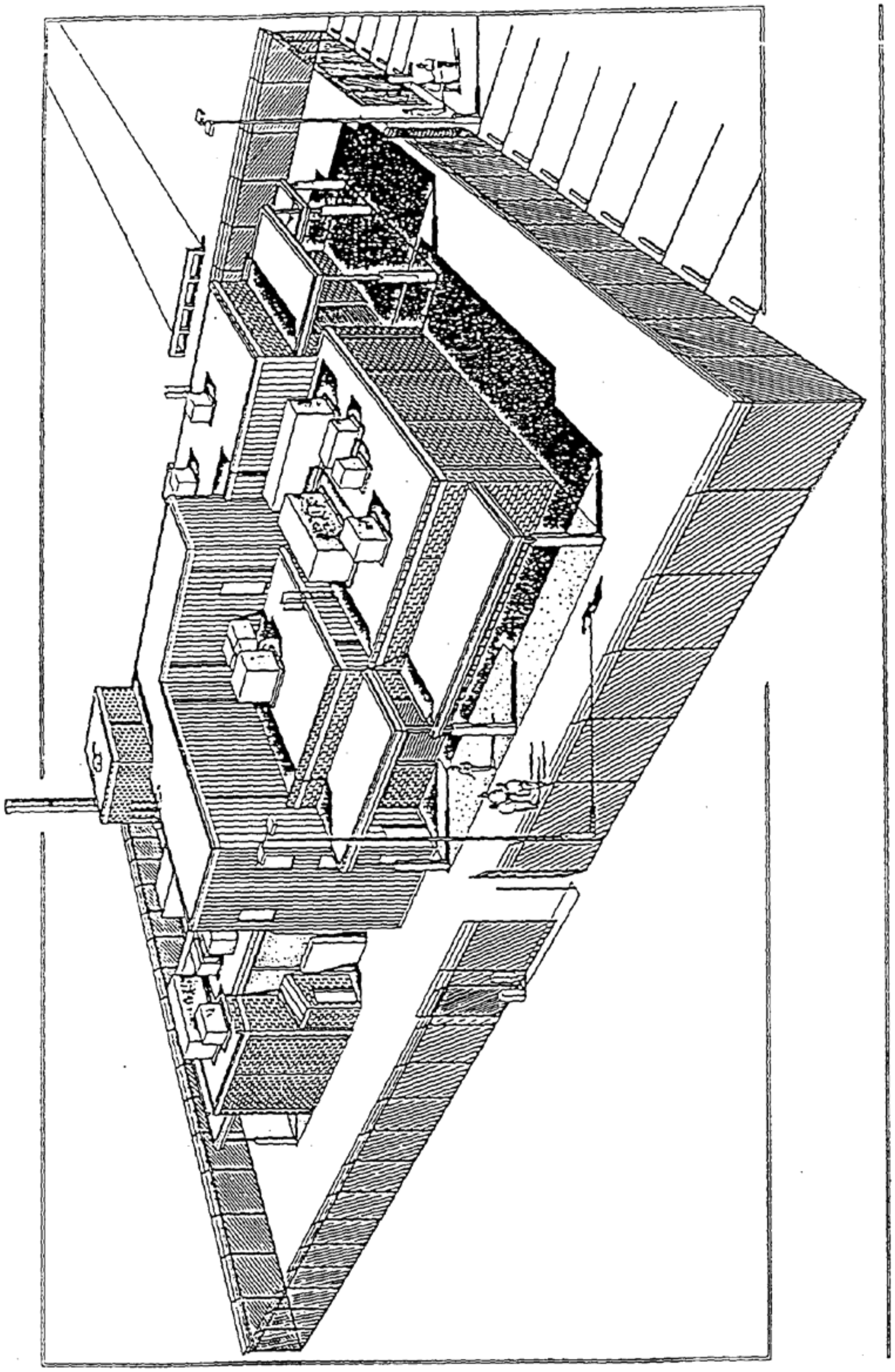

FIGURE 1.1 ARTIST'S AXONOMETRIC VIEW OF MAIN UCD/MNRC FACILITY

[redacted]

FIGURE 1.2 UCD/MNRC PLAN VIEW - MAIN FLOOR

[redacted]

FIGURE 1.3 UCD/MNRC PLAN VIEW - SECOND FLOOR

[redacted]

FIGURE 1.4 UCD/MNRC ELEVATION SECTION A-A

[redacted]

FIGURE 1.5 UCD/MNRC ELEVATION SECTION B-B

[redacted]

FIGURE 1.6 UCD/MNRC ELEVATION SECTIONS

The facility exhaust systems are designed to maintain the reactor room and radiography bays at a slightly negative pressure with respect to surrounding areas to prevent the spread of radioactive contamination. These systems also maintain concentrations of radioactive gases in the reactor room and the radiography bays to levels that are below the 10 CFR Part 20 limits for restricted areas. The reactor and radiography control rooms each have their own air handling systems.

There is a system of interlocks and warning devices to prevent personnel from inadvertent exposure to high radiation levels. Interlocks prevent personnel from entering the radiography bays whenever the beam tube shutters are open and the reactor is operating. This system also prevents the beam tube shutters from being opened when the reactor is operating and personnel are in the radiography bays when the bay doors are open. There are “Reactor On” lights throughout the facility that indicates the reactor operating status. Beam tube shutter positions are monitored in the reactor and radiography control rooms. Audible and visual alarms are sounded in the radiography bays when the shutters are opening. Manual and automatic reactor shutdown devices are located in the reactor room, and each radiography bay, so immediate reactor shutdown can be initiated by anyone occupying these areas should it become necessary.

The UCD/MNRC contains the electrical, water, and sewer utilities required for operation. In addition, the facility has both fire detection and suppression systems, intercom systems, radiation monitoring systems, security systems, parts positioning equipment, irradiation and radiography equipment.

1.2.2 Reactor

The UCD/MNRC reactor is a 1.0 MW steady state, natural-convection-cooled TRIGA® reactor with a graphite reflector presently designed to accept the source ends of the four neutron radiography beam tubes which terminate in four separate neutron radiography bays. The reactor is located near the bottom of a water-filled aluminum tank 7 ft in diameter and about 24.5 ft deep (Figure 1.7). Direct visual and mechanical access to the core and mechanical components are available from the top of the tank for inspection, maintenance, and fuel handling. The water provides adequate shielding for personnel standing at the top of the tank. The control rod drives are mounted above the tank on a bridge structure spanning the diameter of the tank. The reactor is monitored and controlled by a computer-based instrumentation and control system featuring color graphics display and automatic logging of vital information. Both manual and automatic control options are available to the operator.

The reactor console is located in the reactor control room and manages all control rod movements, accounting for such things as interlocks and choice of particular operating modes. It processes and displays information on control rod positions, power level, fuel temperatures, pulse characteristics, and other system parameters. The reactor console performs many other functions, such as monitoring reactor usage and storage of historical operating data for replay at a later time.

[redacted]

FIGURE 1.7 UCD/MNRC REACTOR

Fuel for the UCD/MNRC reactor is standard TRIGA® reactor fuel having 20%, or 30% by weight of uranium enriched to less than 20% ^{235}U. TRIGA® reactor fuel is characterized by inherent safety, high fission product retention, and the demonstrated ability to withstand water quenching with no adverse reaction from temperatures to 1150°C. The inherent safety of TRIGA® reactors has been demonstrated by extensive experience acquired from similar TRIGA® systems throughout the world. This safety arises from the large prompt negative temperature coefficient that is characteristic of uranium-zirconium hydride fuel-moderator elements used in TRIGA® systems. As the fuel temperature increases, this coefficient immediately compensates for reactivity insertions. This results in a mechanism whereby reactor power excursions are limited/terminated quickly and safely.

Heat produced by the reactor core is removed by the primary and secondary cooling systems. The primary system circulates tank water through a water-to-water heat exchanger. The secondary water system gains heat in the heat exchanger and rejects it by use of a cooling tower. A purification system circulates a small amount of tank water through a filter and resin tanks to maintain purity and optical clarity. All of these systems contain the necessary instruments and controls for operations and monitoring performance.

1.3 Relation of UCD/MNRC to Other TRIGA® Reactors

The design of the UCD/MNRC fuel is similar to those of approximately 30 TRIGA® type reactors currently operating world-wide with 16 in the United States. Most of these reactors were constructed in the late 1950s and 1960s. Since a large number of these reactors have been in operation for many years, considerable operational information is available and their characteristics are well documented.

Four of the ten TRIGA® reactors licensed for 1 MW steady-state operation in the United States have characteristics similar to the UCD/MNRC reactor. These four reactors are located at Penn State (1966), the U.S. Geological Survey Center - Denver (1969), Oregon State University - Corvallis (1967), and the University of Texas - Austin (1990). Worldwide, there have been five TRIGA® reactors operated at powers equal to or above 2 MW.

Table 1-1 lists the principal design parameters for the 1 MW UCD/MNRC Reactor and the Thailand 2 MW Reactor. It should be noted that these parameters may vary slightly depending on the use and core loading.

TABLE 1-1 TYPICAL PRINCIPAL DESIGN PARAMETERS

Parameter	UCD/MNRC	Thailand
Maximum steady-state power level	1000	2000
Fuel-Moderator material	U-$ZrH_{1.6-1.7}$	U-$ZrH_{1.6}$
Uranium Enrichment	Up to 20% U^{235}	Up to 20% U^{235}
Uranium Content	20 to 30 wt %	8.5 wt %
Shape	Cylindrical	Cylindrical
Length of Fuel	38 cm (15 in) overall	38 cm (15 in) overall
Diameter of Fuel	3.63 cm (1.43 in) OD	3.63 cm (1.43 in) OD
Cladding Material	0.051 cm (0.020 in) 304 SS	0.051 cm (0.020 in) 304 SS
Number of Fuel Elements	100[a]	100
Maximum Excess Reactivity	7.0[a]% Δk/k (cold, clean)[a]	6.3% Δk/k (cold, clean)
Number of Control Rods	6	5
Regulating	1	1
Safety-Transient	1	3
Shim	4	1
Total Reactivity Worth of Rods	8.7% Δk/k[a]	10.12% Δk/k
Reactor Cooling	Natural Convection of Pool Water	Natural Convection of Pool Water

(a) = approximate value.

The functional characteristics of the UCD/MNRC Reactor's Instrumentation and Control (I&C) System are the same as for the approximately 30 TRIGA® reactors operating in the United States and throughout the world. However, the standard instrument and control system has been replaced by one with a computer-based design incorporating the use of a GA developed, multifunction, NM-1000 microprocessor based neutron monitor channel and a NPP-1000 analog-type neutron monitoring channel. The channels are completely independent and provide redundant safety channels. In addition, the NM-1000 channel provides wide-range log power, period, and multi-range linear power. The control system logic is contained in a separate control system computer with a color graphics display which is the interface between operator and the reactor.

Both the control rod and transient rod drives are slightly different than those used on the earlier standard TRIGA® systems. The UCD/MNRC control rod drives, with the exception of the motor are essentially the same as the drives used on other TRIGA® systems. The UCD/MNRC drives use a stepping-type motor rather than the non-synchronous, single- phase motors used on earlier drives. The design and operation of the stepping motor type drive has been fully developed and has been used on the University of Texas - Austin, U.S. Geological Survey Center - Denver, and the Armed Forces Radiobiological Research Institute TRIGA® Reactor Systems.

The adjustable fast transient rod drive used on the UCD/MNRC is a modified version of the standard fast transient rod drive. The modified design consists of a combination of the standard rack-and-pinion control rod drive and the standard fast transient control rod drive and is used on the Sandia National Laboratory TRIGA® Reactor System. This design has been thoroughly developed, tested, and operationally proven.

The only other significant difference between the UCD/MNRC reactor and others is that the reflector has been modified to accept the source-end of the beam tube. This modification is of minor significance and discussed in more detail in Chapter 10.

1.4 Safety Summary

1.4.1 Nuclear

The analyses presented in this report demonstrate that the UCD/MNRC reactor has been designed and constructed and can be operated, as described herein, without undue risk to the health and safety of UCD/MNRC employees and the general public.

The approach taken in this document to demonstrate the safety of the UCD/MNRC reactor is to:

(a) Show that the UCD/MNRC reactor fuel and instrumentation and control systems are of proven design, based on past operating experience of systems with the same or similar designs, which have been approved for operation by U.S. Government agencies;

(b) Show that the operating and accident conditions of the UCD/MNRC reactor are no greater than those of other similar reactors using the same fuel systems, and therefore present no undue risk to the health and safety of the public.

The UCD/MNRC reactor fuel, control-rod drives, control rods, and experimental systems are similar to many other systems used throughout the United States. These items have well-established operating experience and no new significant reactor-design activity was required.

The UCD/MNRC facility has been specially designed to accommodate the reactor. The tank is embedded in a massive reinforced-concrete block, which is, in turn, surrounded by the reinforced concrete walls and roofs of the radiography bays. The core is approximately 4.5 ft below ground level. The reactor shielding configuration is similar to other TRIGA® reactors. The reactor bulk shielding and the radiography bay walls and reinforced roofs provide biological shielding to keep personnel exposures as low as reasonably achievable, and protects the reactor from natural phenomena. The reactor room air handling system maintains the reactor room at a negative pressure with respect to surrounding areas to control and prevent the spread of airborne radioactive materials. The air from the reactor room passes through a HEPA filter prior to being discharged to the atmosphere. In the event of a release of radioactive material within the reactor room, the reactor room air handling system automatically isolates the room preventing the release of activity to the atmosphere. The room air can then be recirculated within the reactor room and through the HEPA and charcoal filters to remove particulates.

The reactor operates at a nominal steady-state power of 1.0 MW. The average power density is approximately 10 kW/element, whereas the same fuel has successfully operated at other facilities with power densities in excess of 30 kW/element.

The inherent safety of the reactor lies primarily in the large, prompt negative temperature coefficient of reactivity characteristic of the TRIGA® fuel-moderator material. Thus, even when large sudden insertions of reactivity are made and the reactor power rises on a short period, the prompt negative reactivity feedback produced by an increase in temperature causes the power excursion to be terminated before the fuel temperature approaches its safety limit. The prompt shutdown and safety characteristics of reactors fueled with TRIGA® fuel have been demonstrated during transient tests conducted at GA in La Jolla, California as well as other facilities. This demonstrated safety has permitted the location of TRIGA® fueled reactors in urban areas in buildings without the pressure-type containment usually required for power reactors. Chapters 4 and 13 discuss this characteristic in detail.

Abnormal conditions or postulated accidents discussed in this report (See Chapter 13) include:

a. Maximum Hypothetical Accident (MHA);
b. Reactivity insertion;
c. Loss of coolant;
d. Loss of heat-removal system;
e. Fuel cladding failure;
f. Aircraft crashes;
g. Pyrotechnic detonation.

In the first three postulated accidents (using actual measured rod worths), fuel and cladding temperatures remain at levels below those sufficient to produce cladding failure, and thus, no release of fission products would occur.

The limiting fault condition (i.e., the Maximum Hypothetical Accident (MHA), which assumes failure of fuel clad and an air release of fission products from one fuel element, will result in radiation doses to operations and base personnel and the general public for both thyroid and whole body that is orders of magnitude of those of ANS 15.7 (see Section 2.1.2 for boundary definitions). Chapter 13 contains a detailed discussion of this accident scenario.

The calculations of the probability of an airplane impacting the facility and damaging the reactor has been analyzed. It has been found that the probability of such an accident is less than 10^{-8}/year and is, therefore, considered incredible. The aircraft impact accident analysis is summarized in Chapter 13. The complete bounding probabilistic assessment of an aircraft impact risk at the former McClellan AFB is contained in Appendix C.

The amount of explosive material allowed in the radiological bays at any given time will be limited to prevent damage to the reactor (Chapters 10 and 13).

Radiation exposures to personnel working in the UCD/MNRC from both direct and airborne radiation during normal operation have been analyzed. This analysis and measurements show that the highest exposures occur when personnel are working in the radiography bays when the reactor is operating (beam tube bulk shutters closed).

All personnel entering the areas will be closely monitored, exposures kept as low as possible, and in no case will they be allowed to exceed the 10 CFR Part 20 guidelines.

The effects of Ar-41 and N-16 concentrations during normal operation of the reactor have also been evaluated for both operations personnel and the general public. These isotopes result in exposures of only a few mrem/yr to operating personnel. Their release to the atmosphere, through the UCD/MNRC stack, results in a maximum downwind concentration well below the 10 CFR Part 20 guidelines for unrestricted areas, see Chapter 11 and Appendix A for analysis.

The effects of a single fuel element clad failure in air have been evaluated for both operations personnel and the general public. The results show exposures below the 10 CFR Part 20 limits, see Chapter 13 and Appendix B for analysis.

Radiation-monitoring equipment has been installed at key locations to monitor radiation levels and to sound alarms if preset values are exceeded. Also, a system of reactor scrams, interlocks and administrative controls have been provided to prevent operating personnel from entering high radiation areas, namely the radiography bays. Included in the reactor scram chains are a number of ripcords in the radiography bays. These rip cords allow personnel in the radiography bays to terminate reactor operations if radiation levels become abnormally high.

1.4.2 Building

The UCD/MNRC reactor is housed in a building specifically designed for reactor operation. It includes the many systems needed to support this type of operation. The UCD/MNRC Reactor Facility consists of one building which houses the reactor, radiography bays, and support areas. The UCD/MNRC is a three-story facility. The exterior walls are constructed from reinforced concrete and block to a height of 24 ft, and the remaining superstructure is covered with corrugated steel. The roof is a weather-sealed steel deck. The interior walls of the radiography bays are constructed of reinforced standard concrete ranging from 2 to 3 ft thick. The roof of these areas is constructed of 2-ft thick reinforced concrete. The reactor room is constructed of standard reinforced concrete block with a built-up roof.

The structural design of the UCD/MNRC conforms to the Air Force General Design Criteria (AFM 88-15), the Uniform Building Code, the AISC Specifications, and to the ACI Code. The UCD/MNRC design seismic load is in accordance with Uniform Building Code Zone 3 criteria. The massive concrete walls and roof that surround the reactor tank provide protection from natural phenomenon. This, coupled with the fact that the reactor can withstand reactivity-insertion and loss-of-coolant accidents without release of fission products, and the low exposures associated with the design-basis accident, demonstrates that the structure is adequate for housing the UCD/MNRC reactor.

Fire detection and suppression systems have been installed throughout the facility. In addition, the instrument cabinets and the reactor and radiography control consoles have been equipped with fire suppression systems.

The reactor room and equipment room cranes have been designed and constructed in accordance with OSHA 29 CFR Part 1910.184, Overhead and Monorail Cranes. All parts have been designed for resultant static loads based on rated capacity with a factor of safety of at least five based on the ultimate strength of the material used. The fuel transfer cask lifting lugs have been designed using the ANSI/ASME code as guidelines. The design analysis shows a margin greater than six when the entire load of the cask is on one lifting lug. In addition, all of the fuel transfer hoisting equipment will be load tested, maintained and operated in accordance with ANSI/ASME during all fuel handling operations. This design, fabrication and testing approach coupled with the low exposures associated with fuel element clad failures shows that this system is adequate for its intended use (Section 9.1).

1.4.3 Shared utilities

Below is a list of various utilities and their suppliers that the MNRC requires for normal reactor operations. Given the fact the MNRC reactor is a natural convection open pool reactor, no offsite utilities are required to place the reactor in a safe and secure configuration.

Electric: The MNRC's electricity is provided by Sacramento Municipal Utilities District (SMUD). The switchgear that supplies Building 258 (MNRC) is located in Building 248 (adjacent building) electrical distribution room 48A. The switchgear in building 248 receives power from the two transformers on the north side of building 248. Each transformer feeds a section of the switchgear. Either section may provide power to building 258. Only one section may be feeding the switchgear at a time. The main circuit breaker cabinet is located on the south side of Building 258 in a gray colored cabinet. The cabinet has two sections, one for instrumentation and the other for a 3 phase, 480 AC volts, 800 ampere circuit breaker. The main circuit breaker furnishes power to a number of feeder panels throughout the facility.

Phone and Internet: Landline phones and internet are provided by AT&T and Consolidated Communication Services respectively. Though these systems are a de facto requirement to operate the facility for logistical purposes, they are not required for safe operation or shutdown of the reactor.

Water: Water used for all purposes (including the primary and secondary cooling system) is provided to the facility by a network owned and operated by Sacramento County Water District. Sewer services are also provided by Sacramento County Water District. The temporary loss of water supply does not place the reactor in an unsafe configuration. If the water supply were to be lost, the reactor operator would receive a "low water level" in the cooling tower warning in the reactor control room. A relatively quick investigation could determine if the warning was truly the result of loss of municipal water supply. If this were to be the case the reactor would quickly be shutdown to avoid excessive heating in the primary cooling system.

Natural Gas: Natural gas is used at the facility for heating and is supplied via a gas distribution network owned and operated by Pacific Gas and Electric.

1.5 Operational history

While the MNRC reactor was licensed for continuous 2.0 MW steady state operation it can be seen in Figure 4.1 that since 2007 the MNRC has essentially operated as a single shift 1.0 MW reactor. This change in reactor operation is due to historical decrease in workload and little need for higher fluxes (i.e. silicon doping). For the foreseeable future the MNRC will primarily function to support commercial neutron radiography and education/outreach programs. These programs can be accomplished by 1.0 MW single shift operations.

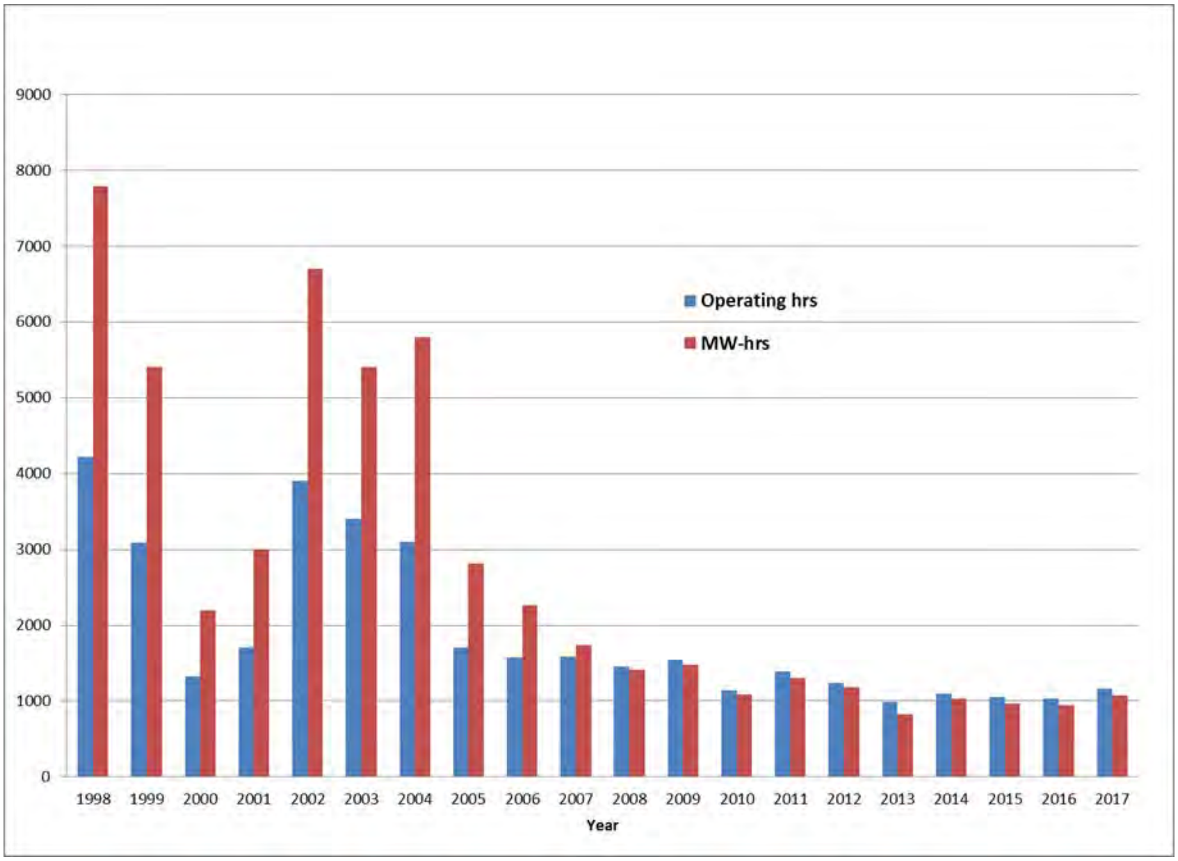

Figure 1.9 Summary of MNRC Operational History

1.6 Facility modifications

Over the past 20 years the MNRC has undergone relatively few facility modifications. This can be attributed to the fact MNRC was purpose built for neutron radiography and has not deviated significantly from that mission. In this section, facility modification of low significance and facility modification requiring NRC approval are not included. Facility modifications of low significance include air conditioning unit replacement, reroofing of the main building, upgrading internet network, and other equipment replacement unrelated to structures, systems, and components related to the operation of the reactor.

Bay 4 Reflector Insert: In February of 1999 the bay 4 beamline insert located in the MNRC reflector assembly was changed out with an insert that contained a sapphire crystal. This new reflector insert produces a much more thermalized (and attenuated) neutron beam in bay 4. The old beamline insert remains in shielded storage in radiography bay 1. The sapphire containing bay 4 beamline insert has remained in place to this day.

Bay 2 Fuel Storage Area: In January of 2003 a fuel storage area was made for a subcritical assembly containing LEU. The assembly was intended to be a flux booster placed in the reactor tank between the core and bay 5. The goal of the assembly was to boost neutron flux to provide shorter irradiation times for boron neutron capture therapy. The assembly was never placed in the MNRC reactor tank. The facility modification was closed out in 2007 when the assemble was returned to Department of Energy.

I-125 Production System: An I-125 production system was approved by the NRC is the early 2000s and operated several dozen times before operational issues became too serious to continue. The system was subsequently decommissioned in 2006. A second I-125 was planned for in the early 2010s. The program was ultimately abandoned before a license amendment was submitted to the NRC. MNRC currently has no capability or planned capability to produce I-125.

CHAPTER 2

SITE CHARACTERISTICS

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2.18. California Geological Survey Interactive Map Historical Earthquakes, Retrieved January 01, 2018, from http://maps.conservation.ca.gov/cgs/historicearthquakes/

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2.20. Toppozada, T. R. and D. L. Parke (1982). Areas damaged by California earthquakes, 1900-1949, Calif. Div. Mines Geol. Open-File Rept. 82-17, 65 pp.

2.21. USGS Quaternary Faults and Folds Database Interactive Map, Retrieved January 01, 2018,http://usgs.maps.arcgis.com/apps/webappviewer/index.html?id=db287853794f4555b8e93e42290e9716

2.0 SITE CHARACTERISTICS

This chapter provides information on the site characteristics of Sacramento County and vicinity as they relate to the safety considerations for operation of the UCD/MNRC reactor.

The conclusion reached in this chapter and throughout this document is that the selected site is well suited for the UCD/MNRC facility when considering the relatively benign operating characteristics of the reactor including the Maximum Hypothetical Accident (MHA). This is consistent with the conclusions reached for the other TRIGA® reactors operating throughout the world. Many of them are located on university campuses and other highly populated areas.

2.1. Geography and Demography

2.1.1. Site Location and Description

The UCD/MNRC reactor is located a few miles northeast of downtown Sacramento, California on the former site of McClellan AFB. Sacramento lies in the Central Valley between the coast range and the Sierra Nevada, about 90 miles northeast of San Francisco, California (Figure 2.1). The adjacent lands are located in the Great Valley subdivision of the Pacific Border Physiographic Province (Reference 2.1). The area is situated on the alluvial plains of the Sacramento River and its tributaries (Reference 2.2). The land is relatively flat, ranging in elevation from 50-75 ft (15-23 m) above mean sea level. Soil cover of about 4 ft (1.2 m) consists of sandy loam (Reference 2.3). The surface soil is moderately permeable but the subsoil has low permeability. The soils have moderate water-holding capacity and pose a slight erosion hazard.

The UCD/MNRC reactor is located approximately eight miles northeast of downtown Sacramento, California in the city of North Highlands (Figure 2.2). The reactor and the city of North Highlands are in Sacramento County, California located northwest of the intersection of Watt Avenue, Roseville Road, and I-80 and is between the communities of North Highlands- Foothills Farms, Arden-Arcade, and Rio Linda-Elverta.

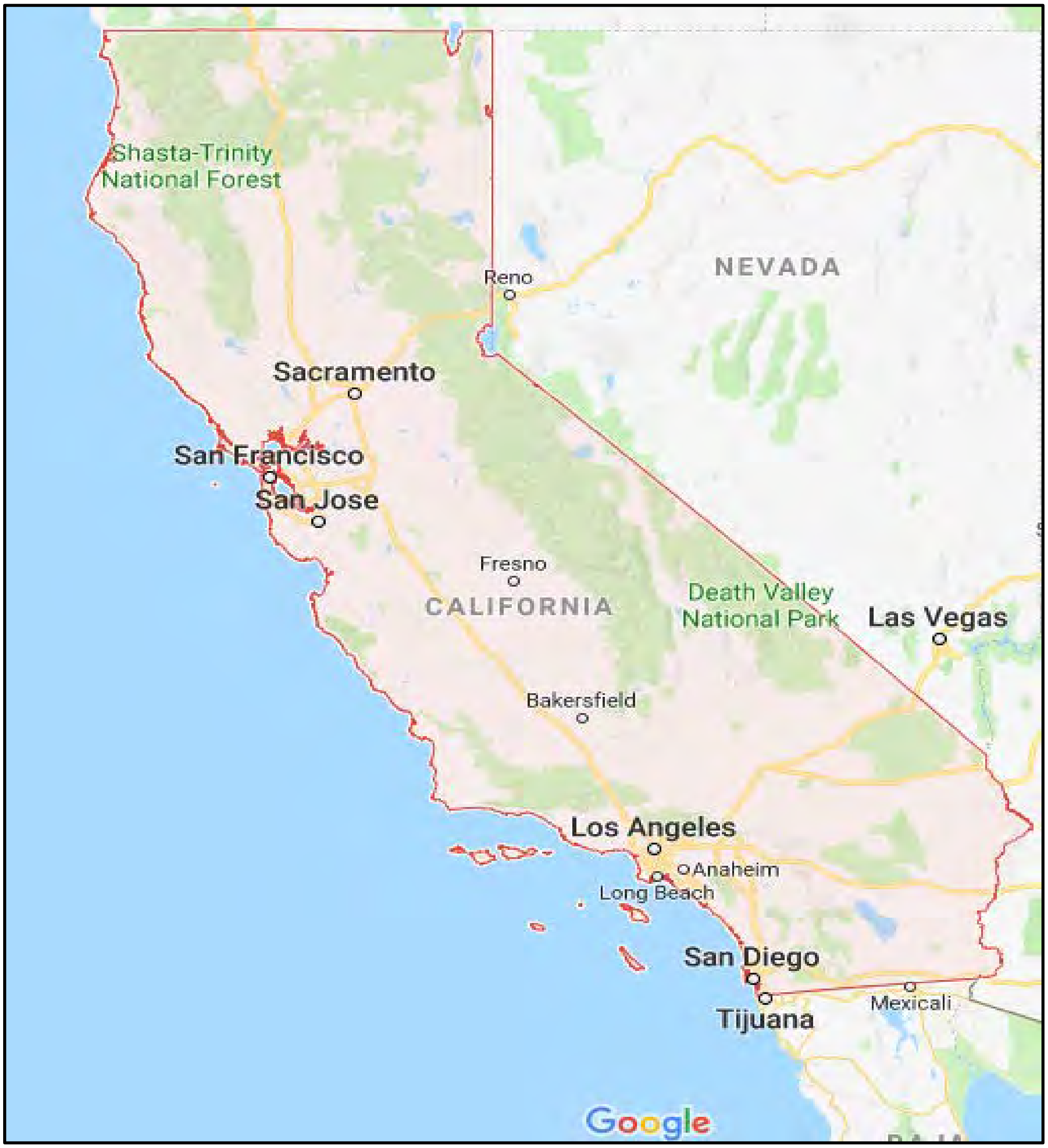


FIGURE 2.1 CALIFORNIA MAP

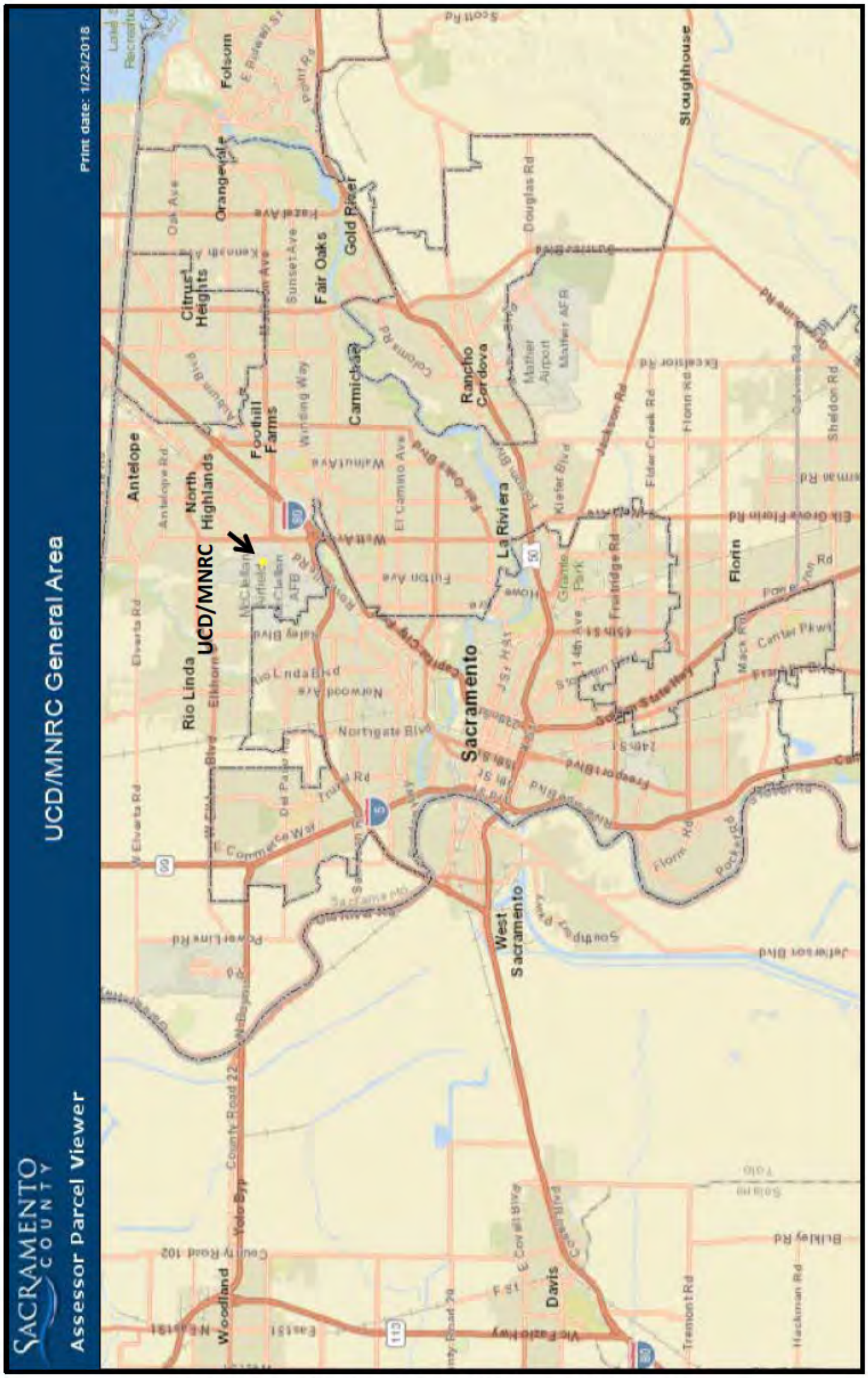


FIGURE 2.2 UCD/MNRC GENERAL LOCATION MAP

The former McClellan AFB had one active runway, 10,600 ft long and 200 ft wide, made of concrete. The south end has a 1,100 ft asphalt overrun, while the north end has a 1,000 ft asphalt overrun. The runway was capable of handling any aircraft in the Air Force inventory. The taxi- way system consists of 383,276 square yards of paved surface. Aircraft aprons total 18.9 acres.

The Air Force maintained a 1,000 ft safety zone on each side of the runway centerline, 3,000 x 3,000 ft clear zones at the ends of the runway, a 200 ft safety zone from the center of each taxi- way, and 125 ft minimum safety zone from the outside of aprons. Hazardous cargo pads are located nearby, with a 1,250 ft safety distance required between hazardous cargos and inhabited structures.

Navigational aids include high intensity runway lights, high intensity approach lighting, Visual Approach Slope Indicator (VASI) lights, Solid-State Instrument Landing System (SSILS), Area Surveillance Radar (ASR), VHF Omni-range and Tactical Navigation Station (VORTAC), and UHF transmitters and receivers.

During the 26 years, from 1970 to 1995, annual aircraft operations at McClellan AFB varied from a low of 43,516 to a high of 104,955. During the 26 year period, there were a total of 1,955,788 operations, which is an annual average of 75,223. The following table summarizes these operations.

ANNUAL AIRCRAFT OPERATIONS 1970-1995[1]

1970	104,955	1979	68,858	1988	83,333
1971	99,927	1980	76,467	1989	85,826
1972	98,125	1981	82,985	1990	78,811
1973	91,081	1982	87,713	1991	59,055
1974	84,720	1983	79,251	1992	52,138
1975	75,404	1984	76,381	1993	58,593
1976	58,734	1985	72,160	1994	50,717
1977	57,180	1986	90,175	1995	43,516
1978	58,822	1987	80,861		

26 year total = 1,955,788
Annual Average = 75,223

[1]Source:.McClellan Control Tower

During the last 17 years, from 2000 to 2016, annual aircraft operations at McClellan Air Field are substantially lower than shown in the previous table. During this latest 17 year period, there were a total of 103,518 operations, which results in an annual average of 6,089, a factor of 12 less than the average annual operations from earlier years. The following table summarizes these latest numbers (Reference 2.17).

ANNUAL AIRCRAFT OPERATIONS 2000 – 2016[2]

2000	1,404	2009	6,507
2001	2,396	2010	7,264
2002	4,716	2011	6,216
2003	6,028	2012	6,008
2004	6,039	2013	5,823
2005	6,712	2014	6,094
2006	7,456	2015	7,340
2007	8,283	2016	7,534
2008	7,698	2017	NA

17 year total = 103,518
Annual Average = 6,089

Use of airspace around the former McClellan AFB is controlled by the Federal Aviation Administration (FAA). In the Sacramento area, responsibility for control of terminal airspace lies with four local air traffic control (ATC) towers located at Metropolitan, Mather Field, Executive airports, and Beale Air Force Bases. Responsibility for transitional area airspace lies with the Terminal Radar Approach Control Facility (TRACON) located adjacent to the former McClellan AFB. The Air Route Traffic Control (ARTCC), located in Fremont, is responsible for enroute airspace.

The UCD/MNRC facility is located as shown in Figures 2.3 and 2.4. It is approximately 1,800 ft (550 m) to the east of the former main runway and 3,000 ft (915 m) from the nearest urban area, Watt Avenue to the east. The next closest urban area is “E” street approximately 4,500 ft (1,375 m) to the north. This area of the former base is the location of other large repair shops.

[2]Source: Federal Aviation Administration, Traffic Flow Management System Counts (TFMSC), Aviation System Performance Metrics (ASPM) https://aspm.faa.gov/tfms/sys/Airport.asp

2.1.2. Exclusion Area Authority and Control

From the UCD/MNRC normal operations, safety, and emergency action standpoint there are two areas of concern (Figures 2.5, and 2.6). The first is the area inside the perimeter fence (with outriggers and barbed wire) surrounding the UCD/MNRC enclosure. This area is the exclusion area. It is a "restricted access" area and control of activities within this area during normal operations and emergencies is the responsibility of the UCD/MNRC Director/Emergency Director.

The second area of concern is the area outside the UCD/MNRC perimeter fence. The general public has free access to this area and direction of emergencies is by local city/county civilian authorities. This is defined as the unrestricted area. The closest urban area to the UCD/MNRC is about 3000 m to the east, Watt Avenue.

The area definitions for purposes of addressing radiation exposure in Chapter 11 are restricted and unrestricted areas. The operations boundary (i.e., the UCD/MNRC perimeter fence) is the boundary between these areas; inside the fence is the restricted area and outside the fence is the unrestricted area.

FIGURE 2.3 MCCLELLAN INDUSTRIAL PARK GENERAL AREA

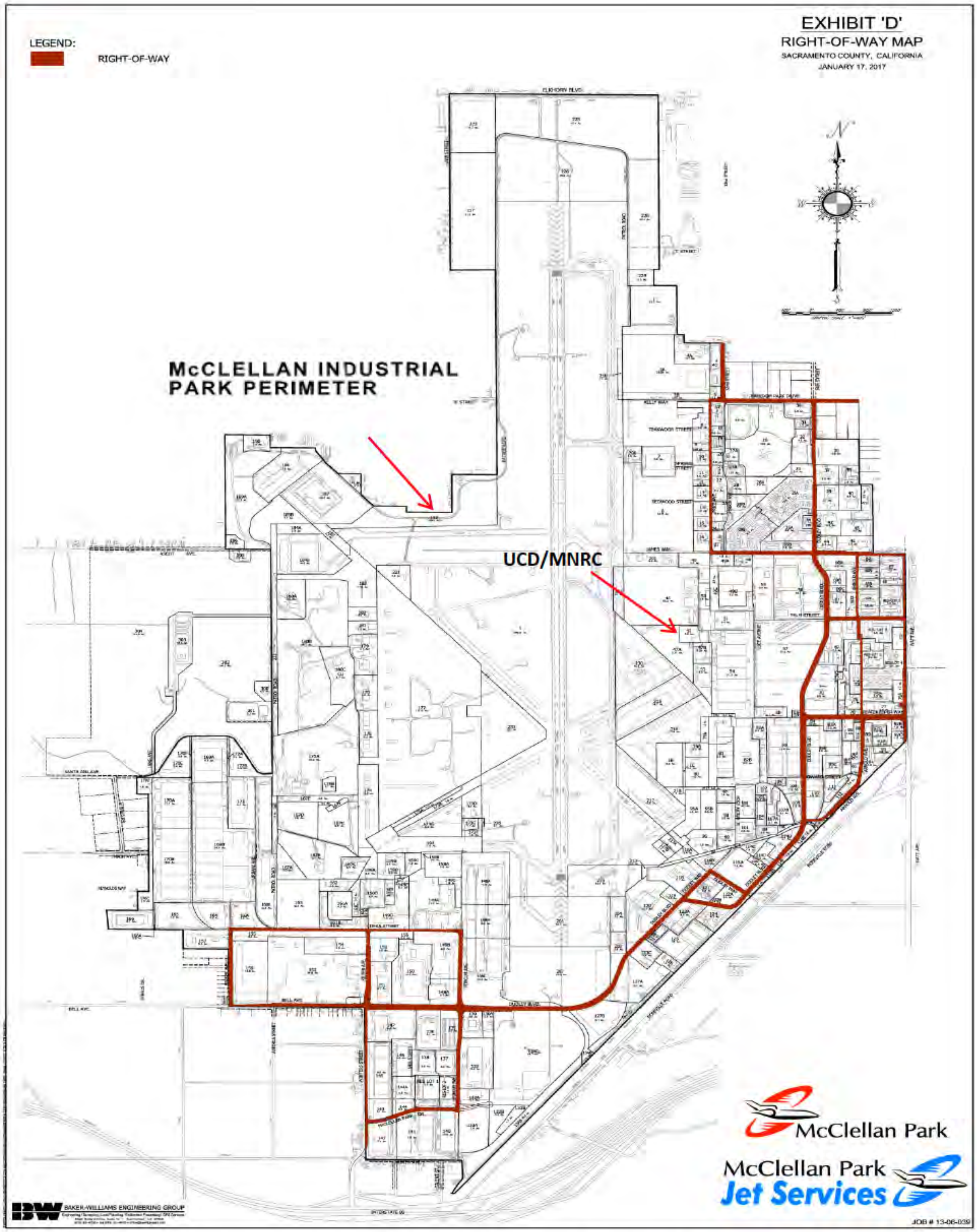


FIGURE 2.4 MAP OF MCCLELLAN INDUSTRIAL PARK

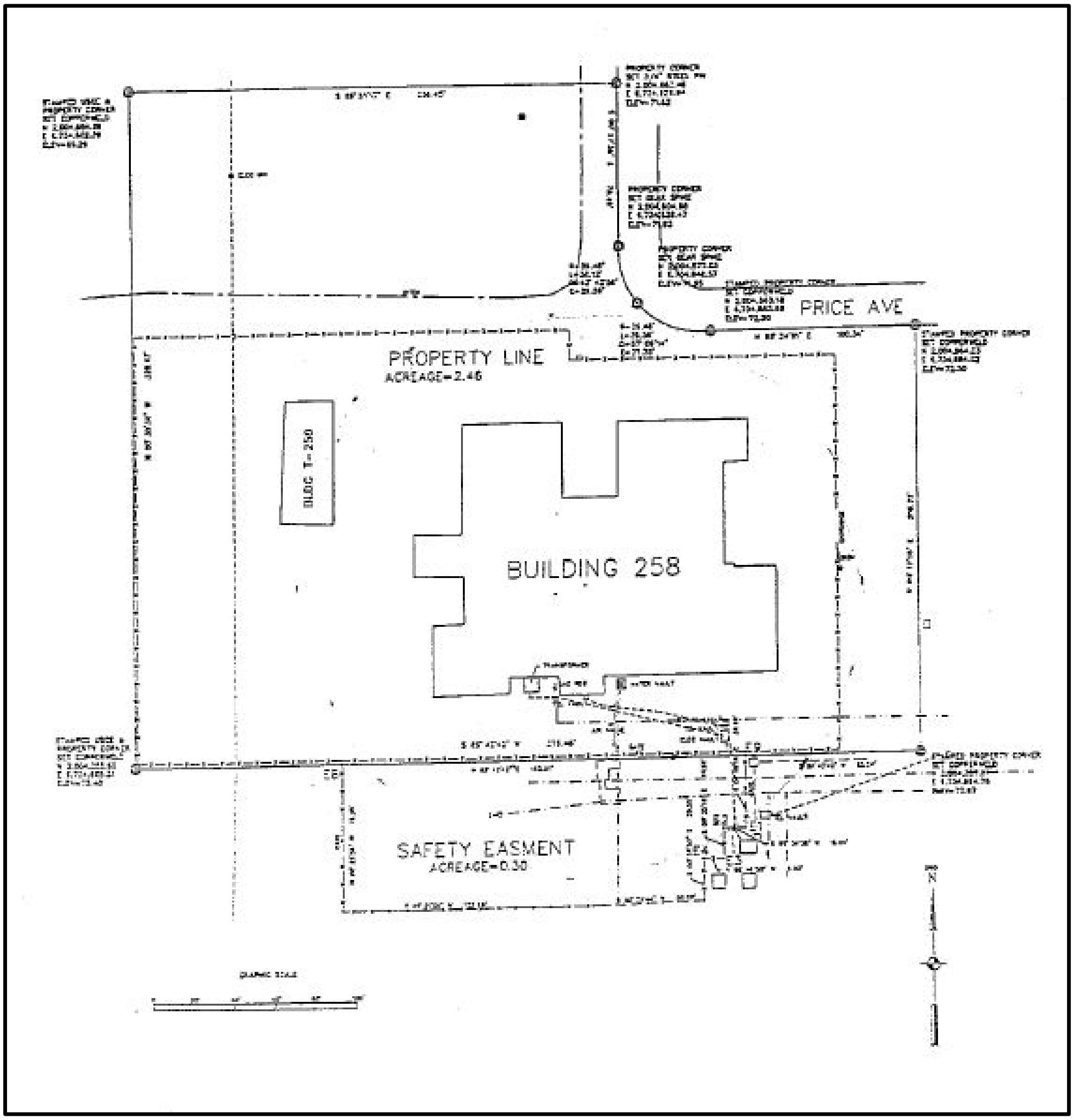


FIGURE 2.5 UCD/MNRC PLOT PLAN

FIGURE 2.6 TOP AND AXONOMETRIC VIEW MAIN UCD/MNRC FACILITY

2.1.3. Population Distribution

The UCD/MNRC is situated approximately 8 miles (13 km) north-by-northeast of downtown Sacramento, California. Sacramento county has a population of about 1,418,788 (2010 census) , an increase of about 23% since the 1992 census which reported a population of about 1,093,000, which was in turn an increase of about 26% since 1970 (References 2.4, and 2.16). The major population center of Sacramento lies south-by southwest of the McClellan Business Park. (References 2.15, 2.16).

The UCD/MNRC is surrounded by communities. To the east and northeast is North Highlands; to the northwest, Rio Linda; to the west is North Sacramento and Del Paso Heights (which are generally considered as part of Sacramento City according to the US Census Bureau); and to the south is Arden-Arcade. The highest density developments are directly to the east, in North Highlands; to the southwest, in the Del Paso Heights area of the city of Sacramento; and to the south in Sacramento County.

Existing land uses, according to the Sacramento Area Council of Governments Comprehensive Land Use Plan (1987), around the base were generally:

<u>North Highlands</u>: Mostly single family residential development of about six units per acre, with retail and other business uses centered along Watt Avenue and Elkhorn Boulevard. There are also some commercial and light industrial uses along Elkhorn Boulevard. Scattered in the residential areas within about a mile of the UCD/MNRC, are 12 elementary schools. Some multi-family housing is scattered in the area. Industrial development is centered along Orange Grove Avenue.

<u>Rio Linda</u>: Mostly single family residential uses with few retail or business uses. There are four elementary schools, a junior high school, and a high school located in this community. There is also a small airport located about two miles to the west of the UCD/MNRC.

<u>North Sacramento</u>: Mostly single family residential uses, with lower densities near Rio Linda and high densities in Del Paso Heights. There are some commercial and business uses along Marysville Boulevard and Grand Avenue. There are nine elementary schools, a junior high school, a high school, and a hospital located in this area.

<u>Arden-Arcade</u>: A highly urbanized area with single family and multiple family residential uses, with retail, commercial, and business uses centered on arterial streets: Marconi Avenue, El Camino Avenue, Howe Avenue, Auburn Boulevard, Fulton Avenue, Watt Avenue, and Arden Way. There are 13 elementary schools, three high schools, a hospital, major shopping centers, and a community park located in the area.

No significant population variations due to transient population or transient land use occur in the area surrounding the UCD/MNRC. Although there are some recreational areas within 10 miles (16 km) of the UCD/MNRC, none attract large numbers of people and most are used by local residents.

Table 2-1 portrays the 2010 census demographics of the surrounding counties, and neighboring areas. Figures 2.7 through 2.11, and 2.12 depict the general areas of interest shown in Table 2-1, the regional airport locations (which will be discussed in more detail in the next section), and current general land use according to Sacramento County.

TABLE 2-1 Sacramento Valley Region Census Data

	Sacramento city	Arden-Arcade	North Highlands	Rio Linda	Sacramento County	Placer County	Sutter County	Yolo County	Solano County
Population, Census, April 1, 2010	466,488	92,186	42,694	15,106	1,418,788	348,432	94,737	200,849	413,344
Persons under 5 years, percent, April 1, 2010	7.50%	6.40%	8.20%	6.70%	7.10%	6.00%	7.60%	6.30%	6.50%
Persons under 18 years, percent, April 1, 2010	24.90%	20.90%	27.70%	27.10%	25.60%	24.40%	27.60%	22.70%	24.60%
Persons 65 years and over, percent, April 1, 2010	10.60%	15.70%	11.10%	10.40%	11.20%	15.40%	12.70%	9.80%	11.30%
Veterans, 2012-2016	23,480	5,586	2,922	903	85,177	27,830	6,393	8,827	33,653
Housing units, April 1, 2010	190,911	44,813	16,093	5,129	555,932	152,648	33,858	75,054	152,698
Owner-occupied housing unit rate, 2012-2016	46.80%	43.90%	43.10%	74.70%	55.20%	70.10%	57.10%	51.30%	59.20%
Median value of owner-occupied housing units, 2012-2016	$259,400	$309,300	$157,600	$194,200	$271,300	$380,900	$213,100	$346,200	$305,900
Households, 2012-2016	179,865	41,028	15,244	4,747	527,335	136,730	32,028	72,544	145,315
Persons per household, 2012-2016	2.65	2.33	3.05	3.13	2.76	2.68	2.95	2.78	2.88
In civilian labor force, total, 2012-2016	62.90%	61.80%	55.40%	54.40%	62.30%	60.50%	59.20%	60.10%	62.10%
Total retail sales, 2012 ($1,000)	4,363,259	1,986,474	550,287	51,730	15,227,280	6,437,879	1,069,489	1,968,762	5,106,627
Mean travel time to work (minutes),2012-2016	24.9	23.9	26.6	27.9	26.5	27.1	26.2	22.4	30.8
All firms, 2012	38,244	8,360	3,531	1,064	110,172	31,846	5,461	13,884	25,724
Population per square mile, 2010	4,764.20	5,170.60	4,834.60	1,525.20	1,470.80	247.6	157.3	197.9	503
Land area in square miles, 2010	97.92	17.83	8.83	9.9	964.64	1,407.01	602.41	1,014.69	821.77
FIPS Code	"0664000"	"0602553"	"0651924"	"0660942"	"06067"	"06061"	"06101"	"06113"	"06095"

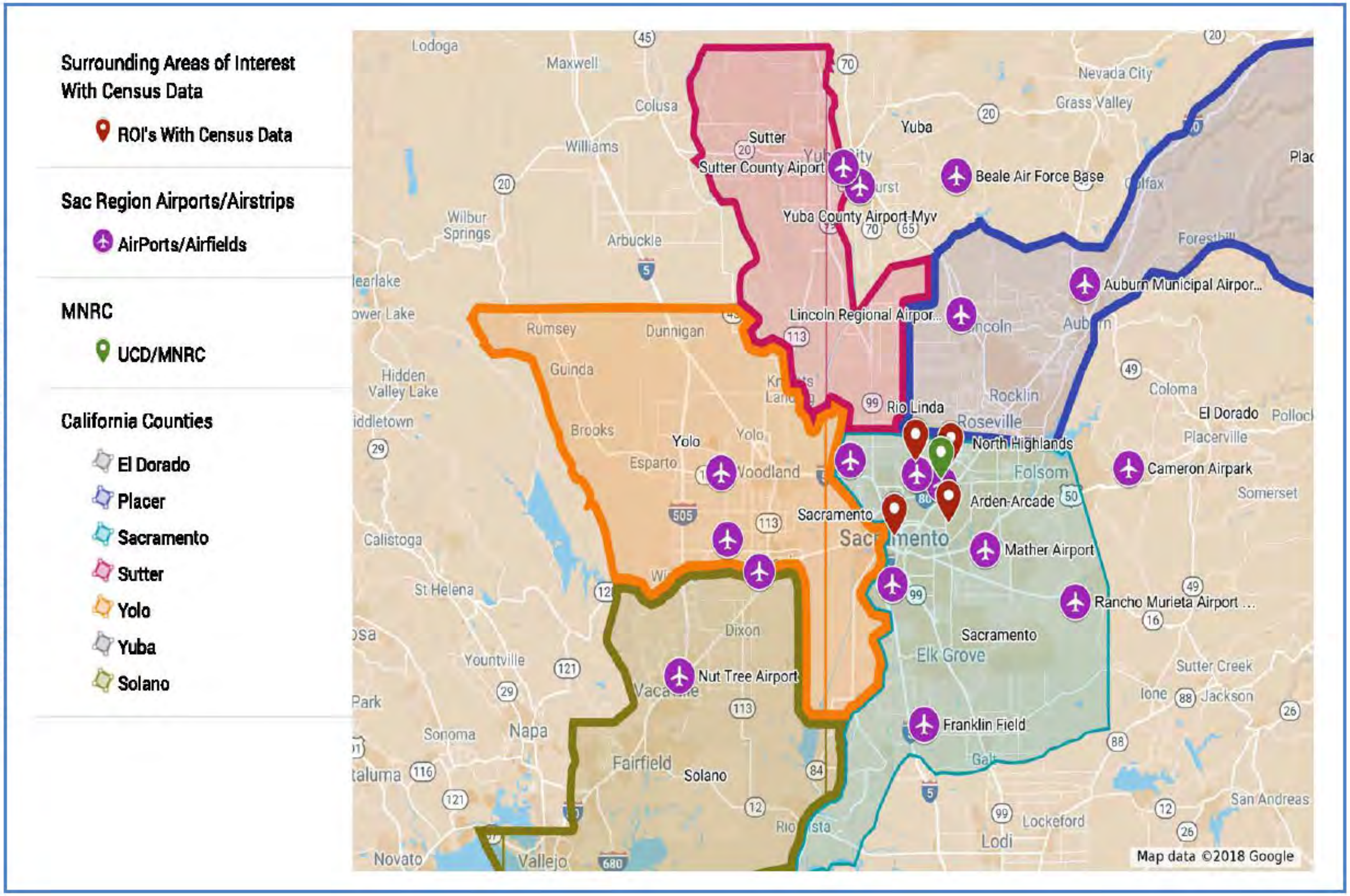
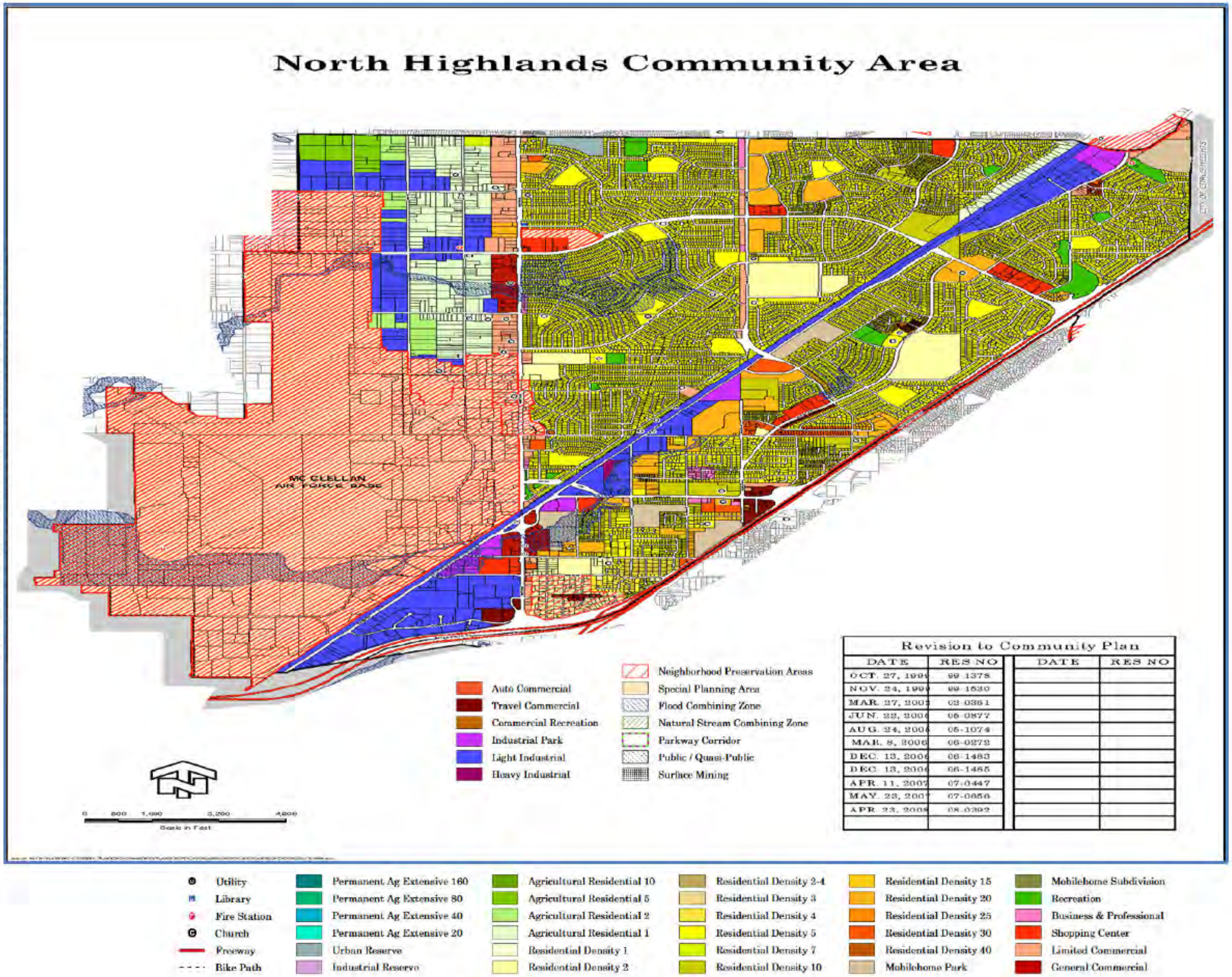


FIGURE 2.7 REGIONS OF INTEREST FOR CENSUS DATA



Revision to Community Plan			
DATE	RES NO	DATE	RES NO
OCT. 27, 1999	99 1378		
NOV. 24, 1999	99-1630		
MAR. 27, 2003	03-0361		
JUN. 22, 2006	06-0877		
AUG. 24, 2005	05-1074		
MAR. 8, 2006	06-0272		
DEC. 13, 2006	06-1483		
DEC. 13, 2006	06-1485		
APR. 11, 2007	07-0447		
MAY. 23, 2007	07-0656		
APR. 23, 2008	08-0392		

FIGURE 2.8 NORTH HIGHLANDS LAND USE MAP

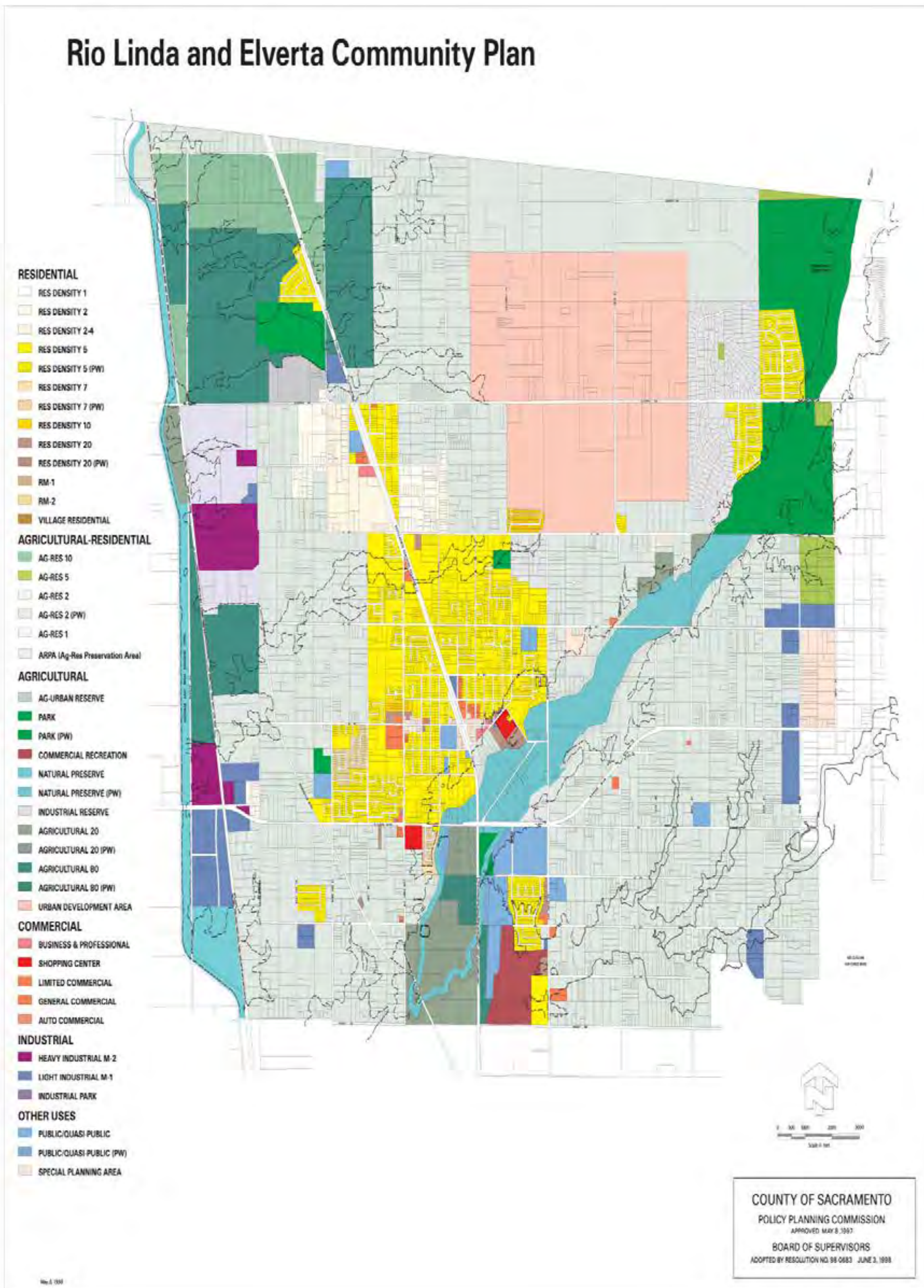


FIGURE 2.9 RIO LINDA LAND USE MAP

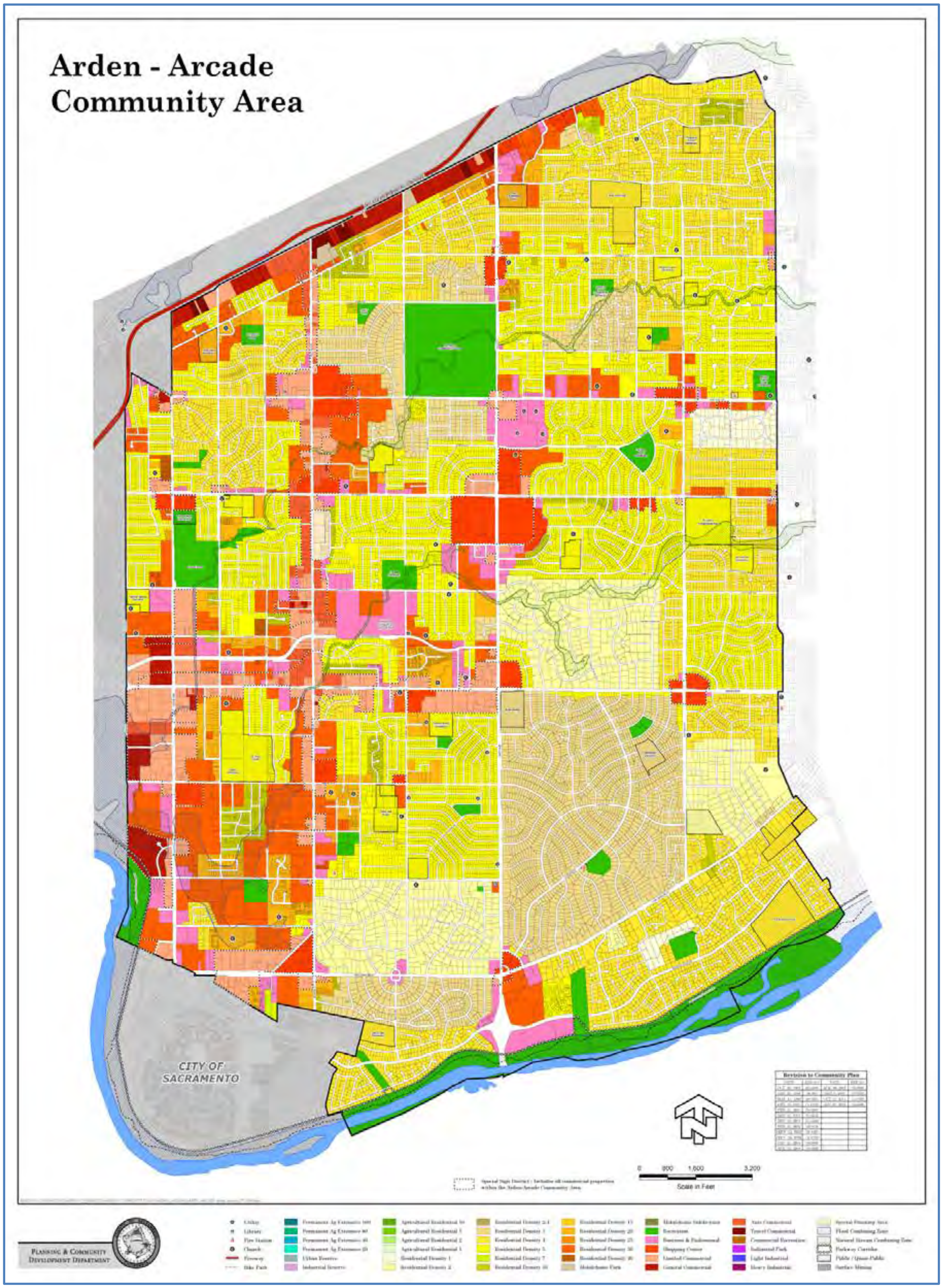


FIGURE 2.10 ARDEN ARCADE LAND USE MAP

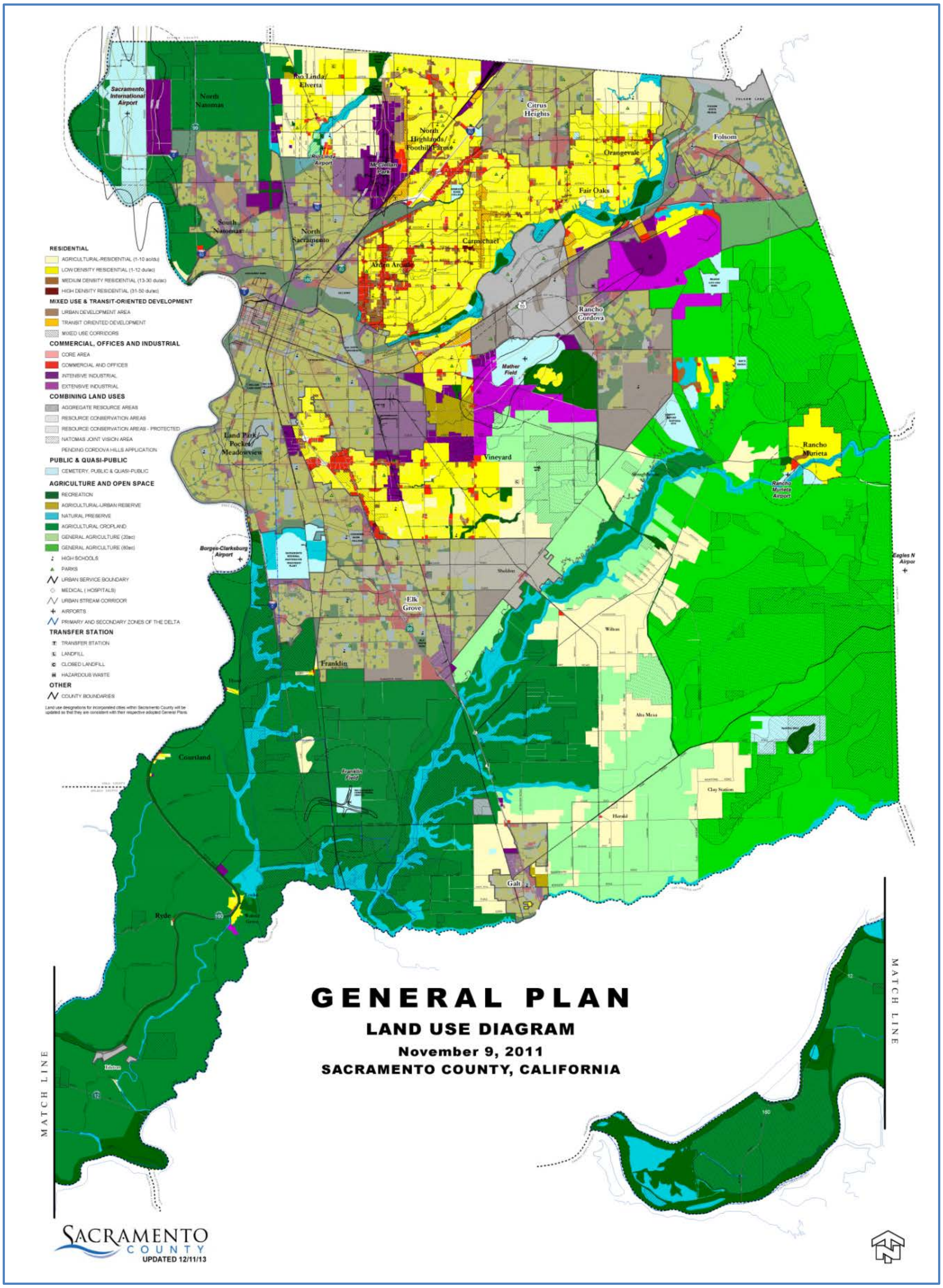


FIGURE 2. 11 SACRAMENTO COUNTY LAND USE

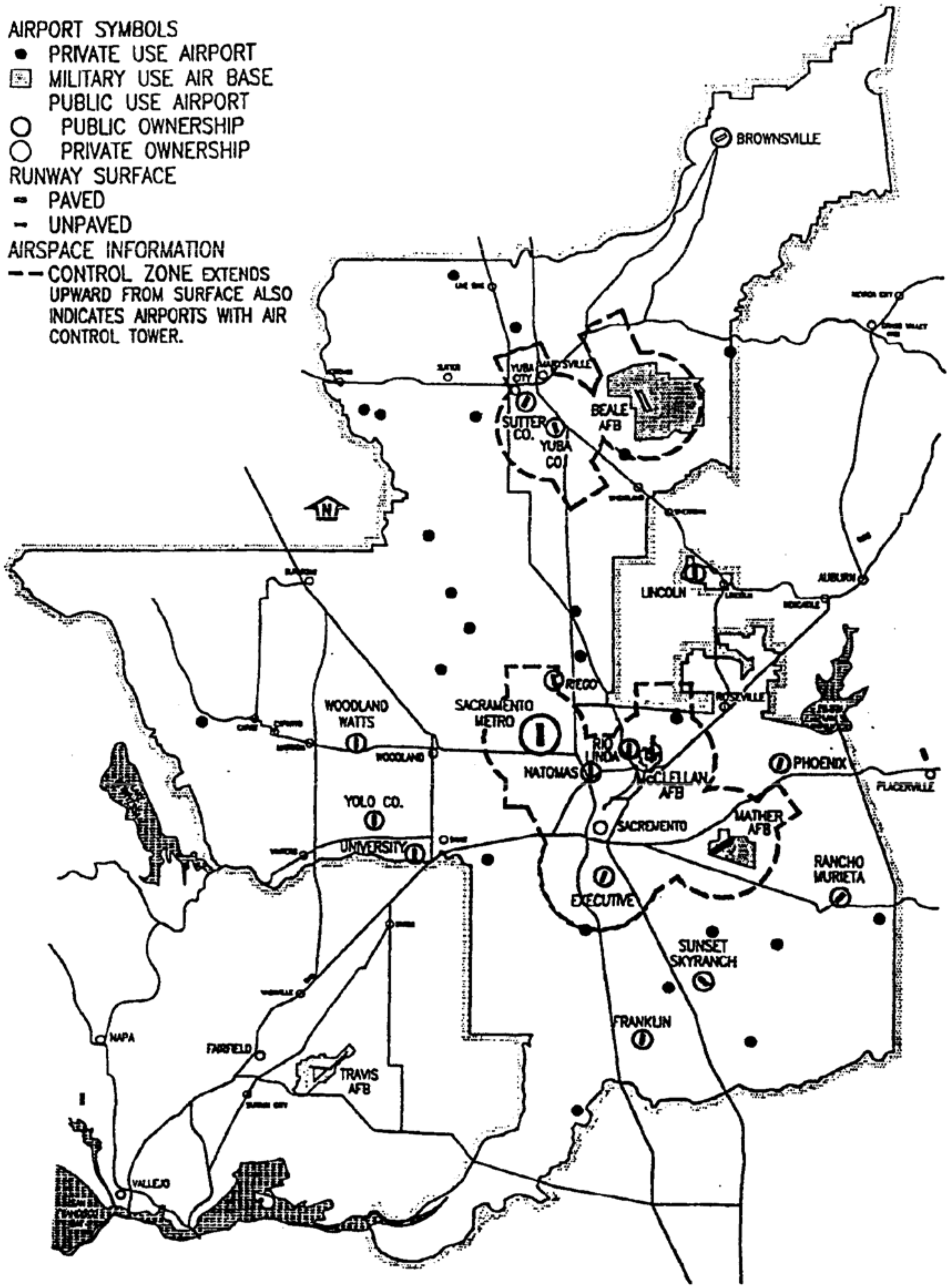


FIGURE 2. 12 REGIONAL AIRPORT SYSTEM – 1983

2.2. Nearby Industrial, Transportation, and Military Facilities

2.2.1. Industry

There are no major industrial facilities in the Sacramento area that need be of concern from the UCD/MNRC safety standpoint. The area's economy is primarily based on agriculture and government with much smaller contributions by such things as mining, manufacturing of durable goods, lumber and wood products, and metal fabrication. The closest oil refinery is located at Martinez, California, approximately 85 miles to the southwest.

The majority of local business within 5 miles of MNRC containing hazardous materials are site are associated with retail gas stations, dry cleaners, and other light manufacturing industry. None of these business could produce an accident that would be detectable at MNRC. There are no large industrial facilities such as chemical producers, fertilizer plants, explosives manufactures, etc. that could have a significant enough accident that could produce adverse effects at MNRC.

Though it is not thought that any facility on the McClellan Business Park could produce an accident that could adversely affect safe operation at MNRC, these operations are nonetheless described below.

Sacramento Municipal Utility District maintains 250 MWe gas turbine power plant that is operated when demand is high on the local power grid. The facility is located approximately 1 mile south of the MNRC. The facility contains various lubricant oil and transformer oil associated with the plant's operations. The natural gas used to operate the plant is piped in and not stored onsite.

Both Cal Fire and the US Coast Guard are located between 1 and 2 miles north of the MNRC. Both facilities perform various air operations and contain hazardous materials required to maintain aircraft. Both facilities and other (primarily civilian) aircraft are refueled by McClellan Jet Services (MJS). MJS maintains up to 12,000 gallons of aviation gasoline and up to 1,200,000 gallons of jet fuel at various locations around the airfield. The risk of fire from various aircraft operations at the McClellan Business Park is similar to the risk at other airfields. Meaning a fire is possible but generally considered to be very unlikely. The closest refueling point to MNRC is 0.5 miles north (Cal Fire) and 0.5 miles south (civilian aircraft). There is little combustible material between MNRC and these refueling points. If a significant fire broke out in the vicinity of MNRC it would pose little threat to MNRC, though the reactor would likely be secured as a precaution prior to evacuating the facility.

Interstate Oil (located 2 miles south of MNRC) maintains storage tanks that hold up to 1,200,000 gallons of diesel oil at a given time in above ground tanks. Given the low flammability of diesel and the significant distance between Interstate Oil and MNRC, there is no credible accident at Interstate Oil that could produce adverse effects at MNRC.

Pyro Spectacular North is a fireworks company that operates a manufacturing facility approximately 2 miles west of MNRC. The facility contains up to 100,000 pounds of pyrotechnic material at a given time. As with any pyrotechnic facility of this size, the risk of a large scale accident is not trivial. However, given the 2-mile separation distance along with the fact that little blast over pressure would be generated by even the most severe accident at the firework facility, there appears to be no credible mechanism that could adversely affect safe operations at MNRC.

2.2.2. Transportation

- Highway Transportation

The Sacramento area is at the cross-roads of two interstate highways: the transcontinental I-80, and N/S I-5. I-80 goes to San Francisco to the west, and to Reno to the east. Business 80 passes through the downtown area and connects with I-80 in west Sacramento, and in northeast Sacramento at Watt Avenue. Three main gates into the Industrial Park are located on Watt Avenue about a mile north of the I-80/Watt Avenue intersection.

Interstate 5 passes through downtown near the Sacramento River; traveling north, it leads to Oregon and Washington; south I-5 leads to Los Angeles and San Diego.

Highway 50 links the downtown area to points east; Rancho Cordova, Folsom, El Dorado Hills, Placerville, and South Lake Tahoe.

State Highway 99 generally parallels I-5 to southern California, joining I-5 south of Bakersfield.

- Airports

As of 1983 there were 71 airports within the Sacramento Area Council of Governments, SACOG, the Region on which records were kept. Of those, 16 were public use, 53 were private, and two (including the former McClellan AFB) were military. Since the early 80's the use of the aircraft runways to present day, at the former McClellan AFB site, continued to involve military; however, it now includes commercial usage. The current usage of the runways was shown earlier in this chapter to be significantly less than the former military usage in the 80's.

Table 2-2 and 2-3 (Reference 2.17) summarizes the items of interest for the public use and military airports in the SACOG region as of 1983 and then 2016 for comparison. The location of these airports with respect to the UCD/MNRC is shown in Figure 2.7 and 2.12.

- Water Transportation

Sacramento has the largest river system in California. A ship channel between Rio Vista and Sacramento was dredged by the Army Corps of Engineers where there was an existing lake area. It is the Port of Sacramento, operated by the Sacramento-Yolo Port district, and lies 79 nautical miles from the Pacific Ocean and approximately 11 miles from the UCD/MNRC.

Since its opening in 1963, the port has developed extensive cargo storage and handling facilities, largely focusing on rice, wheat, and wood chips commodities.

- Rail Transportation

Union Pacific operates the tracks that parallel Roseville Road, and along the south-east corner of the Industrial Park. The closest approach to the reactor facility is approximately 3500 feet. Union Pacific connects Sacramento with 21 western, central, and southern states. On a daily basis there are nine scheduled AMTRACK passenger trains and fourteen freight trains that utilize the tracks just southeast of the reactor facility. All shipments aboard these trains are in accordance with the Code of Federal Regulations 49 CFR - Transportation. All normal shipments are not expected to threaten the reactor facility. The California State Office of Emergency Services has the UCD/MNRC listed as a critical facility for notification during planning of any hazardous material shipments along this route.

There are other feeder, connector and inter-tie services provided to the Sacramento area by Sacramento Northern and Central Traction Company and Western Pacific Railroad. However, these facilities are all to the south and beyond consideration.

2.2.3. Military Facilities

There is one military facility in the vicinity of Sacramento: Beale AFB. Beale AFB is located in Yuba County approximately 13 miles east of Marysville and 75 miles from McClellan. The present 12,000 ft x 300 ft runway was completed in 1959. As of the late 90's, early 2000's, Beale AFB employed approximately 4,800 military and civilian personnel.

Three very different major operations are housed at Beale AFB. These are an air refueling mission, a reconnaissance wing, and missile warning squadron. These operations use four different types of aircraft, each with varying speeds and airspace requirements. The aircraft presently used are: U-2 High Altitude Photographic, T-38 Trainer, and RC-12 SIGINT.

The facilities at Beale AFB are not available for use by the general aviation or air carrier operators. There were an estimated 85,000 aircraft operations at Beale AFB in 1981.

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TABLE 2-2 SUMMARY OF PUBLIC USE AND MILITARY AIRPORTS IN SACOG REGION 1983

FAA ID - Airport	Use	Runways	1983 Total Operations
BAB - Beale AFB	Military	12,000' x 300'	85,000
BRO - Brownsville Airport*	Public	2,240' x 40'	3,800
F72 - Franklin Field	Public	3,105' x 60' 3,030' x 60'	70,000
LHM - Lincoln Regional	Public	6,000' x 100'	40,000
MHR - Mather	Public	11,300' x 300'	25,600
MCC - McClellan Airfield	Military	10,600' x 200'	88,000
Natomas Field*	Public	2,600' x 30'	20,000
RIU - Rancho Murieta	Public	3,800' x 75'	15,500
Riego Flight Strip*	Public	2,380' x 35'	4,500
L36 - Rio Linda	Public	2,620' x 30'	34,000
SAC - Sacramento Executive	Public	5,503' x 150' 3,482' x 150' 3,834' x 100'	115,000
SMF - Sacramento	Public	8,600' x 150'	175,000
Q40 - Sunset Skyranch*	Public	2,780' x 150'	18,000
O52 - Sutter County	Public	3,040' x 75'	352,000
EDU - UC Davis	Public	3,185' x 50'	37,300
O41 - Watts-woodland	Public	3,770' x 60'	63,000
DWA - Yolo County	Public	6,000' x 100'	60,000
MYV - Yuba County	Public	6,000' x 150'	63,000

*No longer in operation, existence, or no current FAA records exist as of 2016

TABLE 2-3 SUMMARY OF PUBLIC USE AND MILITARY AIRPORTS IN SACOG REGION 2016

FAA ID - Airport	***Use***	***Runways***	***2016 Total Operations***
AUN - Auburn Muni	Public	3700' x 75'	935
BAB - Beale AFB	Military	12,000' x 300'	7806
DWA - Yolo County	Public	6000' x 150'	870
EDU - UC Davis	Public	3176' x 50'	359
F72 - Franklin Field	Public	3123' x 60' 3031' x 60'	7
L36 - Rio Linda	Public	2625' x 42'	17
LHM – Lincoln Regional	Public	6,001' x 100'	1130
MCC - McClellan Airfield	Public	10,600' x 200'	7534
MHR - Mather	Public	11301' x 150'	16622
MYV - Yuba County	Public	6007' x 150'	678
O41 - Watts-woodland	Public	3759' x 60'	234
O52 - Sutter County	Public	3045' x 75'	24
O61 - Cameron Airpark	Public	4051' x 50'	276
RIU - Rancho Murieta	Public	3798' x 75'	99
SAC - Sacramento Executive	Public	5503' x 150' 3837' x 100' 3505' x 150'	7120
SMF - Sacramento	Public	8605' x 150' 8598' x 150'	104673
VCB - Nut Tree	Public	4700' x 75'	1043

2.2.4. Evaluation of Potential Accidents

There are no nearby industrial, transportation, or military facilities with the potential of causing a credible accident that would result in a release of radioactive material from UCD/MNRC that would exceed the general public exposure limits of 10 CFR Part 20.

The basic UCD/MNRC design and structure provide significant protection for the reactor. As described in Chapter 1, the reactor core is below grade and surrounded by a monolithic block of reinforced concrete from six to nine ft thick. Also, the above grade structures of the facility that surround the reactor tank are constructed of reinforced concrete and reinforced concrete block.

The accident, from sources outside the UCD/MNRC that is worthy of further discussion is one involving an aircraft since the facility is located near an airstrip. The possibility of an aircraft impact involving the UCD/MNRC reactor has been evaluated (assuming airstrip operation numbers from the early 80's as they are more conservative that current values), see Chapter 13, and it has been determined that the probability of such an event occurring is less than 10^{-8} per year. Therefore, this type of accident is considered incredible.

2.3. <u>Meteorology</u>

2.3.1. Regional Climatology

Sacramento is situated in California's Central Valley between the Sierra Nevada and Coastal Range. The area is characterized by hot summers (July mean maximum temperature 105°F) and cold winters (January mean minimum temperature 28°F) (Reference 2.6). As in most of California, the majority of the annual average precipitation, about 17 in. (40 cm), falls in the winter months as rain. The prevailing winds in the area are from the south to south-by- southeast.

The eastern most mountain chains form a barrier that protects much of California from the extremely cold air from the Great Basin in the winter. There are occasions when cold air from an extensive high pressure area spreads westward and southward over California. Even in these cases, the warming by compression as the air flows down the slopes of the mountains into the valleys prevents severe cold damage. The ranges of mountains to the west offer some protection to the interior from the strong flow of air off the Pacific Ocean. Between the two mountain chains and over much of the desert area the temperature regime is intermediate between the maritime and the continental models. Hot summers are the rule while winters are moderate to cold.

2.3.2. Local Meteorology

The summary of meteorological conditions for the UCD/MNRC site is based on the records obtained by officials of the National Oceanic and Atmospheric Administration, U.S. Department of Commerce and published in Volume II of "Climates of the States." The specific data, for the most part, is from the weather station at the Sacramento Executive Airport.

2.3.2.1. Temperatures

The normal and extreme temperatures for the Sacramento area are shown in Table 2-4. The normal temperatures are climatological standard normal (1931-1960). The normal daily minimum temperature of 37.2°F occurs in January and the normal daily maximum temperature, 93.4°F, occurs in July. Extreme temperatures have ranged from a low of 23°F in January of 1963 to 115°F in June of 1961.

2.3.2.2. Precipitation

The normal precipitation for the Sacramento area is 16.29 in./yr with the highest amounts, approximately 3.2 in. occurring in the months of December and January. The maximum monthly rainfall, 12.64 in., fell in December 1955. The maximum rainfall over a 24-hour period of time, 5.59 in., occurred in October 1962.

2.3.2.3. Humidity

The humidity in the Sacramento area range from a low of 28% in July to a high of 91% in December and January.

2.3.2.4. Winds and Stability

The annual wind rose for the Sacramento area is shown in Figure 2.13 and 2.14. The data to prepare these two wind roses was collected for the periods 1953-57 and 1992-96. As can be seen, the prevailing winds in the area are from the south to south-by-southeast.

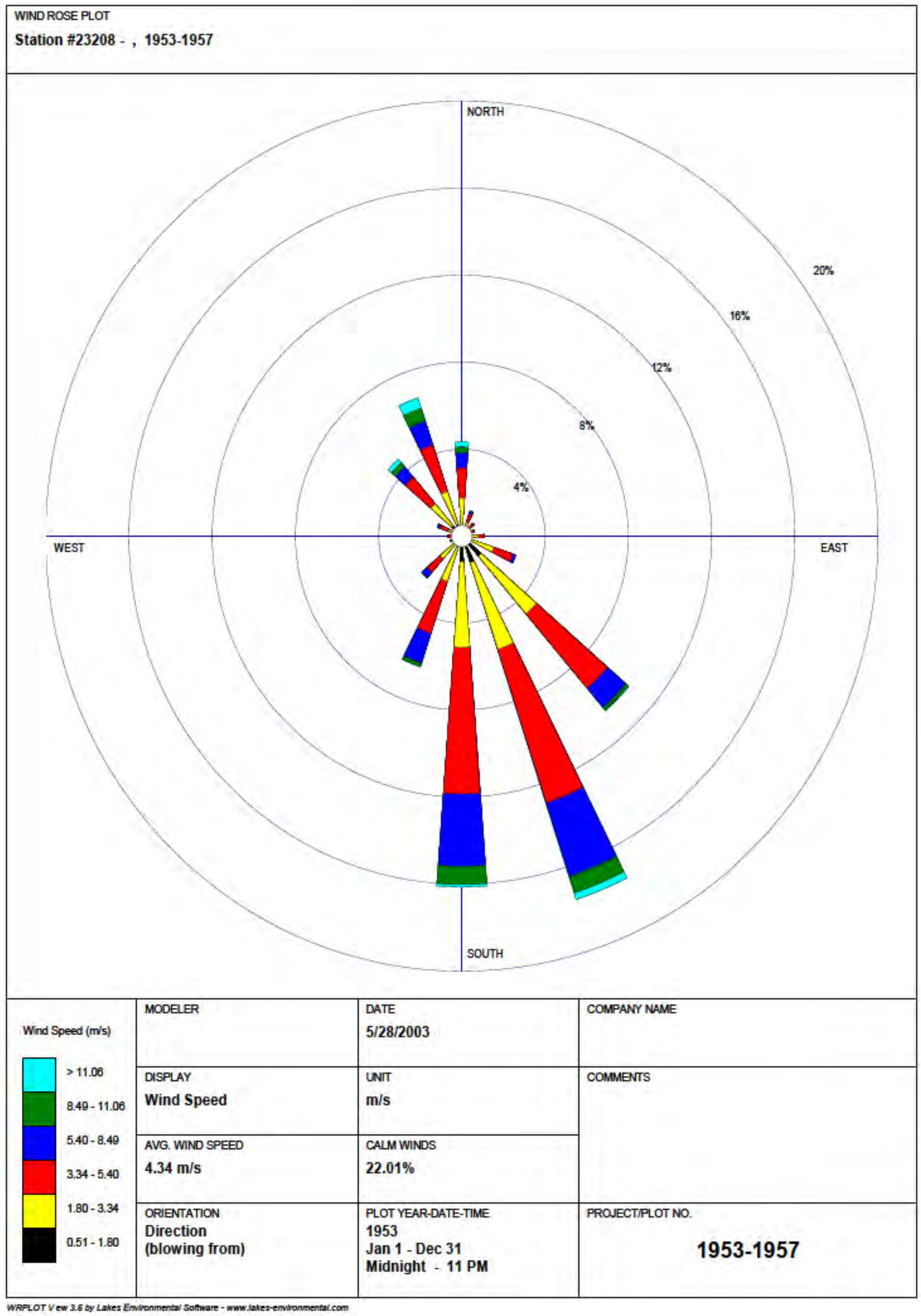

FIGURE 2.13 ANNUAL WIND ROSE FOR THE FORMER MCCLELLAN AFB 1953-1957

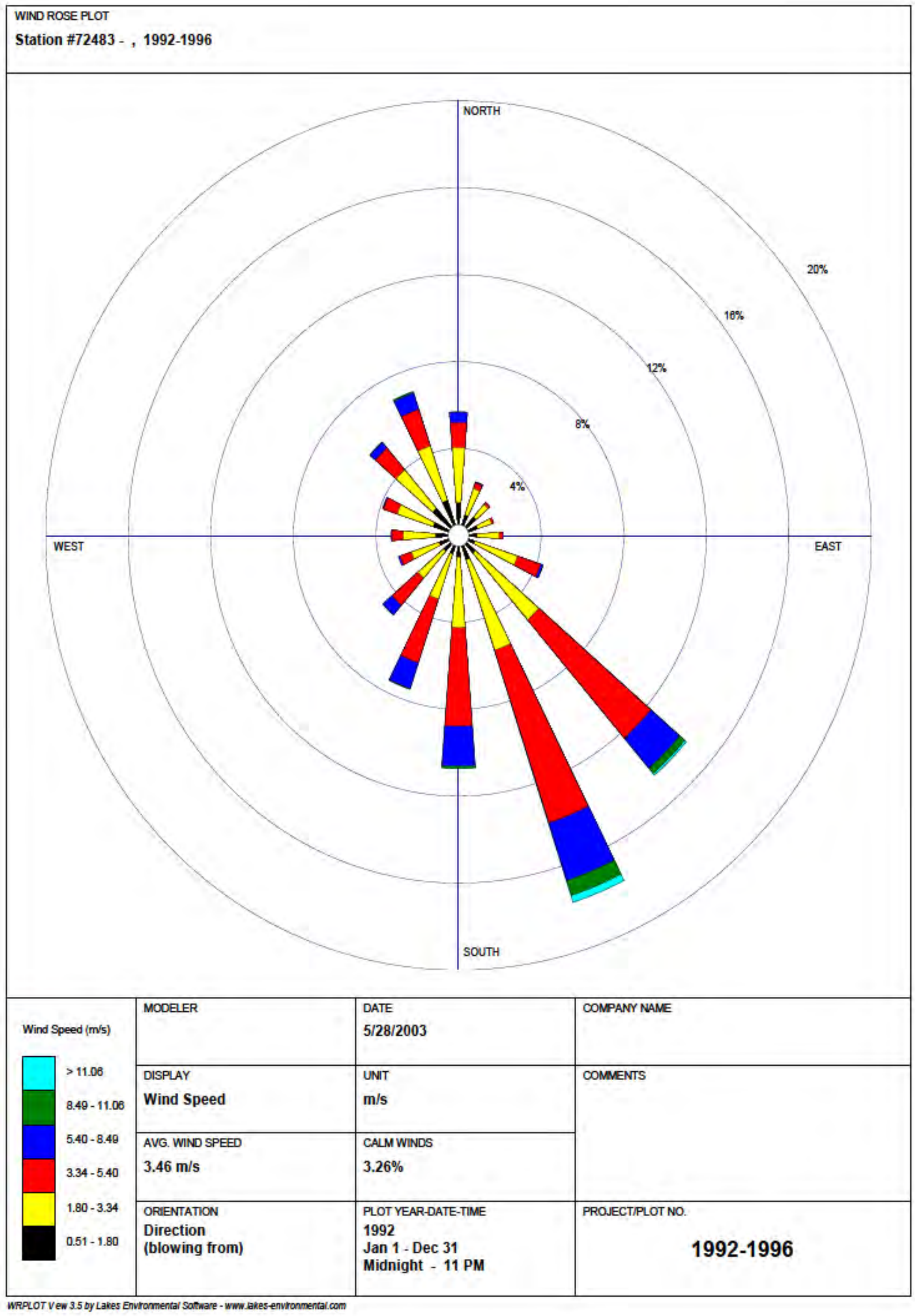

FIGURE 2. 14 ANNUAL WIND ROSE FOR THE FORMER MCCLELLAN AFB 1992-1996

TABLE 2-4 NORMAL AND EXTREME TEMPERATURES

Temperature							
Normal				Extreme			
Month	Daily Maximum	Daily Minimum	Monthly	Record High	Year	Record Low	Year
J	53.2	37.2	45.2	67	1966	23	1963
F	58.6	39.8	49.2	76	1964	28	1962
M	64.8	42.0	53.4	86	1966	28	1966
A	71.4	45.3	58.4	91	1968+	34	1967+
M	78.2	49.7	64.0	101	1967	37	1964
J	86.5	54.4	70.5	115	1961	43	1966
J	93.4	57.4	75.4	113	1961	50	1960
A	91.9	56.3	74.1	107	1968+	49	1966
S	88.2	55.0	71.6	104	1969+	43	1965
O	77.6	49.4	63.5	99	1963	38	1969+
N	64.2	41.6	52.9	87	1960	26	1961
D	54.6	38.1	46.4	72	1967	24	1965
YR	73.6	47.2	60.4	115	June 1961	23	Jan. 1963

2.3.2.5. Severe Weather

Tornadoes have been reported in California, but they are infrequent. They are generally not severe and most cases cause only minor damage to trees or light buildings.

2.4. Hydrologic Engineering

2.4.1. Hydrologic Description

The base and adjacent lands are located in the Great Valley subdivision of the Pacific Border Physiographic Province (Reference 2.1). They are situated on the alluvial plains of the Sacramento River and its tributaries (Reference 2.2). The land is relatively flat, ranging in elevation from 50-75 ft above mean sea level. Soil cover of about 4 ft consists of sandy loam (Reference 2.3). The surface soil is moderately permeable but the subsoil has low permeability. The soils have moderate water-holding capacity and pose a slight erosion hazard.

The UCD/MNRC site is underlain by a thick (>1,000 ft) section of unconsolidated sediments deposited by streams draining the Sierra Nevada. The uppermost deposits are termed the Victor Formation which is approximately 50 to 100 ft thick at the UCD/MNRC site. The Victor Formation is composed of the heterogeneous shifting streams that drained the Sierra Nevada in Pleistocene time. These streams left sand and gravel in channel-like structures that grade laterally and vertically into silt and clay in a manner that provides little correlation of materials from area to area. This is characteristic of floodplain or low-sloping alluvial fan deposits.

Underlying the Victor Formation is a series of alluvial deposits, termed the Laguna or Fair Oaks Formations. These alluvial deposits are composed of a heterogeneous assemblage of beds of silt, clay, and sand with lenticles of gravel deposited on westward-sloping floodplains by meandering, sluggish streams. Some of the sands are clean and well sorted while some of the gravels are extremely silty and poorly sorted. Sediments of the Laguna are variable; for example, in one area the formation consists of compact silt, clay with lenses of poorly sorted gravel, sand, and silt, and in others it contains sand with only a few interbeds of clay and silt.

Underlying the Victor, Laguna, and Fair Oaks Formations is a volcanic unit termed the Mehtren Formation. In the vicinity of the UCD/MNRC site, this formation is composed of sedimentary deposits derived from reworking of andesitic tuff-breccias which issued from volcanic vents in the Sierra Nevada. Typically, these are referred to as “black sands” in drillers logs. The black sands generally are fairly soft and well sorted. They are formed as fluvial deposits, having been derived from andesitic detritus washed down the slopes of the Sierra Nevada. Beds of black sand are commonly about 2 meters thick, although beds up to 6 meters or more have been reported. Where exposed in road cuts, these beds exhibit crossbedding, indicating a steam-laid mode of origin. Associated with the black sands are lenticular beds of stream gravel containing andesitic cobbles and boulders up to a meter or more in diameter. Also associated with the sands are beds of brown to blue clay and silt. In addition to these sedimentary units, volcanic mudflow units have apparently also been encountered.

The Mehtren Formation is the major aquifer of the Sacramento area. The thickness of the Mehtren formation in the vicinity of the base is unknown, but probably exceeds 300 ft.

2.4.2. Floods

The natural surface drainage around the UCD/MNRC site has been altered by construction of a series of storm drains. The North Sacramento, Del Paso Heights, Robla, Rio Linda, and Elverta areas drain storm water runoff to the west through Arcade Creek, Magpie Creek, Rio Linda Creek, Dry Creek, and a series of shallow natural ditches and swales (Figure 2.15). Rather than emptying onto the flat farmland of the Natomas area, as they once did, these creeks and ditches are intercepted by the East Natomas Main Drainage Canal and carried via Bannon Slough to the Sacramento River. In the area, elevation above sea level ranges from about 90 ft in the northeast to 50 ft in the southwest. The extensively-wooded, double channel Dry Creek is the most important component of the natural drainage system serving the study area. Dry Creek begins to the east in Placer County where it collects from a large watershed in the Roseville vicinity.

Two rivers, the Sacramento and American, flow through the Sacramento area (Figure 2.16). The American River flows approximately five miles south of the UCD/MNRC site. There are two flood control dams on this river approximately 20 miles upstream. The major dam which forms Folsom Lake is an earthen structure. Directly downstream of Folsom Dam is Nimbus Dam. This is a concrete structure and forms Lake Natomas. The Sacramento River flows approximately five miles west of the UCD/MNRC site. This river handles the runoff from areas north of Sacramento.

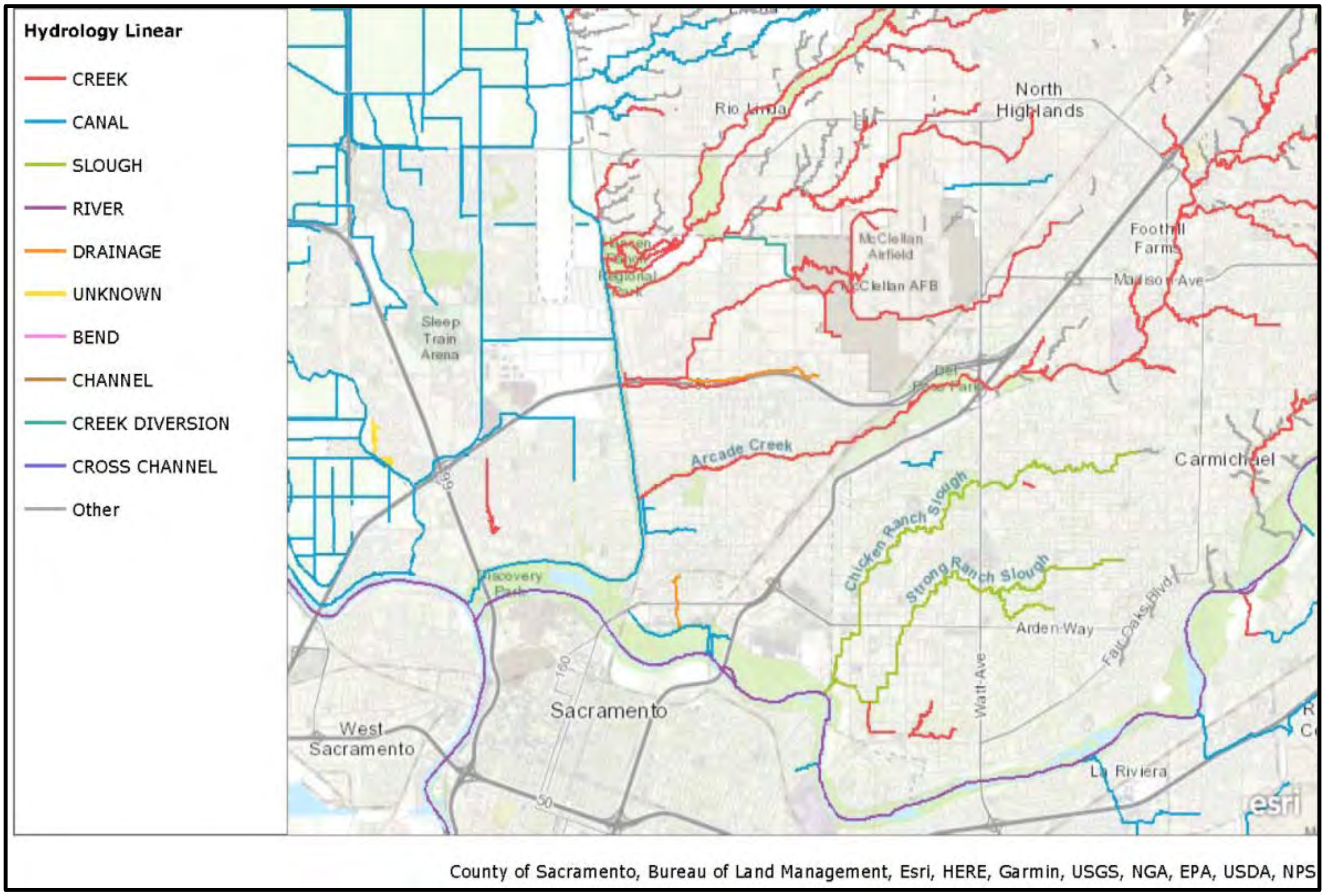

FIGURE 2.15 UCD/MNRC SITE HYDROLOGY

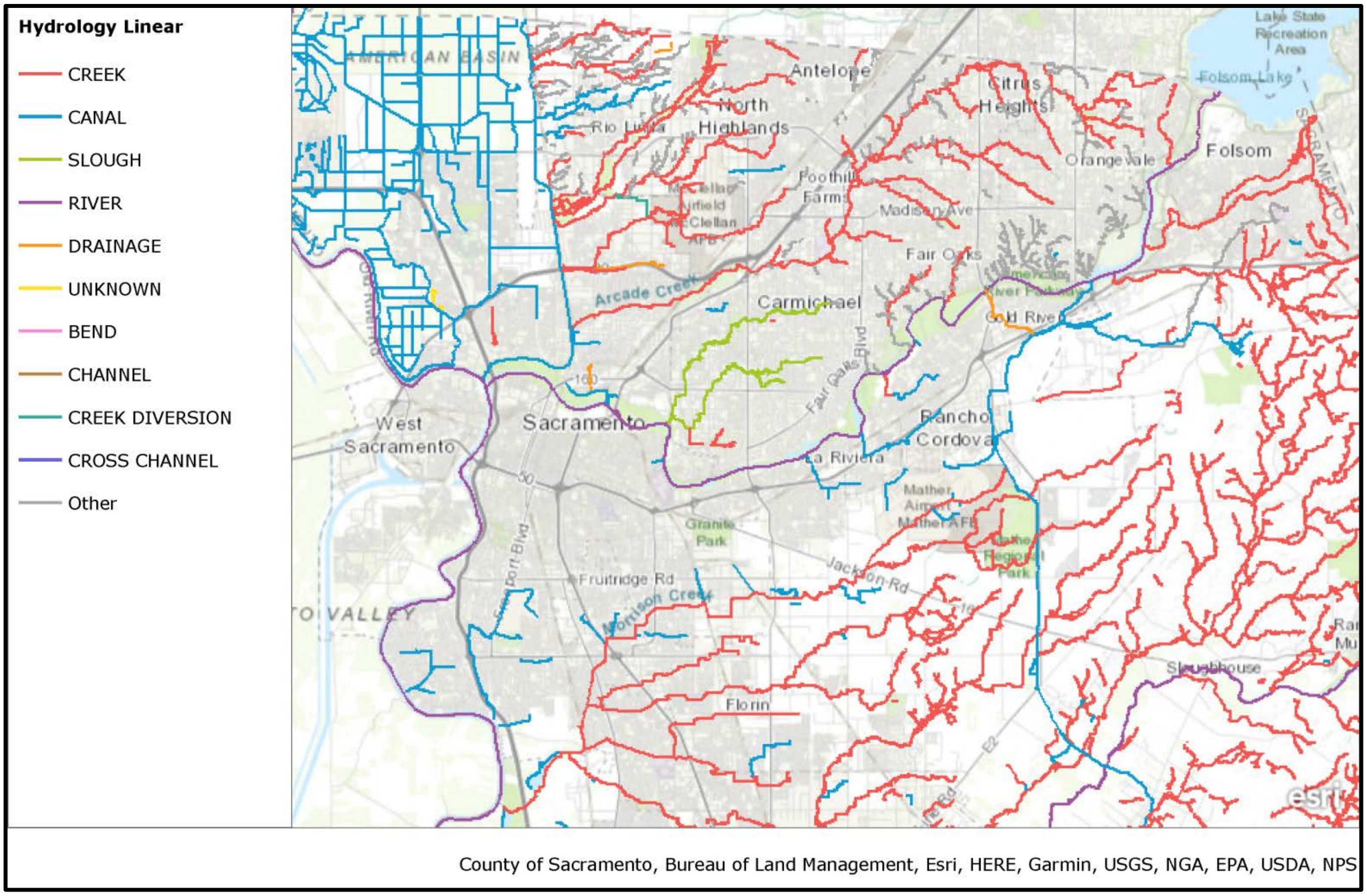

FIGURE 2.16 SACRAMENTO COUNTY HYDROLOGY

Neither of these rivers presents a flood hazard to the UCD/MNRC facility. The nearest 100 yr floodplain is about 3,400 ft (1,037 m) from the site of the UCD/MNRC (Figures 2.17 through 2.20).

2.4.3. Accidental Release of Liquid Effluents in Surface Waters

The probability of an accidental release of radioactive liquid effluents from the UCD/MNRC in surface waters is extremely low. Two (2) UCD/MNRC systems may contain radioactive liquid: the reactor primary cooling system and the water purification systems. All of the components for these systems; reactor tank, pumps, heat exchangers, filters, resin tanks, valves, and piping, are located within the UCD/MNRC reactor and equipment rooms. Any contaminated water leakage from this equipment will be wiped up and disposed of as discussed in Chapter 11. The only other areas where contaminated water may be encountered is in the radiography bays and the men's washroom. The radiography bays have a drain system that leads to a sump in Bay 1. Any water collected in the sump is pumped into an above ground liquid storage tank. The decontamination shower located in the men's washroom also drains into the storage tank. There are no floor drains in the men's washroom that lead to the industrial waste. Any water entering the tank, even if other than the reactor systems, will be analyzed for radioactive materials. If radioactive materials are found it will be disposed of as discussed in Chapter 11.

2.5. Geology, Seismology, and Geotechnical Engineering

The Sacramento area is located in Seismic Zone 3 of the Uniform Building Code. In general, seismic activity is not as great in the area as it is in the coastal areas (References 2.10, 2.11, 2.12 and 2.13). Based on a review of historical records, the maximum-intensity earthquake in Sacramento in historical times has been about VII on the Modified Mercalli scale (References 2.12 and 2.13). This intensity was the result of earthquakes centered about 20 mi (32 km) west of Sacramento with an estimated magnitude of 6.0 to 6.5 on the Richter scale. Earthquakes of the intensity of VII are characterized by collapse of weak chimneys, moderate damage to masonry walls, fall of cornices from high buildings, and fall of some nonstructural, unreinforced brick walls (References 2.12 and 2.13). However, earthquakes of higher intensity could have occurred prior to the coverage of the historical record, and higher intensity earthquakes are possible in the future Figure 2.21 (References 2.18 through 2.20) is a historical summary of the seismic activity in the area.

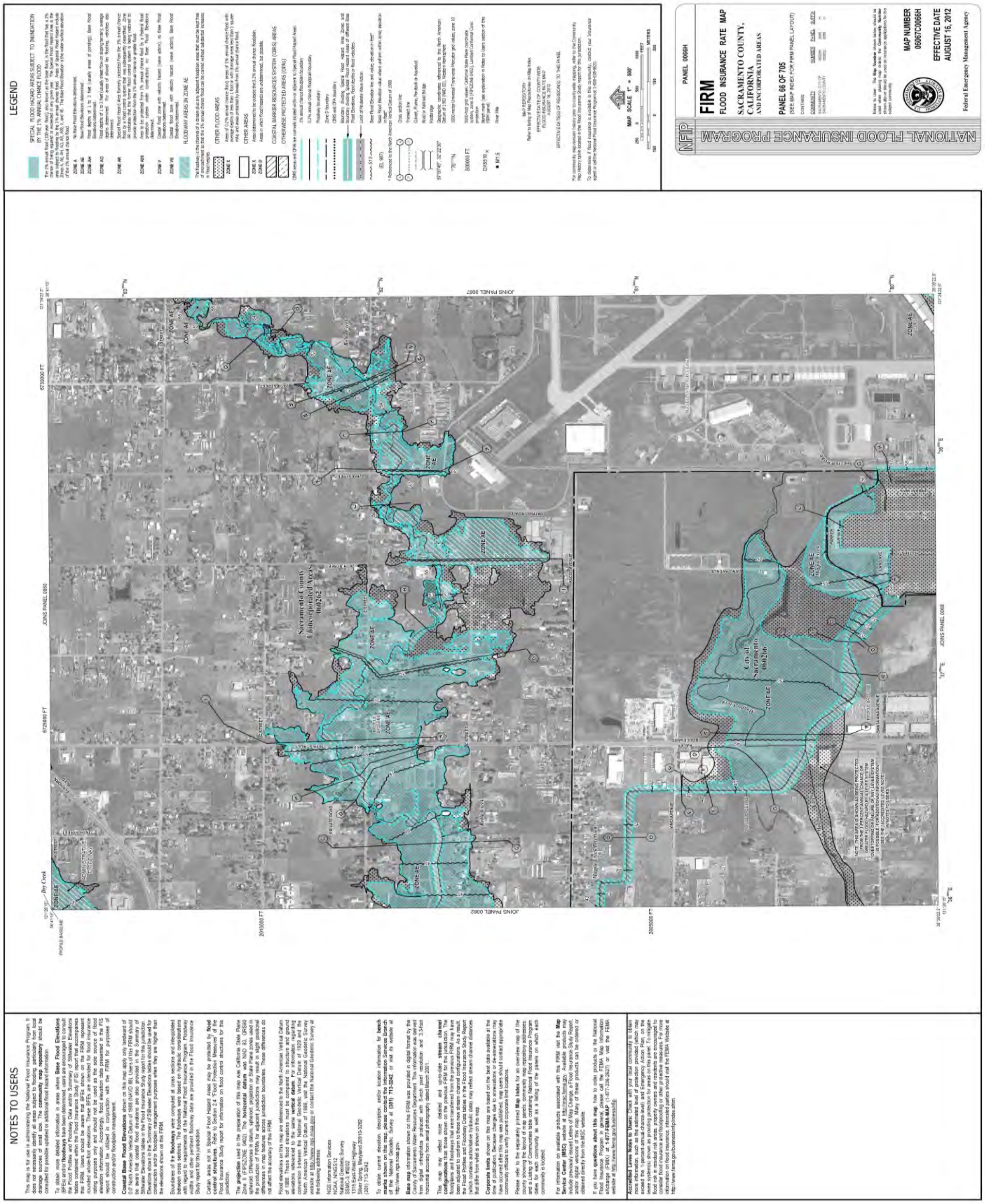

FIGURE 2.17 MCCLELLAN PARK - 100 YEAR FLOODPLAIN NORTH WEST QUADRANT

FIGURE 2.18 MCCLELLAN PARK - 100 YEAR FLOODPLAIN NORTH EAST QUADRANT

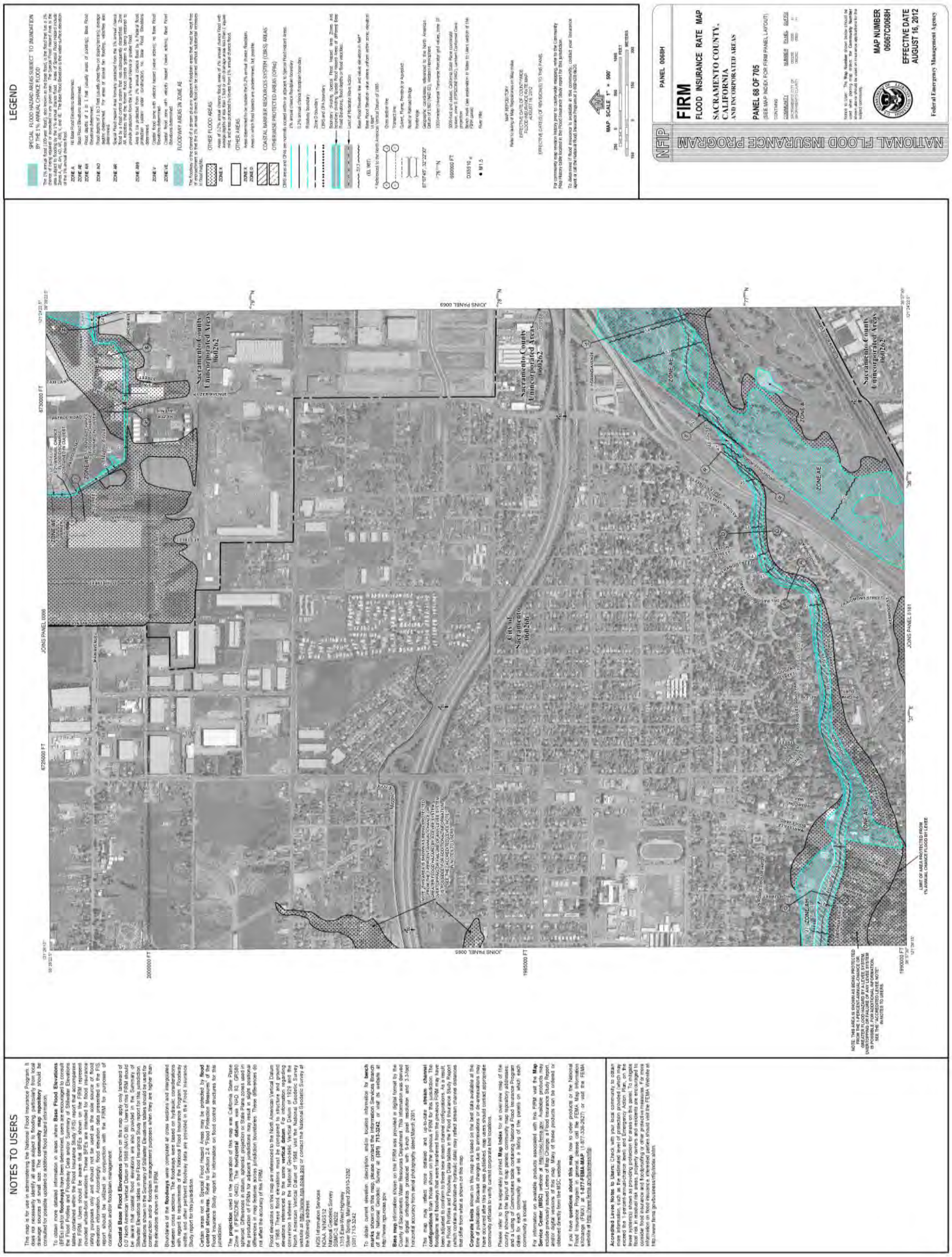

FIGURE 2. 19 MCCLELLAN PARK - 100 YEAR FLOODPLAIN SOUTH WEST QUADRANT

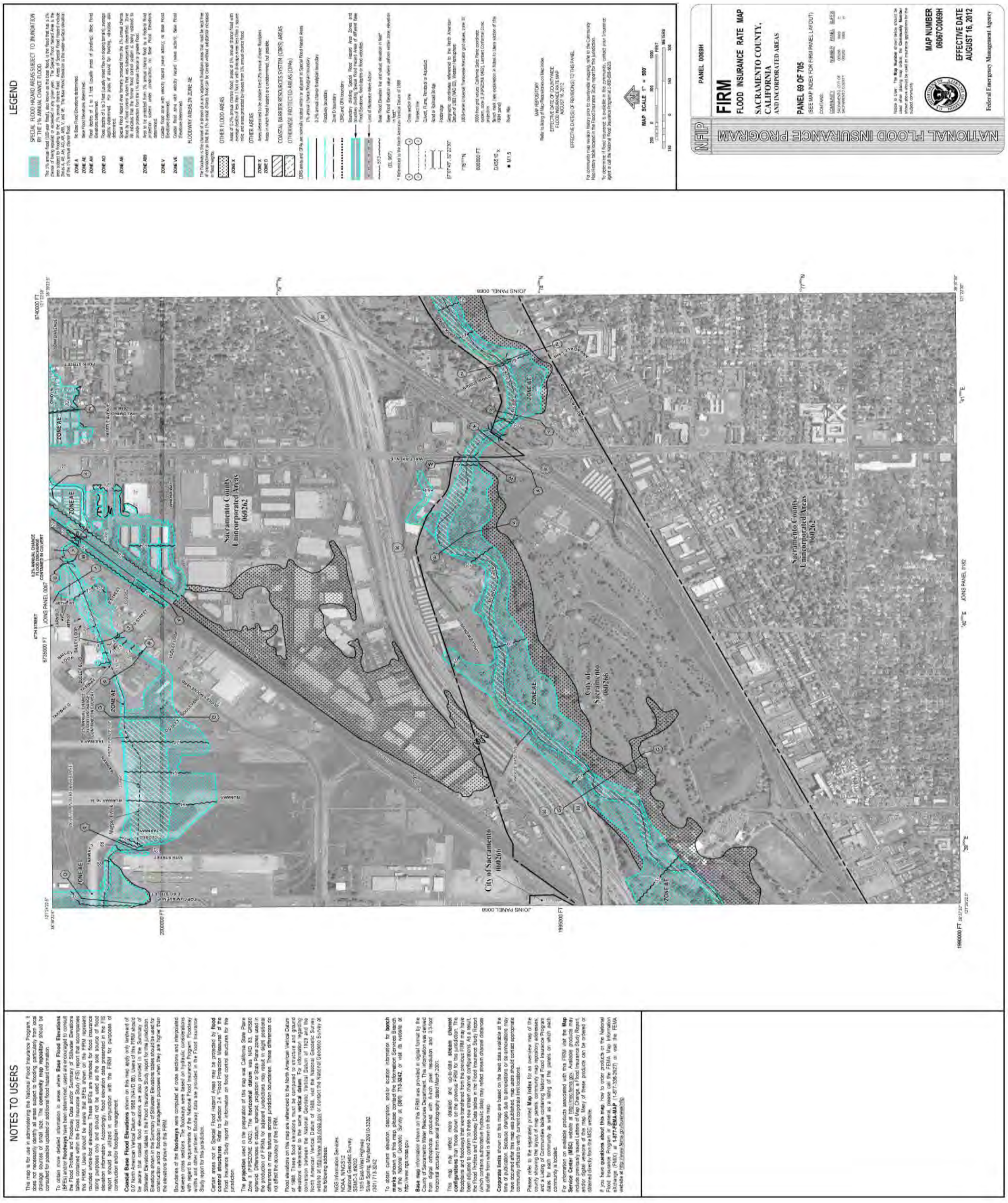

FIGURE 2.20 MCCLELLAN PARK - 100 YEAR FLOODPLAIN SOUTH EAST QUADRANT

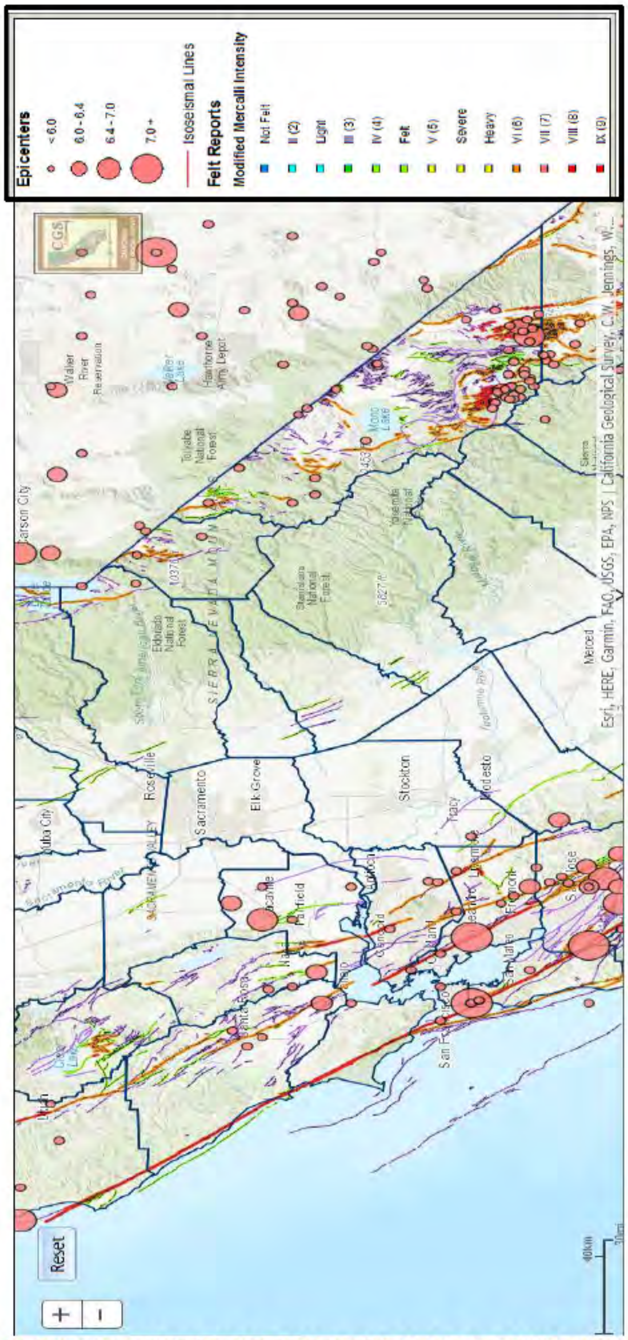

FIGURE 2.21 EARTHQUAKE FAULT EPICENTER MAP OF CALIFORNIA (PARTIAL)

California contains innumerable earthquake faults. Some of these faults are shown in Figure 2.22, including the known faults around Sacramento (Reference 2.14, 2.21). It is quite probable that other surface and subsurface faults also exist; however, this can only be positively determined by adequate explorations. The fact that no surface faults appear on the map in the Sacramento or San Joaquin Valleys may only indicate that sediments laid down during late geologic time cover the fault scars. On the other hand, rock or the firmer sediments usually found in the hill and mountain areas retain the evidence of faults over long time periods.

As shown in the figure, surface faulting has been identified in the Bear Mountain fault zone some 25 miles east of Sacramento and in the Rumsey Hills area west of Woodland. A number of subsurface faults have been found during explorations for gas near Sacramento as reported by the Division of Oil and Gas of the California Department of Conservation. Such subsurface faulting is reported near Freeport and Clarksburg just to the south of Sacramento; in the Todhunter Lake area a few miles north and east of Davis; and in the Rio Vista area, to identify a few areas near Sacramento. Data are not available to indicate the existence of subsurface faulting nearer to or within the City of Sacramento.

Geologic investigations to date have not discovered evidence indicating movement on subsurface faults in the Sacramento Valley more recent than Eocene time, about 40 million years ago. Eocene rocks extend generally from the surface of the ground to 0.5 to 0.75 kilometer depth. One fault in the Folsom area, recently mapped by the California Division of Mines and Geology, has been interpreted as having moved during the Quaternary Period.

One conclusion based on this evidence is that except for the possibly more recent movement on the fault in the Folsom area, there has been no near surface fault displacement in, or within close proximity of Sacramento during the past 40 million years. The focal depth of California earthquakes (the depth below the surface of the earth to the start of the rupture in the rock that provides the energy for the quake) ranges from a few kilometers to 15 to 20 kilometers, and therefore earthquakes of a smaller magnitude could have originated here during the past 40 million years, but the faulting might not have extended into or through this layer of post-Eocene rocks.

A second conclusion is that faulting did extend to the surface, but the evidence for this surface breaking either has not yet been found or is undiscernible in the sediments which fill the valley.

USGS Quaternary Faults and Folds Database

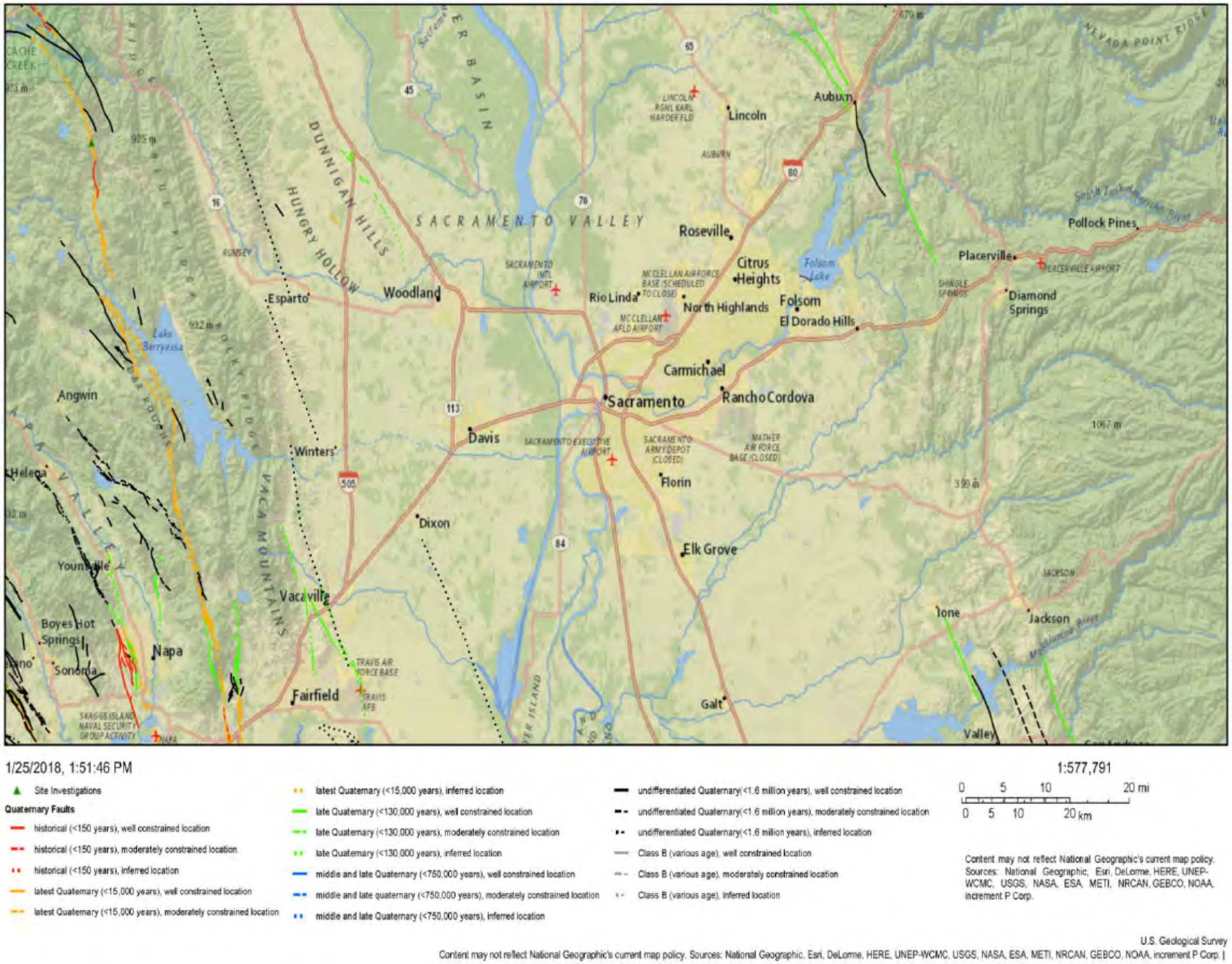

FIGURE 2.22 SACRAMENTO AREA SIGNIFICANT FAULTS

California's approximately 200-year recorded history is short, indeed, compared with the estimated 4.5 billion year age of the earth. It is a certainty that the Sacramento area has experienced violent earthquake motion during a part of this geologic time. From recorded information readily available for the past 200 years, however, it appears that Sacramento has not experienced violent earthquake motion of a nature compared with that experienced by several other areas within California.

Probably the greatest amount of earthquake shaking experienced in Sacramento during the recent past occurred on April 21, 1892. This earthquake produced extensive damage to towns some 25 miles west of Sacramento.

As noted above, the April 21, 1892 earthquake, along with the quake two days earlier, probably produced the most vigorous earthquake shaking in Sacramento during recorded history. There is some evidence that the epicenters of these shocks were in the area between Winters and Vacaville. Both of these towns, as well as Davis, Dixon, and Woodland experienced significant damage to many structures. Although the location for the fault responsible for the 1892 earthquakes is not known, the California Division of Mines and Geology and the U.S. Geological Survey have recently found (May 1972) that the Green Valley fault, west of Fairfield, is showing active fault creep or slip movements just to the south of Interstate 80 highway.

A lineament on the east flank of the Dunnigan Hills has been mapped recently by the U.S. Geological Survey. It may be the surface expression of a fault that has moved recently.

In recent time there was about $10,000 damage at the Sacramento Filtration Plant resulting from the Dixie Valley earthquake, east of Fallon, Nevada, December 16, 1954 - a Richter magnitude 7.2 earthquake. This was about 185 miles northeast of Sacramento and clearly indicates that the long period earthquake waves resulting from distant earthquakes can have definite effects upon structures or their contents. Damage also occurred to the digestion tanks at the Sacramento Sewage Treatment Plant and to a clarifier tank at the Campbell Soup Company.

There appears to be a strong northwesterly structural “grain” to California geology. Earthquakes having epicenters towards the west have not affected Sacramento in the past to the same extent as those centered east and south of Sacramento. The 1892 Winters earthquake appears to be an exception to the general statement. To explain further, the April 18, 1906, San Francisco shock of Richter magnitude 8.25 with its epicenter about 80 miles west of Sacramento was probably felt in Sacramento with about the same intensity as the Owens Valley quake of March 12, 1872, which has been estimated to be between 8.0 and 8.25 Richter magnitude and was about 230 miles southeast of Sacramento. Also, the Boca Reservoir earthquake of Richter magnitude 6.0 on August 12, 1966, 95 miles northeast of Sacramento was strongly felt in the Sacramento area as well as the above mentioned Dixie Valley earthquake 185 miles northeast of Sacramento.

The University of California Seismographic Station Reports that since 1932 there have been approximately 700 earthquakes of Richter magnitude 4 and greater in the area bounded between longitudes 118°W and 124°W and between latitudes 36.5°N and 40.5°N. In general, this area is from Eastgate, located in west central Nevada, to the Pacific Ocean and from south of Fresno to Redding. Also within this area there were approximately 90 earthquakes of magnitude 5 and some 15 earthquakes of magnitude 6 during this period.

As noted above, the distance of the closest fault to the UCD/MNRC site far exceeds the siting requirements of ANSI 15.7, Section 3.2, which states “no proposed facility shall be located closer than 400 meters from the surface location of a known capable fault.”

CHAPTER 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

Chapter 3 - Valid Pages
Rev. 6 06/10/20

all	Rev. 6	06/10/20

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3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 Conformance with NRC General Design Criteria

3.1.1 Introduction

This chapter discusses the “General Design Criteria for Nuclear Power Plants” as set forth in 10CFR50, Appendix A, as they apply to the UCD/MNRC. These General Design Criteria were formulated for the purpose of establishing minimum requirements for the principal design criteria to be utilized for water-cooled nuclear power plants. Further, they are to be applied to new design and construction of plants similar in design for which construction permits have been previously issued. Since the UCD/MNRC is a research facility, many of the systems cannot be logically categorized according to power-plant application. Therefore, the discussions here are oriented with regard to the individual criterion, rather than toward identification of areas of noncompliance and corrective actions.

The nominal UCD/MNRC steady-state power level is 1 MW. Thus, the fission-product inventory is orders of magnitude less than those of the conventional power reactors for which General Design Criteria were primarily prepared. A conservative upper limit of energy released for an entire year of operation would be about 350 MW-days (or <100 MW-days for single shift operation). These comparisons illustrate why the UCD/MNRC may be placed in a category of much lower risk, and treated accordingly, in a rigorous review for compliance with the General Design Criteria.

The accidents described in Chapter 13 conservatively demonstrate that instrumented shutdown actions and building confinement are not necessary to ensure that radiological doses will not exceed allowable limits. In the event of an improbable instantaneous loss of coolant from the reactor tank, an ECCS system has been provided for which ample time is available for initiation through operator action. Table 3-1 presents a synopsis of the conclusions regarding application of the General Design Criteria to the UCD/MNRC.

3.1.2 Overall Requirements (Criteria 1-5)

Criterion 1: Quality Standards and Records

Original structures, systems, and components important to safety were designed, fabricated, constructed, and/or tested to design specifications (MAN-NDI-86-03) and associated standards.

TABLE 3-1 APPLICABILITY OF COMPLIANCE WITH GENERAL DESIGN CRITERIA

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

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

Criterion Number and Title	Compliance	Compliance Not Required	Conditional Noncompliance	Conditional Compliance
PROTECTION AND REACTIVITY CONTROL SYSTEMS				
20. Protection Systems Functions	X			
21. Protection System Reliability and Testability	X			
22. Protection System Independence	X			

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

Criterion Number and Title	Compliance	Compliance Not Required	Conditional Noncompliance	Conditional Compliance
FLUID SYSTEMS				
30. Quality of Reactor Coolant Pressure Boundary		X		
31. Fracture Prevention of Reactor Coolant Pressure Boundary		X		
32. Inspection of Reactor Coolant Pressure Boundary	X			
33. Reactor Coolant Makeup	X			
34. Residual Heat Removal	X			
35. Emergency Core Cooling	X			
36. Inspection of Emergency Core Cooling System	X			
37. Testing of Emergency Core Cooling System	X			
38. Containment Heat Removal	X			
39. Inspection of Containment Heat Removal System		X		
40. Testing of Containment Heat Removal System		X		
	X			
41. Containment Atmosphere Cleanup		X		
42. Inspection of Containment Atmosphere Cleanup System		X		
43. Testing of Containment Atmosphere Cleanup System	X			
44. Cooling Water				X
45. Inspection of Cooling Water System				X

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

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

All design and construction work was monitored by McClellan AFB engineers to assure that the specifications incorporated appropriate standards, and the design and construction were in accordance with these specifications. Modifications have been made in accordance with existing standards and requirements.

<u>Criterion 2: Design Bases for Protection Against Natural Phenomena</u>

Hurricanes, tsunamis, and seiches do not occur in the Sacramento area. Flooding in the area could be caused by run-off from local rainstorm activity or by a catastrophic failure of Folsom Dam. However, the UCD/MNRC is situated some 3,400 ft from the 100-yr floodplain.

Only a small number of tornadoes, one or two per year, have been reported in California. Based on the small probability of occurrences, postulated low intensity, the intermittent type of reactor operation and low fission-product inventory, no criteria for tornadoes have been established for the UCD/MNRC structure. However, the buildings are designed to withstand the area wind loads.

The Sacramento area is classified as being in Seismic Zone 3 as defined in the Uniform Building

Code. The UCD/MNRC structures have been designed and constructed in accordance with this code, with an importance factor of 1.5 and to AFM-88-15, Chapter 13. Seismic activity in the region has registered as high as Richter 6.0-6.5 in historical time which indicates an upper limit on the most likely seismic event (Section 2.5). Since the UCD/MNRC is designed to the Uniform Building Code for Zone 3 with an importance factor of 1.5, there is ample conservatism in the design for the maximum expected event. The UCD/MNRC structures may suffer some damage from a seismic event of the highest possible yield, but, as previously noted, even in the event of the incredible scenario that a seismic event would cause instantaneous loss of tank coolant water, an ECCS has been provided for which there is ample time for operators to initiate actions to minimize the consequences of such an event, and resultant radiological doses would be within the ranges evaluated in Chapter 13.

Criterion 3: Fire Protection

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

Several features reduce both the likelihood and the consequences of a fire. First, periodic fire-safety inspections are made by Fire Safety engineers. Second, periodic in-house inspections are made for the explicit purpose of reducing nonessential combustible material inventory. Third, fire detection and suppression systems are installed in the facility. If these systems are activated, or a fire alarm is tripped, the Sacramento Metropolitan Fire Department is automatically alerted and will respond to the UCD/MNRC within a few minutes. Fourth, a closed circuit television camera in the reactor room with a monitor in the control room permits the reactor operator to continuously observe the reactor room, so that immediate action can be taken to minimize the effects of a fire; established emergency procedures will be put into effect in the event of a fire.

Fifth, the large volume of water in the reactor tank would protect the core from any conceivable fire. Sixth, the reactor is fail safe and will shutdown should a fire damage the instrumentation or control system.

Criterion 4: Environmental and Missile Design Bases

The construction of the facility precludes catastrophic rupturing of the reactor tank. There is no source in the reactor room for generating large, sustained positive pressure differentials which would breach the reactor room integrity.

The amount of explosive materials allowed in the radiography bays has been limited to preclude damage to the reactor should they detonate. Plates covering the entrance to beam tubes have been sized, as discussed in Chapters 10 and 13, so that these will not fail from a pressure pulse explosive generated from the maximum allowable quantities of explosives. Further, each experiment containing explosives will be analyzed to show that detonation will not produce pressure or fragments that will damage the reactor. The reactor core is protected from missiles by being below ground level and surrounded by a large block of reinforced concrete. Dynamic effects of such conditions as whipping pipes are not a problem because there are no high pressure systems. The piping systems are anchored and do not penetrate the tank walls and they could not conceivably affect the reactor. The probability of an event or condition resulting from dynamic effects of missiles, aircraft, etc., causing a reactor incident, is very small. "Probability Assessment of the Airplane Crash Risk for McClellan Air Force Base TRIGA® Reactor" shows that the probability of an aircraft accident impacting the facility is 10^{-8}/year and is, therefore, considered incredible (Appendix C).

Criterion 5: Sharing of Structures, Systems, and Components

Electrical power constitutes the only system shared by the UCD/MNRC. Sharing is based on the fact that the UCD/MNRC electric power is supplied from a distribution point within the adjacent facility. Loss of power results in the shutdown of the reactor since all control circuits are fail-safe, and no power is required for safe shutdown or to maintain safe shutdown conditions. An electric power failure at any point in the UCD/MNRC network will not detrimentally affect the reactor.

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

Criterion 10: Reactor Design

The safety limit placed on the temperature of the reactor fuel for UCD/MNRC operations is 1100°C when the clad is less than 500°C and 930°C when the clad temperature is equal to the fuel temperature.

Accident analyses presented in Chapter 13 show that under credible accident conditions, the safety limit on the temperature of the reactor fuel will not be exceeded. Consequently, there would be no fission product release that would exceed 10CFR Part 20 allowable radiation levels.

Criterion 11: Reactor Inherent Protection

Because of the fuel material (U-ZrH) and core design, there is a significant prompt negative temperature reactivity coefficient. Routine steady-state power operation is performed with the transient, shim, and regulating rods partially withdrawn. As shown in Chapters 4 and 13, the most rapid possible reactivity insertion rates are adequately compensated for by the negative temperature reactivity coefficient of 0.01 %/°C (1 x $10^{-4}\Delta k/k/°C$).

Criterion 12: Suppression of Reactor Power Oscillations

Due to the small dimensions of the core and low power levels, the reactor is inherently stable to space-time and xenon oscillations.

Criterion 13: Instrumentation and Control

The instrumentation and control system for the UCD/MNRC TRIGA® reactor is a computer- based system incorporating the use of a GA-developed, multifunction, NM-1000 microprocessor-based neutron monitoring channel and a NPP-1000 analog type neutron monitoring channel (refer to Chapter 7 for further detail). The NM-1000 system provides a safety channel (percent power with scram), a wide-range log percent power channel (below source level to full power), period indication, and a multirange linear power channel (source level to full power). The NPP-1000 system provides a second safety channel for redundancy (percent power with scram). In the pulse mode of operation, the Data Acquisition Computer (DAC) makes a gain change in the NPP-1000 safety channel to provide NV and NVT indication along with a peak pulse power scram. The NM-1000 is bypassed once a pulse has been initiated.

The control system logic is contained in a separate Control System Computer (CSC) with a color graphics display. While information from the NM-1000, NPP-1000, and fuel temperature channels is processed and displayed by the CSC, each is direct wired to its own output display, and the safety channel connects directly to the protective system scram circuit. That is, signals to the scram circuits are not processed by the Data Acquisition computer or the control computer.

The nuclear information goes directly from the detectors to either the NM-1000 or NPP-1000 where it is processed. The processed signals connect directly to the scram circuit switches. Fuel temperature information goes directly to "action pack modules" for amplification and then to the scram circuit switches.

The NM-1000 digital neutron monitor channels were developed for the nuclear power industry and are fully qualified for use in the demanding and restrictive conditions of a nuclear power generating plant. Their design is based on a special GA-designed fission chamber and low-noise ultra-fast pulse amplifier. The NPP-1000 safety channel was designed to the same criteria as the NM-1000 channels.

The CSC manages all control rod movements, accounting for such things as interlocks and choice of particular operating modes. It also processes and displays information on control rod positions, power level, fuel and water temperature, and pulse characteristics. The CSC also performs many other functions, such as monitoring reactor usage, and storing historical operating data for replay at a later time. A computer-based control system has many advantages over an analog system: speed, accuracy, reliability, the ability for self-calibration, improved diagnostics, graphic displays, and the logging of vital information.

The UCD/MNRC reactor can be operated in four modes: manual, automatic, square wave, and pulse. The operations are controlled from the reactor console mode control and the rod control panels. The manual and automatic modes are steady-state reactor conditions; the square-wave and pulse modes are the conditions implied by their names and require the use of the pulse rod. ***Note that square wave and pulse mode are no longer utilized at MNRC.***

The manual and automatic reactor control modes are used for reactor operation from source level to 100% power. These two modes are used for manual reactor startup, change in power level, and steady-state operation.

Interlocks prevent the movement of the rods in the up direction under the following conditions:

1. Scrams not reset;
2. Source level below minimum count;
3. Two UP switches depressed at the same time;
4. Mode switch in the PULSE position;
5. Mode switch in the AUTOMATIC position [servo-controlled rod(s) only];
6. Mode switch in the SQUARE WAVE position.

Automatic power control can be obtained by switching from manual operation to automatic operation on the mode control panel. All the instrumentation, safety, and interlock circuitry for steady operation applies to this mode. However, the servo-controlled rod(s) is (are) controlled automatically to a power level and period signal. The reactor power level is compared with the demand level set by the operator, on the mode control panel, and used to bring the reactor power to the demand level on a fixed preset period. The purpose of this feature is to maintain automatically the preset power level during long-term power runs. The square-wave mode allows the reactor power to be quickly raised to a desired power level. In a square-wave operation, the reactor is first brought to criticality below one kW in the manual mode, leaving the transient rod partially in the core. The desired power level is set by the reactor operator using the power demand selector located on the mode control panel. All of the steady-state instrumentation is in operation. The transient rod is ejected from the core by means of the transient rod FIRE pushbutton located on the rod control panel. When the power level reaches the demand level, it is maintained in the automatic mode.

Reactor control in the pulsing mode consists of manually establishing criticality at a flux level below one kW in the steady-state mode. This is accomplished by the use of the control rods, leaving the transient rod either fully or partially inserted. The pulse mode selector switch located on the mode control panel is then depressed. The MODE SELECTOR switch automatically causes the DAC to make a gain change in the NPP-1000 safety channel to monitor and record peak flux (NV), energy release (NVT), and to provide a peak pulse power scram. The pulse is initiated by activating the FIRE pushbutton. Once a pulse has been initiated and it is detected by the DAC, the NM-1000 safety scram is bypassed. Pulsing can be initiated from either the critical or subcritical reactor state.

Criterion 14: Reactor Coolant Pressure Boundary

The reactor tank and cooling systems operate at low pressure and temperature. The vessel is open to the atmosphere, and there are no means for pressurizing the system. The reactor tank is constructed of aluminum and the primary coolant system components are aluminum or stainless steel. The system components outside the reactor tank have a low probability of serious leakage or of gross failure.

Criterion 15: Reactor Coolant System Design

The reactor tank is an open system and the maximum pressure in the primary system is that due to the static head (about 23-1/2 ft). The primary cooling system, the secondary cooling system, and the purification system are pressurized by small capacity pumps. The secondary system water pressure is maintained slightly higher than the primary system. This feature prevents any radioactive primary water from entering the secondary system, and the environment, should a leak develop in the heat exchanger. There are no instrumentation systems that derive signals from any portion of the reactor coolant systems to initiate either control or protection actions. Piping and valves in the primary and purification systems are stainless steel or aluminum and of such size to provide adequate operating margins. The secondary system components are carbon steel. Chapter 5 describes the cooling system in detail.

Criterion 16: Containment Design

The structure surrounding the reactor constitutes a confinement building rather than providing absolute containment. Because of the low fission-product inventory, leakage from the structure can be tolerated.

Criterion 17: Electric Power Systems

An uninterruptible power supply (UPS) provides electrical power to the reactor console, DAC, and translator rack during normal reactor operations. An additional emergency generator is provided to supply power to the Auxiliary Make Up Water System (AMUWS) and the reactor room exhaust fan (EF-1) should these systems be called upon to provide backup to the reactor ECCS system. The UPS provides a filtered and regulated power source to the computers and electronic components of the reactor control systems. If there was a loss of electrical power the UPS will supply electrical power to all components for fifteen minutes. Because the reactor is cooled by natural convection, and there is no requirement to provide forced cooling flow for the removal of heat, there is sufficient time for the reactor operator to shut down the reactor and confirm the reactor is shutdown. The UPS also provides an additional four hours of power to the stack continuous air monitor (CAM) and all remote area radiation monitors (RAM).

Criterion 18: Inspection and Testing of Electric Power Systems

The primary power distribution system supplying commercial power to UCD/MNRC is maintained by electrical utility maintenance crews. Routine inspections of the systems are performed.

The UCD/MNRC can tolerate a total loss of electric power with no adverse effects on the safety of the facility. There are no electrical power (distribution) systems designated as necessary to provide power to the UCD/MNRC during either normal or abnormal conditions.

Criterion 19: Control Room

In the event of an accident where operations instructions require shutdown of the reactor, continuous or even partial occupancy of the control room is not a requirement since the reactor has been shut down and experiments in progress terminated. The control room is equipped with a separate exhaust system and can monitor those accidents which do not result in a breach of the control room structure. Exposure levels from radiation sources resulting from an accident would be significantly reduced in magnitude (due to the location of the control room with respect to the reactor room). Consequently, control room radiation levels may not be higher than the allowable tolerance levels. In the event that the ECCS requires activation, this system is actuated by the operator through coupling a quick connect on the reactor building roof. The backup systems to the ECCS, can be controlled from either the reactor room or other remote locations. Nevertheless, the UCD/MNRC Emergency Plan describes actions for mitigating accident situations which require control room evacuation.

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

Criterion 20: Protection System Functions

The UCD/MNRC Reactor Protection System has been designed to initiate automatic actions to assure that fuel design limits are not exceeded by anticipated operational occurrences or accident conditions. The automatic actions are initiated by two nuclear power channels and two fuel temperature channels. The Reactor Protective System automatically scrams the control rods when trip settings are exceeded (Chapter 7). There are no other automatic actions required by UCD/MNRC systems to keep fuel temperature limits from being exceeded. The Reactor Protective System satisfies the intent of IEEE-323-1974 in the areas of redundancy, diversity, power-loss fail-safe protection, isolation and surveillance.

Criterion 21: Protection System Reliability and Testability

The UCD/MNRC Reactor Protection System is designed to be fail-safe: any sub-channel loss that causes the channel to lose its ability to perform its intended function results in initiation of shutdown action. Protective action is manifested through several independent scram inputs arranged in series such that action by any one interrupts current to the scram magnets resulting in shutdown of the reactor. Redundancy of channels is provided and in addition, a loss of any channel due to open circuit or loss-of-power will result in a scram. Scram action is, therefore, on a one-out-of-one basis. All instrumentation is provided with testing capability. The Reactor Protective System satisfies the intent of the IEEE-323-1974 standard.

Criterion 22: Protection System Independence

The protective system satisfies the intent of IEEE-323-1974 "Criteria for Protective Systems for Nuclear Power Generating Stations." Protective functions are initiated through two independent nuclear and two independent fuel temperature channels, and there is a diversity of scram modes. Furthermore, the protective system is fail-safe upon loss of power.

The Reactor Protective System and the magnet power supply are, for the most part, physically and electrically isolated from the remainder of the control system. The cables between the control room and reactor room are enclosed in conduit. There is a separate conduit for each safety channel and one for the magnet power supply.

Criterion 23: Protection System Failure Modes

The reactor protective system is designed and constructed to be fail safe in event of a failure of a safety channel. Failure of a safety channel will result in removal of power to the control rod and transient rod magnets, allowing the control rods to fall into the core. Simultaneously, loss of a safety channel causes the transient rod's solenoid valve to de- energize thus removing any gas pressure that may be on the pneumatic cylinder. This causes the transient rod to fall into the core. The reactor protective system contains no control functions. Therefore, loss of a protective function will not necessarily affect the operation of the reactor, such as initiating an uncontrolled reactivity

Criterion 24: Separation of Protection and Control Systems

The UCD/MNRC has two nuclear instrumentation and two fuel temperature channels. One of the nuclear channels utilizes a fission chamber and a GA NM-1000. This channel provides signals for both safety (scram) and control action as well as signals for monitoring operations over a wide power range. The second channel utilizes an ion chamber and a GA NPP-1000. This channel provides % power for safety (scram) action as well as neutron monitoring capability for pulse operation. Fuel temperature is measured by thermocouples placed within the special instrumented fuel elements. While information from these channels is processed and displayed by the control system computer, each channel is independent, has its own output displays, and connects directly to the safety system scram circuit, see Criterion 13, second paragraph, for technique used to separate protection and control functions. The ability of this configuration to meet the intent of protection system requirements for reliability, redundancy, and independence for TRIGA®-type reactors has been accepted by the NRC.

Separate conduits are used for the safety channel and control system cabling from the NM- 1000 and the NPP-1000 (located in the reactor room) to the control console.

Finally, the control and safety systems are fail safe and will scram the reactor should they malfunction. No control or safety systems are required to maintain a safe shutdown condition.

Criterion 25: Protection System Requirements for Reactivity Control Malfunction

The UCD/MNRC Protection System is designed to assure that fuel temperature limits are not exceeded for any single malfunction of the reactivity control system. However, Chapter 13 shows that accidental runout of all rods simultaneously from the core at their normal drive speed will not result in exceeding fuel temperature limits.

Criterion 26: Reactivity Control System Redundancy and Capability

The UCD/MNRC has six independent reactivity control rods: four shim rods, one regulating rod and one transient rod. Each of the rods has its own drive mechanism and control circuit and they are operated individually.

The shim and regulating rods and drives are similar. The regulating rod is used to control power either manually or by automatic control.

Upon receipt of a scram signal, all six rods are released from their drives and dropped into the core. Insertion of five of the six rods ensures reactor shutdown.

Criterion 27: Combined Reactivity Control System Capability

The addition of a neutron absorbing material in the emergency cooling water is not required as the total worth of the rods is more than adequate to maintain the core at a subcritical level, with the most reactive rod stuck out of the core.

Criterion 28: Reactivity Limits

As shown in Chapter 13, neither continuous rod withdrawal nor loss of coolant will cause undue heating of the fuel. Identified accidents will not result in significant movement of adjacent fuel elements or otherwise disturb the core so as to add reactivity to the system.

Since the primary coolant system operates at atmospheric pressure, control-rod ejection is not a credible event. The shim rods, the regulating rod, and the transient rod cannot drop out of the core because the rods in the full down position are approximately one inch above the safety plate located near the bottom of the tank. The possibility of traveling out of the core in the downward position is therefore eliminated. The transient-rod system is specially designed for rapid reactivity insertion, and an accidental rod system ejection might cause a reactivity accident however, the pressurized gas supply to drive the transient rod has been permanently disconnected.

Criterion 29: Protection Against Anticipated Operational Occurrences

There are two scram loops, using different input signals, to provide redundancy in scram capability. The protection and reactivity control systems satisfy all existing design standards. Periodic checks (i.e., startup, shutdown, and maintenance procedures) of all reactor protective system channels and reactivity control systems demonstrate that they perform their intended function.

If there was a loss of electrical power the UPS will supply electrical power to all components for fifteen minutes. Because the reactor is cooled by natural convection, and there is no requirement to provide forced cooling flow for the removal of heat, there is sufficient time for the reactor operator to shut down the reactor and confirm the reactor is shut down.

3.1.5 Fluid Systems (Criteria 30-46)

Criterion 30: Quality of Reactor Coolant Pressure Boundary

The reactor tank is open to the atmosphere and is subjected only to ambient conditions. All components containing primary coolant (i.e., reactor tank, primary coolant system, and the purification system) are constructed of aluminum and stainless steel, using standard codes for quality control. There is no requirement for leak detection in the primary coolant or purification loop since no conceivable leak condition can result in the tank water level to lower more than approximately three feet.

Criterion 31: Fracture Prevention of Reactor Coolant Pressure Boundary

Since the coolant system is open to the atmosphere, no reactor coolant pressure boundary exists.

Criterion 32: Inspection of Reactor Coolant Pressure Boundary

The reactor tank is surrounded by a thick reinforced-concrete block which prevents external forces from being directly transmitted to the tank and precludes movement of the tank. The tank wall cannot be inspected by any means other than visual observation through the water from inside the tank. All piping, valves, meters, etc., associated with the primary water system are located in open spaces and are readily accessible for periodic inspections. There are no penetrations in the tank wall.

The UCD/MNRC operates at relatively low powers and temperatures. Because of the moderate fluence levels and low temperature factors, no significant change in material properties is expected.

Criterion 33: Reactor Coolant Makeup

The UCD/MNRC water purification system design includes an Auxiliary Make Up Water System (AMUWS) for makeup of primary coolant water. This system is manually operated and contains 4600 gal water. This system can also act as a backup to the ECCS system should the need ever arise (Chapter 5).

Criterion 34: Residual Heat Removal

Natural convection cooling is adequate to dissipate core afterheat. Many years of operations with TRIGA® reactors have shown that natural convection will provide adequate flow for the removal of heat after several hours of maximum steady-state operation.

Calculations performed for loss of coolant show that an ECCS connected to the domestic water supply is sufficient to assure that fuel temperatures will not reach the safety limit even under loss-of-coolant conditions (Chapter 13).

The AMUWS and the reactor room exhaust fan also provide a back-up capability to the ECCS system sufficient to provide the emergency cooling function should the domestic water supply also fail.

Criterion 35: Emergency Core Cooling System

An emergency core-cooling system has been provided in the case of the unlikely probability that an accident such as a severe seismic event occurs which results in the instantaneous loss of all reactor coolant. Analyses presented in Chapter 13 show that sufficient capability resides in simply providing outside air to cool the core post LOCA.

Criterion 36: Inspection of Emergency Core Cooling System

All components of the ECCS system are located in open spaces and are readily available for periodic inspection. Verification of the availability of the domestic water system is checked on a daily basis.

Criterion 37: Testing of Emergency Core Cooling System

The UCD/MNRC ECCS is routinely checked, tested, and maintained.

Criterion 38: Containment Heat Removal

There are no systems, devices, equipment, experiments, etc., with sufficient stored energy to require a primary containment heat-removal capability.

Criterion 39: Inspection of Containment Heat Removal System

This criterion is not applicable.

Criterion 40: Testing of Containment Heat Removal System

This criterion is not applicable.

Criterion 41: Containment Atmosphere Cleanup

Post-accident activities are not contingent upon maintaining the integrity of the building structure. Accident analyses in Chapter 13 have shown that downwind doses would not exceed 10 CFR Part 20 or ANSI 15.7 guidelines in any credible accident.

Routine operations result in two isotopes of concern being produced: Argon-41 in the reactor room and radiography bays and Nitrogen-16 from the irradiation of oxygen in the tank water. Analysis in Chapter 11 show concentrations to be below ANSI 15.7 guidelines for accident situations and below 10 CFR Part 20 guidelines for normal operations.

Criterion 42: Inspection of Containment Atmosphere Cleanup Systems

This criterion is not applicable.

Criterion 43: Testing of Containment Atmosphere Cleanup Systems

This criterion is not applicable.

Criterion 44: Cooling Water

A coolant system is utilized to cool reactor tank water during normal operation of the reactor.
The UCD/MNRC requires no auxiliary cooling system for cooling of reactor tank water upon shutdown.

Criterion 45: Inspection of Cooling Water System

Cooling equipment used in normal operation of the reactor is located either in the reactor room, equipment room, or outside the building with adequate space provided to permit inspection and testing of all components. Operation of the bulk coolant and cooling tower system is checked on a daily basis prior to reactor operation. During this checkout, the performance of each system is monitored with emphasis on pump outlet pressures, pressure differentials and system flow rates.

Criterion 46: Testing of Cooling Water System

UCD/MNRC reactor cooling systems are routinely checked, tested, and maintained.

3.1.6 Reactor Containment (Criterion 50-57)

Criterion 50: Containment Design Basis

Under the conditions of a loss of coolant, it is conceivable that the temperature at the reactor room could increase slightly due to heating of the air flowing through the core. However, since the building is not leak tight, it will not pressurize from the heating of the air.
Further, there is no requirement from a radiological-exposure viewpoint for a containment structure; hence, only confinement capability is provided. In addition, there is no source of energy (from an accident) which would provide a significant driving force (ΔP) if no corrective action were taken.

Criterion 51: Fracture Prevention Boundary

The confinement structure (the reactor room) is a reinforced filled concrete block structure with a conventional built-up roof. The entire structure is exposed to only normal external environmental conditions and internal environmental conditions are maintained at regulated conditions.

The structure will not be subjected to significant internal pressures during normal operations. Postulated accident conditions cannot result in significant changes in the pressure differential due to the non-leak tightness of the structure.

Criterion 52: Capability for Containment Leakage Rate Testing

This criterion is not applicable.

Criterion 53: Provisions for Containment Testing and Inspection

The reactor room confinement capability is checked on a daily basis prior to reactor operation and routinely during reactor operations. This check involves monitoring the pressure differentials between the reactor room and the surrounding areas. The reactor room exhaust recirculation system is checked monthly to confirm proper operation.

Criterion 54: Piping Systems Penetrating Containment

Piping systems which involve penetrations through the reactor building walls have no effect on the safety of operation; therefore, isolation, redundancy, and secondary containment of these systems is not required.

Criterion 55: Reactor Coolant Pressure Boundary Penetrating Containment

The reactor room was neither designed nor constructed as a containment structure but provides only a minor confinement capability. As pointed out in the responses to previous criteria, there are no requirements for containment (or confinement) capabilities. The only systems that penetrate the reactor room are the ventilation system, primary coolant system, ECCS/AMUWS, demineralizer system, helium pressurization system, Ar -41 production lines, electrical wiring ports, and air monitor lines for CAM and remote sampling. Reactor room wall penetrations are packed with fill material.

Criterion 56: Primary Containment Isolation

Penetrations through the building walls have no effect on the safety of reactor operations, therefore, isolation systems are not required in the UCD/MNRC.

Criterion 57: Closed System Isolation Valves

The UCD/MNRC reactor building was designed to provide only confinement capability; isolation valves are not required.

3.1.7 Fuel Radioactivity Control (Criteria 60-64)

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

There is no readily available path for liquid waste to be discharged directly to the environment. Liquids in the reactor room which could subsequently be injected into the environment may result from spills, washdown of the floor, etc. These liquids are collected in a storage tank within the UCD/MNRC, analyzed for radioactivity, and disposed of accordingly.

Significant dilution of gaseous materials released to the atmosphere, and soil permeability coefficients are such that transmissibility times of ground fluids are very long (Chapter 13).

Criterion 61: Fuel Storage and Handling and Radioactivity Control

The major concern relative to storage, handling, and control of radioactivity of irradiated fuel is shielding. All irradiated fuel elements are either stored in special racks (Criterion 62) in the reactor tank or storage pits in the reactor room (Chapter 9). When fuel is stored in the reactor tank, the water provides a minimum shield thickness of at least 18 ft. This amount of water also provides scavenging of any fission products should any escape from the fuel elements. Lead covers provide shielding for elements stored in the reactor room storage pits. Cooling is not required due to low burnup and no large decay heat source is present in the UCD/MNRC fuel. Irradiated fuel elements are handled either under water or with a cask. The elements are transferred one at a time so they are in a criticality-safe configuration (Chapter 9).

Some spare, unirradiated, UCD/MNRC fuel elements may be stored in a criticality-safe configuration in the reactor room. These elements require no special handling arrangements or radiation shields.

For some experiments, special core loadings may be required. Fuel elements removed from the core can be placed in a criticality-safe fuel storage rack attached to the tank wall.

Criterion 62: Prevention of Criticality in Fuel Storage and Handling

Fuel-storage capability is provided by storage racks mounted in the tank and fuel storage pits which are located in the reactor room floor. The storage locations are criticality safe due to the geometry utilized and the limited quantity of fuel elements which can be stored (Chapter 9).

Since only one fuel element can be handled at a time, handling does not present a criticality problem.

Criterion 63: Monitoring Fuel and Waste Storage

No residual heat removal or temperature measuring capability is required for irradiated UCD/MNRC fuel elements. Fuel burnup is low, therefore, only a minimum decay heat source is present.

The reactor room and the UCD/MNRC fuel storage area radiation level is monitored with both a RAM system and a CAM system.

Criterion 64: Monitoring Radioactivity Releases

The radiation monitoring system for the UCD/MNRC consists of the RAM's and CAM's. RAM's monitor the reactor room, and selected areas outside the reactor room for gamma activity. There are three CAMs in the facility.

The UCD/MNRC exhaust stack is equipped with a CAM which provides continuous readings of radiation from Ar-41 and beta/gamma particulates released from the facility.

The reactor room CAM monitors the air exhausted from the reactor room for radioactive iodine, beta/gamma particulates, and noble gases. Actions initiated to reduce the release of radioactivity if the set points of this instrument are exceeded are discussed in Chapter 9 and Chapter 11. The sample lines to this unit are manifolded and valved so that one bay may be monitored at a time. In addition to providing routine surveillance of the bays, this unit will be used to help determine the source of activity should the stack monitor alarm. All three of these CAMs have local readouts and alarms as well as remote readouts and alarms in the reactor control room.

3.2 Classification of Structures, Components, and Systems

The UCD/MNRC reactor does not have structures, components, or systems that are important to safety in the same context as nuclear power plants. For the UCD/MNRC, a loss of coolant event, failure of the protection system, or any other credible accident does not have the potential for causing off-site exposure comparable to those listed in the guideline for accident exposures of ANSI 15.7.

Thus, the UCD/MNRC facility does not have structures, components, and systems requiring a

Category I classification. However, certain structures, components, and systems have been designed vital to MNRC operations though their failure would not result in a potential emergency. These systems are discussed below as wells as how these systems are expected to age for the remaining life of the facility.

3.2.1 MNRC Systems

The following reactor systems and components were considered for their potential susceptibility to degradation with respect to their safety functions as a result prior utilization:

1. Structures
    a. Reactor Tank
    b. Grid Plate
    c. Core Support Structure
    d. Graphite Reflector
2. Instrumentation and Control
    a. NM-1000
    b. NPP-1000
    c. DAC (Data Acquisition Cabinet)
    d. CSC (Control System Console)
    e. Control Rod Drives
    f. Reactor Protective System (SCRAM Circuit)
3. Electrical Power Systems
    a. UPS (Uninterruptible Power Supply)
4. Reactor Coolant Systems
    a. Primary Coolant System
    b. Secondary Coolant System
    c. Primary/Secondary Heat Exchanger
    d. Cooling Tower System
    e. Reactor Water Purification System
5. Air Handling Systems
    a. Reactor Room Ventilation System
    b. Reactor Equipment Room Ventilation System
    c. Radiography Bays' Ventilation System
6. Radiation Safety
    a. Reactor CAM (Continuous Air Monitor)
    b. Stack CAM (Continuous Air Monitor)
    c. Facility RAMs (Radiation Area Monitors)

The UCD/MNRC reactor operates at relatively low powers and temperatures, resulting in low neutron fluence levels and temperature factors that have not structurally affected the reactor tank in any observable manner. Furthermore, the reactor tank is surrounded by a thick, reinforced concrete structure that prevents external forces from being directly transmitted to the tank and precludes movement of the tank, and clearance has been provided between neutron beam tubes connected to the tank's exterior and the tank's surrounding concrete structure to prevent structural loading of the tank wall from thermal expansion. Interior surfaces of the tank wall are visually inspected for damage and corrosion on a regular basis, and the exterior surface of the tank is coated with epoxy and tar-saturated roofing felt to prevent corrosion. There are no penetrations of any kind in the reactor tank.

The UCD/MNRC grid plate has been in use since the completion of the 2 MW upgrade in 1998. It was designed to have a thickness and hole pattern identical to those of other TRIGA® reactors with hexagonal grid patterns. It is visually inspected for wear and corrosion annually during routine fuel element and control rod inspections. A small crack (has not worn through) was observed in the thin section between two fuel element positions of the upper grid plate during an annual fuel inspection approximately 10 years ago. The formation of the crack is believed to be linked to reactor operation at 2 MW when fuel vibrated from natural convection coolant flow. The crack has been monitored since its discovery to ensure that it does not get worse. No changes have been observed in the crack's appearance since its discovery, and it does not compromise fuel element configuration (i.e. coolant channel low) or present a structural risk at this time. However, if the crack were to wear through, the grid locations around the crack can be populated with two graphite elements to eliminate the possibility of reducing coolant flow around fuel.

The UCD/MNRC core support structure was retrofitted to receive the hexagonal grid plate during the 2 MW upgrade in 1998. The core support structure is visually inspected during routine reactor operations, and more thorough visual inspections of the structure are carried out during annual fuel element inspections. No core support structure damage or wear has been identified, and any change in the core support structure that could degrade its ability to perform its function would result in the suspension of reactor operation until suitable repair procedures could be identified and implemented.

The UCD/MNRC reactor's graphite reflector has been in use for the entire life of the facility. No damage, wear, corrosion, or degradation in the performance of the reflector has been observed over its lifetime. Any change in the graphite reflector that could degrade its ability to perform its safety function would result in the suspension of reactor operation until suitable repair procedures could be identified and implemented.

The UCD/MNRC reactor's instrumentation and control system implements NM-1000, NPP-1000, DAC (Data Acquisition Cabinet), and CSC (Control System Console) systems that are configured in an accepted format that meets the intended protection system requirements for reliability, redundancy, and independence for TRIGA®-type reactors. Furthermore, the control and safety systems of these devices are fail-safe and will SCRAM the reactor, should they malfunction, and no control or safety systems are required to maintain a safe shutdown condition. These systems are tested regularly, and they have undergone maintenance as required throughout their service life. However, they do not show any significant signs of degradation beyond that expected of typical systems of a similar age and service history. Failure of any reactor instrumentation or control device would result in the immediate suspension of reactor operation until any damaged components could be repaired or replaced.

The control rod drives are standard TRIGA® drive mechanisms manufactured by GA. They are inspected and maintained regularly, and they do not show any significant signs of degradation beyond that expected of typical systems of a similar age and service history. A control rod drive failure would result in the immediate suspension of reactor operation until the damaged unit could be repaired or replaced.

The reactor protective system implements a series of relays to interrupt control rod magnet current and immediately insert all control rods in the event of any SCRAM action. This system is designed to be failsafe and will SCRAM the reactor if a system component malfunctions. Furthermore, the reactor protective system employs two SCRAM loops that use different input signals to provide redundancy in the SCRAM capability. Failure of any of reactor protective system component would result in the immediate suspension of reactor operation until all effected components could be repaired or replaced.

The UPS (Uninterruptible Power Supply) system is not required to safely operate the UCD/MNRC reactor, but it does provide additional operating power that can facilitate the continued observation of reactor conditions and radiation monitors. The UPS is cleaned, inspected, and load-tested regularly, and batteries are replaced as needed. The system does not show any significant signs of degradation beyond that expected of typical systems of a similar age and service history, and the unit can be replaced if it fails.

The reactor coolant systems have not shown any significant signs of degradation beyond that expected of typical systems of a similar age and service history. All piping, valves, meters, and other cooling system components are subject to regular inspection, and consumable items, such as the reactor water purification system's resin tanks, are replaced as needed. However, some system components, such as the pumps and certain cooling tower components have finite service lives and may need to be replaced within the lifetime of the facility. Any observed degradation in a cooling system component that could compromise reactor operations would result in the suspension of reactor operation until suitable repair procedures could be identified and implemented.

The UCD/MNRC air handling system components associated with reactor room, reactor equipment room, and radiography bay ventilation serve safety roles (see Chapter 9, Section 9.5 for more details) that could be diminished as a result of component failure. The ducting, dampers, exhaust fans, and air conditioning units that comprise these systems are inspected and maintained regularly, and they do not show any significant signs of degradation beyond that expected of typical systems of a similar age and service history. However, some system components, such as the exhaust fans that service the reactor room (EF-1) and radiography bays (EF-2) and the air conditioning unit that ensures positive pressure in the reactor equipment room with respect to the reactor room during recirculation mode in the reactor room (AC-2), have finite service lives and may need to be replaced within the lifetime of the facility.

The reactor CAM, stack CAM, and facility RAMs all serve critical radiation safety roles. These systems are inspected, maintained, and calibrated regularly. They do not show any significant signs of degradation beyond that expected of typical systems of a similar age and service history.

3.3 Wind and Tornado Considerations

The UCD/MNRC reactor core is protected from damage by high winds or tornadoes by virtue of its below grade location and the thick reinforced concrete structure surrounding the reactor tank. The superstructure of the UCD/MNRC has been designed for area wind loads.

3.4 Flood Protection

As discussed in Chapter 2, flooding is not expected at the UCD/MNRC site. However, even if flooding occurred, reactor safety would not be an issue since the core is located in a water pool.

3.5 Missile Protection

Missile protection is provided for the UCD/MNRC reactor by virtue of the building design and the below grade location of the core which is surrounded by a seven (7) ft thick reinforced concrete block (see Chapter 1 for building design). Chapter 13 also shows that an aircraft accident damaging the facility is not probable.

3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

There are no pipes in the UCD/MNRC facility capable of flooding the radiography bays to the first floor level. Furthermore, the lowest elevation in the UCD/MNRC (Bay 1 floor) contains a sump. If a predetermined water level is reached, the sump pump will automatically start.

3.7 Seismic Design

The UCD/MNRC site is in a UBC Zone 3 risk area (Chapter 2). The UCD/MNRC building, reactor foundation, shielding structure, reactor tank, and core support structure are designed in accordance with AFM-88-15, Chapter 13 and UBC Zone 3 requirements with an importance factor of 1.5. Meeting these requirements will ensure that the reactor can be returned to operation without structural repairs following an earthquake likely to occur during the plant lifetime. Furthermore, failure of the reactor tank and loss of the coolant in the event of a very large earthquake has been considered in Chapter 13 and the consequences found acceptable from the standpoint of public safety.

3.8 Design of Category I Structures

The UCD/MNRC facility does not have any Category I structures.

3.9 Mechanical Systems and Components

3.9.1 Control-Rod Drives

The control-rod-drive assemblies for all control rods are mounted on the reactor bridge structure. The drives are standard TRIGA® drive mechanisms manufactured by GA. A drive mechanism for the shim and regulating rods is shown in Figure 3.1. The mechanism consists of a stepping motor and reduction gear, a rack and pinion, an electromagnet and armature, a dashpot assembly, and a control-rod extension shaft. Rod-position data are obtained from potentiometers. Limit switches are provided to indicate the up and down positions of the magnet and the down position of the rod. The drive motor is of the stepping type and is instantly reversible. The nominal drive speed for the shim and the regulating rods is 24 in./min.. The stepping motor speeds are adjustable with a maximum rod speed of 42 in./min.. The ability to change the rod drive speed is administratively controlled and access to the area is limited to authorized personnel only. Rod withdrawal reactivity insertion accidents use this maximum rate (Chapter 13).

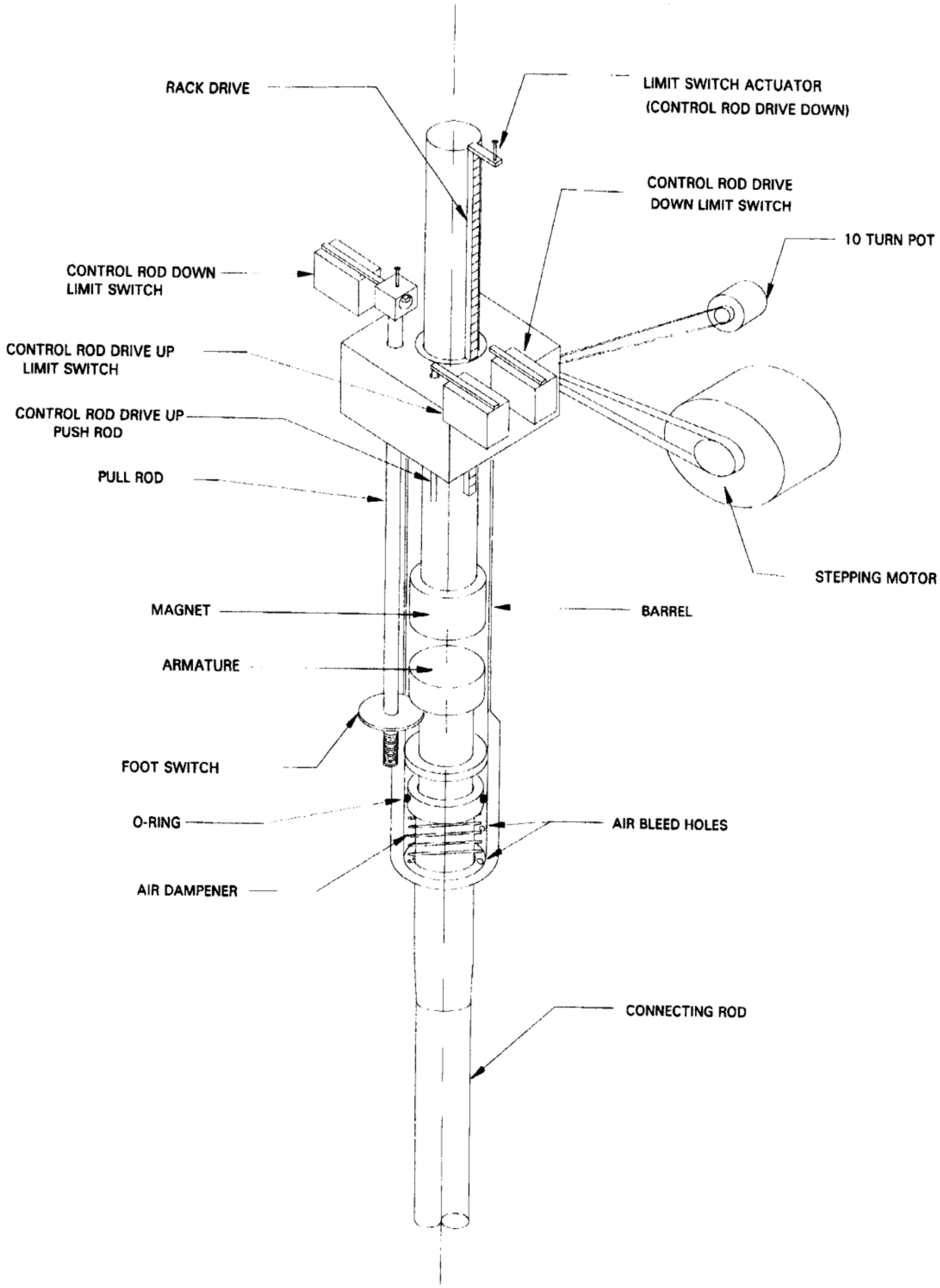


FIGURE 3.1 TYPICAL RACK-AND-PINION CONTROL-ROD-DRIVE MECHANISM

During a scram, the control rod, rod extension, and magnet armature are detached from the electromagnet and drop by gravity. The dashpot assembly slows the rate of insertion near the bottom of the stroke to limit deceleration forces.

The transient rod drive mechanism is shown in Figure 3.2. This is an adjustable fast transient TRIGA® pneumatic pulse drive system. The operability and reliability of this system has been proven over many years of use at Sandia National Laboratories.

3.9.2 Core-Support Structure

The fuel elements and graphite assemblies are supported by the core-support structure shown in Figure 3.3. The UCD/MNRC grid plate has been designed to have a thickness and hole pattern identical to those of other TRIGA® reactors with hexagonal grids.

3.9.3 Instrument Guide Tubes

The nuclear instrument chamber guide tubes for UCD/MNRC are supported by the core support structure as shown in Figure 3.4. There are three guide tubes, but only two are used during normal operations.

3.9.4 Neutron Source

The startup source is approximately 4 Ci of Americium-Beryllium held in a triple encapsulated stainless steel container approximately 3 in. long by 1 in. in diameter. The capsule is held in a container that positions the source near the reactor core centerline (Figure 3.3). Chapter 4 gives a detailed description of the source capsule and holder.

3.9.5 Fuel Storage Assemblies

Five fuel storage racks capable of holding 20 elements each are mounted in the reactor tank. Out-of-tank storage for a complete core is provided by five pits within the reactor room. Each pit has a storage capacity for 38 elements. The storage systems are described in Chapter 9.

3.9.6 Beam-Tube Assemblies

Four beam tubes originate within the graphite reflector approximately 90° apart. The beam tubes are described in Chapter 10 and mounting arrangements are shown in Figure 3.4.

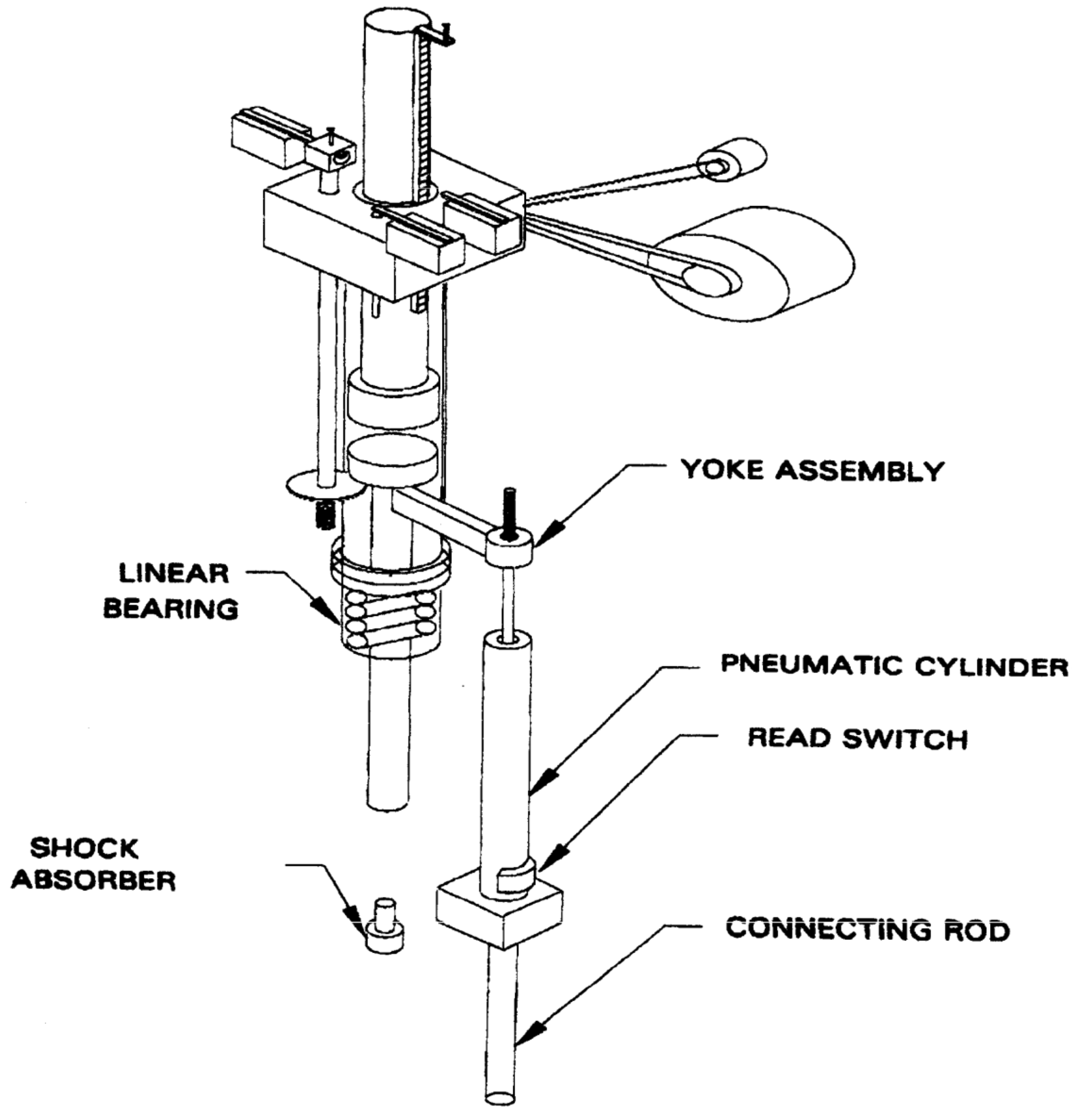
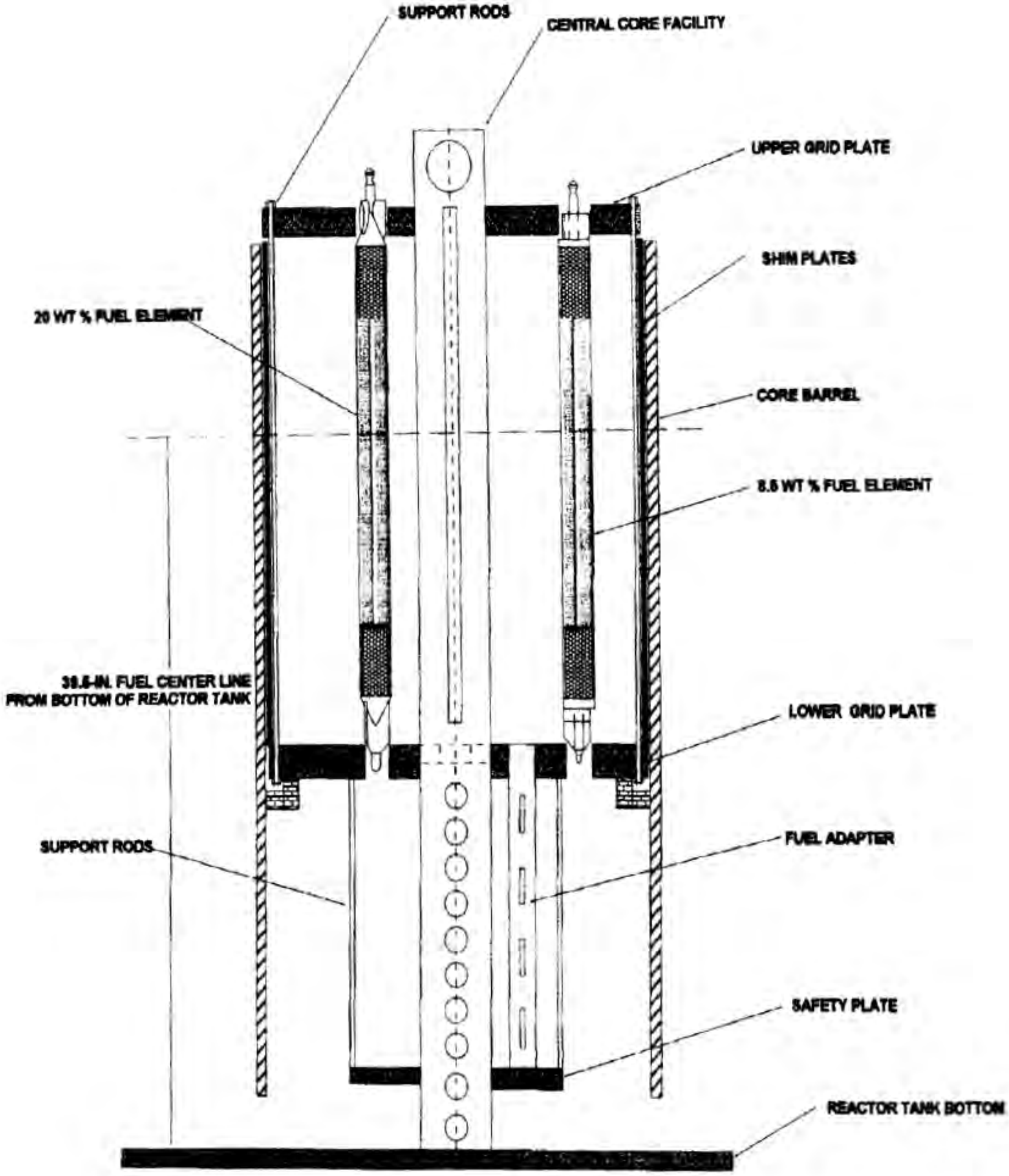


FIGURE 3.2 TYPICAL ADJUSTABLE FAST TRANSIENT ROD DRIVE



NOTE: CORE BARREL IS SUPPORTED BY THE REFLECTOR ASSEMBLY BASE SUPPORT (SEE FIG. 3.4)

FIGURE 3.3 TRIGA® REACTOR

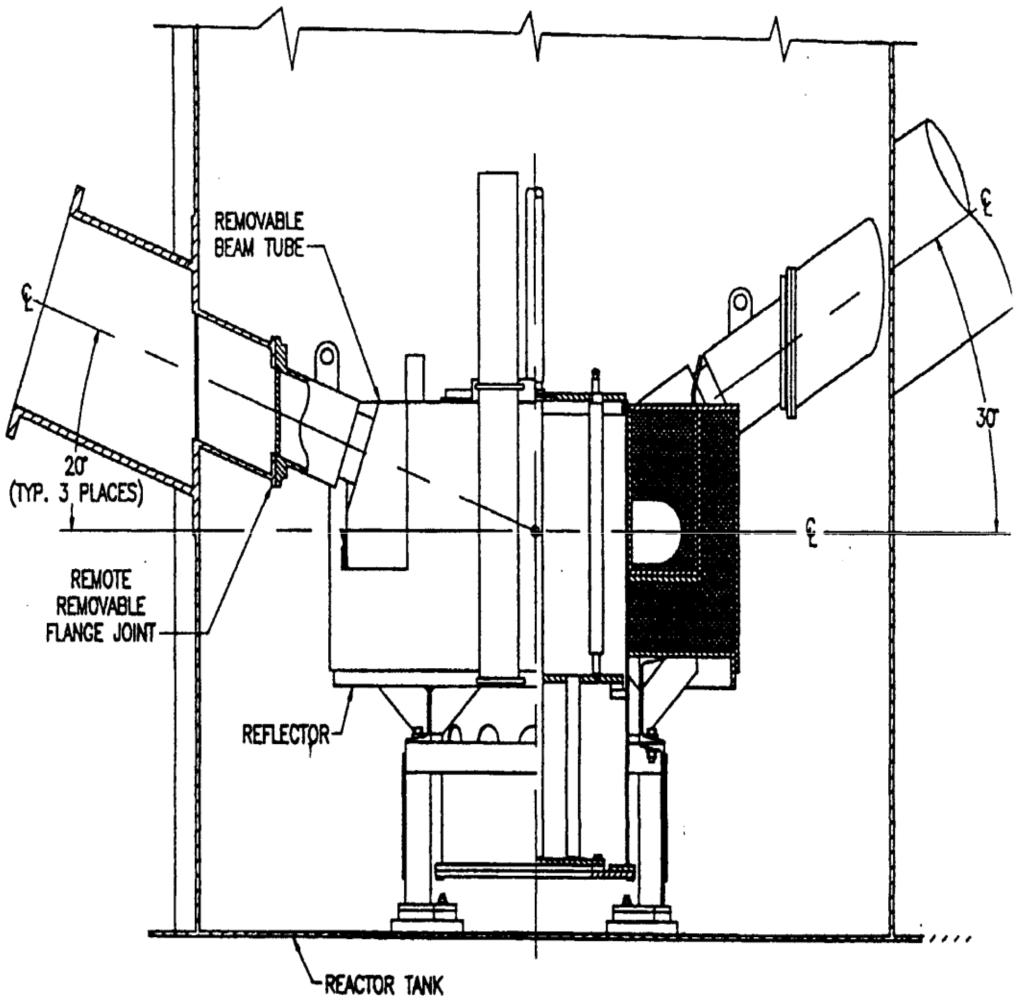


FIGURE 3.4 TYPICAL IN-TANK REACTOR CORE AND BEAM TUBE ASSEMBLY

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CHAPTER 4

UCD/MNRC TRIGA® REACTOR

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4.0 UCD/MNRC TRIGA® REACTOR

4.1 Introduction

The UCD/MNRC reactor is a hexagonal grid, natural convection water cooled TRIGA® reactor originally designed to operate at a nominal 2 MW steady state power as well as pulse and square wave operation. MNRC has not operated routinely above 2 MW nor has the reactor routinely pulsed for more than a decade. For the foreseeable future the MNRC will primarily function to support commercial and research neutron radiography and education/outreach programs. These programs can be accomplished by 1 MW single shift operations without pulsing. In order to operate the MNRC reactor with the largest operational safety margins as possible the reactor is no longer operated in pulse or square-wave mode. Furthermore, the maximum steady-state power is reduced to 1.0 MW. The analysis in this chapter is for 1.0 MW steady-state operation. An analysis for the maximum pulsing magnitude is also provided in this chapter. This analysis is provided, not to inform the largest pulse the reactor may perform (as pulsing is no longer permitted), but to inform the largest amount of negative reactivity experiments that may be place in core and not result in damaging the core under a rapid experiment ejection accident.

The reactor utilizes a specially designed graphite radial reflector to currently accept the source ends of four neutron radiography beam tubes. These beams terminate in four separate neutron radiography bays. The reactor core is located near the bottom of a water-filled aluminum tank 2.13 m (7.0 ft) in diameter and 7.47 m (24.5 ft) deep. The water provides adequate radiation shielding at the top of the tank.

Standard TRIGA® fuel of two types 20, and 30 wt% uranium, each having an enrichment slightly less than 20 % ^{235}U, can be utilized in UCD/MNRC reactor core loadings. Data on 8.5, 12, and 45 wt% will be provided in this chapter as reference only. These fuel types are sometimes referred to here as 20/20 and 30/20 fuel, respectively, where the number preceding the slash is the weight percent of uranium and the number following it is the nominal percent enrichment. Mixed core loadings utilizing 12 wt% standard TRIGA® fuel can also be operated safely at 1 MW; however, no core loading of this type is presently planned and shall be analyzed (requiring a license amendment) prior to use. Likewise, 8.5 wt% is unlikely to be used again at MNRC and shall be analyzed (requiring a license amendment) prior to use. Although reference data for fuels of up to 45 wt% are presented to envelope the fuels authorized at the UCD/MNRC and to quote actual representative data, no fuel above a nominal 30 wt% is authorized.

TRIGA® fuel is characterized by inherent safety, high fission product retention, and the demonstrated ability to withstand water quenching with no adverse reaction from temperatures up to 1100°C (2012°F). The inherent safety of this TRIGA® reactor has been demonstrated by the extensive experience acquired from similar TRIGA® systems throughout the world. This safety arises from the large prompt negative temperature coefficient that is characteristic of uranium-zirconium-hydride fuel-moderator elements used in TRIGA® systems. As the fuel temperature increases, this coefficient immediately compensates for reactivity insertions. The negative compensation results in a mechanism whereby reactor power excursions are terminated quickly and safely (Section 4.5.4.2).

Power distributions were predicted by neutronics calculations and these values were input to thermal-hydraulic calculations so that the fuel and coolant temperatures could be predicted. The main thrust of the reactor physics analysis, as far as safety is concerned, is to identify reactor grid loading patterns that have acceptable values of peak power (temperature), excess reactivity and shutdown reactivity. The reactor physics analysis includes evaluation of the flux that will be attained there, as well as reactivity and power peaking issues associated with this facility.

A typical mixed core arrangement of reactor fuel elements, graphite elements, and control rods is shown in Figure 4.2. The core is made up of approximately 100 fuel-moderator elements including fuel followed control rods and approximately 10 graphite elements.

The reactor core structure is shown in Figure 4.3. The reactor grid plates and fuel/loading are contained within a core barrel approximately 24 inches in diameter by 40 inches in height. The reactor core structure and reflector assembly, shown in Figure 4.4, is a cylinder approximately 43 in. in diameter and 23 in. high currently accommodating four tangential neutron radiography beam tubes. Submerged in the reactor tank, the reflector assembly rests on a platform, which raises the lower edge of the reflector assembly about 2 ft above the tank floor. Coolant water occupies about one-third of the core volume. Cooling of the reactor fuel elements is provided by natural convection of the tank water. The heat dissipated to the tank water is removed by circulating the tank water to a primary to secondary heat exchanger. The heat from the primary system (reactor tank water) is removed by a secondary system cooling tower.

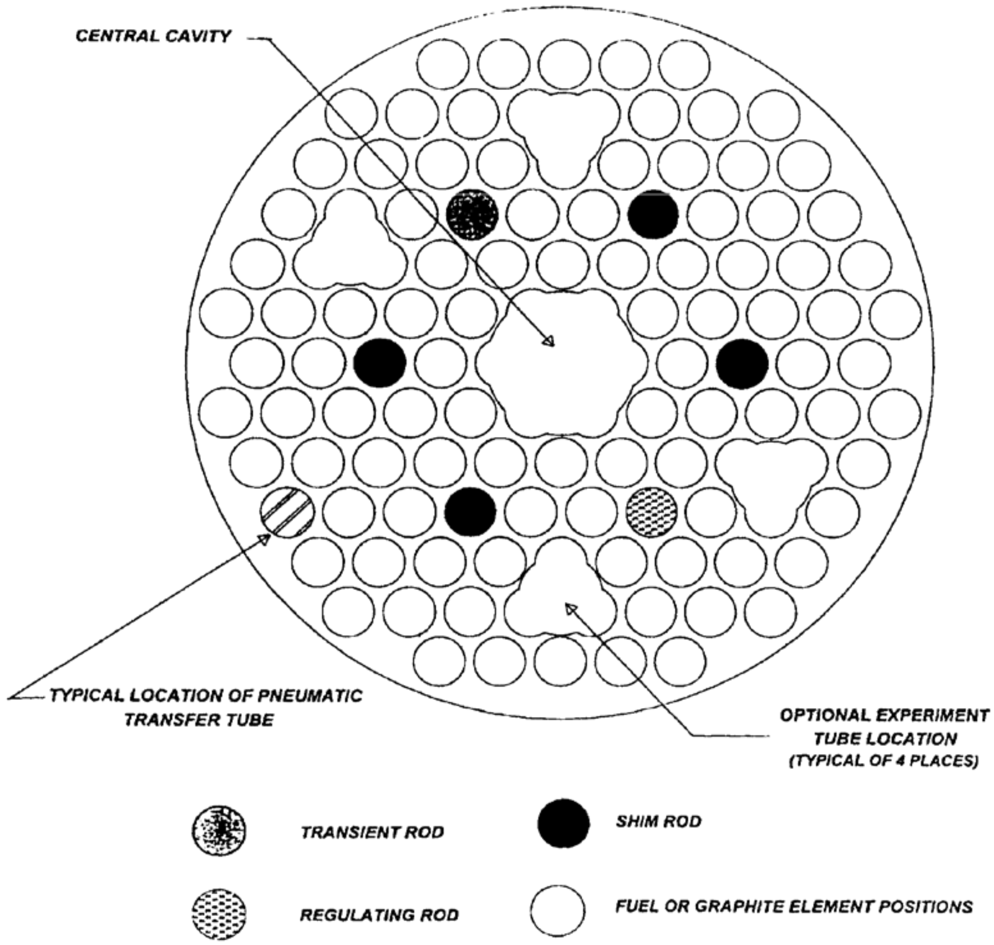
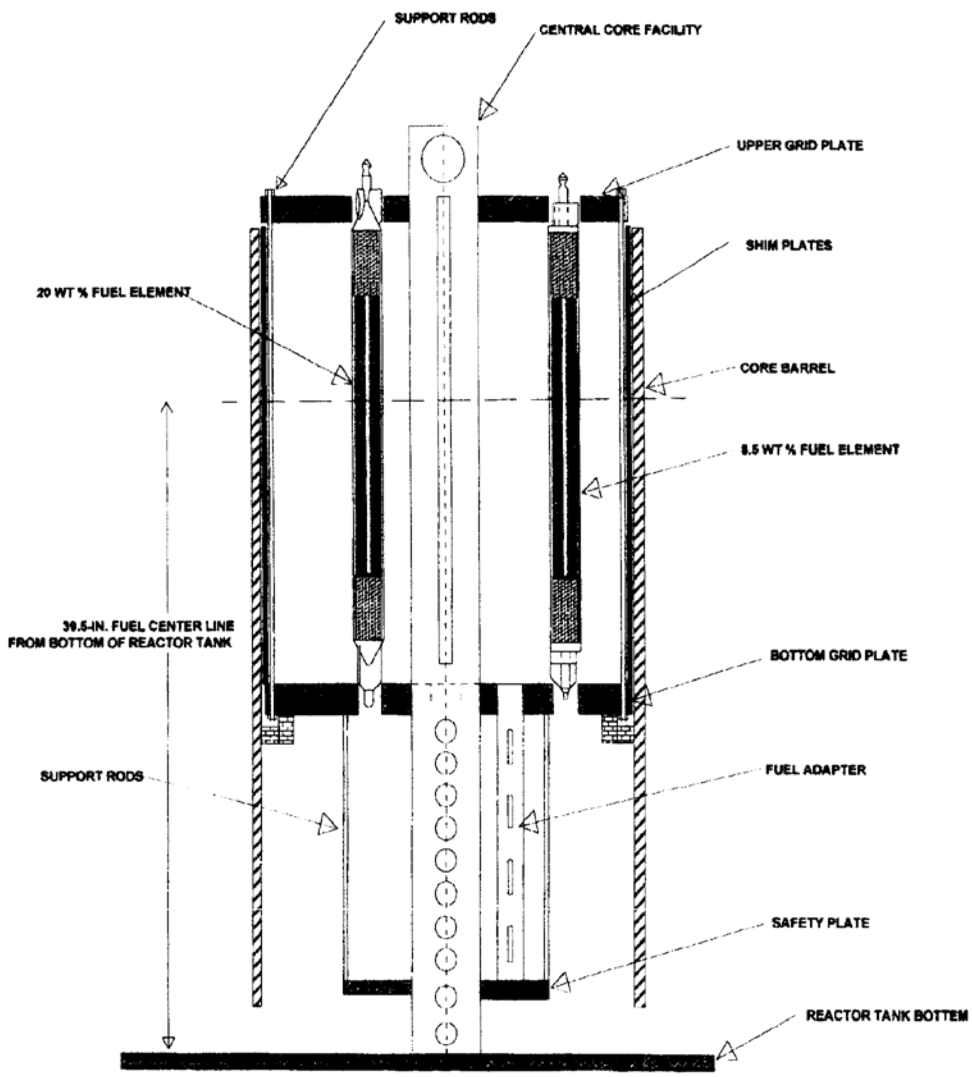


FIGURE 4.1 TYPICAL SIX-CONTROL-ROD CORE ARRANGEMENT



NOTE: CORE BARREL IS SUPPORTED BY THE REFLECTOR ASSEMBLY BASE SUPPORT (SEE FIG. 4.15)

FIGURE 4.2 REACTOR CORE STRUCTURE - ELEVATION

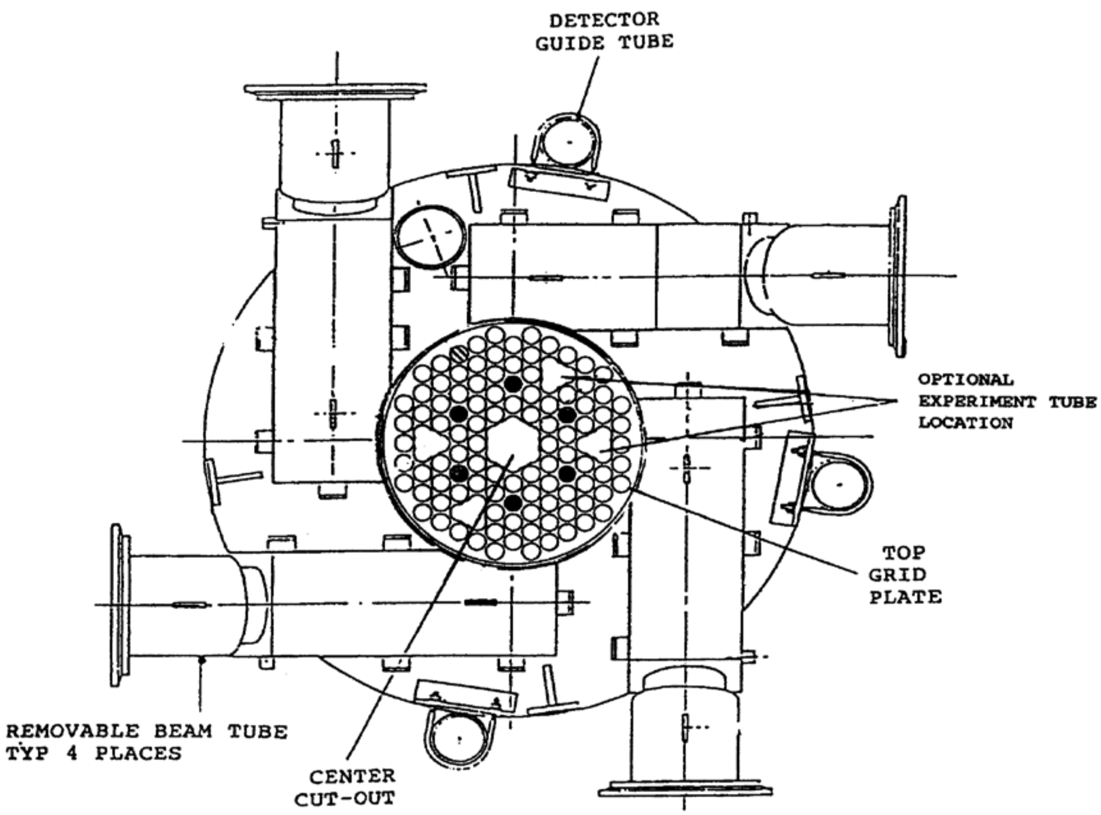


FIGURE 4.3 REACTOR CORE STRUCTURE AND REFLECTOR ASSEMBLY - PLAN VIEW

[redacted]

During steady-state operation, the reactivity in the reactor core is controlled by up to five standard control rods and drives and one transient rod and drive. The standard control rods have fuel followers and are sealed in Type 304 stainless steel tubes approximately 43 in. long by 1.35 in. in diameter. The uppermost section contains a neutron absorber (boron carbide in solid form) and immediately below the neutron absorber is a fuel follower section containing 20, or 30 wt % uranium enriched to less than 20% ^{235}U. These control rods are attached to drive assemblies mounted on a bridge that spans the tank top. The drive assembly consists of a motor and reduction gear driving a rack-and-pinion. The control rod together with its segmented connecting rod is connected to the rack through an electromagnet and armature.

Though the MNRC reactor no longer pulses a transient rod and drive is used as essentially a non-fuel followed control rod. The transient rod drive is also mounted on the reactor bridge and is a combination of the standard GA rack-and-pinion control rod drive and the standard pneumatic fast transient rod drive. The rod is a 44.25 in. long by 1.25 in. in diameter aluminum tube. The top portion of the rod contains solid boron carbide for neutron absorption; the bottom is an air followed section. A complete description of both the pulse and steady-state control rods and drives is contained in Section 4.2.3 and Chapter 7.

The Instrumentation and Control System (ICS) for the TRIGA® reactor is a computer-based design incorporating a GA-developed, multifunction microprocessor-based neutron monitor channel and an analog-type neutron monitoring channel. These two units provide redundant safety channels (percent power with scram), wide-range log power (below source level to full power), period, and multirange linear power (source level to full power). The control system logic is contained in a separate Control System Computer (CSC) with a color graphics display which is the interface between the operator and the reactor. Details of the control system logic can be found in Chapter 7.

4.2 Reactor Core, Associated Structures, and Reactor Experiment Facilities

This section describes, and where appropriate, evaluates the following: the reactor core assembly, the reflector assembly, the grid plates, the safety plate, the fuel-moderator elements, including instrumented elements, the neutron source, the graphite dummy elements, the control rods and drives, the experiment facilities, and the beam tubes. A detailed description of the control rod system can be found in Chapter 7.

4.2.1 Reactor Fuel

4.2.1.1 Fuel-Moderator Element

The active part of each fuel element, shown in Figure 4.5, is approximately 1.43 in. in diameter and 15 in. long. The reactor fuel is a solid, homogeneous mixture of uranium-zirconium hydride alloy containing 20, or 30 % by weight of uranium enriched to 20% ^{235}U. The hydrogen-to-zirconium atom ratio within the UCD/MNRC fuel varies from 1.6 to 1.7. To facilitate hydriding, a small hole is drilled through the center of the active fuel section and a zirconium rod is inserted in this hole after hydriding is complete.

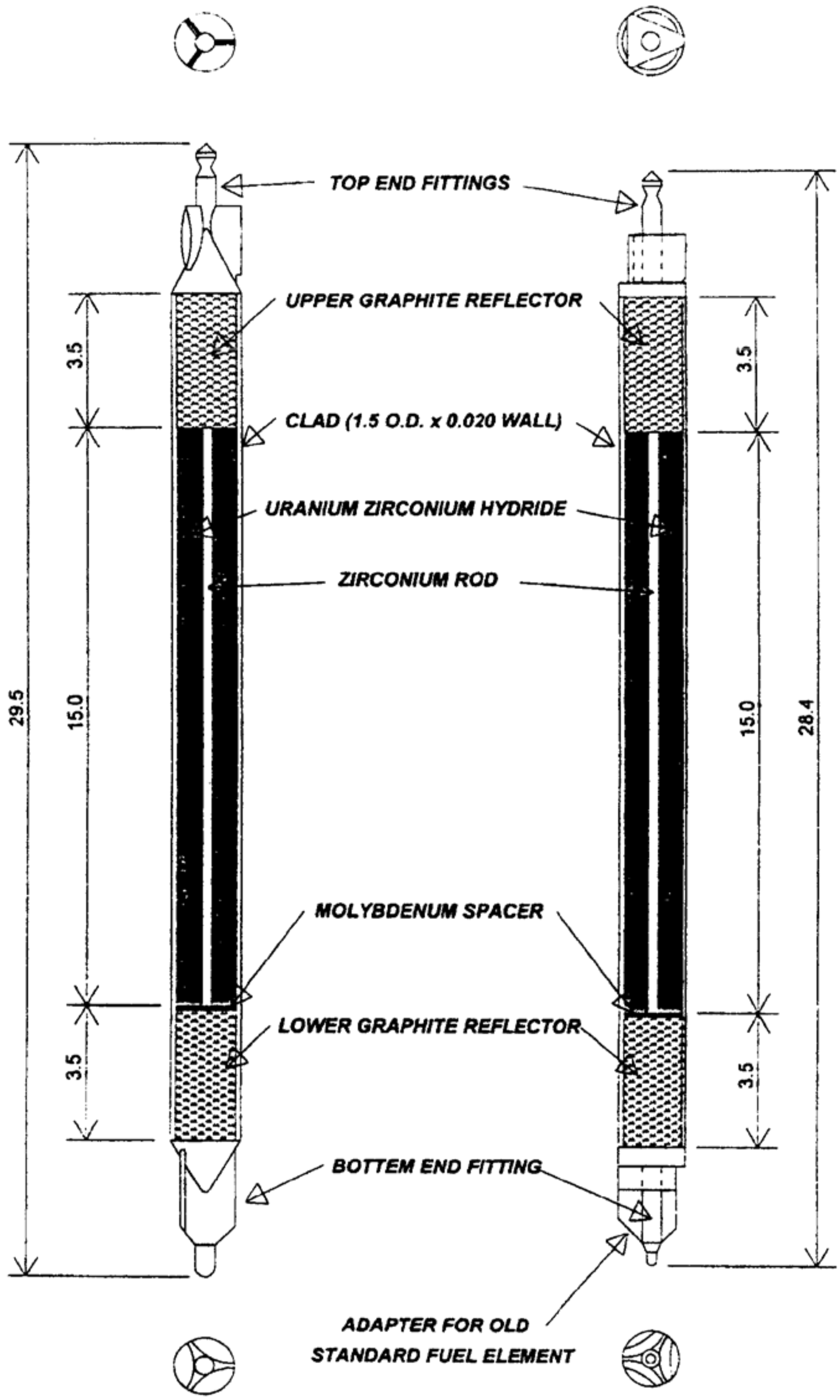


FIGURE 4.5 TYPICAL TRIGA® FUEL ELEMENT ASSEMBLY

Each element is clad with a 0.020 in. thick stainless steel can, and all closures are made by heliarc welding. Two sections of graphite are inserted in the can, one above and one below the fuel, to serve as top and bottom reflectors for the core. Stainless steel end fixtures are attached to both ends of the can, making the overall length of the fuel-moderator element approximately 29.0 inches. Table 4-1 gives a summary of fuel element specifications.

The lower end fixture supports the fuel moderator element on the bottom grid plate. The upper end fixture consists of a knob for attachment of the fuel-handling tool and a triangular spacer (tri-flute), which permits cooling water to flow through the upper grid plate. The total weight of a fully-loaded fuel element is about 3.18 kg (7 lb).

4.2.1.2 Instrumented Fuel-Moderator Element

An instrumented fuel-moderator element has three thermocouples embedded in the fuel. As shown in Figure 4.7, the sensing tips of the fuel element thermocouples are located about 0.3 in. radially from the vertical centerline.

The thermocouple leadout wires pass through a seal in the upper end fixture. A leadout tube provides a watertight conduit carrying the leadout wires above the water surface in the reactor tank. Thermocouple specifications are listed in Table 4-2. In other respects, the instrumented fuel-moderator element is identical to the standard element.

4.2.1.3 Evaluation of Fuel Element Design

The safety limits are determined by the chemical stability of the fuel-moderator material, U-ZrH_x, described in Section 4.5.4.1. At sufficiently high temperatures the zirconium hydride dissociates, creating hydrogen gas pressure that the cladding must be able to contain without deforming or failing. It is the pressure-induced stress compared to the tensile strength of the cladding that determines the safety limits. The chemical stability has been shown to be nearly independent of the uranium weight percent over a range that encompasses the 20% to 30% range for the UCD/MNRC reactor fuel (Section 4.5.4.1.4). The chemical stability of ZrH_x depends on x as well as temperature and, for the high- hydride ($1.6<x<1.7$) fuel used in the UCD/MNRC reactor, the temperature dependence is known quantitatively. Likewise, the tensile strength of Type 304 stainless steel, which is the cladding material for the UCD/MNRC reactor fuel, is known quantitatively as a function of temperature. Using this information, the limit of safe operation, which depends only on the temperatures of the fuel and the cladding, has been determined. One safety limit has been established in the form of a maximum fuel temperature for steady- state operation, where the cladding is assumed conservatively to be at the same temperature as the fuel. It is shown in other sections of this Safety Analysis Report that fuel temperatures remain below this limit in normal and abnormal operation. The dimensional stability of the overall fuel element has been excellent for the TRIGA® reactors in operation. Dimensional stability results from experimental irradiations are summarized in Reference 4.1.

TABLE 4.1 SUMMARY OF FUEL ELEMENT SPECIFICATIONS

	Nominal Value	
Fuel-Moderator Material		
H/Zr ratio	1.6 to 1.7 (actual)	
Uranium content	8.5, 12, 20, 30 wt%	
Enrichment (U-235)	≤ 20%	
20 wt%-Uranium-235	[redacted]	
30 wt%-Uranium-235	[redacted]	
Diameter	1.43 in.	
Length	15 in.	
Graphite End Reflectors	Upper	Lower
Porosity	20%	20%
Diameter	1.35 in.	1.35 in.
Length	Varies	3.47 in.
Cladding		
Material	Type 304 SS	
Wall thickness	0.020 in.	
Length	22.10 in.	
End Fixtures and Spacer	Type 304 SS	
Overall Element		
Outside diameter	1.47 in.	
Length	28.4 in. and 29.5 in.	
Weight	7 lb	

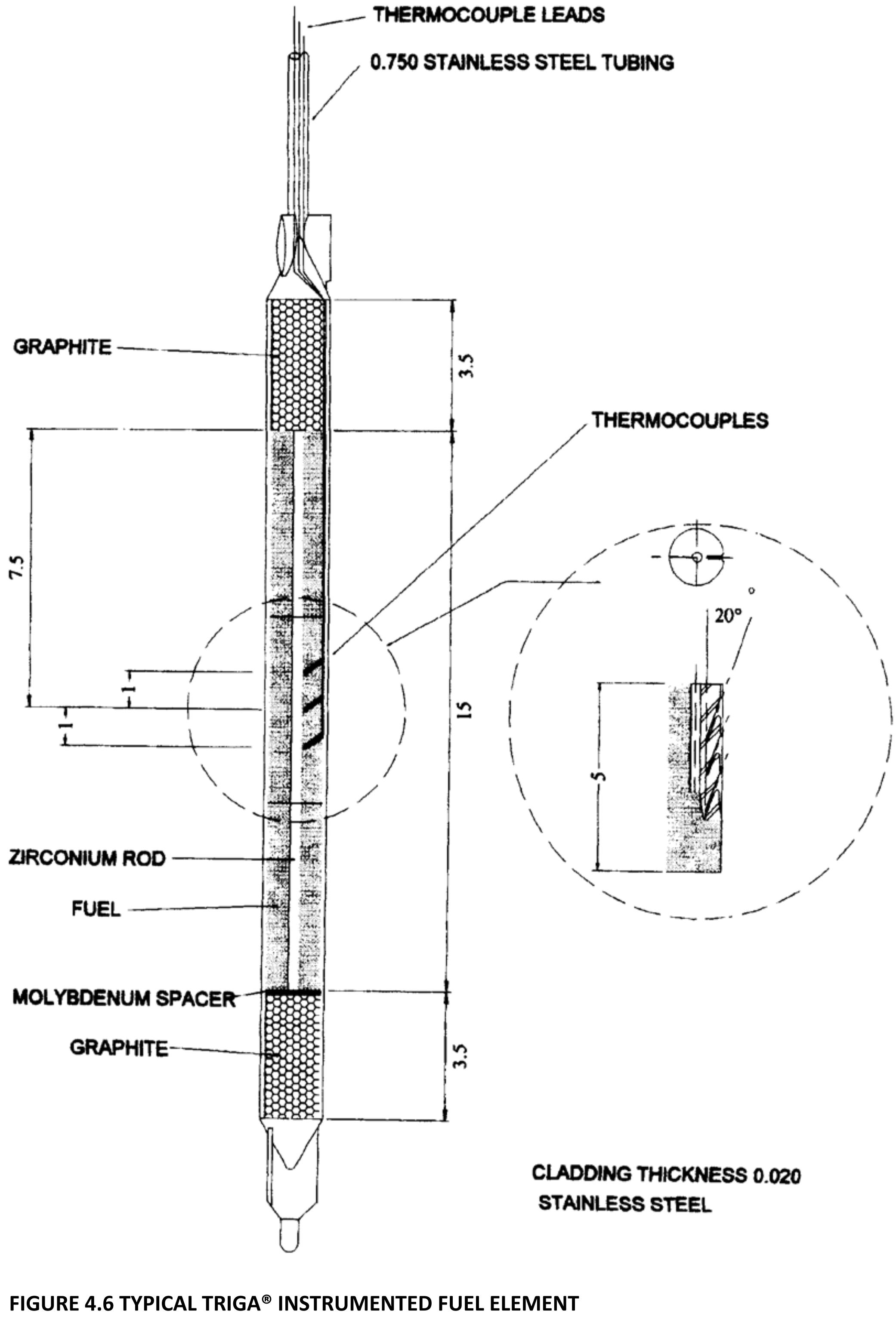


FIGURE 4.6 TYPICAL TRIGA® INSTRUMENTED FUEL ELEMENT

TABLE 4.2 THERMOCOUPLE SPECIFICATIONS

Type	Chromel-alumel
Wire diameter	0.005 in.
Resistance	24.08 ohms/double foot at 68°F
Junction	Grounded
Sheath material	Stainless Steel
Sheath diameter	0.040 in.
Insulation	MgO
Lead-out wire	
Material	Chromel-alumel
Size	20 AWG
Color code	Chromel - yellow (positive)
	Alumel - red (negative)
Resistance	0.59 ohms/double foot at 75°F

4.2.2 Graphite Dummy Elements

Graphite dummy elements, shown in Figure 4.7, may be used to fill grid positions not filled by the fuel-moderator elements or other core components. They are of the same general dimensions and construction as the fuel-moderator elements, but are filled entirely with graphite and are clad with aluminum. Graphite dummy elements can be an integral part of core loadings.

4.2.3 Control Rods

4.2.3.1 Control Function

The reactivity of the UCD/MNRC reactor is controlled by up to five standard control rods and a transient rod. The control and transient rod drives are mounted on a bridge at the top of the reactor tank. The drives are connected to the control and transient rods through a connecting rod assembly.

Every potential core loading includes fuel-followed control rods, i.e., control rods that have a fuel section below the absorber section. The uppermost section is a solid boron carbide neutron absorber. Immediately below the absorber is the fuel section consisting of $U\text{-}ZrH_{1.7}$ enriched in ^{235}U to less than 20%. The weight percent of uranium in the fuel is either 20, or 30, depending on the core loading. The bottom section of the rod has an air-filled void. The fuel and absorber sections are sealed in Type 304 stainless steel tubes approximately 43 inches long by 1.35 inches in diameter.

A detailed description of the control rod system, control rods, and drives is provided in Chapter 7.

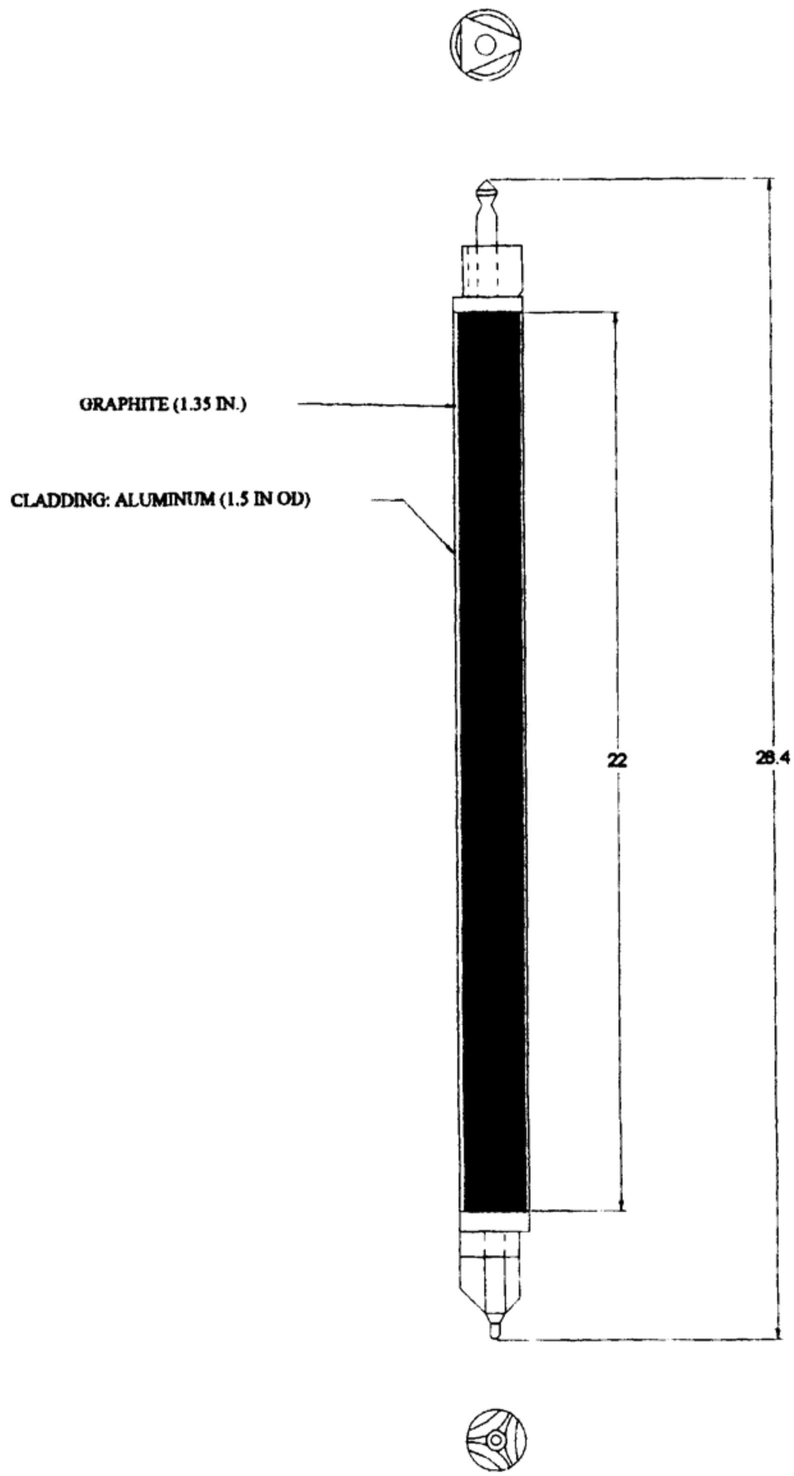


FIGURE 4.7 GRAPHITE DUMMY ELEMENT

4.2.3.2 Evaluation of Control Rod System

The reactivity worth and speed of travel for the control rods are adequate to allow complete control of the reactor system during operation from a shutdown condition to full power. The scram times for the rods are quite adequate since the TRIGA® system does not rely on speed of control as being paramount to the safety of the reactor. The inherent shutdown mechanism of the TRIGA® prevents unsafe excursions and the control system is used only for the planned shutdown of the reactor and to control the power level in steady-state operation.

The reactivity worth of the control system can be varied by the placement of the control rods in the core. The control system has been configured to provide for the excess reactivity needed for 1 MW operation 24 hours per day (including xenon override) and will be capable of providing a shutdown reactivity greater than 50 cents, even with the most reactive control rod in its most reactive position and moveable experiments in their most reactive position.

The nominal speed of the control rods is about 24 in. per minute and the travel is about 15 in. However, the drive system is capable of moving the rods at a maximum speed of 42 in. per minute. Changing the rod speeds is administratively controlled and can only be accomplished by authorized personnel. The area where the control rod drives are located is a restricted access area, only authorized personnel are allowed in the area. The system is fail-safe, that is, multiple failures are required to get uncontrolled rod withdrawals at the maximum speed. It is very unlikely that control rod speed would ever be increased to its maximum.

4.2.4 Reflector Assembly

The reflector, shown in Figure 4.8, is a ring-shaped block of graphite that surrounds the core radially. The graphite is 12.625 in. thick radially with an inside diameter of 21.5 in. and a height of about 22.125 in. The graphite is protected from water penetration by a leak-tight welded aluminum can. Vertical tubes attached to the outer diameter of the reflector assembly permit accurate and reproducible positioning of fission and ion chambers used to monitor reactor operation.

The reflector currently accommodates four tangential neutron radiography beam tubes. This design provides space for a removable in-tank beam tube section referred to as the reflector insert. Each insert begins the shaping of a gradually widening, conical neutron beam from the reactor core to the plane of radiography. Each insert is constructed from a block of graphite surrounded by a leak tight aluminum can. The inserts fit into four perpendicular cutouts in the reflector assembly with each perpendicular cutout being tangential to the reactor core.

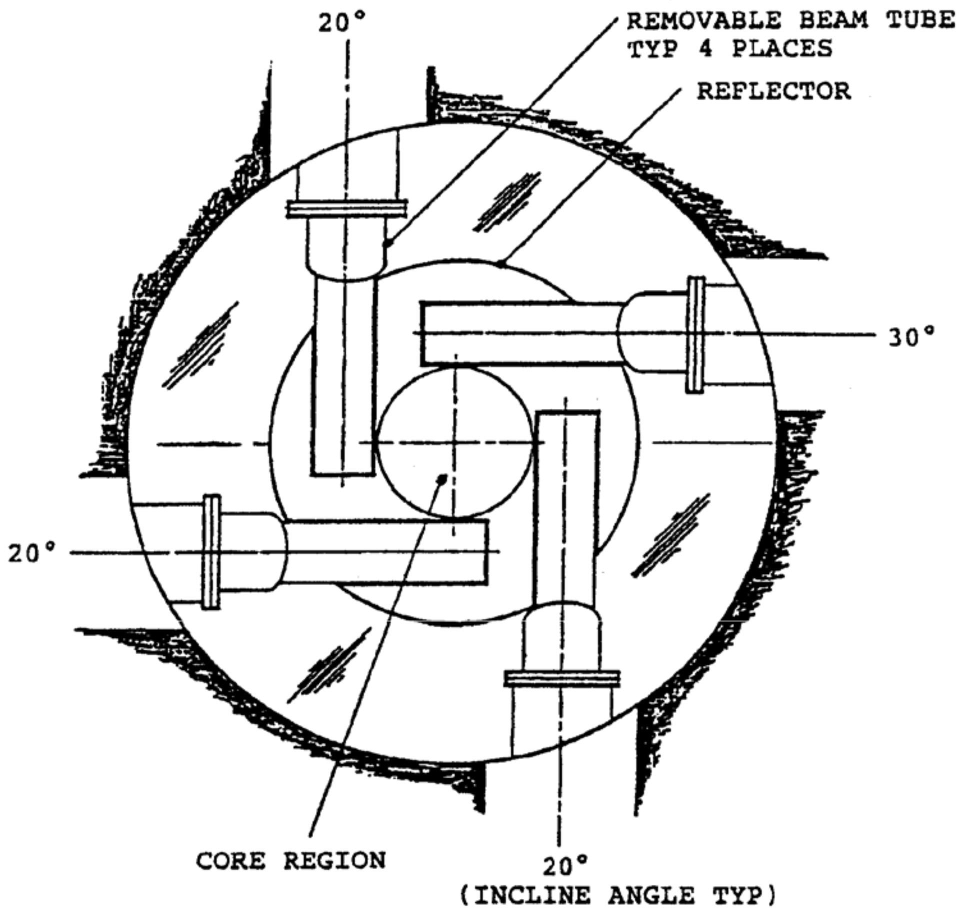


FIGURE 4.8 REACTOR AND BEAM TUBE ASSEMBLY

Since the reactor is located below grade level, the inserts are inclined to direct neutrons upward towards the plane of radiography. Three of the inserts are inclined at a 20° angle and one at a 30° angle.

The reflector assembly rests on an aluminum platform at the bottom of the tank. Four lugs are provided for lifting the assembly.

4.2.5 Neutron Source and Holder

A 4 curie (~9 x 10^6 n/sec) americium beryllium neutron source is used for reactor startup. The source material is triple encapsulated in welded stainless steel. The capsule is approximately 1 in. in diameter and approximately 3 in. long. The neutron source is contained in an aluminum cylindrical shaped source holder. The source holder can be installed in any fuel location in the top grid plate. A shoulder at the upper end of the holder supports the assembly on the upper grid plate. The neutron source is contained in a cavity in the lower portion of the source holder at the horizontal centerline of the core. The upper and lower portions of the holder are screwed together and pinned. Since the upper end fixture of the source holder is similar to that of the fuel element, the source holder can be installed or removed with the fuel handling tool. In addition, the upper end fixture has a small hole through which one end of a stainless steel wire may be inserted to facilitate handling operation from the top of the tank.

4.2.6 Grid Plates

4.2.6.1 Top Grid Plate

The top grid plate is an aluminum plate 21 in. in diameter and 1 1/4 in. thick (3/4 in. thick in the central region) that provides accurate lateral positioning for the core components. The top grid plate is supported by six 1/2 in. stainless steel rods that are attached to the bottom grid plate. Both plates are anodized to resist wear and corrosion.

One hundred twenty one (121) holes, on a 1.714 in. hexagonal pitch are drilled into the top grid plate in seven hexagonal rings to locate the fuel-moderator and graphite dummy elements, the control rods, guide tube, and the pneumatic transfer tube (Figure 4.9)*. The 121 holes includes those associated with the hexagonal and triangular sections described below.

A hexagonal section can be removed from the center of the upper grid plate for the installation of irradiation fixtures into the region of highest flux; this displaces the central seven fuel element positions (Hex Rings A and B, or Grid Positions F-06, F-07, G-05, G- 06, G-07, H-07, and H-06).

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*Two grid numbering systems have been utilized for describing individual positions in the hexagonal grid. The traditional system has numbered the hexagonal rings of the grid starting from A for the inner ring to G for the outer ring, with individual positions sequentially numbered. This grid nomenclature was utilized for the majority of the calculations performed. An operational grid pattern has however been created whereby grid columns are designated by letters and grid rows numerically. Specific grid positions in this document have been referenced by the column, row format for operational ease (Figures 4.10). Some reference calculations refer to the hexagonal ring system to simplify their explanation. A cross reference table has been provided (Table 4-3).

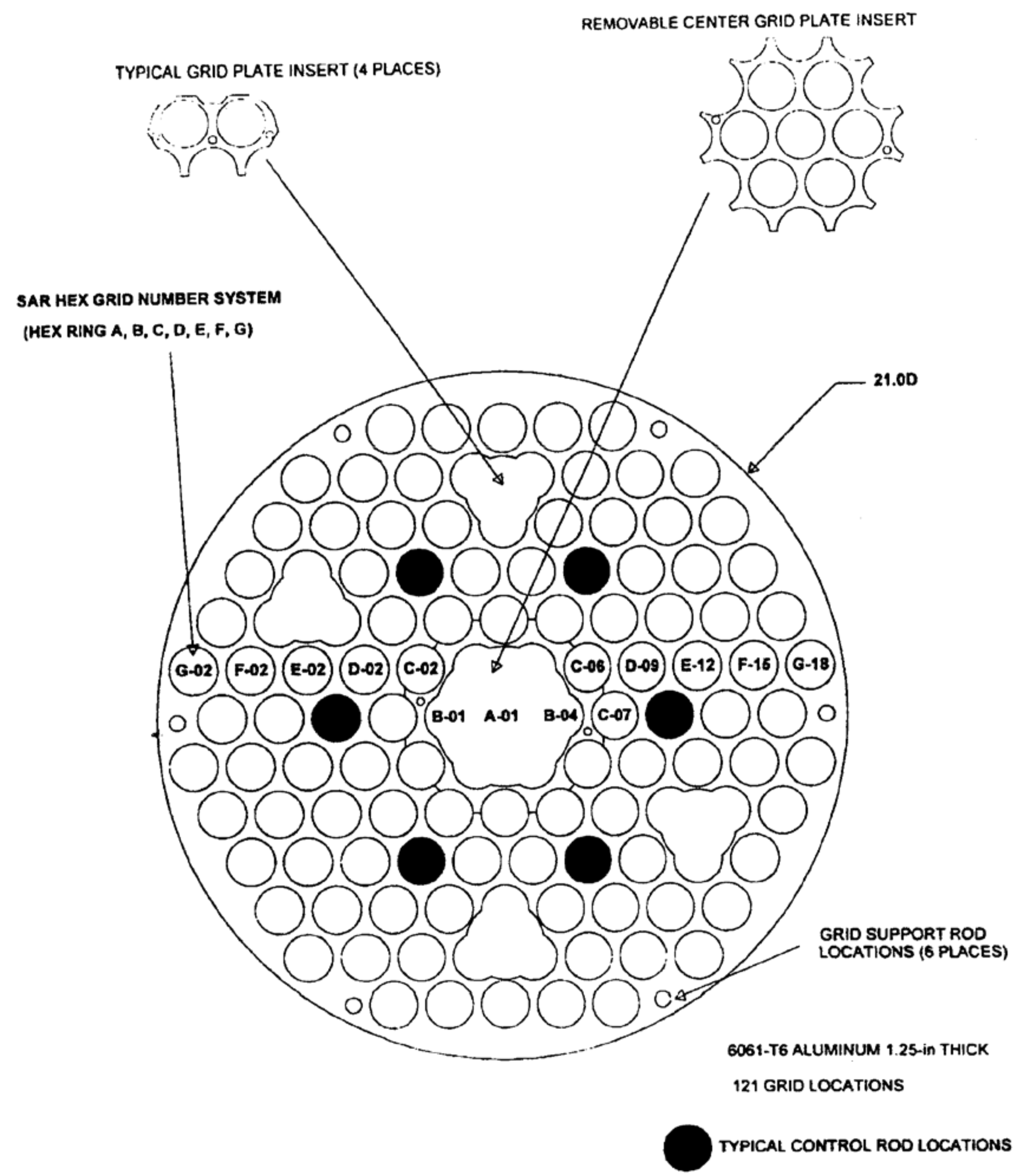

FIGURE 4.9 TOP GRID PLATE

TABLE 4.3 GRID POSITION CONVERSION TABLE

Hexagonal Grid No.	Operational Grid No.	Hexagonal Grid No.	Operational Grid No.	Hexagonal Grid No.	Operational Grid No.
A-01	G-06	E-06	H-10	F-24	D-02
B-01	F-07	E-07	I-09	F-25	C-02
B-02	G-07	E-08	J-08	F-26	B-02
B-03	H-07	E-09	K-07	F-27	B-03
B-04	H-06	E-10	K-06	F-28	B-04
B-05	G-05	E-11	K-05	F-29	B-05
B-06	F-06	E-12	K-04	F-30	B-06
C-01	E-07	E-13	K-03	G-02	B-08
C-02	F-08	E-14	J-03	G-03	C-09
C-03	G-08	E-15	I-03	G-04	D-10
C-04	H-08	E-16	H-03	G-05	E-11
C-05	I-07	E-17	G-02	G-06	F-12
C-06	I-06	E-18	F-03	G-08	H-12
C-07	I-05	E-19	E-03	G-09	I-11
C-08	H-05	E-20	D-03	G-10	J-10
C-09	G-04	E-21	C-03	G-11	K-09
C-10	F-05	E-22	C-04	G-12	L-08
C-11	E-05	E-23	C-05	G-14	M-05
C-12	E-06	E-24	C-06	G-15	M-04
D-01	D-07	F-01	B-07	G-16	M-03
D-02	E-08	F-02	C-08	G-17	M-02
D-03	F-09	F-03	D-09	G-18	M-01
D-04	G-10	F-04	E-10	G-20	L-01
D-05	H-09	F-05	F-11	G-21	K-01
D-06	I-08	F-06	G-12	G-22	J-01
D-07	J-07	F-07	H-11	G-23	I-01
D-08	J-06	F-08	I-10	G-24	H-01
D-09	J-05	F-09	J-09	G-25	F-01
D-10	J-04	F-10	K-08	G-27	E-01
D-11	I-04	F-11	L-07	G-28	D-01
D-12	H-04	F-12	L-06	G-29	C-01
D-13	G-03	F-13	L-05	G-30	B-01
D-14	F-04	F-14	L-04	G-32	A-01
D-15	E-04	F-15	L-03	G-33	A-02
D-16	D-04	F-16	L-02	G-34	A-03
D-17	D-05	F-17	K-02	G-35	A-04
D-18	D-06	F-18	J-02	G-36	A-05
E-01	C-07	F-19	I-02		
E-02	D-08	F-20	H-02		
E-03	E-09	F-21	G-01		
E-04	F-10	F-22	F-02		
E-05	G-11	F-23	E-02		

Four triangular-shaped sections are cut out of the upper grid plate. When fuel elements are placed in these locations, their lateral support is provided by a special fixture. When the fuel elements and support are removed, space is provided for the insertion of experiment tubes up to 2.4 in. outside diameter for placement of experiments.

The UCD/MNRC reactor is equipped with a TRIGA®-type pneumatic transfer system for irradiation of small specimens. The in-core section of this system is typically located in the outer portion of the reactor core.

The differential area between the fitting at the top of the fuel elements and the round holes in the top grid plate permits passage of cooling water through the plate. The grid plate holes are shaped to provide relief at the inlet and outlet edges; there is a taper on both the upper and lower sides of the plate, which reduces the resistance for the coolant flow. All outlet coolant flow is through the flow holes.

4.2.6.2 Bottom Grid Plate

The bottom grid plate is an aluminum plate 1.25 in. thick, which supports the entire weight of the core and provides accurate spacing between the fuel-moderator elements (Figure 4.10). Six adapters are bolted to pads welded to a ring which is, in turn, welded to the core barrel to support the bottom grid plate.

Holes 1.25 in. in diameter in the bottom grid plate are aligned with fuel element holes in the top grid plate. They are countersunk to receive the adaptor end of the fuel-moderator elements and the adaptor-end of the pneumatic transfer tube.

Eight additional 1.505 in. diameter holes are aligned with upper grid plate holes to provide passage of fuel-follower control rods. Those holes in the bottom grid plate not occupied by control rod followers are plugged with removable fuel element adaptors that rest on the safety plate. These adapters are aluminum tubing 1.5 in. OD x 1.25 in. ID by 18 in. long. Slotted channels are machined in the sides of the tubing to provide for coolant flow. At the lower end is a fitting that is accommodated by a hole in the safety plate. The upper end of the cylinder is flush with the upper surface of the bottom grid plate when the adaptor is in place. This end of the adaptor is countersunk similar to that in the bottom grid plate for accepting the fuel element lower end fitting. With the adaptor in place, a position formerly occupied by a control rod with a fuel follower will now accept a standard fuel element.

4.2.6.3 Safety Plate

A safety plate is provided to preclude the possibility of control rods falling out of the core (Figure 4.12). It is a 1 inch thick machined aluminum plate that is suspended from the lower grid plate by 18.25 inch long stainless steel rods.

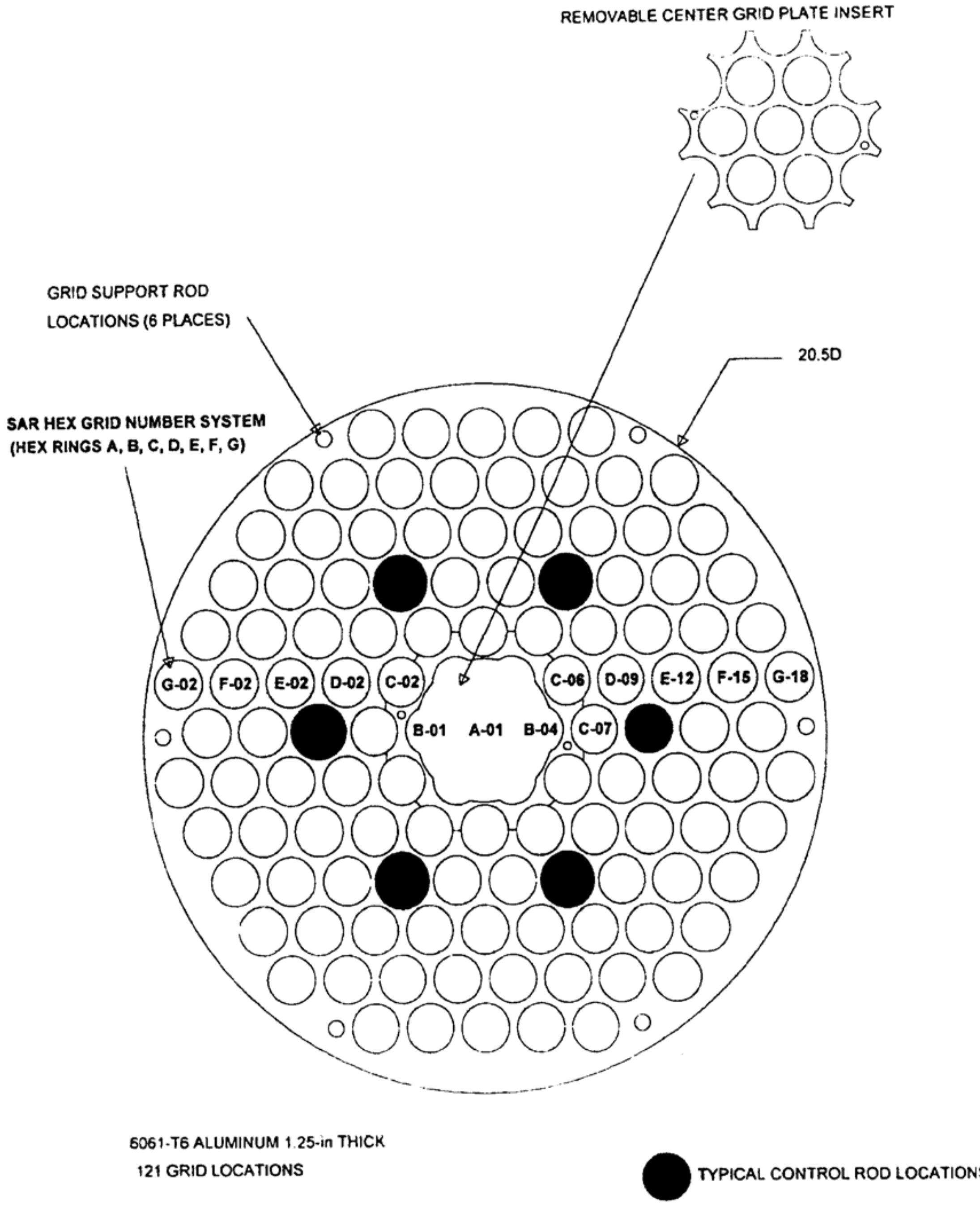

FIGURE 4.10 BOTTOM GRID PLATE

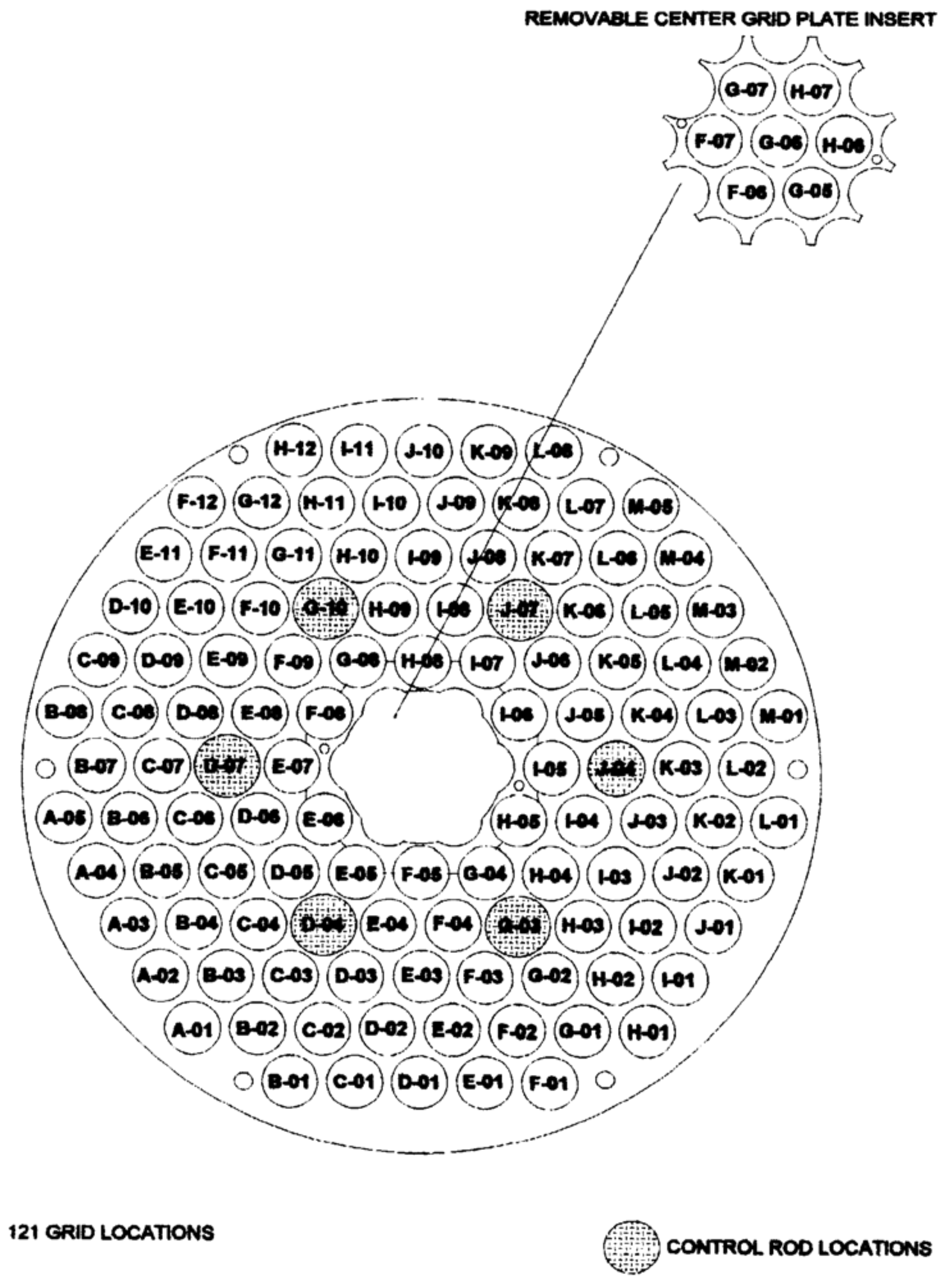

FIGURE 4.11 OPERATIONAL GRID NOMENCLATURE

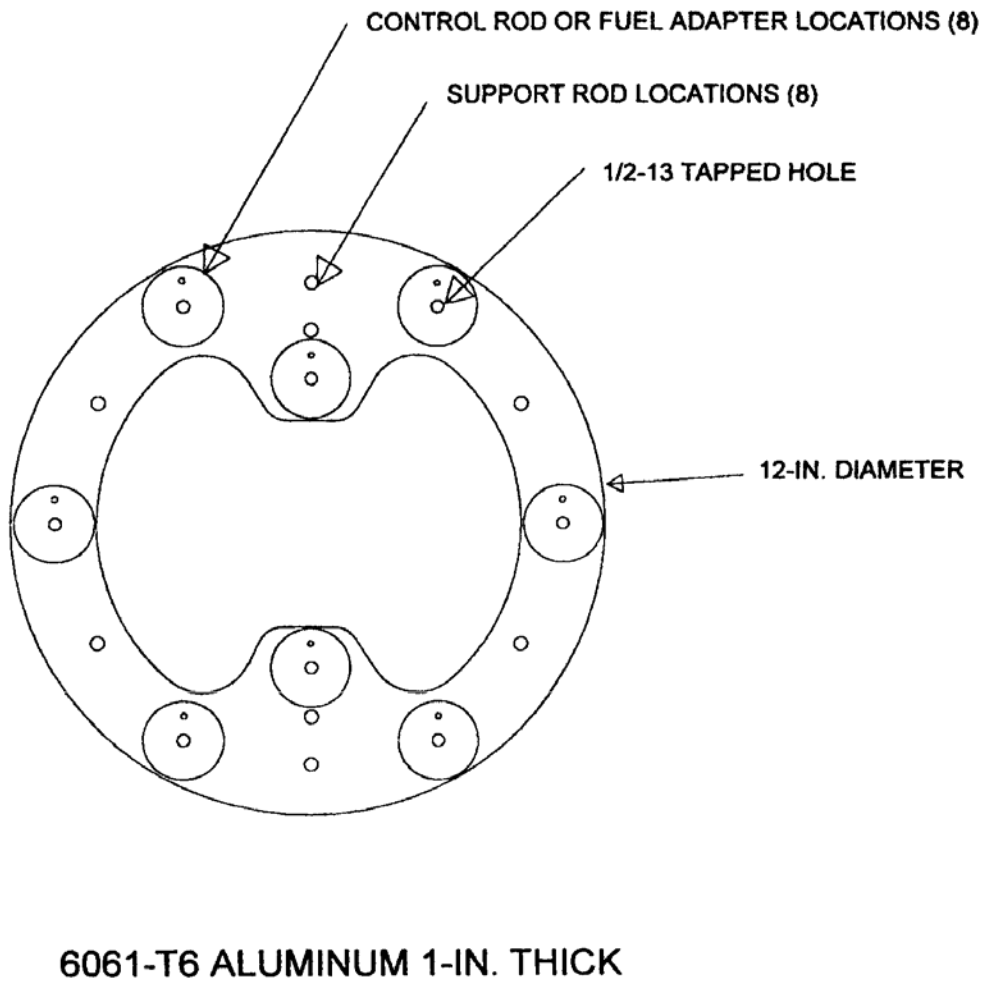

FIGURE 4.12 SAFETY PLATE

4.3 Reactor Tank

The UCD/MNRC reactor core assembly is located near the bottom of a cylindrical aluminum tank surrounded by a reinforced concrete structure (Figure 4.12). The reactor core and beam tube assembly installation is shown in Figure 4.13. The reactor tank is a welded aluminum vessel with 1/4 in. thick walls, a diameter of approximately 7 ft., and a depth of approximately 24-1/2 ft.. The tank is all-welded for water tightness. The integrity of the weld joints is verified by radiographic testing, dye penetrant checking, and leak testing. The outside of the tank is coated for corrosion protection.

Presently four beam tubes clamp onto the reactor tank at 90° interval spacing tangential to the reflector assembly and core (Figures 4.7). The tank wall section of the beam tubes consists of a 12-½ in. diameter pipe welded to the tank wall. These special flanges are welded to the in-tank end for water tightness. The beam tubes clamp onto the tank wall and extend through the bulk shielding concrete that surrounds the reactor tank. Three beam tubes are positioned at a 20° angle from horizontal and a fourth beam tube is positioned at a 30° angle from horizontal as shown in Figure 4.13.

4.4 Biological Shield

The reactor tank is surrounded by a monolithic reinforced standard concrete bulk shield structure. Below ground level, the concrete is approximately [redacted] thick. Above ground level, the concrete varies in thickness from approximately [redacted]., with the smaller dimension at the tank top. The tank is supported by a concrete pad approximately 9-1/2 ft. thick.

The massive concrete bulk shield structure provides radiation shielding for personnel working in and around the UCD/MNRC. Also, the massiveness of the concrete bulk shield structure provides excellent protection for the reactor core against natural phenomena that could result in damage to the reactor core.

4.5 Primary Coolant

The primary coolant of the MNRC is comprised of ultra-pure light water. No chemical shims of any kind are utilized at MNRC. Along with the hydrogen in the fuel meat, the graphite top and bottom caps inside the fuel elements, the graphite reflector, the primary coolant supplies the final moderation for the reactor to achieve criticality. If the reactor were to experience a complete LOCA the reactor would become subcritical even with all control rods withdrawn and the maximum licensed reactor excess reactivity.

As with all TRIGA reactors the fuel spacing results in an under moderated reactor in all conditions. Due to the fact the TRIGA reactor is a natural convection reactor the temperature water inside the core does not change strongly with reactor temperature (e.g. 0.5 MW vs 1.0 MW). The slight increase in core water temperature as the reactor thermal output increases will lower the density of that water slightly and result in a further under moderated reactor, which represents a weak negative reactivity feedback. No safety credit is taken for this feedback as the effect is very small when compared to the negative feedback of other aspects characteristic of TRIGA fuel.

4.6 Nuclear Design

4.6.1 TRIGA® Fuels

This section provides a brief description of TRIGA® fuels followed by evaluations of neutron physics considerations, materials properties, irradiation performance, fission product release, pulse heating, and limiting design basis (Reference 4.1).

4.6.1.1 Description of TRIGA® Fuels

The uranium-zirconium hydride fuel used in TRIGA® reactors is fabricated by hydriding an alloy that is a solid solution of uranium in zirconium. The zirconium is selectively hydrided, and the uranium remains as small metallic inclusions in the zirconium hydride matrix. The size of the uranium particles increases from 1 to 5 µm with increasing uranium content from 8.5 to 45 wt%. Some important parameters for TRIGA® fuels are provided in Table 4-4.

[redacted]

FIGURE 4.13 REACTOR TANK

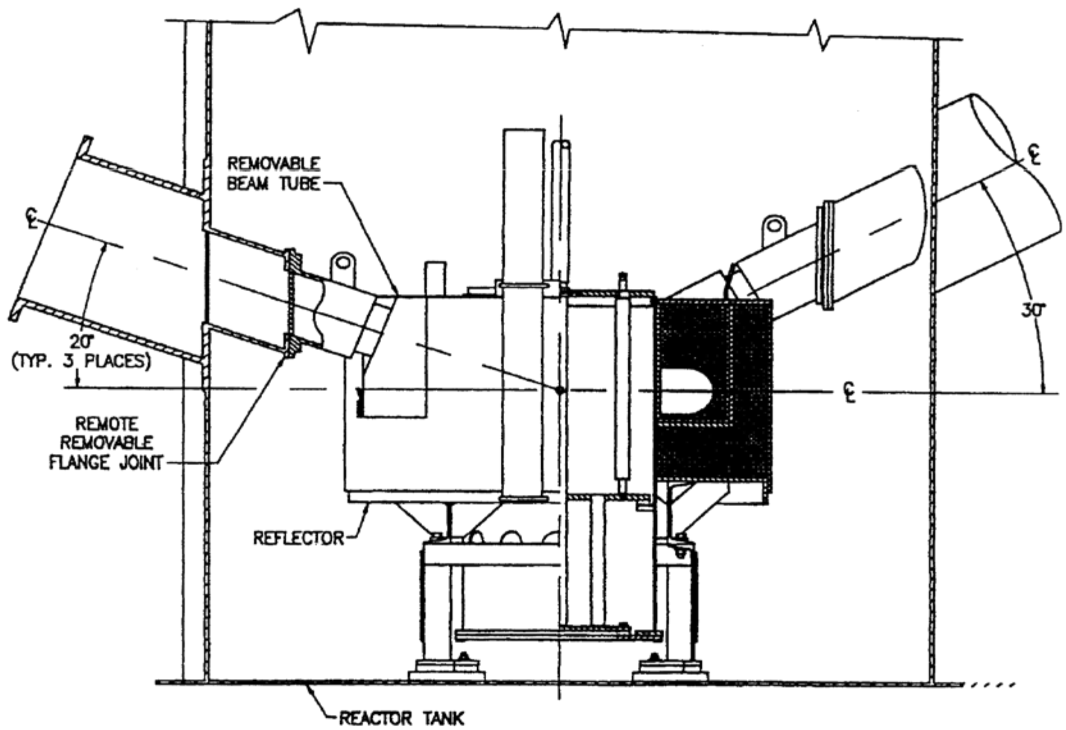

FIGURE 4.14 TYPICAL IN-TANK REACTOR CORE AND BEAM TUBE ASSEMBLY

TABLE 4.4 PARAMETERS FOR TRIGA® LEU FUELS

Type of Fuel	Weight Percent		Uranium-235 (g/element)	Uranium Enrichment (%)	$\alpha \times 10^5$ ($\Delta k/k°C$)	Core lifetime (Mwd)	Uranium Volume (%)
	Uranium	Erbium					
Original LEU*	8.5	0.0	[redacted]	20	9.5	100	2.6
LEU*	12	0.0		20	10	600	3.8
LEU	20	0.5		20	10.5	1200	6.8
LEU	30	0.9		20	8	3000	11.2
LEU*	45	0.4-0.9		20	6-9	4000	19.5

*Given for reference only.
LEU = low-enrichment uranium

The use of erbium burnable poison in conjunction with the higher ^{235}U loadings permits longer core lifetimes than would be obtainable with the original TRIGA® fuel. It also permits maintaining a large prompt negative temperature coefficient of reactivity, α, that is changed little from that of the original fuel through at least the 30-wt% LEU fuel. As shown in Table 4-4, the volume percent (v%) of uranium increases with the increasing uranium loading but remains a small value, increasing from 2.6 v% in the original fuel to 11.2 v% for the 30-wt% LEU fuel and to 19.5 v% for the 45 wt% fuel.

4.6.1.2 Performance of Erbium Loaded Fuels

The primary intent of a GA Technologies Reactor Physics Qualification Program was to show that neutronically the 20-20 and 30-20 TRIGA® LEU fuels behave essentially the same as the TRIGA® Fuel Life Improvement Program (FLIP) fuel. The following was concluded (Reference 4.1):

- The power peaking factors in the LEU and FLIP fuels are very comparable. Any variations are due mainly to differences in the contained U-235 (not the total uranium loading).

- The prompt negative temperature coefficient, the reactivity worth, and the core lifetime of the TRIGA® LEU and FLIP fuels are comparable primarily because of the adjustment of the erbium poison concentration. Also, the reactor kinetics parameters most important to power/burst behavior, prompt neutron lifetime, and effective delayed neutron fraction are similar.

The data in Table 4-4 represent a single average estimate of α. References 4.2 and 4.3 present plots of the effect of fuel temperature on the prompt coefficient for 20 wt% fuel. These sources show that, at low burnup, the feedback coefficient is a strong function of temperature, ranging from about -7×10^{-5} at 200°C to about -15×10^{-5} at 700°C. From the curve in Reference 4.2, an integrated average over the 23 to 750°C range is -10.9×10^{-5}/°C. Similarly, the curve from Reference 4.3 shows that for the ranges 23 - 800° and 23 - 1000°C, the averages are -11.05×10^{-5} and -11.8×10^{-5}/°C, respectively. All of these values demonstrate that the prompt negative feedback characteristics are retained with the erbium additions to the 20 and 30 wt% fuel.

4.6.1.3 Materials Properties

The materials properties of TRIGA® fuels with higher uranium contents were reviewed relative to those 8.5 wt% TRIGA® fuels (Reference 4.1) with the following conclusions.

Measurements were made of the thermal conductivity of 8.5-, 30-, and 45-wt% uranium-zirconium hydride fuels. The data from these measurements, in conjunction with density and specific heat data were used to determine the thermal conductivity of these materials. The thermal conductivity was found to be independent of uranium content within this range.

The specific heat of uranium-zirconium hydride was calculated as a function of uranium content using known specific heats for uranium and zirconium hydride and a linear interpolation. This method is a straightforward and acceptable approach, and the resulting values for heat capacity have been adequately factored into the analyses of kinetic behavior of the higher loaded LEU fuels.

The coefficient of thermal expansion was measured for 45-wt% uranium fuel and compared with that for 8- to 12-wt% fuel. For a maximum power density TRIGA® fuel element, the maximum radial expansion would be about 0.6% for 45-wt% fuel as compared with 0.5% for 8.5-wt% fuel, which is not a significant change.

The monitoring of hydrogen pressure during hydriding in the fabrication of high uranium content fuels showed that the equilibrium hydrogen dissociation pressure of the fuel depends only on the hydrogen/zirconium (H/Zr) ratio and the fuel temperature. It is independent of the uranium content.

Thermal cycling tests were performed on 45-wt% uranium fuel over the temperature range of 500 to 725°C, which includes the orthorhombic-to-tetragonal phase transformation at 653°C. Specimens were cycled 100 times out of pile and then 32 times in a neutron flux of 4 x 10^{12}n/cm^2·s. There were no significant changes in dimensions in the out-of-pile tests, and a small decrease in weight was measured. The in-pile cycling test showed a small decrease in both length and diameter, which may be related to a loss of hydrogen. The dimensional stability of the high uranium content fuel is understandable considering the fine dispersion of the uranium in the zirconium hydride matrix. The dispersion of uranium in particles less than 5 µm in diameter evidently precludes anisotropic growth during cycling through the phase transformation because of accommodation by the matrix, which makes up 80% of the fuel volume in the case of 45-wt% uranium fuel.

Uranium and zirconium form eutectics with iron, nickel, and chromium, the principal constituents of the four alloys (304 or 304 L stainless steel, Incoloy 800, and Hastelloy-x) that are licensed for use for fuel rod cladding. The uranium eutectics have lower melting temperatures than those of zirconium, which is tied up as a hydride in any case. The melting points of the eutectics with uranium are: iron, 725°C; nickel, 740°C; and chromium, 859°C. As the uranium content of the fuel is increased, the potential for the formation of low-melting eutectics is enhanced. Localized fuel melting has been observed in 45-wt% uranium fuel in contact with Inconel 600 thermocouple sheating at temperatures above 1050°C. The extent of potential eutectic melting due to fuel/cladding interaction should be less in the 20- and 30-wt% uranium fuels than in 45-wt% uranium fuel, but more than in the original 8.5 wt% uranium fuel. In all cases, the extent of eutectic melting would be limited by the relatively small volume fraction of uranium in the fuels (11.2 v% or less for the fuels under review). The temperature at which eutectic fuel melting has been observed (1050°C) is 100°C above the lowest temperature at which cladding failure by hydrogen overpressure is predicted under conditions in which the cladding is at approximately the fuel temperature. Therefore, eutectic fuel/cladding melting does not constitute a more severe limit for fuel rod integrity than does hydrogen overpressure. It does, however, have the potential to produce fuel melting at temperatures about 80°C lower than the uranium melting point. This mechanism could lead to somewhat higher releases of fission products from the fuel rod in the temperature range 1050 to 1130°C under some accident conditions (such as loss of coolant) or during film boiling; however, these temperatures are above the safety limit of 930°C.

During sustained irradiation, hydrogen tends to migrate from the hot radial center of the fuel to a cooler annulus near the pellet periphery. Hydrogen/zirconium (H/Zr) ratios can vary by ±10 to 15% of their initial values. The increased H/Zr ratio near the outer radius of the fuel, coupled with high peak fuel temperatures that occur at the outer radius during a pulse, can cause excessive hydrogen pressures in the fuel matrix, which can weaken and deform the fuel matrix and cause excessive swelling and fuel element deformation. Experience suggests that pulse sizes or maximum fuel temperatures should be limited in higher burnup cores to account for the effects of hydrogen redistribution. This effect, however, is independent of uranium content in the TRIGA® fuel, and the evidence suggests that an equilibrium hydrogen distribution is established within a moderate time scale.

A 45-wt% uranium LEU fuel rod that was instrumented for measuring temperature and pressure was subjected to a series of 30 power pulses in a TRIGA® reactor to maximum temperatures in the range of 1050 to 1100°C. Only very modest (generally less than 2 psi) pressure pulses were measured in the rod as a result of the pulsing, in agreement with previous data on negligible hydrogen release during the pulsing of 8.5-wt% uranium fuel to temperatures up to 1150°C. All surveillance examinations on rod deformation were satisfactory. Tests have shown that the pulse response of uranium-zirconium hydride TRIGA® fuel is independent of the uranium content of the fuel and is dominated by the behavior of the zirconium hydride, along with the prompt temperature coefficient of reactivity.

As mentioned earlier, pulse sizes or maximum fuel temperatures should be limited in higher burnup cores to account for the effects of hydrogen redistribution. This potential problem is adequately addressed by imposing limits on maximum operating temperatures in standard TRIGA® fuels. The effects of hydrogen migration will not lead to a fission product release if these restrictions are applied.

4.6.2 Design Bases

The reactor design bases are established by the maximum operational capability for the fuel elements and configurations described in this report. The TRIGA® reactor system has three major areas that are used to define the reactor design basis:

a. fuel temperature;
b. prompt negative temperature coefficient;
c. reactor power.

The ultimate safety limit is based on fuel temperature, while the negative temperature coefficient contributes to the inherent safety of the TRIGA® reactor. A limit on reactor power is set to ensure operation below the fuel temperature safety limit. A summary of the conclusions of the analyses that supports these limits is presented below.

Fuel Temperature

The fuel temperature is a limit in both steady-state and pulse mode operation. This limit stems from the out-gassing of hydrogen from U-ZrH fuel and the subsequent stress produced in the fuel element cladding material. The strength of the cladding as a function of temperature sets the upper limit on the fuel temperature. A Fuel temperature limits of 930°C for U-ZrH with a H/Zr ratio less than 1.70 has been set to preclude the loss of clad integrity (Section 4.5.4.1.3). These temperature limits are less than the basic limits for TRIGA® fuel of 950°C as stated in Reference 4.1.

Prompt Negative Temperature Coefficient

The basic parameter that provides the TRIGA® system with a large safety factor in steady-state operation and under transient conditions is the prompt negative temperature coefficient. This coefficient is a function of the fuel composition and core geometry. The value for the negative temperature coefficient in 8.5/20 fuels is rather constant with temperature and is 0.01 %/°C ($1 \times 10^{-4}\Delta k/k/°C$). For 20/20 and 30/20 fuels, the value is a strong function of temperature and the average value over temperatures of interest is at least as large as the value for 8.5/20 fuel, as described in Section 4.5.1.2.

Reactor Power

Fuel and clad temperature define the safety limit. A power level limit is calculated that ensures that the fuel and clad temperature limits will not be exceeded. The design basis analysis indicates that operation at up to 1000 kW with an ~100 element core (45°C inlet water temperature) natural convective flow will not allow film boiling, and therefore, high fuel and clad temperatures which could cause loss of clad integrity could not occur. One MW reactor operation does not affect the fundamental aspects of the TRIGA® fuel, including the reactivity feedback coefficients, temperature safety limits, and fission product release rates (Section 4.5.4). The thermal-hydraulic performance is discussed in Section 4.7.

4.6.3 Design Criteria – Operating Core Configuration (OCC), Limiting Core Configuration (LCC), Planned Future Operating Core Configuration, and End of Life Planned Future Operating Core Configuration.

Three separate core loadings are defined for use in the safety calculations to arrive at the most limiting thermal hydraulic conditions to be analyzed for later in this chapter. The Operating Core Configuration is the current MNRC reactor core configuration. The OCC and all subsequent cores are assembled with both the 20/20 and 30/20 type TRIGA fuel elements, together with five control rods having 20/20 type fuel as followers (FFCRs), one transient rod, and in some cases dummy graphite elements. The transient rod was used for pulsing runs in the past, but now serves as one additional control rod.

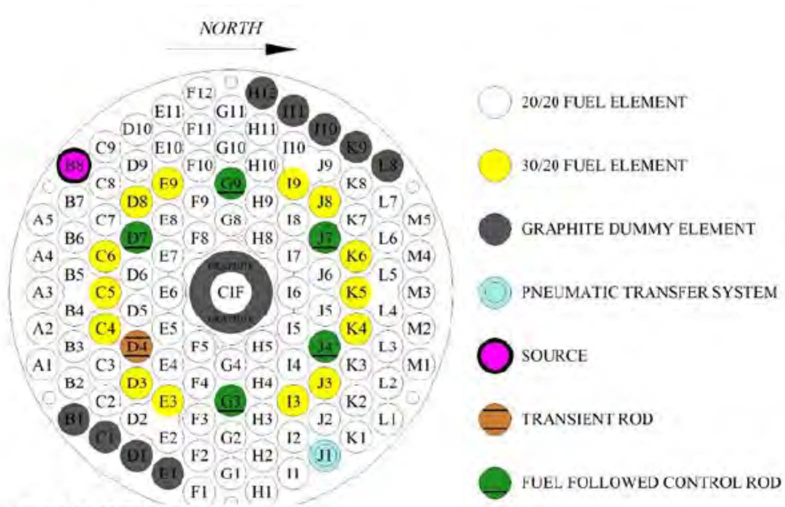

Figure 4.15 MNRC OCC

The limiting core configuration provides the highest peaking and highest element heat output to be used in the thermal hydraulic analysis. When the MNRC was licensed for 2.0 MW steady-state operations 30/20 elements were not permitted in C ring. When looking at 1.0 MW operation the placement of 30/20 elements in C ring does not produces excessive peaking or element heat output. The advantage of this configuration is it is expected to better utilize the remaining fuel and increase excess reactivity by just under $1.00.

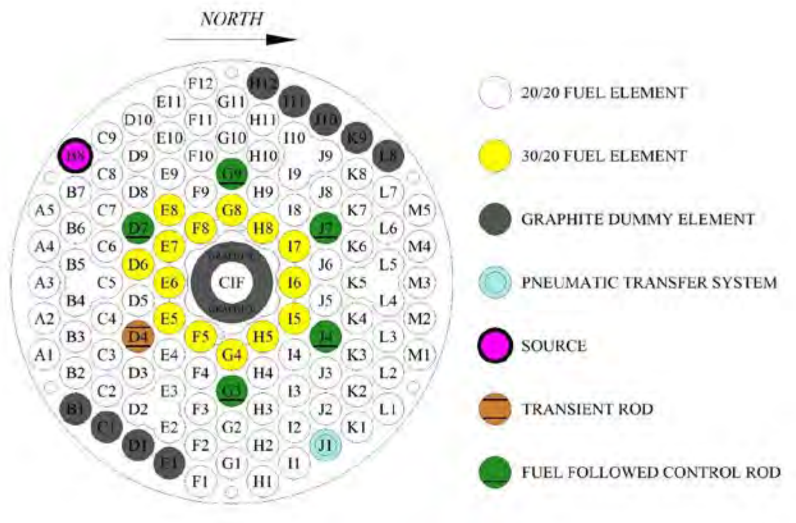

Figure 4.16 MNRC LCC

It is assumed the 30/20 element is fresh with no U-235 burnup in order to maximize peaking in that element.

Core Loading

- No fuel will be loaded into Hex Ring A or B.
- The only fuel types allowed are 20/20 and 30/20.
- 20/20 fuel may be used in any position in Rings C through G.
- 30/20 fuel may be used in any positions in Rings C through G.
- No core configuration is permitted where an element output is expected to exceed 17.69 kW at 1.0 MW based on the MNRC MCNP model.
- Experiment placed in the CIF must be modeled so ensure the per element heat output given above will not be exceeded.

4.6.4 Reactor Core Parameters

4.6.4.1 Reactor Fuel Temperature

The basic safety limit for the TRIGA® reactor system is the fuel temperature. This applies for both the steady-state and pulsed mode of operation.

Limiting temperatures for the two modes of operation are of interest, depending on the type of TRIGA® fuel used. The UCD/MNRC reactor utilizes fuel with H/Zr ratios between 1.6 and 1.7. (i.e., greater than 1.5). This allows operation at a higher fuel temperature limit. Figure 4.17 indicates that the higher hydride compositions are single phase and are not subject to the large volume changes associated with the phase transformations at approximately 530°C in the lower hydrides. It has been noted in Reference 4.6 that the higher hydrides lack any significant thermal diffusion of hydrogen. These two facts preclude concomitant volume changes. The important properties of delta phase U-ZrH are given in Table 4-5.

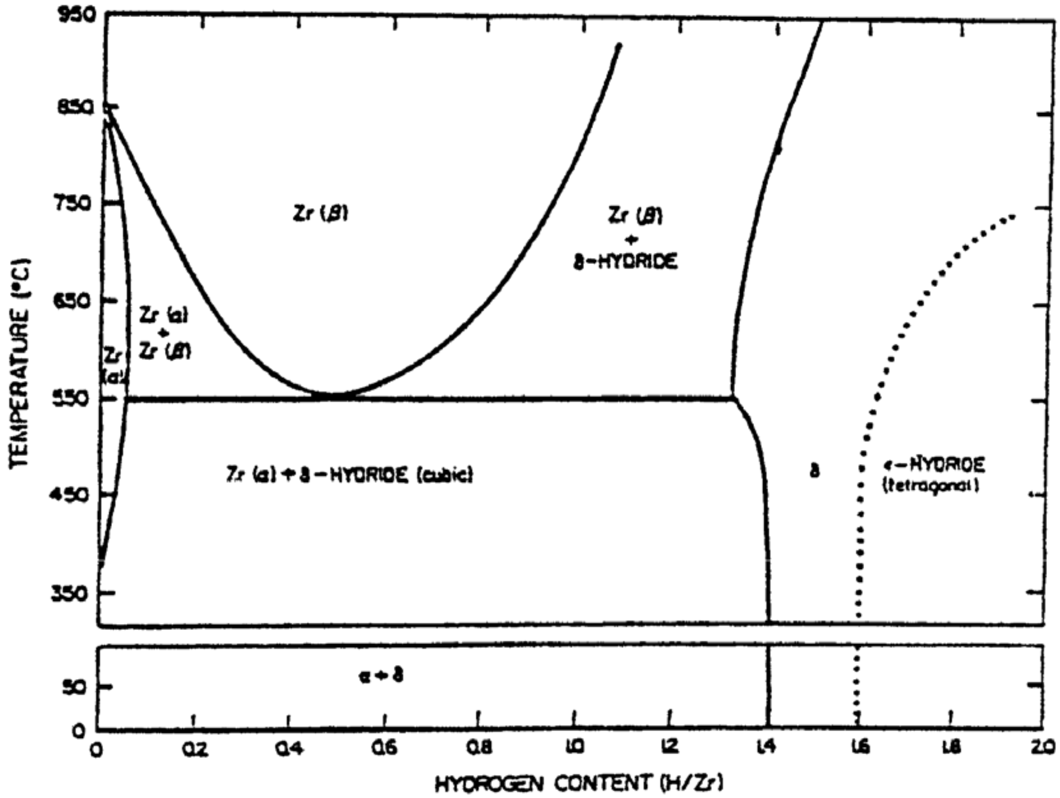

FIGURE 4.17 PHASE DIAGRAM OF THE ZIRCONIUM-HYDROGEN SYSTEM

TABLE 4.5 PHYSICAL PROPERTIES OF DELTA PHASE U-ZrH

Thermal conductivity (93°C-650°C)	13 Btu/hr – ft^2-°F
Elastic modulus: 20°C	9.1 x 10^6 psi
650°C	6.0 x 10^6 psi
Ultimate tensile strength (to 650°C)	24,000 psi
Compressive strength (20°C)	60,000 psi
Compressive yield (20°C)	35,000 psi
Heat of formation ($\delta^{\circ}_{H_f}$ 298°)	37.72 kcal/g-mole

Among the chemical properties of U-ZrH and ZrH, the reaction rate of the hydride with water is of particular interest. Since the hydriding reaction is exothermic, water will react more readily with zirconium than with zirconium hydride systems. Zirconium is frequently used in contact with water in reactors, and the zirconium-water reaction is not a safety hazard. Experiments carried out at GA Technologies show that the zirconium hydride systems have a relatively low chemical reactivity with respect to water and air. These tests (Reference 4.7), have involved the quenching with water of both powders and solid specimens of U-ZrH after heating to as high as 850°C, and of solid U-Zr alloy after heating to as high as 1200°C. Tests have also been made to determine the extent to which fission products are removed from the surfaces of the fuel elements at room temperature. Results prove that, because of the high resistance to leaching, a large fraction of the fission products are retained in even completely unclad U-ZrH fuel.

At room temperature, the hydride is like a ceramic and shows little ductility. However, at the elevated temperatures of interest for pulsing, the material is found to be more ductile. The effect of very large thermal stress on hydride fuel bodies has been observed in hot cell observations to cause relatively widely spaced cracks which tend to be either radial or normal to the central axis and do not interfere with radial heat flow (Reference 4.8). Since the segments tend to be orthogonal, their relative positions appear to be quite stable.

The limiting effect of fuel temperature is the hydrogen gas pressure causing cladding stress. Figure 4.18 relates equilibrium hydrogen pressure in a Zr/H mixture as a function of temperature for three different H/Zr ratios.

The main concern regarding hydrogen pressure is to ensure that the cladding ultimate strength is not exceeded by the stress caused by the pressure. The mechanisms in obtaining temperatures and pressures of concern are different in the pulsing and steady-state mode of operation, and each mechanism will be discussed separately.

The UCD/MNRC fuel consists of U-ZrH with a H/Zr ratio between 1.6 and 1.7 and with 20 or 30 wt% enriched in ^{235}U to approximately 20% ^{235}U. The cladding is 0.020 in. thick stainless steel and has an inside diameter of 1.43 in. The rest of the discussion on fuel temperatures will be concerned with fuel having H/Zr ratios greater than 1.5 (i.e., single phase and not subject to the large volume changes associated with phase transformation at approximately 530˚C in the lower hydrides). Further, it will specifically address fuel with an H/Zr ratio of 1.7 since this is the highest ratio fuel to be used in the UCD/MNRC and will produce the highest clad pressure and stress for a given temperature. Figure 4.19 shows the characteristic of 304 stainless steel with regard to yield and ultimate strengths as a function of temperature.

The stress applied to the cladding from the internal hydrogen gas pressure is given by:

$$S = P\, r/t \quad (1)$$

where:

S = stress in psi;
P = internal pressure in psi;
r = radius of the stainless steel cylinder;
t = wall thickness of the stainless steel clad.

Using the parameters given above:

$$S = 36.7\, P \quad (2).$$

For safety considerations, it is necessary to relate the strength of the cladding material at its operating temperature to the stress applied to the cladding due to the internal gas pressure associated with the fuel temperature. Figure 4.20 gives the ultimate cladding strength and the stress applied to the cladding as a result of hydrogen dissociation for fuel having H/Zr ratios of 1.65 and l.70, both as a function of temperature. This curve shows that the cladding will not fail for fuel with $Zr/H_{1.7}$ if both the clad and fuel temperatures are equal and below about 930˚C. This is conservative since the cladding temperature will be below the fuel temperature. This establishes the safety limit on fuel temperature for steady-state operations. The actual steady-state peak fuel temperature at even 2.0 MW is substantially below the limiting maximum measured fuel temperature of 750°C. The remainder of this section deals with the safety limit for transient operation.

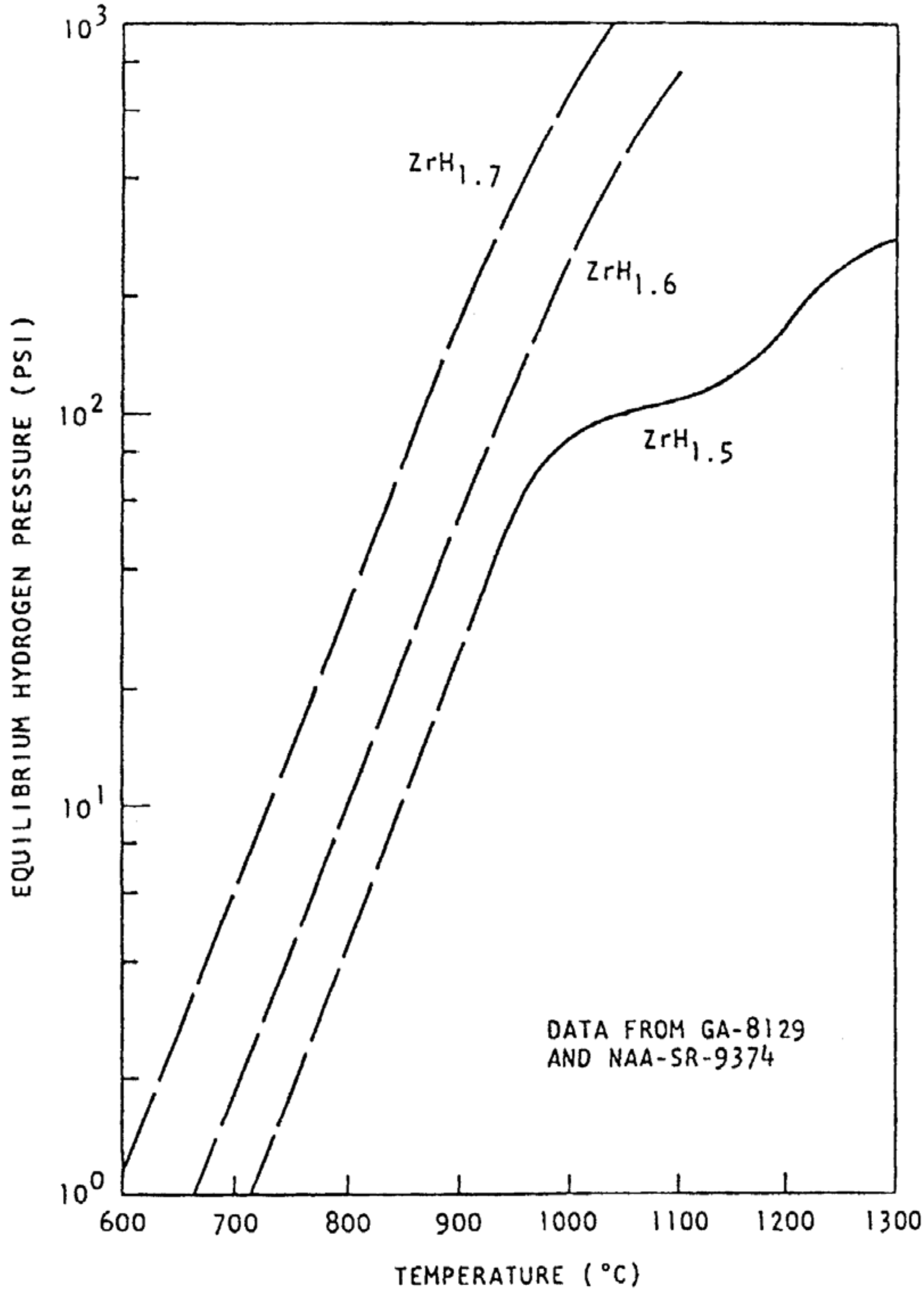

FIGURE 4.1 EQUILIBRIUM HYDROGEN PRESSURES OVER ZrHx VERSUS TEMPERATURE

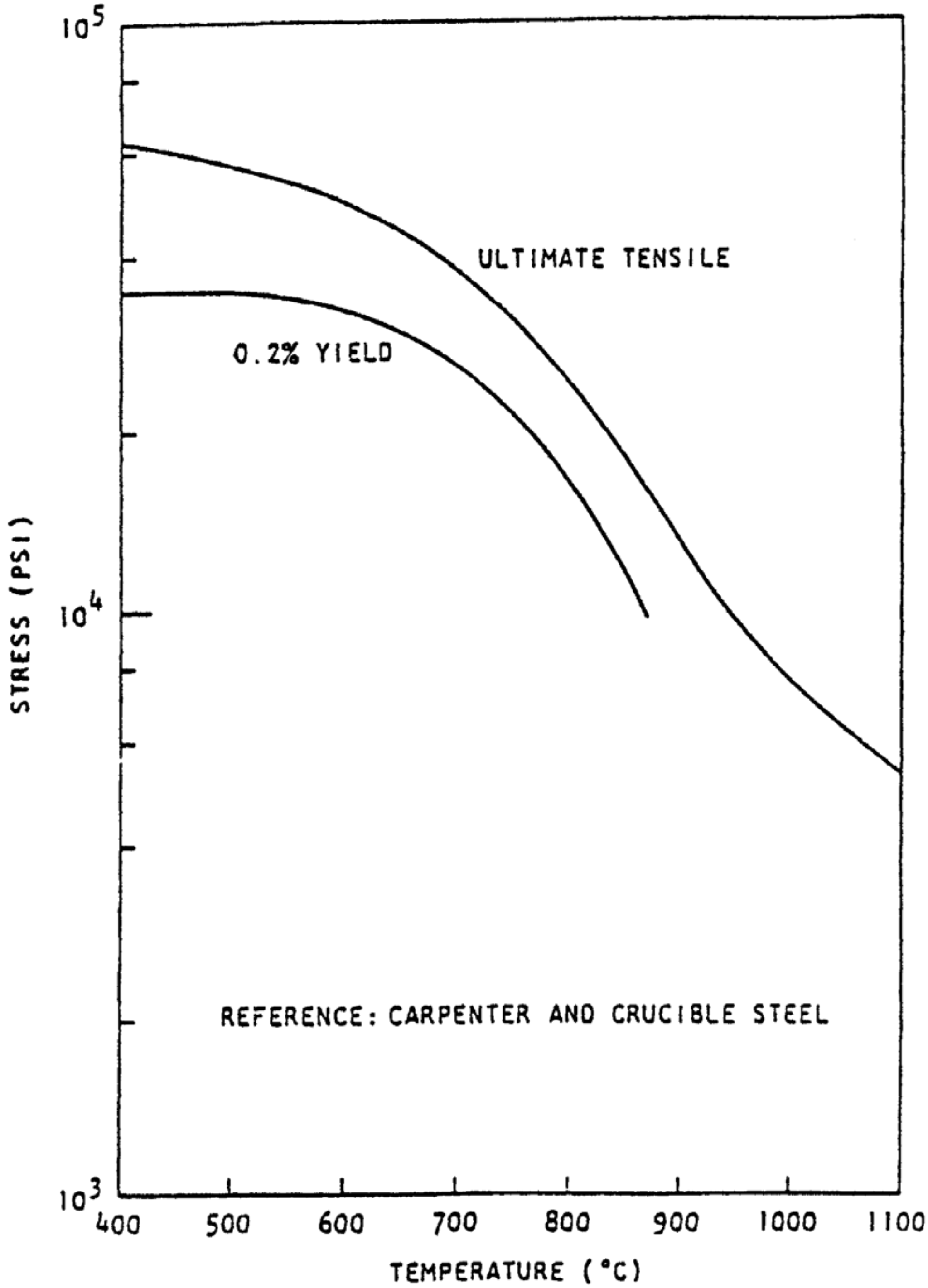

FIGURE 4.19 STRENGTH OF TYPE 304 STAINLESS STEEL AS A FUNCTION OF TEMPERATURE

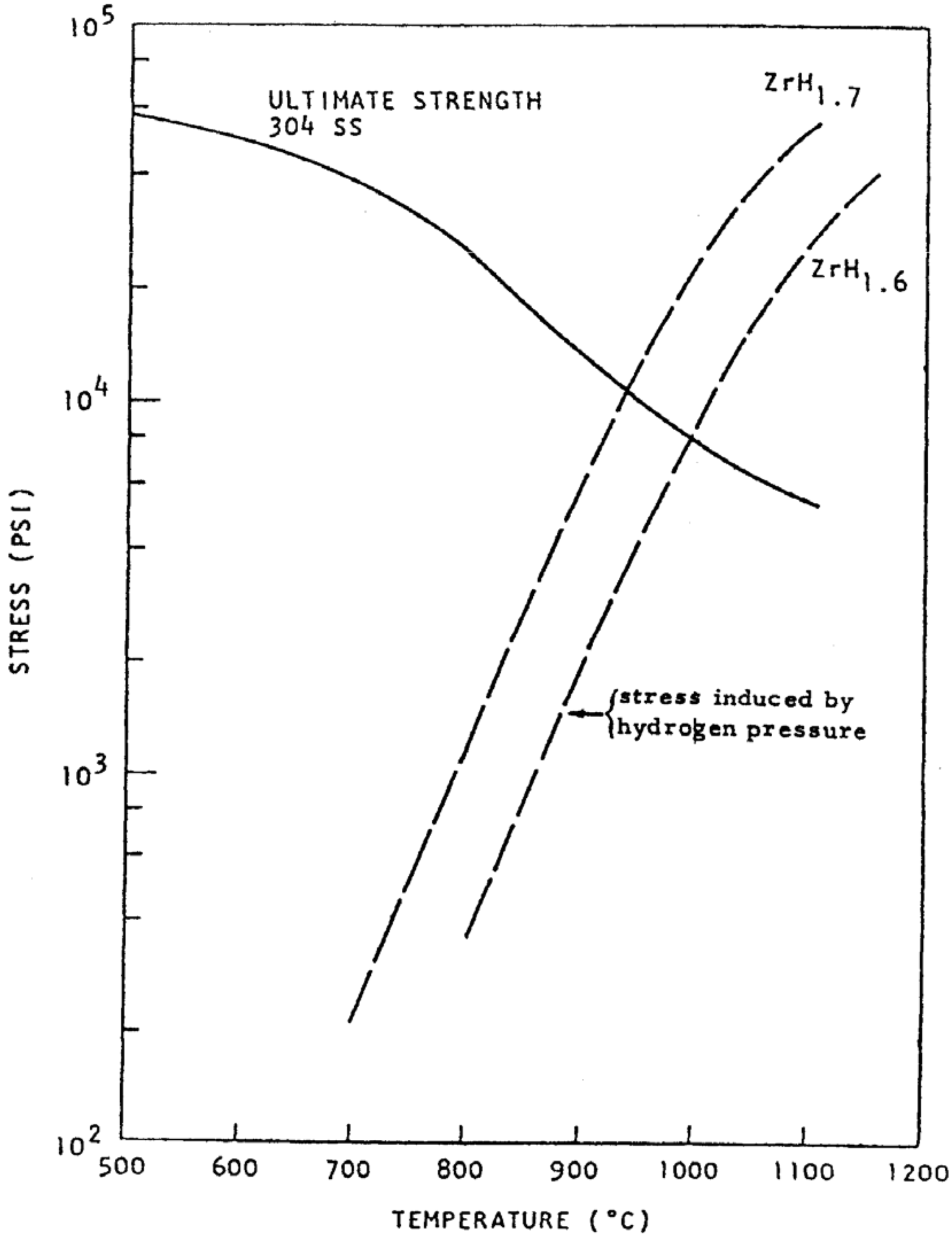

FIGURE 4.20: STRENGTH AND APPLIED STRESS AS A FUNCTION OF TEMPERATURE

In transient operation, it is necessary to account for the difference in fuel and cladding temperatures to establish a safety limit based on fuel temperature. Additionally, the diffusion of hydrogen reduces peak pressures from those predicted at equilibrium at the peak fuel temperatures. The net result of these two points is that a higher safety limit exists for transient operation. An analysis of the two points is given in the following two subsections.

4.6.4.1.1 Fuel and Clad Temperature During Pulsing

The following analysis is provided only to inform the consequences (or lack thereof) for an inadvertent insertion of reactivity in chapter 13. The MNRC will no long pulse the reactor by injection of the transient rod. For the steady-state safety limit, it was assumed that the cladding and fuel temperatures were the same. The following discussion shows that the cladding temperature is well below the maximum fuel temperature after a pulse. This allows a higher safety limit on fuel temperature. The radial temperature distribution in the fuel element immediately following a pulse is very similar to the power distribution shown in Figure 4.21. This initial steep thermal gradient at the fuel surface results in some heat transfer during the time of the pulse so that the true peak temperature does not quite reach the adiabatic peak temperature. A large temperature gradient is also impressed upon the clad which can result in a high heat flux from the clad into the water. If the heat flux is sufficiently high, film boiling may occur and form an insulating jacket of steam around the fuel elements permitting the clad temperature to approach the fuel temperature. Thermal transient calculations were made using the RAT computer code. RAT is a 2D transient heat transport code developed to account for fluid flow and temperature dependent material properties. Calculations show that if film boiling occurs after a pulse, it may take place either at the time of maximum heat flux from the clad, before the bulk temperature of the coolant has changed appreciably, or it may take place at a later time when the bulk temperature of the coolant has approached the saturation temperature, resulting in a reduced threshold for film boiling. Data obtained by Johnson et al., Reference 4.9, for transient heating of ribbons in 100°F water, showed burnout fluxes of 0.9 to 2.0 MBtu/ft^2-hr for e-folding periods from 5 to 90 milliseconds. On the other hand, sufficient bulk heating of the coolant channel between fuel elements can take place in several tenths of a second to lower the departure from nucleate boiling (DNB) point to approximately 0.4 MBtu/ft^2-hr. It is shown, on the basis of the following analysis, that the second mode is the most likely, i.e., when film boiling occurs, it takes place under essentially steady-state conditions at local water temperatures near saturation.

A value for the temperature that may be reached by the clad if film boiling occurs was obtained in the following manner. A transient thermal calculation was performed using the radial and axial power distributions in Figures 4.21 and 4.22, respectively. The thermal resistance at the fuel-clad interface was assumed to be zero.

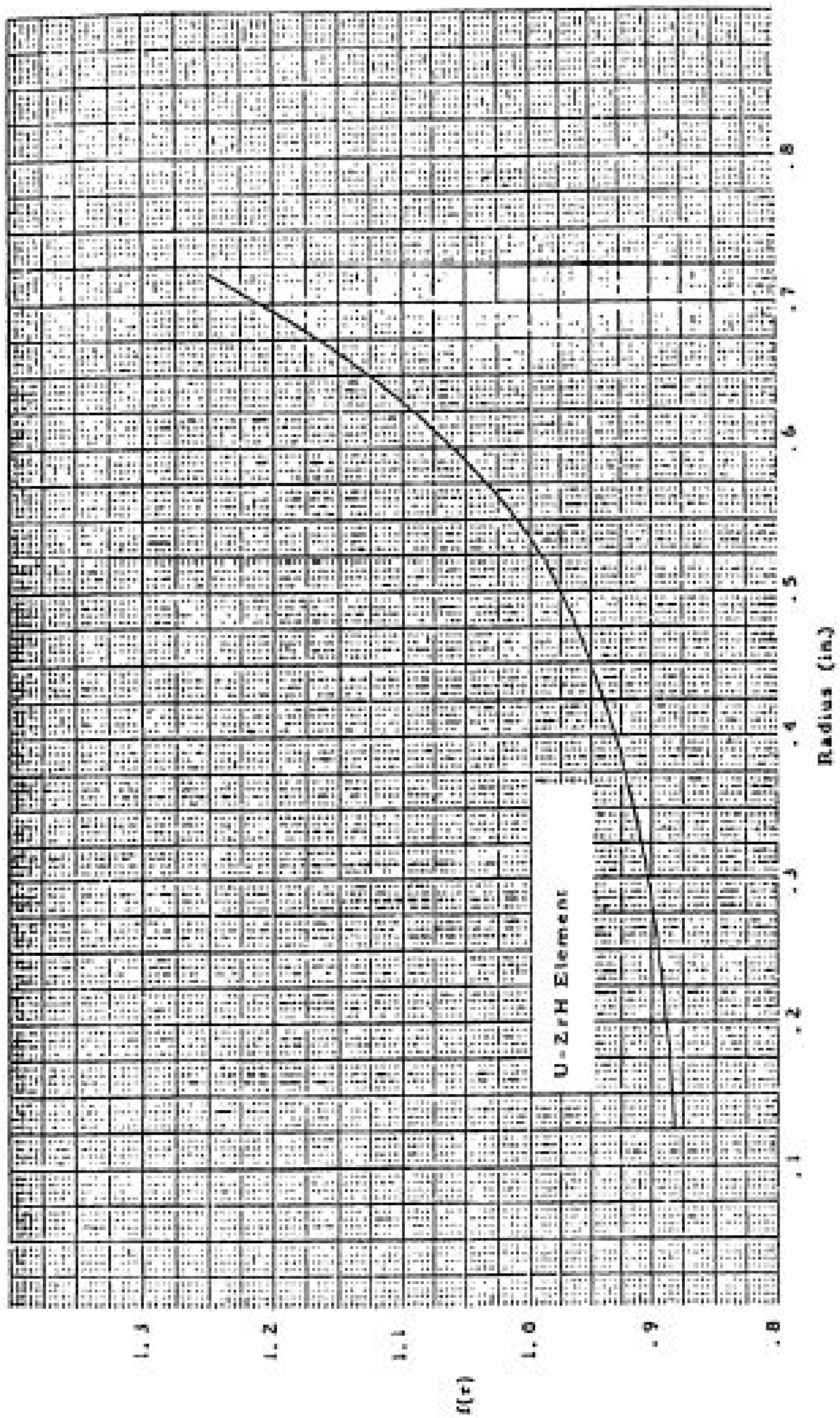

FIGURE 4.21 RADIAL POWER DISTRIBUTION IN U-ZrH FUEL ELEMENT

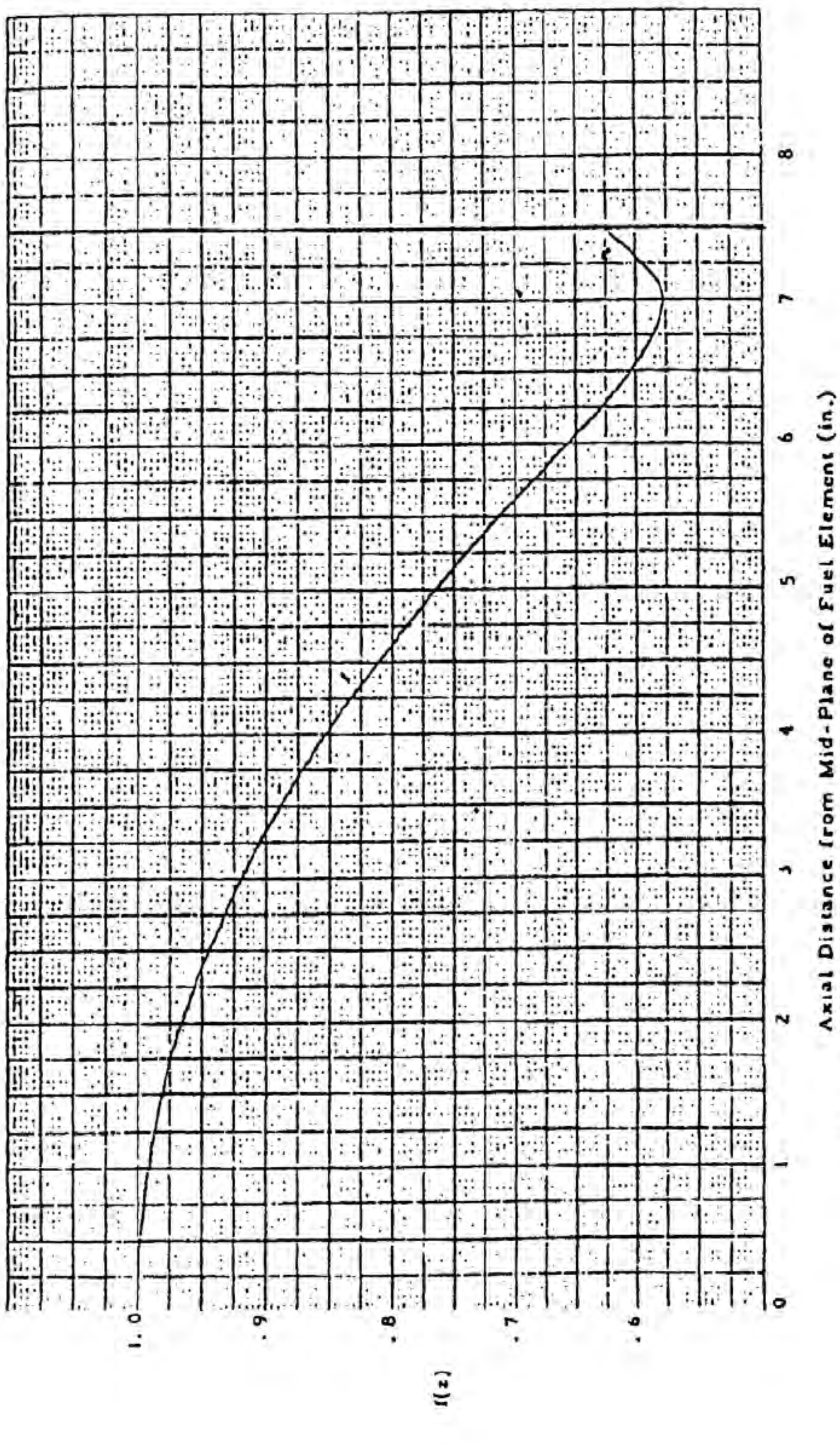

FIGURE 4.22 AXIAL POWER DISTRIBUTION IN A FUEL ELEMENT ASSUMED FOR THEHERMAL ANALYSIS

A boiling heat transfer model, as shown in Figure 4.23, was used in order to obtain an upper limit for the clad temperature rise. The model used the data of McAdams (Reference 4.10), for the subcooled boiling and the work of Sparrow and Cess (Reference 4.11), for the film boiling regime. A conservative estimate was obtained for the minimum heat flux in film boiling by using the correlations of Speigler et al. (Reference 4.12), Zuber (Reference 4.13), and Rohsenow and Choi (Reference 4.14), to find the minimum temperature point at which film boiling could occur. This calculation gave an upper limit of 760°C clad temperature for a peak initial fuel temperature of 1000°C, as shown in Figure 4.24. Fuel temperature distributions for this case are shown in Figure 4.25 and the heat flux into the water from the clad is shown in Figure 4.26. In this limiting case, DNB occurred only 13 milliseconds after the pulse, conservatively calculated assuming a steady-state DNB correlation. Subsequently, experimental transition and film boiling data were found to have been reported by Ellion (Reference 4.15), for water conditions similar to those for the TRIGA® system. The Ellion data show the minimum heat flux, used in the limiting calculation described above, was conservative by a factor of 5. An appropriate correction was made which resulted in a more realistic estimate of 470°C as the maximum clad temperature expected if film boiling occurs. This result is in agreement with experimental evidence obtained for clad temperatures of 400°C to 500°C for TRIGA® Mark F fuel elements which have been operated under film boiling conditions (Reference 4.16). Based on this analysis, the peak cladding temperature will be 470°C for a transient fuel temperature of 1000°C. Further analysis shows that this peak clad temperature is valid for a higher peak fuel temperature.

The preceding analysis assessing the maximum clad temperatures associated with film boiling assumed no thermal resistance at the fuel-clad interface. Measurements of fuel temperatures as a function of steady-state power level provide evidence that after operating at high fuel temperatures, a permanent gap is produced between the fuel body and the clad. This gap exists at all temperatures below the maximum operating temperature (for example, Figure 16 in Reference 4.16). The gap thickness varies with fuel temperature and clad temperature: cooling of the fuel or overheating of the clad tends to widen the gap and decrease the heat transfer rate. Additional thermal resistance due to oxide and other films on the fuel and clad surfaces is expected. Experimental and theoretical studies of thermal contact resistance have been reported, References 4.17-4.19, which provide insight into the mechanisms involved. They do not, however, permit quantitative prediction because the basic data required for input are presently not fully known. Instead, several transient thermal computations were made using the RAT code, varying the effective gap conductance, in order to determine the effective gap coefficient for which departure from nucleate boiling is incipient. These results were then compared with the incipient film boiling conditions of the 1000°C peak fuel temperature case.

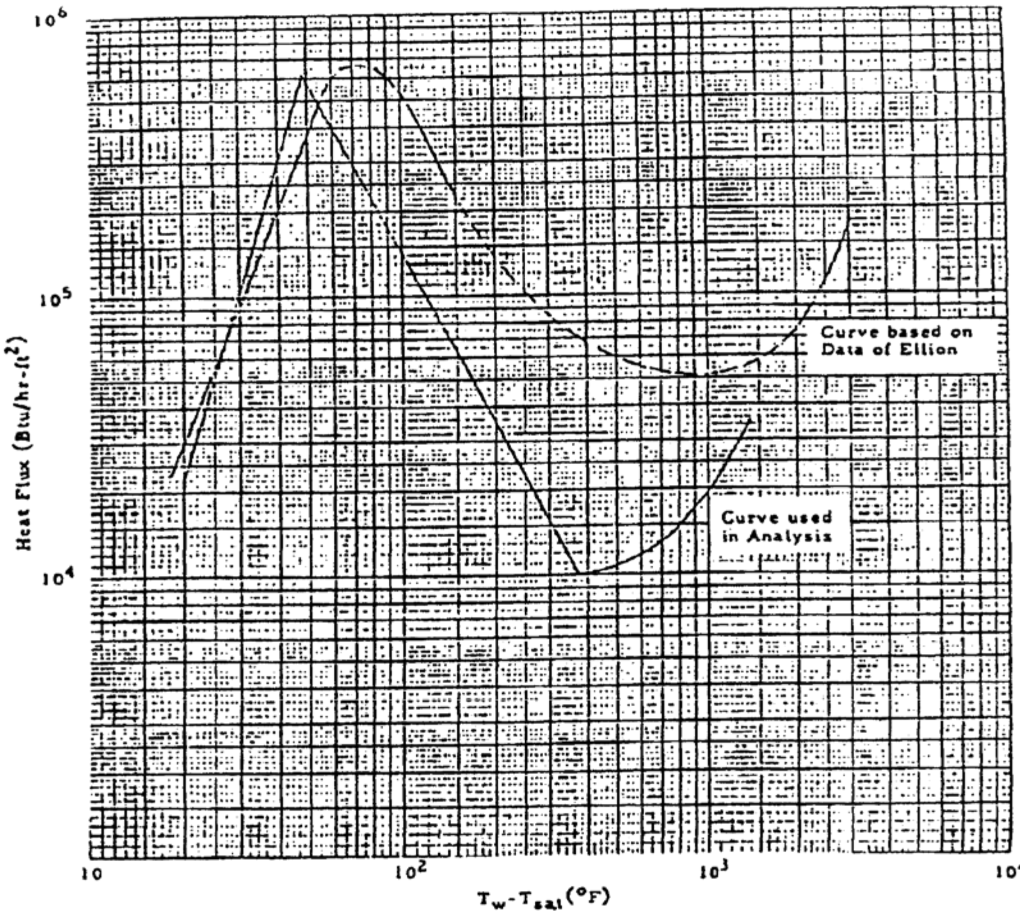

FIGURE 4.23 SUBCOOLED BOILING HEAT TRANSFER FOR WATER

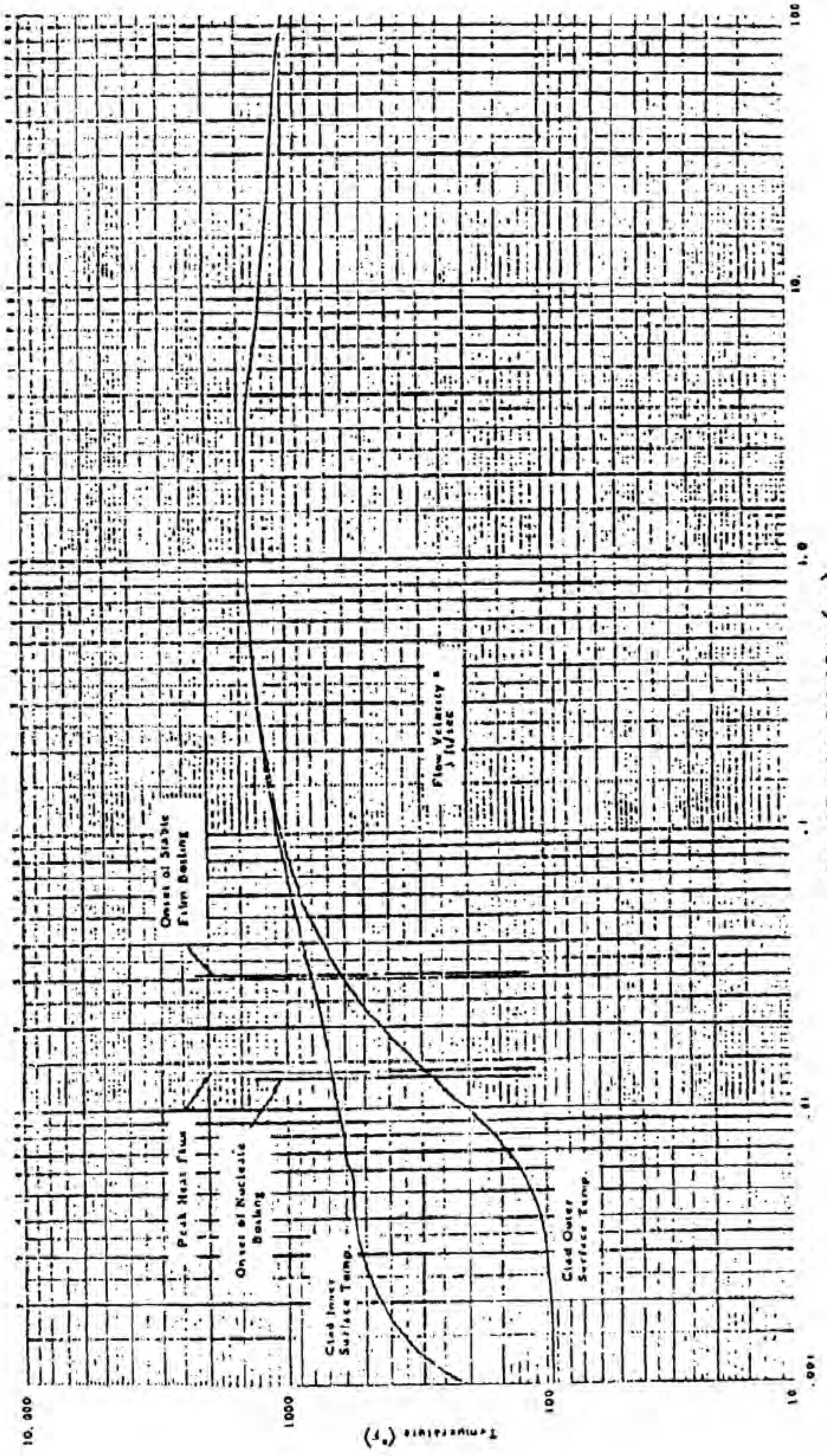

FIGURE 4.24 CLAD TEMPERATURE AT MIDPOINT OF WELL BONDED FUEL ELEMENT

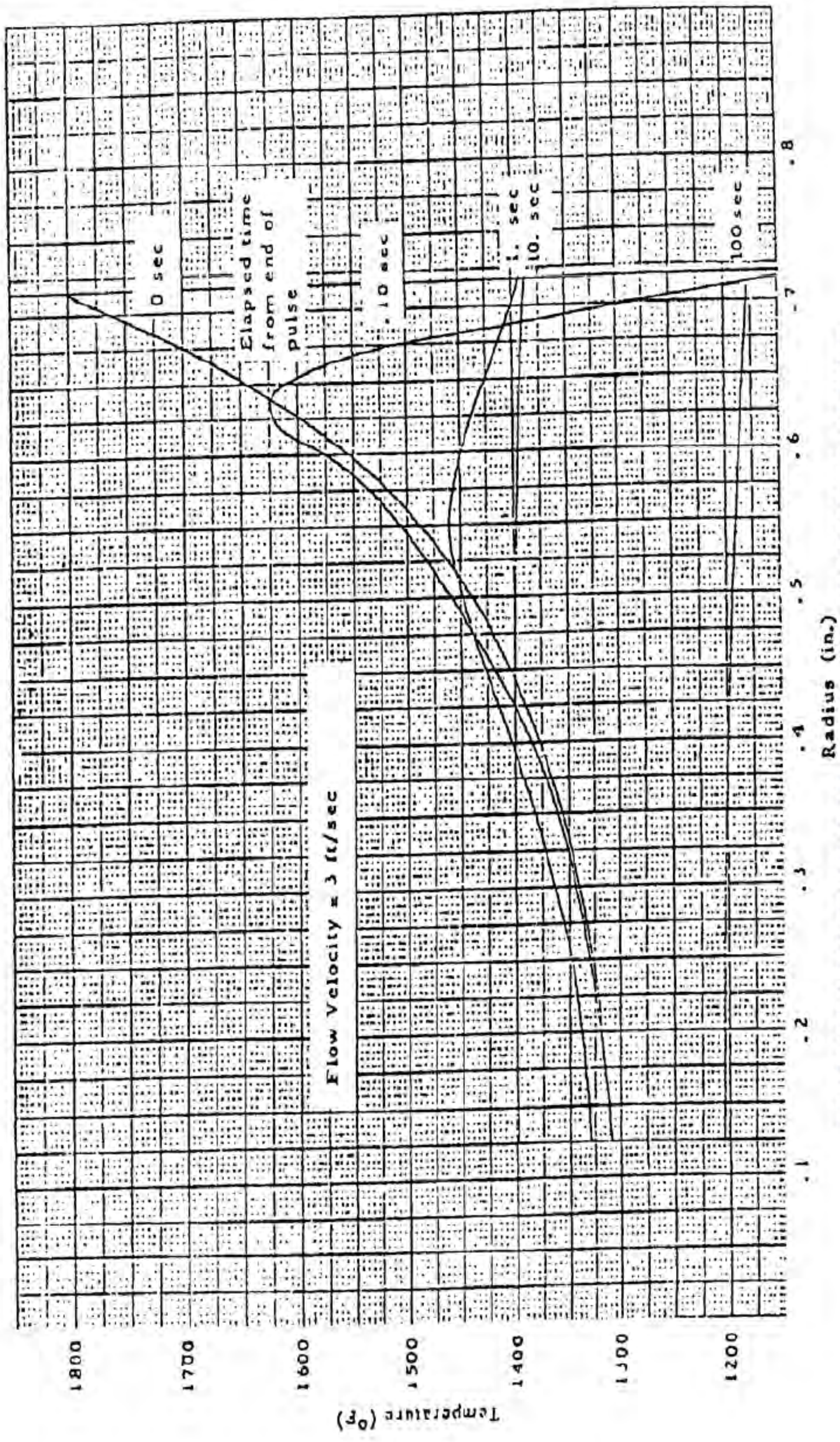

FIGURE 4.25 FUEL BODY TEMPERATURES AT MIDPLANE OF WELL-BONDED U-Zr-H ELEMENT AFTER PULSE

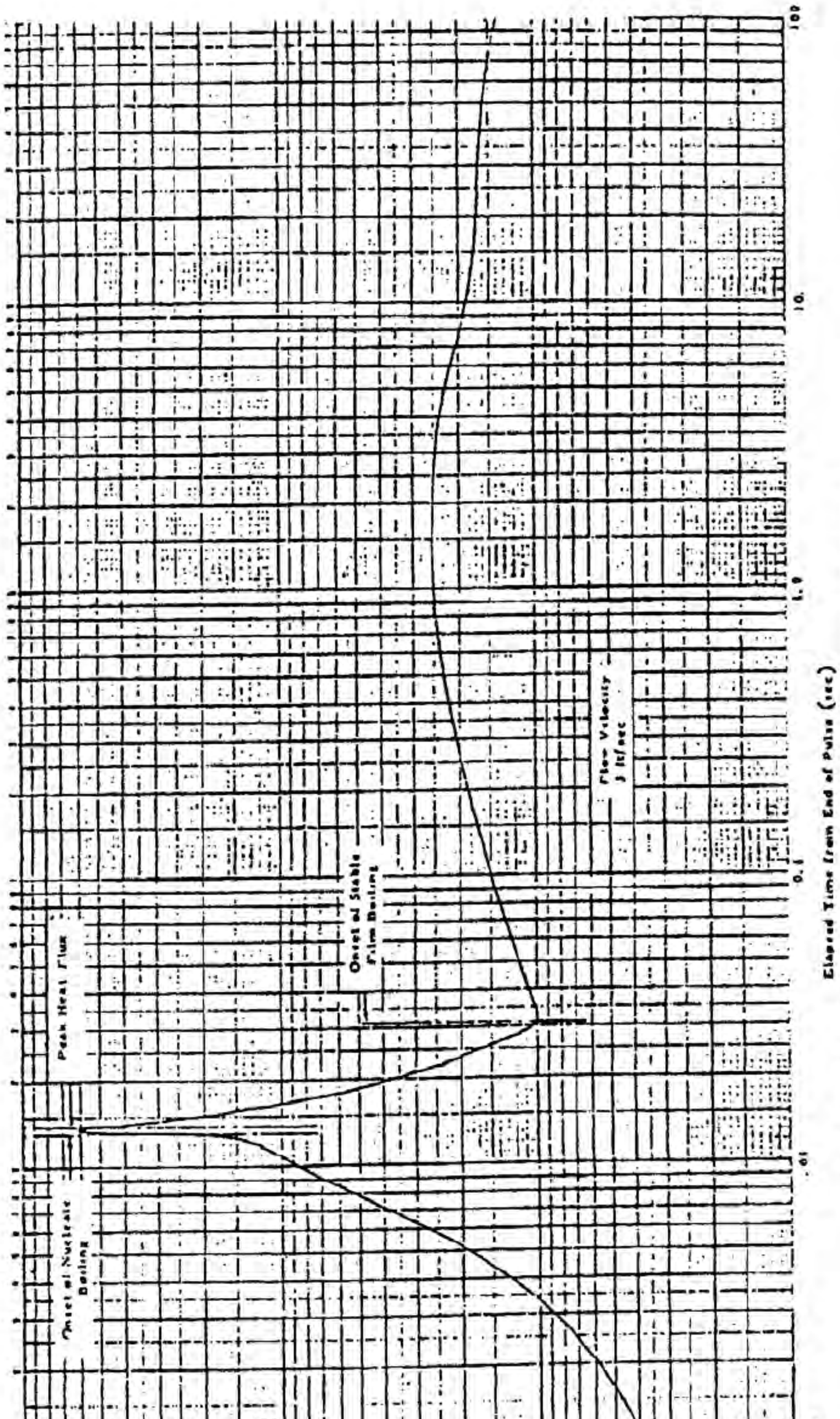

FIGURE 4.26 SURFACE HEAT FLUX AT MIDPLANE OF WELL BONDED U-Zr-H ELEMENT AFTER PULSE

For convenience, the calculations were made using the same initial temperature distribution as was used for the preceding calculations. The calculations assumed a coolant flow velocity of 1 ft per second, which is within the range of flow velocities computed for natural convection under various steady-state conditions for these reactors. The calculations did not use a complete boiling curve heat transfer model, but instead, included a convection cooled region (no boiling) and a subcooled nucleate boiling region without employing an upper DNB limit. The results were analyzed by inspection using the extended steady-state correlation of Bernath (Reference 4.20), which has been reported by Spano (Reference 4.21), to give agreement with SPERT II burnout results within the experimental uncertainties in flow rate.

The transient thermal calculations were performed using effective gap conductances of 500, 375, and 250 Btu/hr-ft^2-°F. The resulting wall temperature distributions were inspected to determine the axial wall position and time after the pulse which gave the closest approach between the local computed surface heat flux and the DNB heat flux according to Bernath. The axial distribution of the computed and critical heat fluxes for each of the three cases at the time of closest approach is shown in Figures 4.27 through 4.29. If the minimum approach to DNB is corrected to TRIGA® Mark F conditions and cross-plotted, an estimate of the effective gap conductance of 450 Btu/hr-ft^2-°F is obtained for incipient burnout so that the case using 500 is thought to be representative of standard TRIGA® fuel.

The surface heat flux at the midplane of the element is shown in Figure 4.30 with gap conductance as a parameter. It may be observed that the maximum heat flux is approximately proportional to the heat transfer coefficient of the gap, and the time lag after the pulse for which the peak occurs is also increased by about the same factor. The closest approach to DNB in these calculations did not necessarily occur at these times and places, however, as indicated on the curves of Figures 4.27 through 4.29. The initial DNB point occurred near the core outlet for a local heat flux of about 340 kBtu/hr-ft^2-°F according to the more conservative Bernath correlations at a local water temperature approaching saturation.

From this analysis, a maximum temperature for the clad during a pulse which gives a peak adiabatic fuel temperature of 1000°C is estimated to be 470°C. This is conservative since it was obtained by assuming no thermal resistance between the fuel and the clad. As was shown above, a value of 500 Btu/hr-ft^2°F for the gap conduction is more realistic.

As can be seen from Figure 4.19, the ultimate strength of the cladding at a temperature of 470°C is 59,000 psi. If the stress produced by the hydrogen over pressure on the clad is less than 59,000 psi, the cladding will not be ruptured. Referring to Figure 4.20, and considering U-$ZrH_{1.7}$ fuel with a peak temperature of 1000°C, one finds the stress on the clad to be 24,000 psi. Analysis in the next section which considers diffusion will show that the actual hydrogen pressure produced in a pulse is less than the equilibrium pressure for the peak temperature. This allows a safe limit on fuel temperature to be 1100°C. TRIGA® fuel with a hydrogen to zirconium ratio of at least 1.6 has been pulsed to temperatures approaching 1150°C without damage to the clad (Reference 4.22).

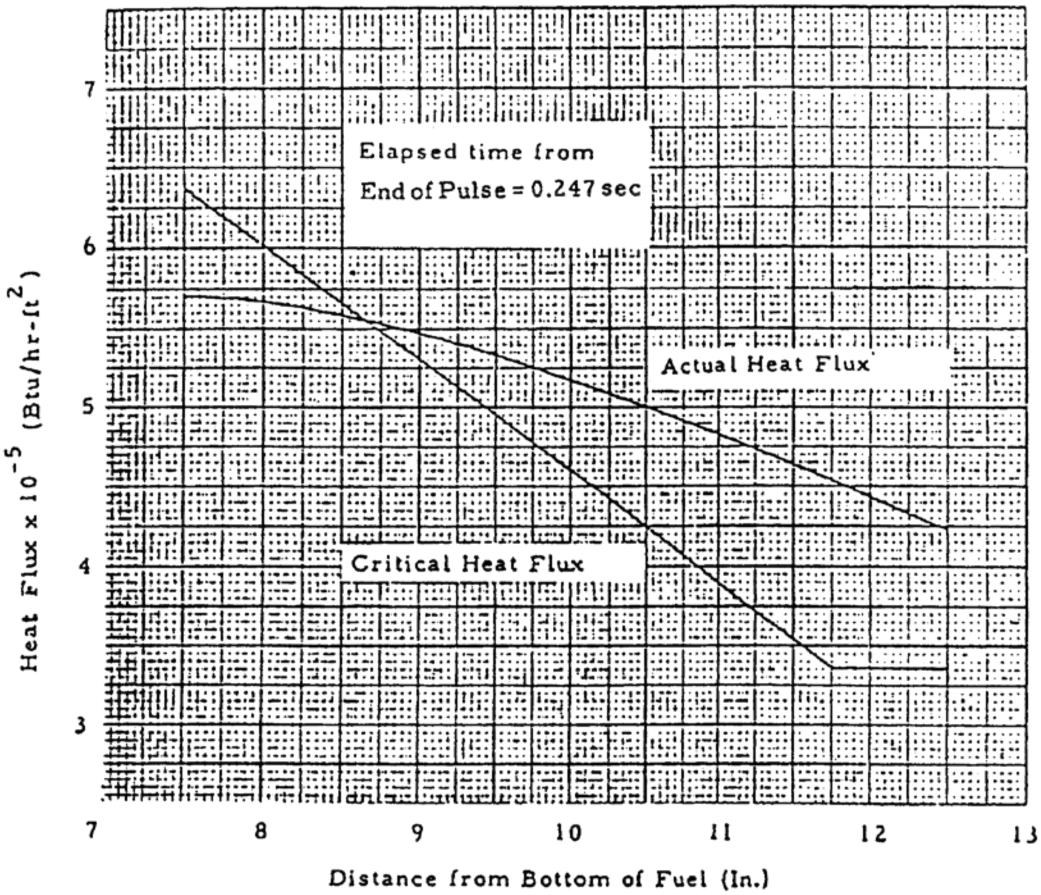

FIGURE 4.2 SURFACE HEAT FLUX DISTRIBUTION FOR STANDARD NON-GAPPED FUEL ELEMENT AFTER PULSE hgap = 500

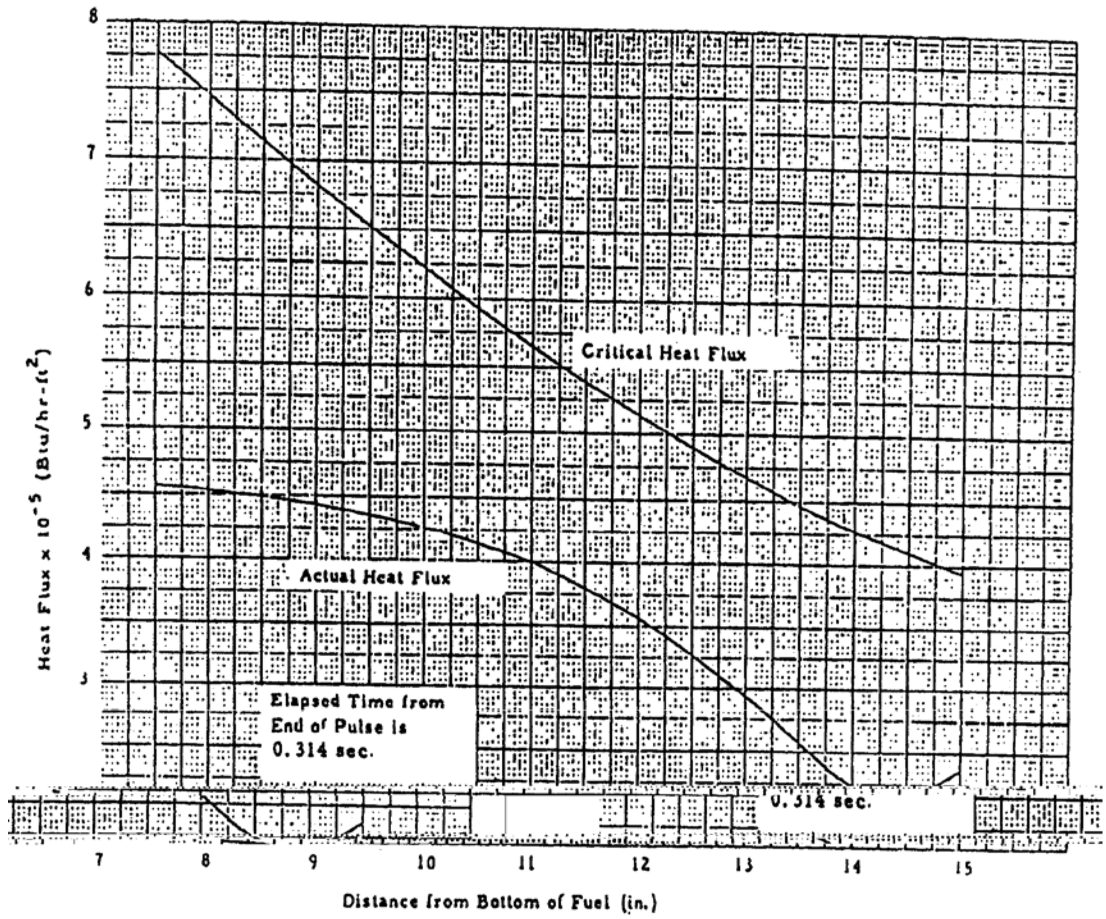

FIGURE 4.28 SURFACE HEAT FLUX DISTRIBUTION FOR STANDARD NON-GAPPED FUEL ELEMENT AFTER PULSE, hgap = 375

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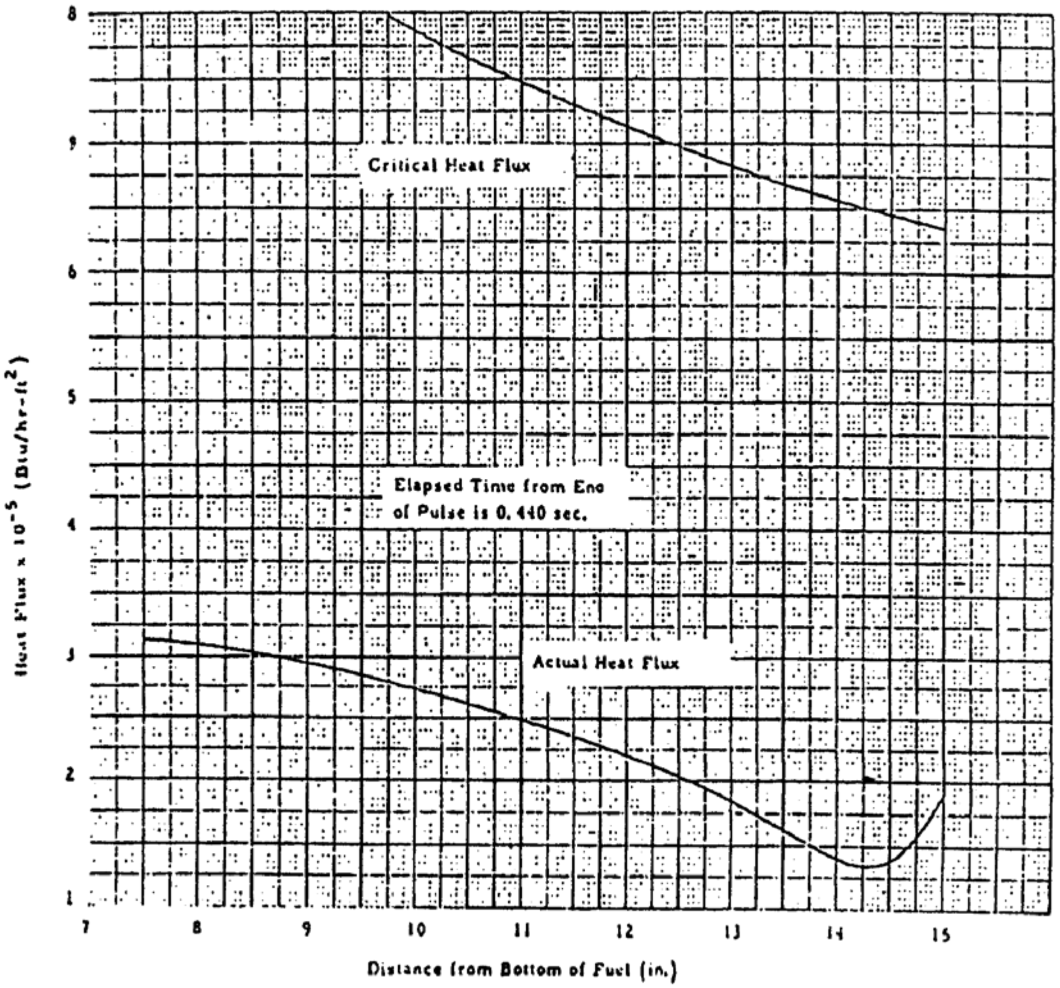

FIGURE 4.3 SURFACE HEAT FLUX DISTRIBUTION FOR STANDARD NON-GAPPED FUEL ELEMENT AFTER PULSE, hgap = 250

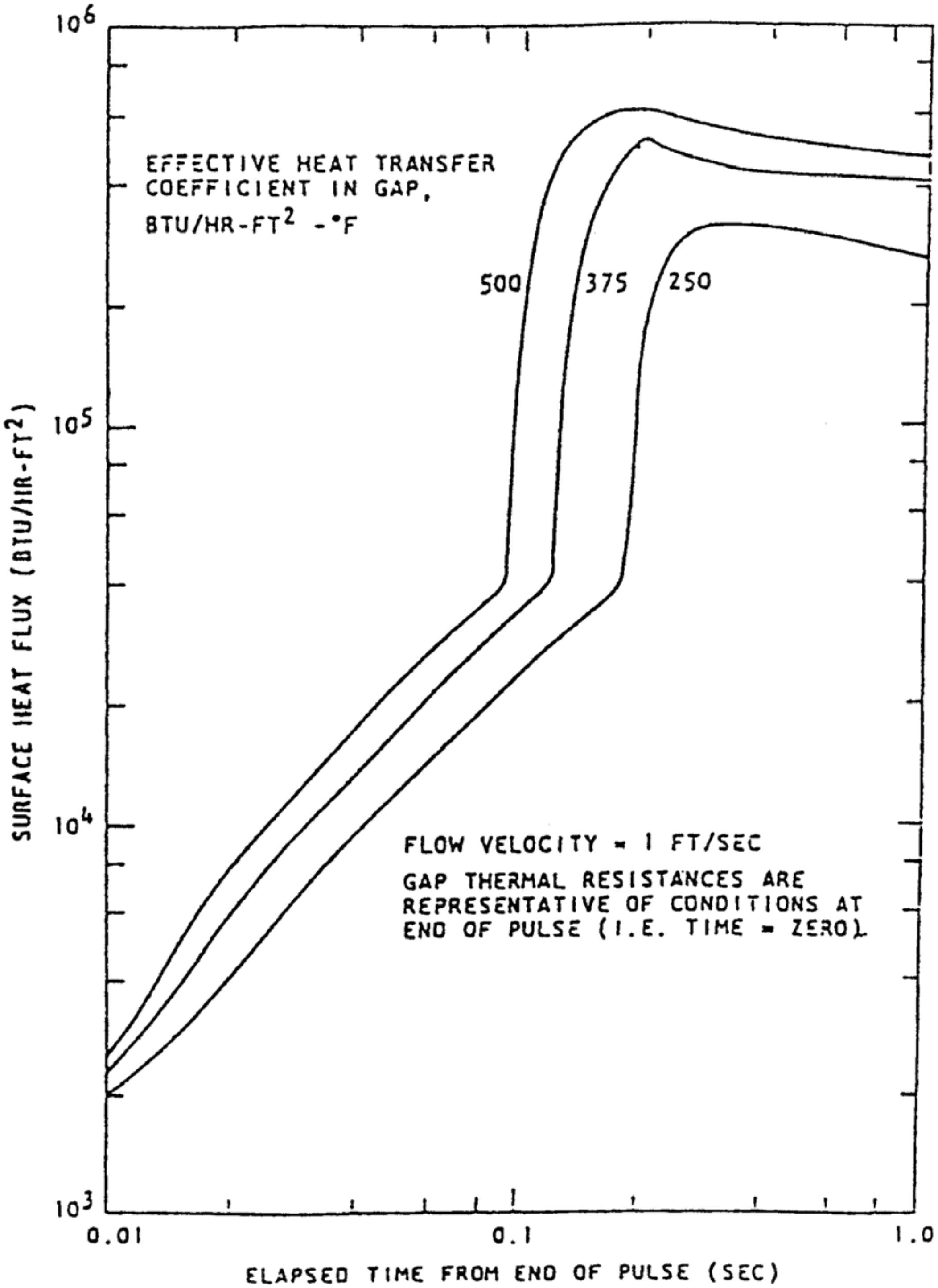

FIGURE 4.30 SURFACE HEAT FLUX AT MIDPOINT VERSUS TIME FOR STANDARD NON-GAPPED FUEL ELEMENT AFTER PULSE

4.6.4.1.2 Hydrogen Pressure in TRIGA® Fuel Elements

To assess the effect of the finite diffusion rate and the rehydriding at the cooler surfaces, the following analysis is presented.

As hydrogen is released from the hot fuel regions, it migrates to the cooler regions and the equilibrium pressure that is obtained is characteristic of some temperature lower than the maximum. To evaluate this reduced pressure, diffusion theory is used to calculate the rate at which hydrogen is evolved and reabsorbed at the fuel surface.

Ordinary diffusion theory provides an expression for describing the time dependent loss of gas from a cylinder:

$$\frac{\bar{c} - c_f}{c_i - c_f} = \sum_{n=1}^{\infty} \frac{4}{Z_n^2} \quad \exp - \frac{Z_n^2 \ Dt}{r_0^2} \; ; \qquad (3)$$

Where:

$\overline{c}$, c_i, c_f	=	the average, the initial, and the final gas concentration in the cylinder, respectively;
Z_n	=	the roots of the Equation $J_0(x) = 0$
D	=	the diffusion coefficient for the gas in the cylinder;
r_0	=	the radius of the cylinder;
t	=	time.

Setting the term on the right-hand side of Equation 3 equal to κ, one can rewrite Equation 3 as:

$$\bar{c}/c_i = {c_f}/{c_i} + \left(1 - {c_f}/{c_i}\right) \kappa \quad (4)$$

and the derivative in time is given by:

$$\frac{d\left({c}/{c_i}\right)}{dt} = \left(1 - {c_f}/{c_i}\right) \frac{d\kappa}{dt} \quad (5)$$

This represents the fractional release rate of hydrogen from the cylinder, f(t). The derivative of the series in the right-hand side of Equation 3 was approximated by:

$$\frac{d\kappa}{dt} = -\left(7.339e^{-8.34\epsilon} + 29.88e^{-249\epsilon}\right)\frac{d\epsilon}{dt} \quad (6)$$

where $\epsilon = {}^{Dt}/_{r_o^2}$

The diffusion coefficient for hydrogen in zirconium hydride in which the H/Zr ratio is between 1.56 and 1.86 is given by:

$$D = 0.25\, e^{-17800/\mathrm{R(T+273)}} \quad (7)$$

where:

R	=	the gas constant; and
T	=	the zirconium hydride temperature in °C

Equation 3 describes the escape of gas from a cylinder through diffusion until some final concentration is achieved. Actually, in the closed system considered here, not only does the hydrogen diffuse into the fuel-clad gap, but also diffuses back into the fuel in the regions of lower fuel temperature. The gas diffuses through the clad at a rate dependent on the clad temperature. Although this tends to reduce the hydrogen pressure, it is not considered in this analysis. When the diffusion rates are equal, an equilibrium condition will exist. To account for this, Equation 5 was modified by replacing the concentration ratios by the ratio of the hydrogen pressure in the gap to the equilibrium hydrogen pressure, P_h/P_e. Thus:

$$f(t) = \frac{d\left({}^{c}/_{c_i}\right)}{dt} = \left(1 - P_h(t)/P_e\right)\frac{d\kappa}{dt} \quad (8)$$

where:

$P_h(t)$	=	the hydrogen pressure, as a function of time; and
P_e	=	the equilibrium hydrogen pressure over the zirconium hydride which is a function of the fuel temperature and H/Zr ratios

The rate of change of the internal hydrogen pressure, in psi, inside the fuel element cladding is:

$$\frac{dP_h}{dt} = \frac{14.7\, f(t)\, N_h}{6.02 \times 10^{23}} \;\frac{22.4}{V_g}\; \frac{T + 273}{273} \quad (9)$$

where:

N_h = the number of molecules of H_2 in the fuel;

T = the gas temperature (°C);

f(t) = the fractional loss rate from Equation 8;

V_g = the free volume inside the fuel clad (liters).

For a fuel volume of 24.4 in^3, the moles of H_2 available from fuel with $ZrH_{1.65}$ and $ZrH_{1.7}$ is 19.9 and 20.6 moles respectively. The free volume is assumed to consist of a cylindrical volume, at the top of the element, 0.125 in. high with a diameter of 1.43 in. for a total of 0.2 in^3. The temperature of the hydrogen in the gap was assumed to be the temperature of the clad. The effect of changing these two assumptions was tested by calculations in which the gap volume was decreased by 90% and the temperature of the hydrogen in the gap was set equal to the maximum fuel temperature. Neither of these changes resulted in maximum pressures different from those based on the original assumptions although the initial rate of pressure increase was greater. For these conditions:

$$P_h = A \times 10^3\,(T+273) \int f(t)\, dt\,; \quad (10)$$

where:

A = 7.29 for $ZrH_{1.65}$ and 7.53 for $ZrH_{1.7}$.

The fuel temperature used in Equation 7 to evaluate the diffusion coefficient is expressed as:

$$T(z) = T_0 \quad ; t<0;$$

$$T(z) = T_0 + (T_m - T_0) \cos[2.45(z-0.5)];\; t \geq 0\,; \quad (11)$$

where:

T_m = the peak fuel temperature (°C);

T_0 = the clad temperature (°C);

z = the axial distance expressed as a fraction of the fuel length;

t = the time after step increase in power.

It was assumed that the fuel temperature was invariant with radius. The hydrogen pressure over the zirconium hydride surface when equilibrium prevails is strongly temperature dependent as shown in Figure 4.20, and for ZrH, can be expressed by:

$$P_e = 2.07 \times 10^9 \, e^{-1.974 \times 10^4/(T+273)} \quad (12)$$

The coefficients have been derived from data developed by Johnson (Reference 4.23). The rate at which hydrogen is released or reabsorbed takes the form:

$$g(t,z) = \frac{[P_e(z) - P_h(t)]}{P_e(z)} f(t,z) \quad (13)$$

where:

f(t,z)	=	the derivative given in Equation 8 with respect to time evaluated at the axial position z;
$P_h(t)$	=	the hydrogen pressure in the gap at time t;
$P_e(z)$	=	the equilibrium hydrogen pressure at the ZrH temperature at position z.

The internal hydrogen pressure is then:

$$P_h(t) = A \times 10^3 \, (T_0 + 273) \int_0^t \int_0^1 g(t, z) \; dz. \quad (14)$$

This equation was approximated by:

$$P_h(t_i) = A \times 10^3 \, (T_0 + 273) \times \sum_{i=i}^{n} \sum_{j=1}^{m} \left[1 - \frac{P_h(t_{i-1})}{P_e(z_j)}\right] 1 \times f(t_i, z_j) \, \delta z \, \delta t \; ; \quad (15)$$

where the inner summation is over the fuel element's length increments and the outer summation is over time.

For the cases where the maximum fuel temperature is 1150°C for $ZrH_{1.65}$ and 1100°C for $ZrH_{1.7}$, the equilibrium hydrogen pressure in ZrH is 2000 psi, which leads to an internal stress of 72,000 psi. Using Equation 14, it is found that the internal pressure for both $ZrH_{1.65}$ and $ZrH_{1.7}$ increases to a peak at about 0.3 sec, at which time the pressure is about one-fifth of the equilibrium value or about 400 psi (a stress of 14,700 psi). After this time, the pressure slowly decreases as the hydrogen continues to be redistributed along the length of the element from the hot regions to the cooler regions.

Calculations have also been made for step increases in power to peak $ZrH_{1.65}$ fuel temperatures greater than 1150°C. Over a 200°C range, the time to the peak pressure and the fraction of the equilibrium pressure value achieved were approximately the same as for the 1150°C case. Similar results were found for fuel with $ZrH_{1.7}$. Thus, if the clad remains below about 500°C, the internal pressure that would produce the yield stress in the clad (35,000 psi) is about 1000 psi and the corresponding equilibrium hydrogen pressure is 5000 psi. This corresponds to a maximum fuel temperature of about 1250°C in $ZrH_{1.65}$ and 1180°C in $ZrH_{1.7}$. Similarly, an internal pressure of 1600 psi would produce a stress equal to the ultimate clad strength (over 59,000 psi). This corresponds to an equilibrium hydrogen pressure of 5 x 1600 or 8000 psi and a fuel temperature of about 1300°C in $ZrH_{1.65}$ and 1240°C in $ZrH_{1.7}$.

Measurements of hydrogen pressure in TRIGA® fuel elements during steady-state operation have not been made. However, measurements have been made during transient operations and compared with the results of an analysis similar to that described here. These measurements indicated that in a pulse in which the maximum temperature in the fuel was greater than 1000°C the maximum pressure ($ZrH_{1.65}$) was only about 6% of the equilibrium value evaluated at the peak temperature. Calculations of the pressure resulting from such a pulse using the methods described above gave calculated pressure values about three times greater than the measured values.

An instantaneous increase in fuel temperature will produce the most severe pressure conditions. When a peak fuel temperature is reached by increasing the power over a finite period of time, the resulting pressure will be no greater than that for the step change in power analyzed above. As the temperature rise times become long compared with the diffusion time of hydrogen, the pressure will become increasingly less than for the case of a step change in power. The reason for this is that the pressure in the clad element results from the hot fuel dehydriding faster than the cooler fuel rehydrides (takes up the excess hydrogen to reach an equilibrium with the hydrogen over pressure in the can). The slower the rise to peak temperature, the lower the pressure because of the additional time available for rehydriding.

4.6.4.1.3 ZrH Fuel Temperature Limits

The foregoing analysis gives a strong indication that the cladding will not be ruptured if fuel temperatures are never greater than in the range of 1200°C to 1250°C, providing that the cladding temperature is less than about 500°C. However, for fuel with a $ZrH_{1.7}$ a conservative safety limit of 1100°C has been chosen for this condition. As a result, at this safety limit temperature the pressure is about a factor of 4 lower than would be necessary for cladding failure. This factor of 4 is more than adequate to account for uncertainties in cladding strength and manufacturing tolerances. As a safety limit, the peak adiabatic fuel temperature to be allowed during transient conditions is considered to be 1100°C for U- ZrH fuel with ratios up to 1.70. Under any condition in which the cladding temperature increases above 500°C, the temperature safety limit must be decreased as the cladding material loses strength at elevated temperatures. To establish this limit, it is assumed that the fuel and the cladding are at the same temperature. There are no conceivable circumstances that could give rise to a situation in which the cladding temperatures are higher than the fuel temperature.

In Figure 4.20, the stress imposed on the clad by the equilibrium hydrogen pressure as a function of the fuel temperature is plotted. Also shown is the ultimate strength of 304 stainless steel at the same temperatures. The use of these data for establishing the safety limit for conditions in which the cladding temperature is greater than 500°C is justified as:

a. the method used to measure ultimate strength requires the imposition of the stress over a longer time than would be imposed for accident conditions;

b. the stress is not applied biaxially in the ultimate strength measurements as it is in the fuel clad

The point at which the two curves in Figure 4.20 intersect (for $ZrH_{1.7}$) is the safety limit, that is, 930°C for conditions in which the cladding temperature is above 500°C. At that temperature, the equilibrium hydrogen pressure would impose a stress on the cladding equal to the ultimate strength of the clad.

The same argument about the redistribution of the hydrogen within the fuel presented earlier is valid for this case. In addition, at elevated temperatures the cladding becomes quite permeable to hydrogen. Thus, not only will hydrogen redistribute itself within the fuel to reduce the pressure, but some hydrogen will escape from the system entirely.

The use of the ultimate strength of the cladding material in the establishment of the safety limit under these conditions is justified because of the transient nature of accidents. Although the high cladding temperatures imply sharply reduced heat transfer rates to the surroundings (and consequently longer cooling times), only slight reductions in the fuel temperature are necessary to reduce the stress sharply. For a fuel with $ZrH_{1.7}$, a 40°C decrease in temperature from 930°C to 890°C will reduce the stress by a factor of 2. The above analyses and limits are generic. They establish the bounds of the element's capability; the limits are not related to any specific fuel element power or fissionable material content. They relate to the temperatures in the element, to the properties of the fuel, and to the strength of and the stress on the cladding that can be allowed without cladding rupture.

4.6.4.1.4 Performance of High Uranium wt% Fuels

A substantial review and evaluation of the performance of 20 and 30 wt% uranium fuel elements was conducted based on the information provided in References 4.1, 4.24 and 4.25. The basic conclusions are:

> *The performance of these higher uranium content fuels is substantially independent of uranium content up to at least 45 wt%. The 20 and 30 wt% fuel are indistinguishable from the 8.5 wt% fuel. Fuel growth is as predicted; there is limited thermal migration of hydrogen; there is no pressure buildup inside the cladding as burnup proceeds; and the fission product release fractions from high-burnup elements is not significantly different from fresh fuel (Reference 4.24). From these studies, the release fractions of fission products were observed not to be related to uranium content; a single correlation serves to describe the gas release behavior over a broad temperature range (Reference 4.25). The basic release fraction for fuel temperatures less than 400°C remains as assessed previously (1.5 x 10^{-5}). These studies covered burnups up to 64% of the uranium-235 content.*

In summary, the prompt negative temperature coefficient, fuel properties, irradiation performance, behavior under pulse heating, and effect of hydrogen disassociation on the fuel element safety limits, for fuel containing up to 45 wt% uranium, were all found to mirror that of the reference 8.5 wt% fuel.

TRIGA® fuels of 20, 30, and 45 wt% uranium, 19.7% enriched, were irradiated in the Oak Ridge Research Reactor (ORR) and thoroughly examined (References 4.24 and 4.26), and evaluated (Reference 4.1). Table 4-6 presents a profile of the irradiation conditions of these elements.

The performance base for the higher wt% LEU fuels irradiated in ORR encompasses burnups to greater than 60% of the contained ^{235}U, exposures as high as 919 full power days and fast neutron fluence of 5 x 10^{21}n/cm^2. The maximum linear power density during the ORR irradiations was comparable to the maximum predicted in Section 4.5 for the worst UCD/MNRC case.*

TABLE 4.6 ORR IN-PILE IRRADIATION PARAMETERS

	20 Wt-%U	30 Wt-%U	45 Wt-%U
Contained U-235 per 22 in. fuel rod (g)	[redacted]	[redacted]	[redacted]
Vol-% U (19.7% Enriched)	7	11	20
Max Calc Rod Power Generation (kW)			
Initial Configuration		36	35
Full Cluster Configuration	41	43	48
45 Wt-% Only Configuration			55
Time at Power (FPD)			
Initial Configuration (Dec 79-Nov 80)	0	278	278
Full Cluster Config (May 81-Nov 82)	295	295	295
45 Wt-% Only Config (July 82-Nov 83)	0	0	328
45 Wt-% Only Config (Aug 84)	0	0	18
Target Burnup of U-235 (%)	35	40	50
Final Burnup Range (%)	45-57	47-57	60-66

It is stated in References 4.24 and 4.26 that the temperatures experienced by the LEU fuel during the ORR irradiations ranged from 25° C to 650° C. The upper end of this temperature range exceeds that for 2 MW operation in the UCD/MNRC. In addition, the performance of these fuels under extended thermal cycling and pulse heating has been reviewed and evaluated (Reference 4.1). The thermal cycling specimens were cycled 100 times out of pile and then 32 times in a neutron flux of 4 x 10^{12}n/cm^2 sec over the temperature range of 500° to 725°C. The fuel displayed outstanding integrity and stability.

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*ORR maximum was 1.26 max/avg x 55 kW/55.9 cm = 1.24 kW/cm, UCD/MNRC maximum is 1.33 max/avg x 33.2 kW/38.1 cm = 1.16 kW/cm.

A 45 wt% uranium LEU fuel rod that was instrumented for measuring temperature and pressure was subjected to a series of 30 power pulses in a TRIGA® reactor to maximum temperatures in the range of 1050° to 1100°C. Only very modest (generally less than 2 psi) pressure pulses were measured in the rod as a result of the pulsing. This is in agreement with previous data showing negligible hydrogen release during the pulsing of 8.5 wt% uranium fuel to temperatures up to 1150°C. All surveillance examinations showed no rod deformation. Tests have shown that the pulse response of uranium-zirconium hydride TRIGA® fuel is independent of the uranium content of the fuel and is dominated by the behavior of the zirconium hydride, along with the prompt temperature coefficient of reactivity. Highly burned fuel does not necessarily have the benign response to power pulsing that was demonstrated in these tests. Hydrogen migrates to the fuel pellet periphery during burnup and a strong pulse under these conditions can produce excessive hydrogen pressure and cladding deformation. The pulse analysis in Section 13.2.2 predicts that highly irradiated fuel can be subjected to a reactivity pulse as large as $1.92 without damage which justifies maximum amount of fixed negative reactivity worth experiments in the core at a given time.

It was concluded in Reference 4.1:

Tests of uranium-zirconium hydride fuels have shown that the limiting design basis for the operation of TRIGA® fuels is independent of uranium content up to at least 45 wt%.

4.6.4.2 Prompt Negative Temperature Coefficient

The basic parameter which provides the greatest degree of safety in the operation of a TRIGA® reactor system is the prompt negative temperature coefficient of reactivity. This temperature coefficient (α) allows great freedom in steady-state operation, since the effect of accidental reactivity changes occurring from experimental devices in the core is minimized.

The prompt negative temperature coefficient for the TRIGA®-LEU core is based on the same core spectrum hardening characteristics that occurs in a standard* TRIGA® core. The spectrum hardening is caused by heating of the fuel-moderator elements. The rise in temperature of the hydride increases the probability that a thermal neutron in the fuel element will gain energy from an excited state of an oscillating hydrogen atom in the lattice. As the neutrons gain energy from the ZrH, the thermal neutron spectrum in the fuel element shifts to a higher average energy (the spectrum is hardened), and the mean free path for neutrons in the element is increased appreciably. For a standard TRIGA® element, the average chord length is comparable to a mean free path, and the probability of escape from the element before being captured is significantly increased as the fuel temperature is raised. In the water, the neutrons are rapidly rethermalized so that the capture and escape probabilities are relatively insensitive to the energy with which the neutron enters the water. The heating of the moderator mixed with the fuel in a standard TRIGA® element thus causes the spectrum to harden more in the fuel than in the water.

*A standard TRIGA® core contains U-ZrH fuel with no erbium. The uranium is 8.5 wt-% with an enrichment of 20%, and the fuel element (rod) diameter is about 1.5 in. (3.8 cm) with a core water volume fraction of about 0.33.

As a result, there is a temperature-dependent disadvantage factor for the unit cell in which the ratio of absorptions in the fuel to total cell absorptions decreases as fuel element temperature is increased. This brings about a shift in the core neutron balance, giving a loss of reactivity.

In the TRIGA®-LEU fuel, the temperature-hardened spectrum is used to decrease reactivity through its interaction with a low-energy-resonance material. Thus, erbium, with its double resonance at ~0.5 eV, is used in the TRIGA®-LEU fuel as both a burnable poison and a material to enhance the prompt negative temperature coefficient. The ratio of the absorption probability to the neutron leakage probability is increased for TRIGA®-LEU fuel relative to the standard TRIGA® fuel because the ^{235}U density in the fuel rod is about 2.5 times greater and also because of the use of erbium. When the fuel-moderator material is heated, the neutron spectrum is hardened, and the neutrons have an increasing probability of being captured by the low-energy resonances in erbium. This increased parasitic absorption with temperature causes the reactivity to decrease as the fuel temperature increases. The neutron spectrum shift, pushing more of the thermal neutrons into the ^{167}Er resonance as the fuel temperature increases, is illustrated in Figure 4.31, where cold and hot neutron spectra are plotted along with the energy-dependent absorption cross section for ^{167}Er. As with a standard TRIGA® core, the temperature coefficient is prompt because the fuel is intimately mixed with a large portion of the moderator; thus, fuel and solid moderator temperatures rise simultaneously, producing the temperature- dependent spectrum shift.

For the reasons just discussed, more than 50% of the temperature coefficient for a standard TRIGA® core comes from the temperature-dependent disadvantage factor, or cell effect, and ~20% each come from Doppler broadening of the ^{238}U resonances and temperature- dependent leakage from the core. These effects produce a temperature coefficient of about 9.5 x 10^{-5}/°C, which is essentially constant with temperature. On the other hand, for the TRIGA®-LEU core, the effect of cell structure on the temperature coefficient is smaller. Over the temperature range 73° to 1292°F (23° to 700°C), about 70% of the coefficient comes from temperature-dependent changes in ηf within the core, and more than half of this effect is independent of the cell structure. Almost all the remaining part of the prompt negative temperature coefficient is contributed by Doppler broadening of the ^{238}U resonances. Over the temperature range 73° to 1292°F (23° to 700°C), the temperature coefficient for the TRIGA®-LEU fuel is about 1.07 x 10^{-4}/°C, thus being somewhat greater than the value for standard TRIGA® fuel. It is also temperature dependent.

The calculation of the temperature coefficient for standard TRIGA® and TRIGA®-LEU cores requires a knowledge of the differential slow neutron energy transfer cross section in water and ZrH, the energy dependence of the transport cross section of hydrogen as bound in water and ZrH, the energy dependence of the capture and fission cross sections of all relevant materials, and a multi-group transport theory reactor description which allows for the coupling of groups by speeding up as well as slowing down.

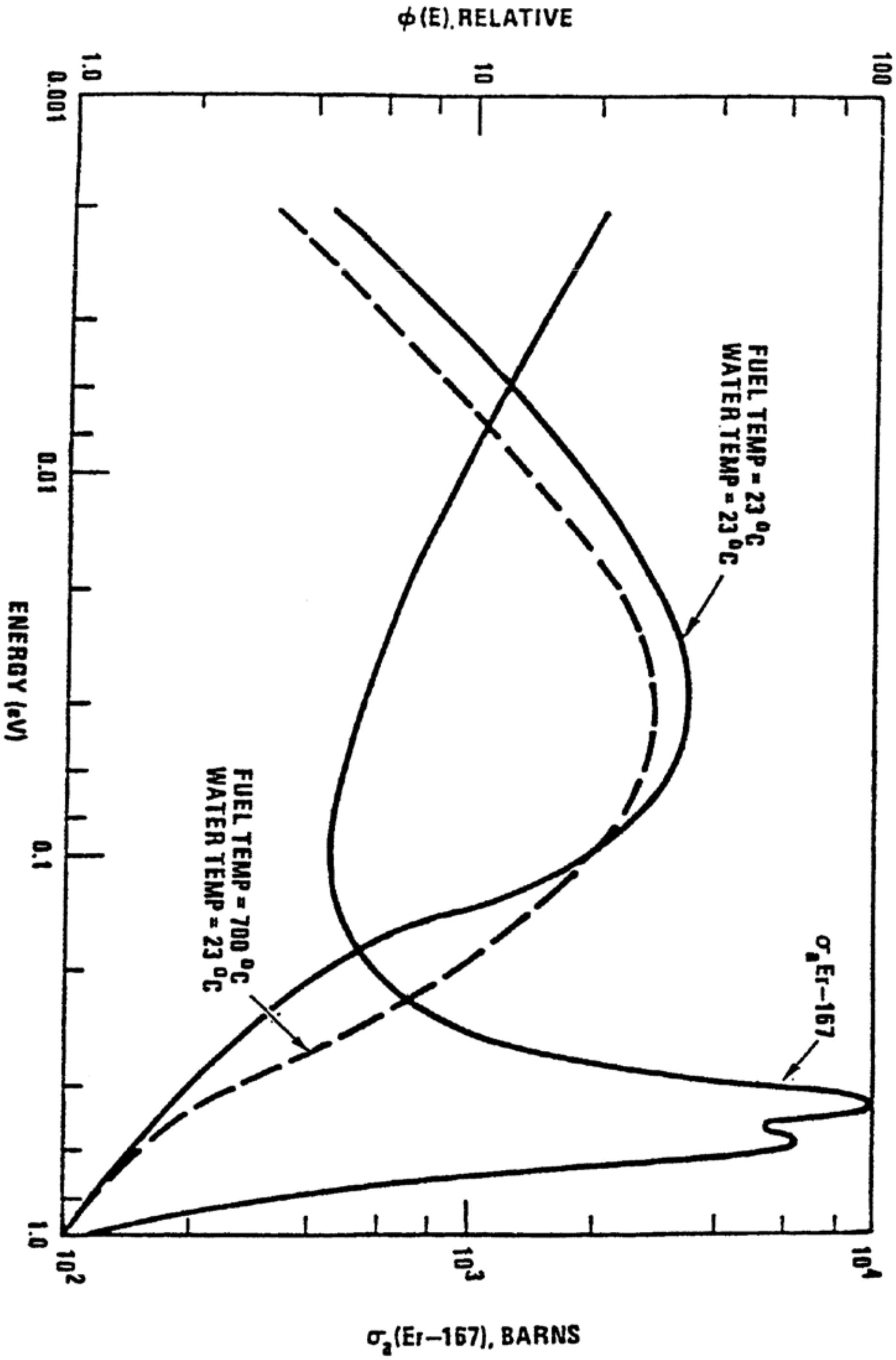

FIGURE 4.31 THERMAL NEUTRON SPECTRA VERSUS FUEL TEMPERATURE RELATIVE TO σa VERSUS ENERGY FOR Er-167

Qualitatively, the scattering of slow neutrons by ZrH can be described by a model in which the hydrogen atom motion is treated as an isotropic harmonic oscillator with energy transfer quantized in multiples of ~0.14 eV. More precisely, the calculational model uses a frequency spectrum with two branches: one for the optical modes for energy transfer for the bound proton and the other for the acoustical modes for energy transfer with the lattice as a whole. The optical modes are represented as a broad frequency band centered at 0.14 eV and whose width is adjusted to fit measured cross-section data. The low-frequency acoustical modes are assumed to have a Debye spectrum with a cutoff of 0.02 eV and a weight determined by an effective mass of 360.

This structure then allows a neutron to thermalize by transition in energy units of ~0.14 eV so long as its energy is above 0.14 eV. Below 0.14 eV, the neutron can still lose energy by the inefficient process of exciting acoustic Debye-type modes in which the hydrogen atoms move in phase with one another. These modes therefore correspond to the motion of a group of atoms whose mass is much greater than that of hydrogen, and indeed even greater than the mass of zirconium. Because of the large effective mass, these modes are very inefficient for thermalizing neutrons; but for neutron energies below 0.14 eV, they provide the only mechanism for slowing down the neutron. (In a TRIGA® core, the water provides for ample neutron thermalization below 0.14 eV.) In addition, in the ZrH it is possible for a neutron to gain one or more energy units of ~0.14 eV in one or several scatterings from excited Einstein oscillators. Since the number of excited oscillators present in a ZrH lattice increases with temperature, this process of neutron acceleration is strongly temperature dependent and plays an important role in the behavior of ZrH-moderated reactors.

The temperature coefficient for the TRIGA®-LEU core increases as a function of fuel temperature because of the steadily increasing number of thermal neutrons being pushed into the ^{167}Er resonance. This temperature-dependent character of the temperature coefficient of a TRIGA® core containing erbium is advantageous in that a minimum reactivity loss is incurred in reaching normal operating temperatures, but any sizable increase in the average core temperature results in a sizably increased prompt negative temperature coefficient to act as a shutdown mechanism. The temperature coefficients computed by GA Technologies (Reference 4.3) at beginning of life and 1000 and 2000 MWd of burnup are shown in Figure 4.32. After 1000 and 2000 MWd of burnup, the coefficient is less temperature dependent and smaller in magnitude than that for the initial clean core because of the sizable burnup of ^{167}Er and the resulting increased transparency of the approximate 0.5-eV resonance region to thermal neutrons.

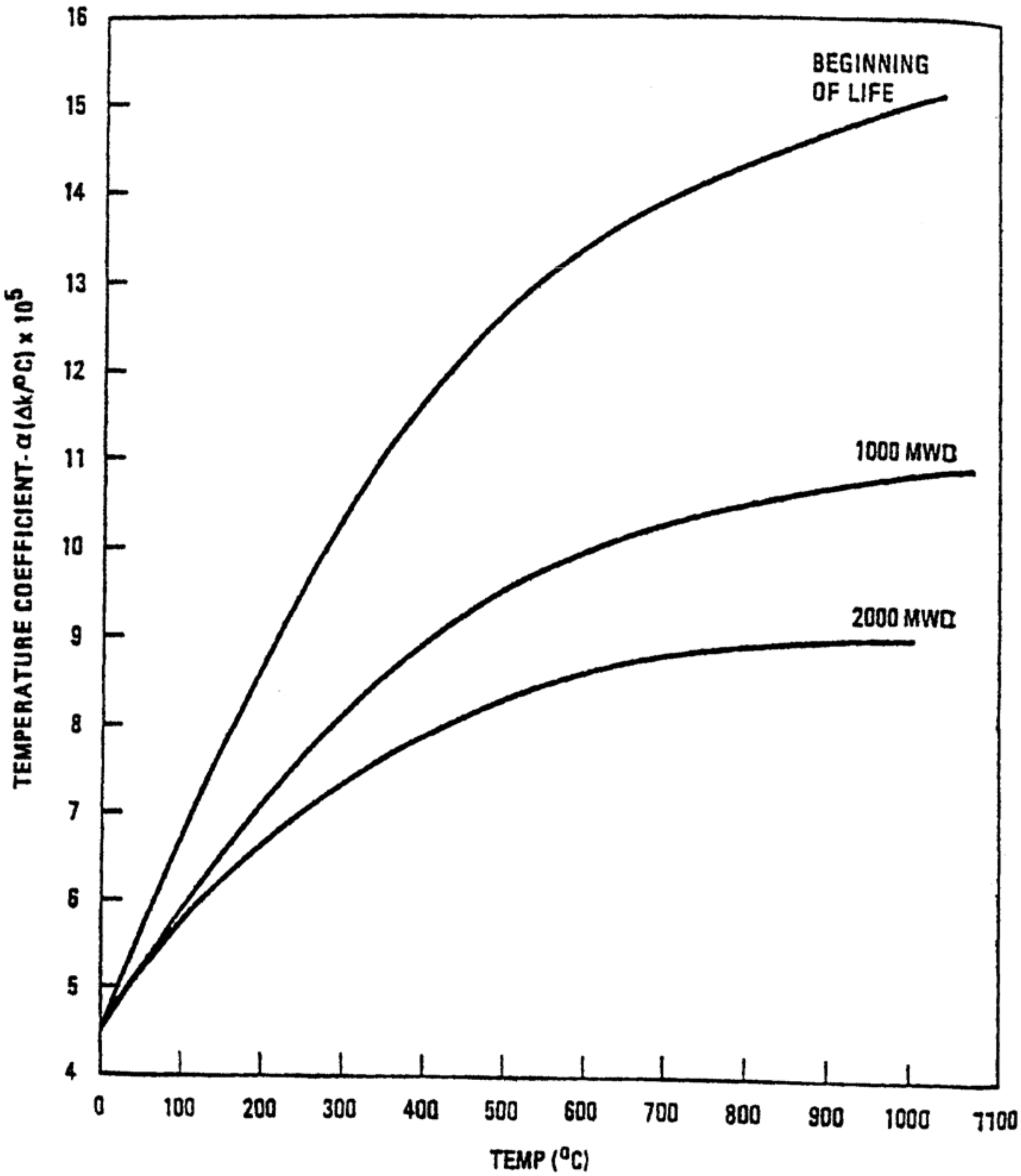

FIGURE 4.32 PROMPT NEGATIVE TEMPERATURE COEFFICIENT FOR TRIGA® LEU FUEL [20 wt-% URANIUM (19.7% ENRICHED), 0.47 wt-% ERBIUM]

The prompt negative temperature coefficient was computed for 20/20 and 30/20 fuel types at three burnups and several temperatures by Chen (Reference 4.28). The WIMS-D4 code and ENDF/B Version 5 cross section data base were used for these calculations. For a given burnup, a WIMS calculation of the eigenvalue was made for fuel at each of the discrete temperatures at which the scattering kernels for H and ZrH are available in WIMS. The scattering kernels in WIMS embody the energy transfer mechanisms described above. The fuel temperatures ranged for 300 °K to 1050 °K while the coolant temperature was kept at 300 °K. A simple finite difference approximation over 50° intervals was used to determine the temperature coefficient. For each fuel type and burnup, the data points from Chen's thesis were fit to a quadratic polynomial for use in the accident analyses in Chapter 13.

Evidence that this procedure yielded accurate prompt temperature coefficients is given in Figure 4.33. It shows the temperature coefficient for 20/20 fuel at approximately 13% ^{235}U burnup computed by two independent approaches, the procedure just described and a GA Technologies calculation. (The GA prediction is the 1000 MWD curve in Figure 4.32, which came from Reference 4.29, and the WIMS prediction is from Chen's 10 MWD/rod data.) The two predictions agree to within a few percent over the entire temperature range. When these two alternatives were used in Nordheim-Fuchs calculations of the peak temperature resulting from a $2.20 step reactivity insertion, the resulting peak fuel temperature differed by only 4 °C.

The prompt negative temperature coefficient was computed for standard TRIGA® fuel (8.5/20) by GA Technologies (Reference 4.27). The temperature coefficient, which is essentially independent of temperature and burnup, is shown in Figure 4.16 of Reference 4.27 for a typical high-hydride TRIGA® core.

4.6.4.2.1 Validation of MNRC MCNP Core Model

Based on reactor operation in October 2018, estimated critical position (ECP) is accomplished when the transient rod at D4 location and 4 other control rods at D7, G3, G9, J4 locations were banked at 60% withdrawal, while the designated regulating rod at J7 location was withdrawn at 48%. The central irradiation facility is occupied by the aluminum thimble and cylindrical graphite sleeve. The MCNP simulated core configuration represents the same OCC with those six control rods withdrawn at the same heights. The K_{ECP} is calculated to be 1.00087 +/- 0.00011. The difference is within 0.001, or 100 pcm.

The MCNP simulation of the OCC (as of 2018) was benchmarked by evaluating each individual control rod worth and compare to the measured values during the annual shutdown for reactor maintenance in August 2018. During the control rod worth measurements, these are accomplished by banking 5 control rods at 60% withdrawal and slowly raising the "evaluated" control rod from 100% insertion to 100% withdrawal. Sequentially, the reactor core begins in subcritical condition with 5 control rods at 60% withdrawal, becomes critical at low power, and continues its power increase up to about 900 W, but <1 kW without adding detectable heat to the reactor core, when the "evaluated" control rod is 100% withdrawal. The effective delayed neutron fraction is 0.0075, chosen from the range of 0.0071 to 0.0075, originally recommended by General Atomics (GA). As listed below between measured and calculated values, each calculated control rod worth, compared to the measured value, is within 10%.

Control Rod at D4 Location: (Transient Rod)

Measured Value: $1.83
Calculated Value: $1.78 +/- $0.03

Control Rod at D7 Location:

Measured Value: $2.49
Calculated Value: $2.65 +/- $0.03

Control Rod at G3 Location:

Measured Value: $2.61
Calculated Value: $2.49 +/- $0.03

Control Rod at G9 Location:

Measured Value: $2.56
Calculated Value: $2.54 +/- $0.03

Control Rod at J4 Location:

Measured Value: $2.91
Calculated Value: $2.78 +/- $0.03

Control Rod at J7 Location: (Regulating Rod)

Measured Value: $2.78
Calculated Value: $2.62 +/- $0.03

Validated OCC Shutdown Margin

Based on the above evaluations of control rod worths, the OCC shutdown margin in October 2018 was:

Total Control Rod Worth:	$15.18
(Subtract) Total Core Excess:	- $5.16
(Subtract) Highest Rod Worth Non-Secured Experiment:	- $1.00
(Subtract) Most Reactive Control Rod Worth:	- $2.91
Validated Shutdown Margin (October 2018)	= $6.11

4.6.4.3 Operating Core Configuration (OCC)

Monte Carlo (MCNP) code, version 5, with ENDF/ B-VI continuous neutron and photon cross-section data, has been used to evaluate the OCC and subsequent limiting core configuration (LCC) with both the 20/20 and 30/20 type TRIGA fuel elements, together with five control rods having 20/20 type fuel as followers (FFCRs) and one transient rod. The transient rod was used for pulsing runs in the past, but now serves as one additional control rod. All MCNP studies are based on actual fuel burnup information updated in October 2018, which completes the official filing for our fuel depletion and inventory.

To evaluate the clean reactor core excess reactivity compared to the reactor operational data for a benchmark, all six control rods, including one transient rod and five FFCRs, are completely out of the reactor core, or in full up position. The clean excess reactivity is calculated to be $5.16, which is consistent with the measured value of $5.13 in the startup of Monday morning on October 1, 2018.

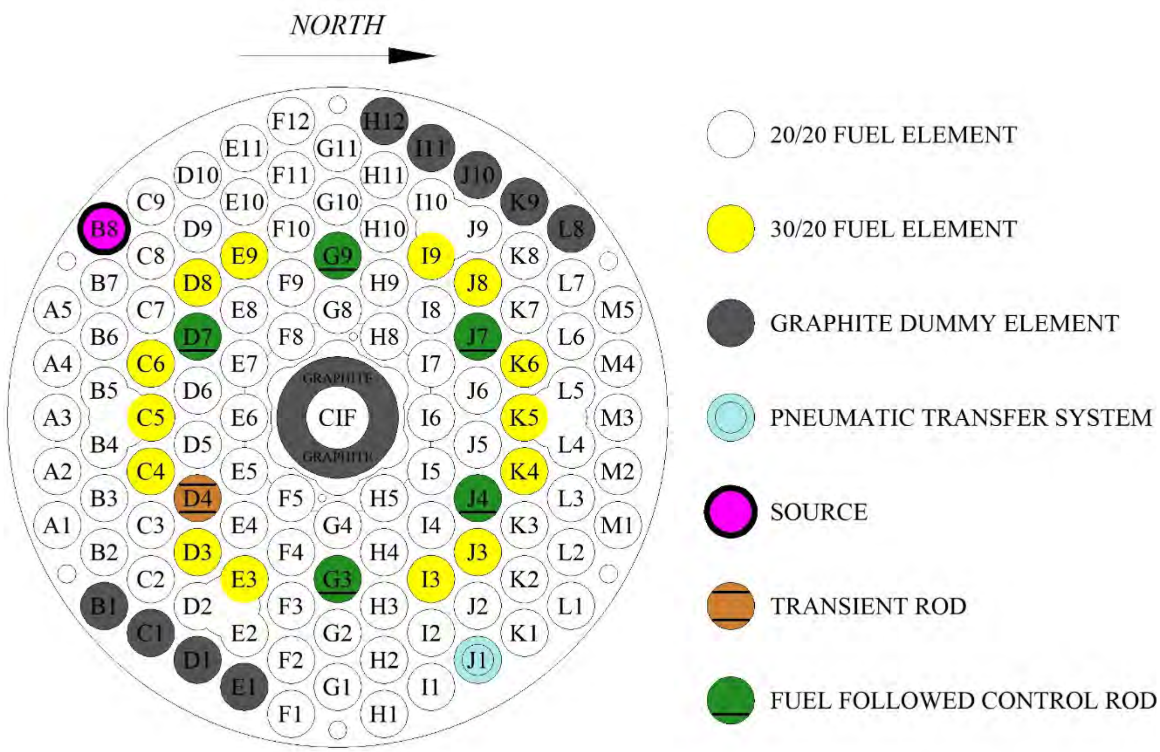

Figure 4.33 OCC's fuel map. There are 121 positions in total in the fuel grid plate; 83 20/20 type fuel elements, 14 30/20 type fuel elements, 1 transient rod and 5 fuel-followed control rods, 9 graphite dummy rods, 1 neutron source, 1 pneumatic transfer system (PTS), and aluminum thimble and cylindrical graphite sleeve in the central irradiation facility, occupying 7 positions.

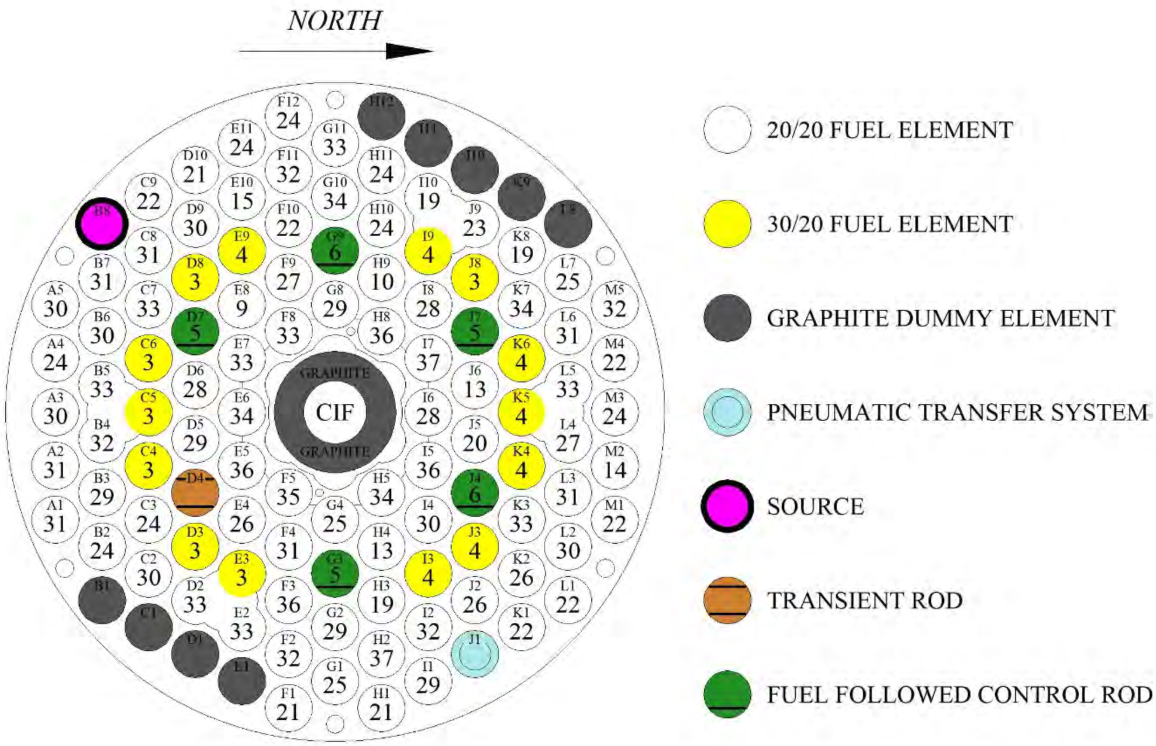

Figure 4.34 OCC fuel burnup rate in percentages. The average burnup rate for all 20/20 type fuel elements is >27%. Fourteen 30/20 type fuel elements were added in year 2010. These fuel elements are placed in E-hex locations, and the burnup rate is about 3% to 4%.

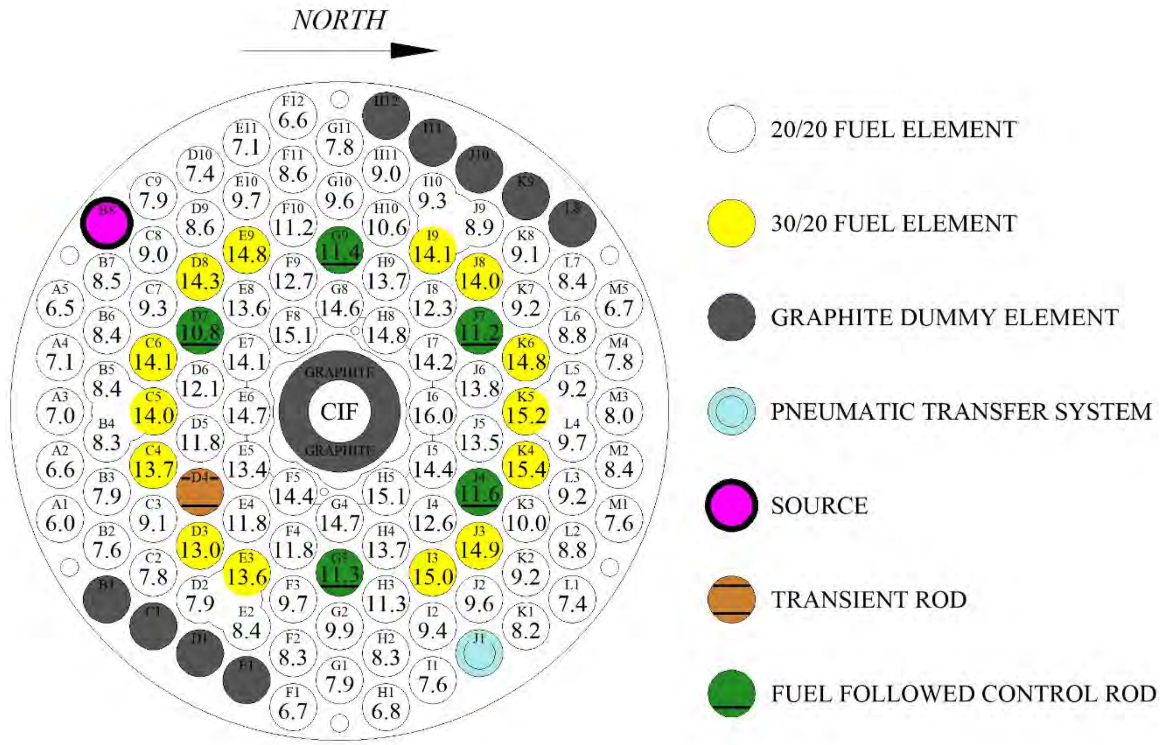

Figure 4.35 OCC's power distribution at 1.1 MW. For all 20/20 type fuel elements, the maximum power per fuel element is 16.0 kW, which is at I6 location in C-hex. For all 30/20 type fuel elements, the maximum power per fuel element is 15.4 kW, which is at K4 location in E-hex. Currently, two instrumented fuel elements (IFEs) are used; one 20/20 type IFE are located at I6 position in C-hex and one 30/20 type IFE are located at E9 position in E-hex. Both IFEs are located in the opposite positions of the reactor core with close to the highest values of power per fuel element. Their normal readings are about 320 to 330 degrees C at 1.0 MW steady state operation.

4.6.4.4 Future Cores and the Limiting Core Configuration (LCC)

The next evolution of the MNRC will take place once the MNRC license has been revised to allow for placement of 30/20 elements in C ring. The purpose of this core evolution is to shift the highest fission rate fuel elements to coincide with the lowest burnup elements. The intent of this fuel shuffle is to increase excess reactivity and prolong core life.

The proposed future cores will begin with relocating those 30/20 type fuel elements into the C-hex locations. By doing so (see Figure 4.36), the excess reactivity increases $ 0.92 compared to the OCC with aluminum thimble and cylindrical graphite sleeve are in place.

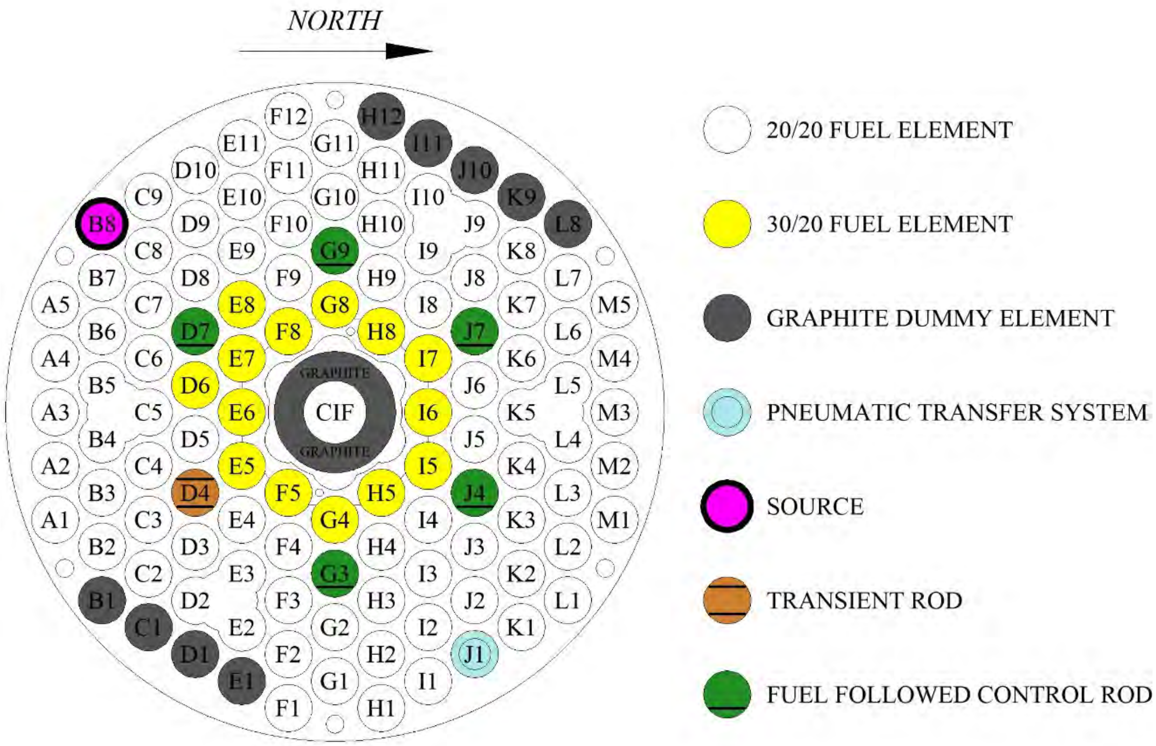

Figure 4.36 LCC's fuel map: 14 30/20 type fuel elements are relocated to 12 positions in C-hex and 2 positions in D-hex.

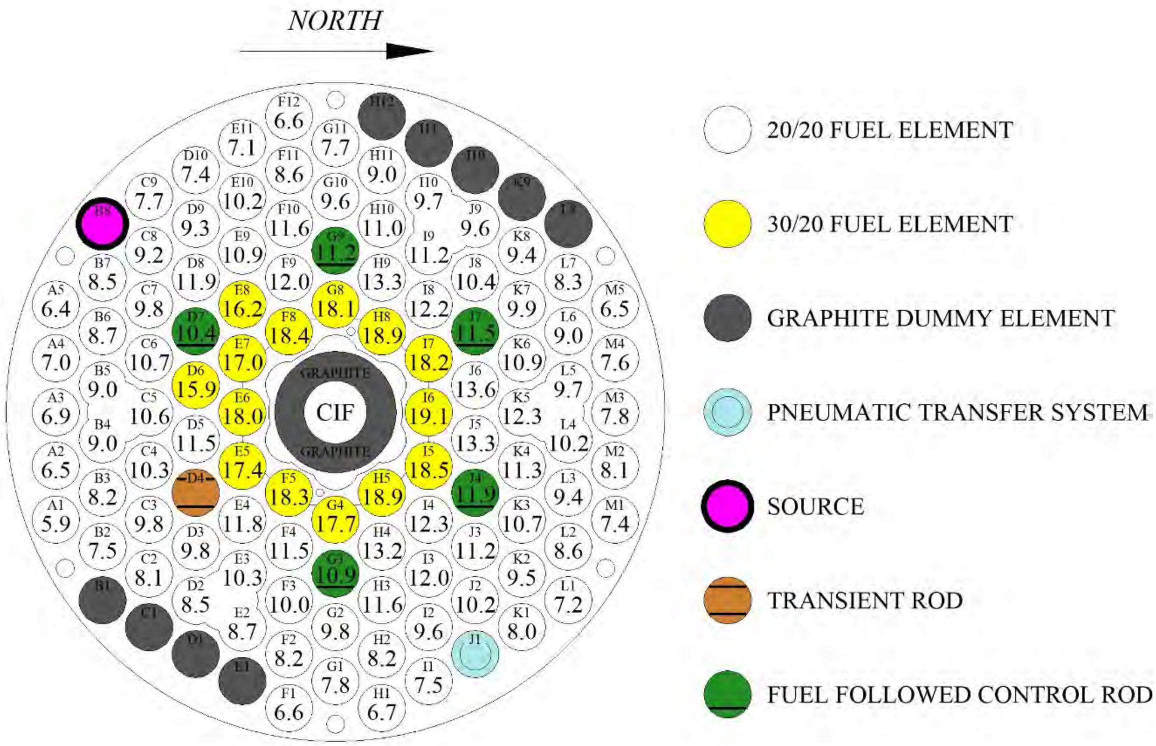

Figure 4.37 LCC's power map: Fuel elements with maximum power per fuel element are now in C-hex, ranging from 17.0 kW to 19.1 kW at 1.1 MW operating power.

The hot fuel rod is found to be 17.7 kW at I6 location in the C-hex at 1.0 MW operating power. This maximum heat output for an element is higher than the in the OCC but is significantly lower than the maximum heat output (>30kW) of the previously license 2.0 MW steady state MNRC core. This hot fuel rod location will become where the 30/20 type IFE should be positioned in the future.

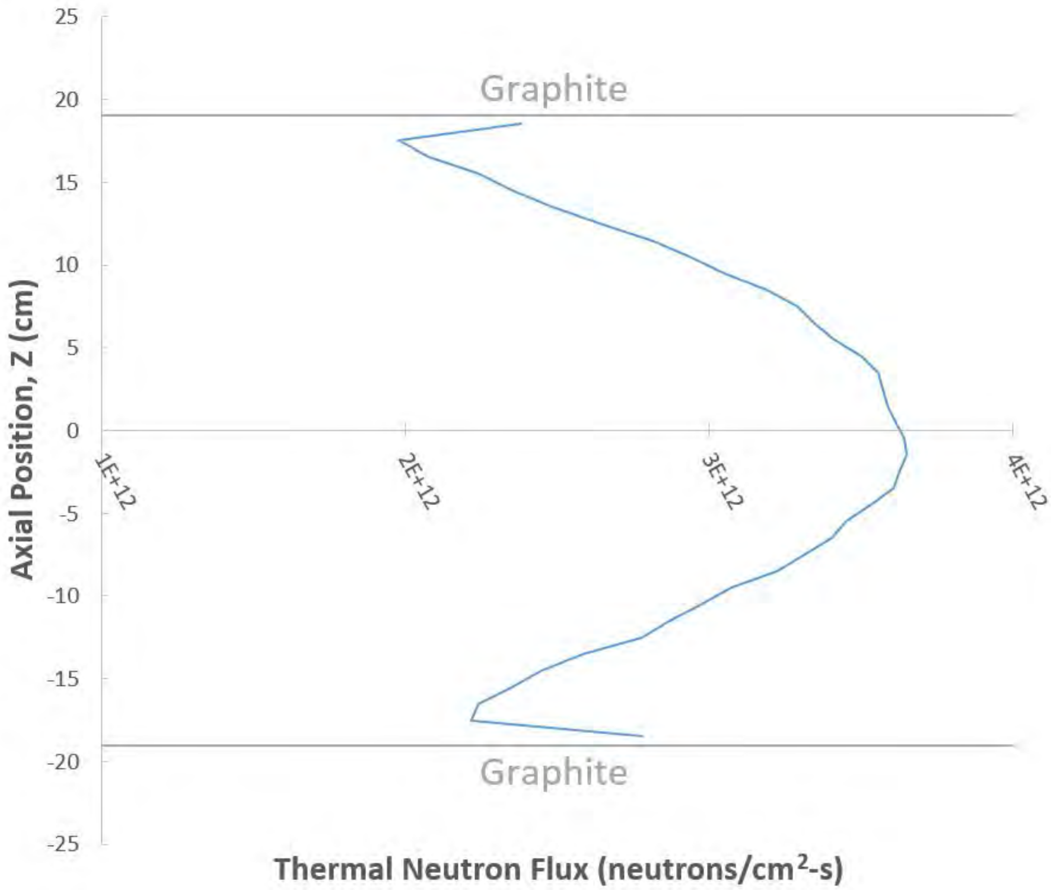

Figure 4.38 Axial thermal neutron flux distribution of the hot fuel rod at I6 location at 1.0 MW operating power. The length of the fuel meat, i.e. U-ZrH, is 15”, or 38.1 cm.

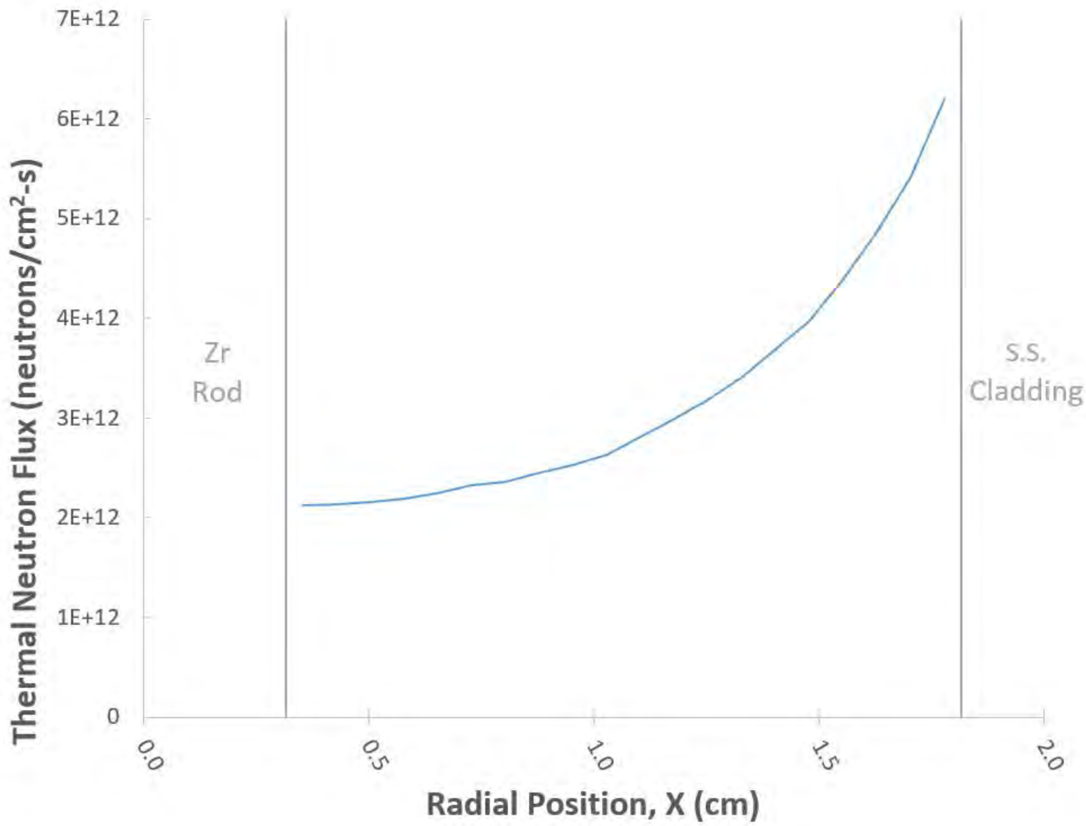

Figure 4.39 Radial thermal neutron flux distribution of the hot fuel rod at I6 location at 1.0 MW operating power. The diameter of the fuel element, including a Zr rod in the center and S.S. cladding, is 1.47", or 3.7338 cm.

The evolution of the future over time should result in slightly wider safety margins as the core progresses to the facility's end of life near 2040. MNRC typically operates 1,200 MWhrs per year, which is expected to continue. To further demonstrate how the power of this hot fuel rod changes over time, an accelerated burnup study was made. By projecting 10 yrs of normal operation into the future, i.e. 12,000 MWhrs, the additional burnups on average of those 5 FFCRs, 14 30/20 type fuel elements, and 83 20/20 type fuel elements are 7.7%, 6.1%, and 5.4%, respectively. As expected, additional burnups of 30/20 type fuel elements, which are located closer to the reactor core center, are higher than those of 20/20 type fuel elements. After additional burnups were included to the future, the hot fuel rod at I6 location in the C-hex remains about 17.6 kW at 1.0 MW operating power. Therefore, the peak power may drift slightly lower over time.

MNRC has requested additional fresh 30/20 type fuel elements to support its long-term operation and its final anticipated core configuration. Figure 4.40 shows additional 10 30/20 type fuel elements, replacing existing 20/20 type fuel elements, are added in D-hex of the LCC. In the meantime, these 20/20 type fuel elements replace those graphite dummy rods in G-hex. In this core configuration, the hot fuel rod is found to be 15.5 kW, which is significantly lower than 17.7 kW. This is because not only additional U-235 from fresh 30/20 type fuel elements is added to the reactor core, but also 9 existing 20/20 type fuel elements are added in G-hex to share, and reduce the power load per fuel element. A similar accelerated burnup study was also made to assess the power change of the hot fuel rod after 10 yrs of normal operation. After 10 yrs of normal operation into the future, the additional burnups on average of those 5 FFCRs, 24 30/20 type fuel elements, and 82

20/20 type fuel elements are 6.9%, 5.3%, and 4.8%, respectively. The hot fuel rod at I6 location drifts slightly lower and becomes 15.4 kW. Is migration the final anticipated MNRC core configuration will take place gradually to avoid any condition where the allowable core excess is exceeded. It is therefore expected that the LCC will remain the bounding configuration, for the thermal hydraulic calculations, for the remainder of the MNRC's lifetime.

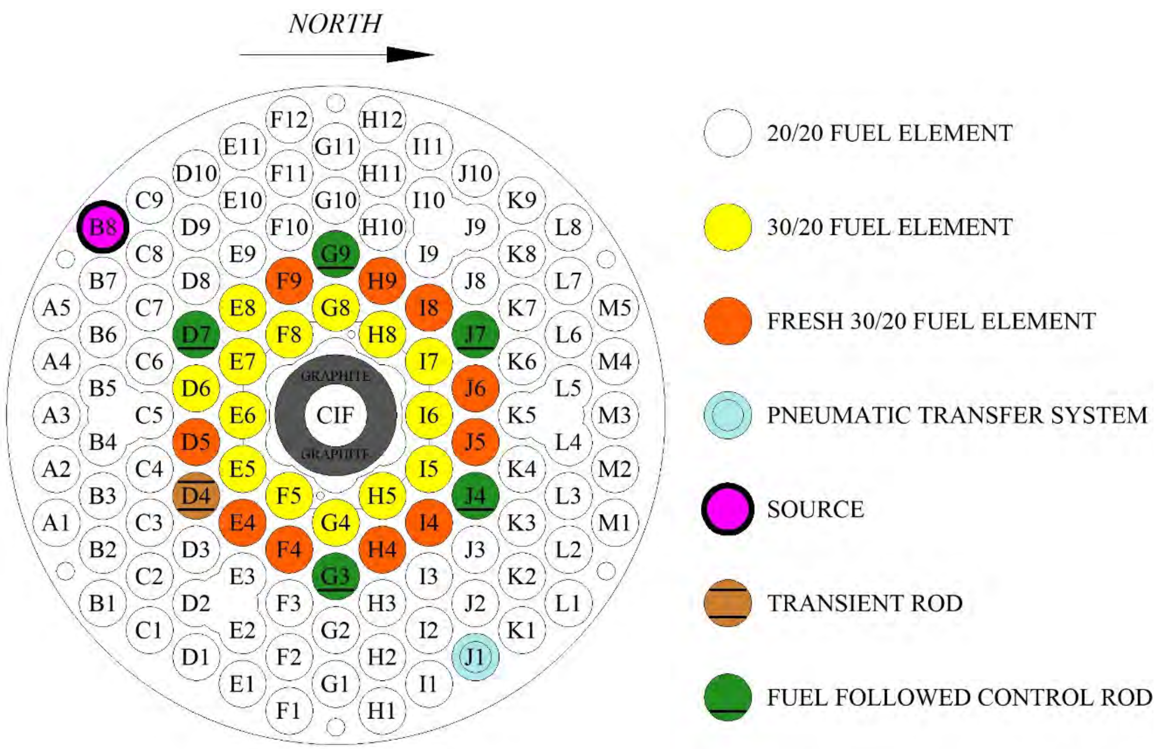

Figure 4.40 Future LCC's fuel map: 10 additional, fresh 30/20 type fuel elements are added in D-hex and 9 graphite dummy rods are replaced by existing 20/20 type fuel elements from D-hex.

4.6.4.5 Fission Product Release Fraction

Considerable effort has been expended to measure and define the fission product release fractions for TRIGA® LEU fuels. Data on this aspect of fuel performance are reported in References 4.24, 4.25 and 4.26 and evaluated in Reference 4.1.

Using these data, GA developed a conservative correlation for fission gas release:

$$\textit{Release Fraction} = 1.5 \times 10^{-5} + 3600e^{-13400/T} \; \textit{(16)};$$

where:

T = fuel temperature in degrees Kelvin.

In characterizing the conservatism, it is stated on page 35 of Reference 4.25, "18At normal TRIGA® operating temperatures (<750°C), there is a safety factor of approximately four between predicted values by the above equation and experimentally deduced values." The same observation is reported in Reference 4.37. This correlation was adopted to predict the release of the inert gases and semi-volatile halogen fission products to the fuel-clad gap.

It is generally accepted that the solid fission products (those having low volatility, such as Cs and Sr) are released at significantly lower rates.

The appropriate temperature to use in the GA correlation is the fuel temperature averaged over the irradiation history. The fuel can be characterized as having two separate temperature histories: the average temperature the fuel experienced during its steady state irradiation and the temperature the fuel may experience during the accident that is presumed to lead to a cladding rupture. To induce rupture, the fuel temperatures must equal or exceed the safety limits. The basis for defining the appropriate fuel temperature, and thus the release fraction, is given in Chapter 5 of Reference 4.8, as follows:

> "The release fraction for accident conditions is characteristic of the normal operating temperature, not the temperature during accident conditions. This is because the fission products released as a result of a fuel clad failure are those that have been collected in the fuel-clad gap during normal operation."

4.7 Thermal and Hydraulic Design

The thermal hydraulic portion of the MNRC relicensing study requires fuel element power values as well as hot channel intra-fuel rod power distributions in order to quantify the appropriate hot channel thermal hydraulic properties. MCNP5 was used to compute the relative contribution of each fuel element to the overall core power. Two unique core configurations were considered as a part of the neutronic and thermal hydraulic analysis with the most limiting core configuration being additionally considered both beginning and end of life states. In each core configuration and core life burnup state analyzed, the power per element was computed at a total core power level of 1.0 MW_{th}. The hottest fuel element (largest thermal power production) within the core provides the most limiting conditions for the thermal hydraulic study in each core configuration. This takes place in the I6 position for both the LCC and OCC Cores.

Additionally, the highest power element ("hottest element") in each of these three core states considered was further detailed to evaluate the intra-rod power distribution in both the radial and axial directions internal to the element.

Figure 4.41, Figure 4.42, and Figure 4.43 display the intra-fuel power distribution for the OCC Beginning of Life Core, LCC Beginning of Life Core, and LCC End of Life Core, respectively in the hot channel fuel element location. More fission occurs in the radial outer portion of the fuel element due to radial volume weighting (equations 17 through 19) and self-moderation; while the axial power is contributed primarily in the axial center of the fuel element.

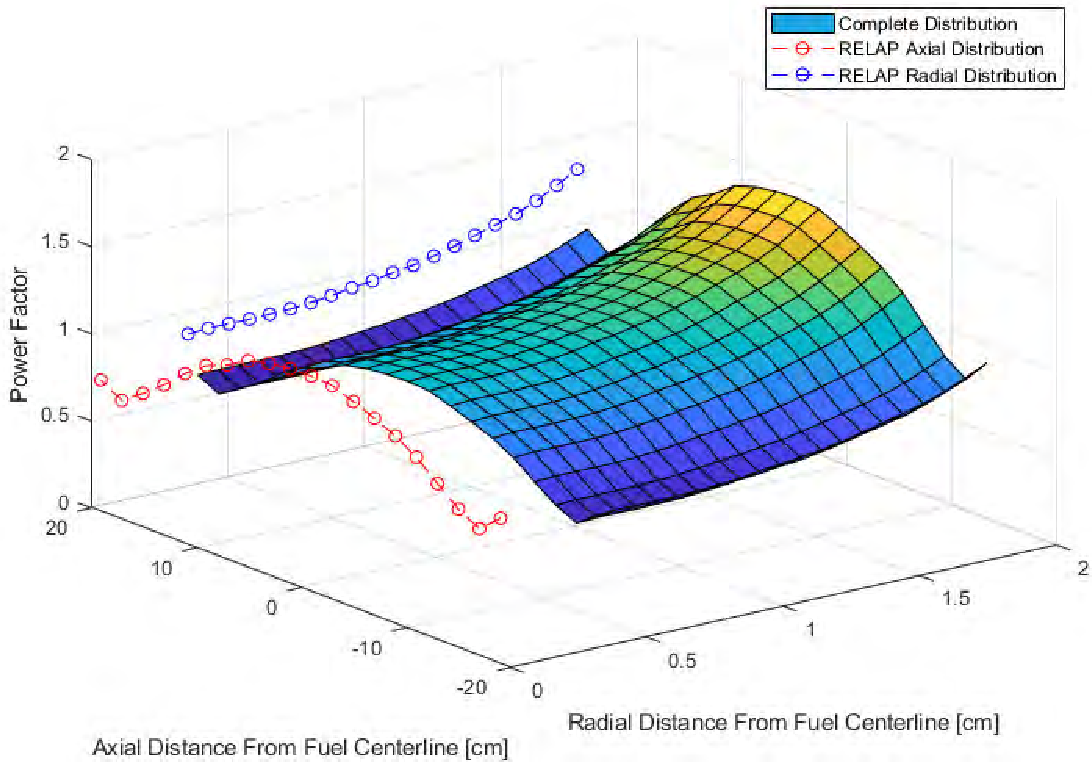

Figure 4.41: Hot element fuel power distribution (OCC-BOL Core)

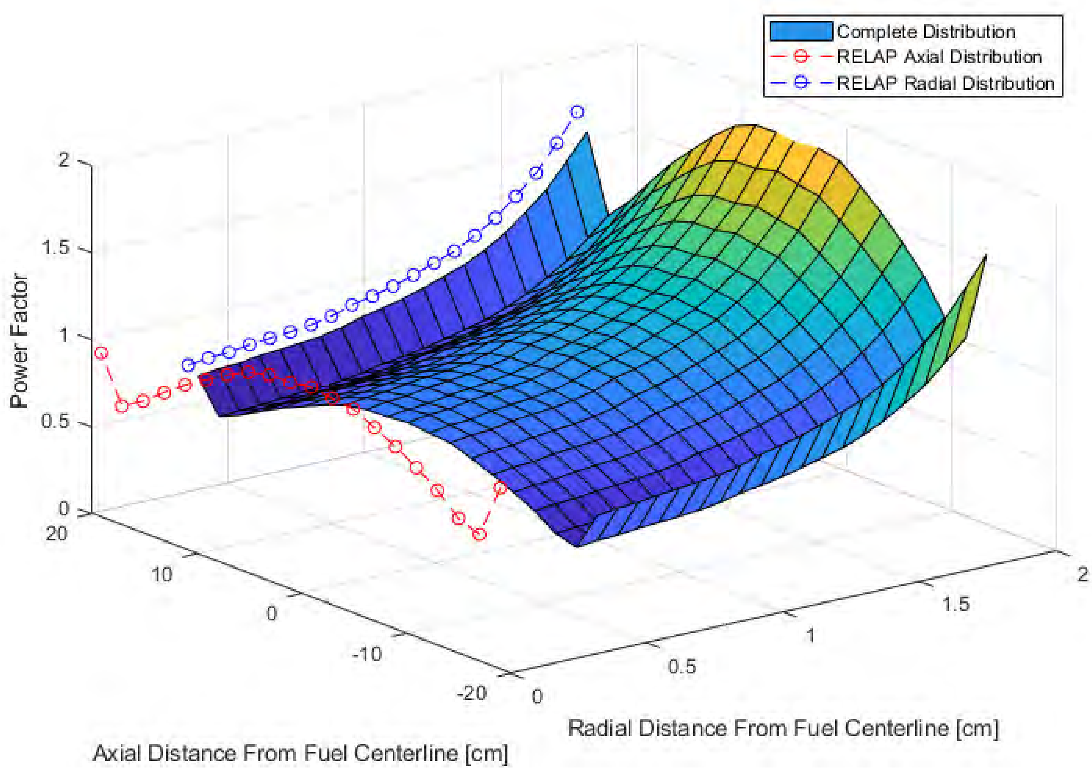

Figure 4.42: Hot element fuel power distribution (LCC-BOL Core)

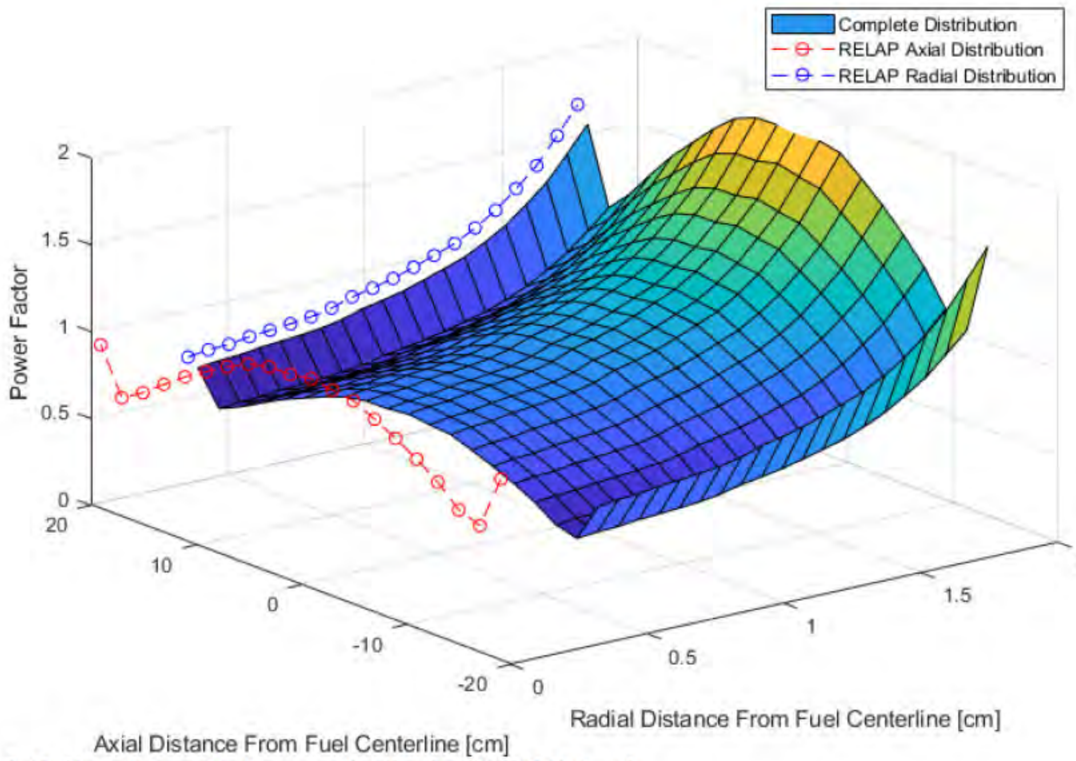

Figure 4.43: Hot element fuel power distribution (LCC-EOL Core)

In order to conduct the thermal hydraulic analysis using RELAP5-3D, the code requires a structured format as it relates to input fuel element or "heat structure" parameters. RELAP5-3D requires that two normalized vectors be identified in its equivalent fuel element. These vectors are located in the radial and axial direction of the fuel element and can be seen as the projected vectors detailed in figures 4.41 to 4.43, respectively.

For this thermal hydraulic analysis, it is required that the most bounding (limiting) conditions of the current (OCC) and LCC Cores be identified. The most limiting condition for the hot channel thermal power output can generally be summed up as the "Effective Hot Channel Peak Factor" presented in table 4.7. The Effective Hot Channel Peak Factor is the product of the Hot Channel Peak Factor, Hot Channel Fuel Axial Peak Factor, and Hot Channel Fuel Radial Peak Factor.

Table 4.7: Hot channel power summary

Hot Channel Power Summary						
Core Configuration	Hot Channel Location	Hot Channel Thermal Power [kW]	Hot Channel Peak Factor [P_{max}/P_{avg}]	Hot Channel Fuel Axial Peak Factor [P_{max}/P_{avg}]	Hot Channel Fuel Radial Peak Factor [P_{max}/P_{avg}]	Effective Hot Channel Peak Factor [P_{max}/P_{avg}]
OCC BOL Core	I6	14.81	1.511	1.230	1.314	2.442
LCC BOL Core	I6	17.69	1.804	1.218	1.681	3.694

LCC EOL Core	I6	17.59	1.794	1.218	1.667	3.643

The hot channel thermal power, axial and radial power profiles were obtained from the MCNP5 analysis. The hot channel power factor was obtained by taking the ratio of the hottest fuel element thermal power to the average fuel element thermal power in each core configuration. It is important to note that the hot channel thermal power found in Table 4. represents the thermal power after applying the hot channel peak factor. Similarly, the axial peak factor was obtained from referring to the axial power distribution procured from the MCNP5 analysis by taking the ratio of the hottest axial nodalized thermal power value to the average axial nodalized thermal power value. The radial peak factor was calculated by normalizing the thermal power in cylindrical coordinates as follows; all three power profiles are graphically presented below in Figure 4.44.

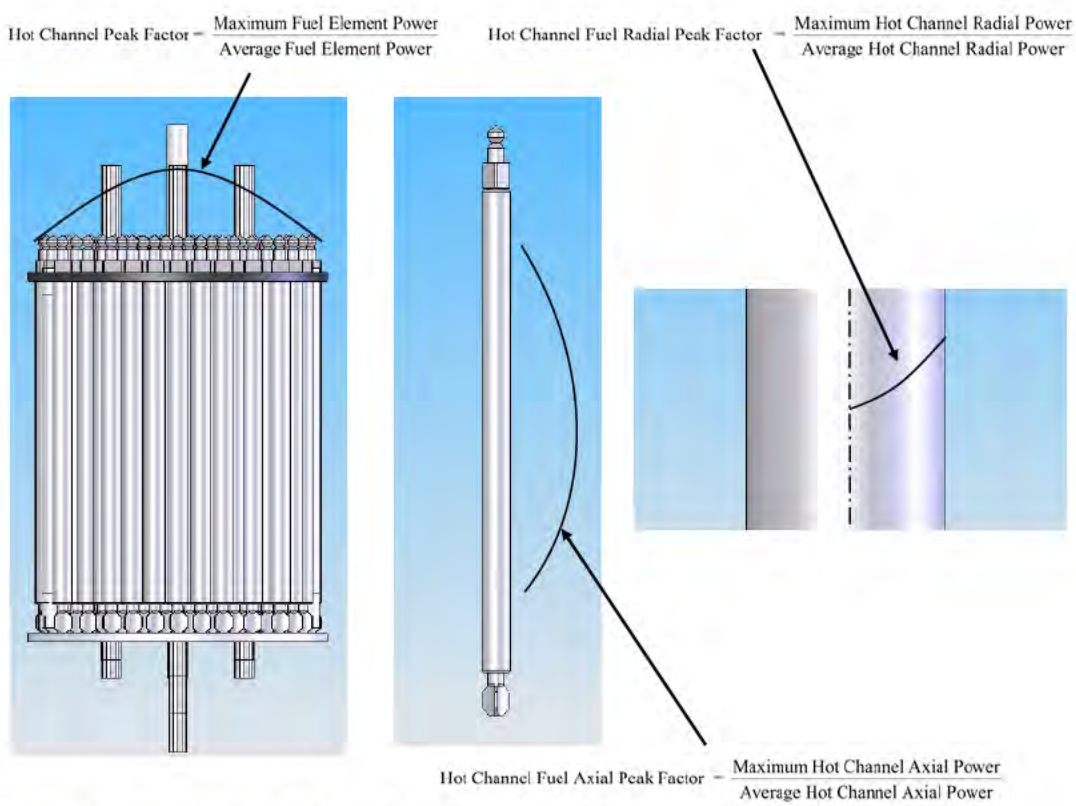

Figure 4.44 Core power, fuel element axial power, fuel element radial power profile

Consider the fission rate in a fuel element as *f(r,z)*, which is defined in cylindrical coordinates. The spatially average value of *f(r,z)* at some given axial position (z_o) is given as:

$$\bar{f}(r,z_o)=\frac{2\pi\int f(r,z_o)rdr}{2\pi\int rdr} \quad (17)$$

Where the integral ranges over the radial region of interest and 2π represents the azimuthal dependence of the function.

Now consider if *f*(*r,z*) is to be discretized at some given axial location z_o. It can be done in a way that preserves the integral of *f*(*r,*z_o) as follows:

$$f\left(r_i, z_o\right) = \frac{\int_{r_i}^{r_i+1} f\left(r_i, z_o\right) rdr}{\int_{r_i}^{r_i+1} rdr} \quad (18)$$

In the case of a discretized function *f*(r_i,z_o), we can change the discretized scheme by converting the integral in the numerator to a summation:

$$f\left(r_i, z_o\right) = \frac{\sum_i f\left(r_i, z_o\right) \pi \left(r_{i+1}^2 - r_i^2\right)}{\pi \left(r_{n+1}^2 - r_n^2\right)} \quad (19)$$

Where the r_n and r_{n+1} correspond to the transformed nodal discretization and the summation runs over all cells (full or partial) that exist in the original discretization. In the case of a partial cell, only that portion that lies within the transformed nodal space is considered. Twenty radial nodal locations were defined within the fuel meat when calculating the radial peak factor. The radial peak factor was calculated using the methodology found in equations (17) through (19) and applying them to equation (20).

$$PF\left(z_o\right) = \frac{f\left(r, z_o\right)_{\max}}{\bar{f}\left(r, z_o\right)} \quad (20)$$

4.7.1 Description of the RELAP5-3D Model

The development of a single RELAP5-3D model is presented herein This model is based on a single-channel analysis assumed to represent the hottest channel via combination of smallest hydraulic geometry and highest-power element in the core. Previous studies have demonstrated that a single channel analysis is limiting over multi-channel analysis approaches [4.50]. These studies have led to the successful licensing of numerous reactors including the Reed Research Reactor and the Oregon State TRIGA® Reactor.

4.7.1.1 Single Channel Model Description

The RELAP5-3D model seen in figure 4.45 consists of a coolant source, cold leg, horizontal connector, hot channel, and coolant sink. This model is representative of a single MNRC core hot channel.

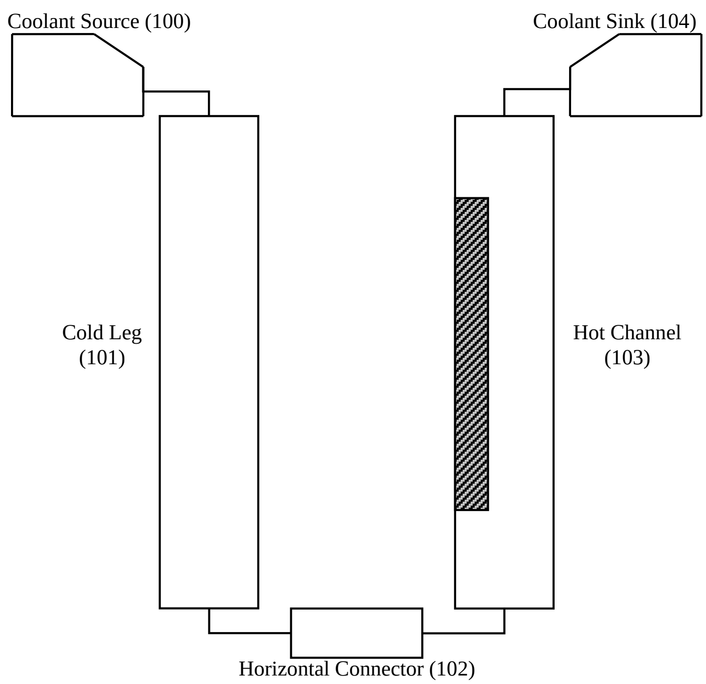

Figure 4.45: Single channel RELAP5-3D model schematic

The coolant source is modeled as a time dependent volume in RELAP5-3D allowing for an inlet pressure and temperature boundary condition to be imposed on the system during the analysis. The cold leg is incorporated into the RELAP5-3D model in order to create a pressure differential between the cold coolant entering the subchannel and the heated coolant passing through the subchannel. This drives the natural circulation flow. The horizontal connector serves no physical purpose in the MNRC, but is rather a nonphysical connector between the cold leg and hot channel to allow for communication between Volumes 101 and 103 during the computational process.

The hot channel (Volume 103) is the volume which contains the fuel element of a single MNRC subchannel. In the RELAP5-3D model that the hot channel has the most conservative thermal hydraulic parameters found in the MNRC core and that it is located in the B Ring.

Referring to Figure 4.46, the 24 nodes that occupy the hot channel volume incorporate all geometric thermal hydraulic parameters into its axial nodalized locations. Node 01 represents the volume which is occupied by the lower grid plate in the MNRC. Node 02 denotes the lower reflector comprised of graphite material in the lower portion of the MNRC fuel element. Nodes 03 through 22 are comprised of the fissionable fuel element material (U-ZrH), and constitute the RELAP5-3 heat structure. In these 20 axial nodal locations a modified cosine axial heat distribution has been applied, based on the results produced from the MCNP5 model. Node 23 is the upper reflector, similar to the lower reflector it is also comprised of graphite. Node 24 is the upper grid plate location in the hot channel volume.

The coolant sink (Volume 104) exactly parallels the geometric input parameters in the coolant source (Volume 100) time dependent volume. These Volumes are incorporated in the RELAP5-3D model in order to impose boundary conditions on the subchannel.

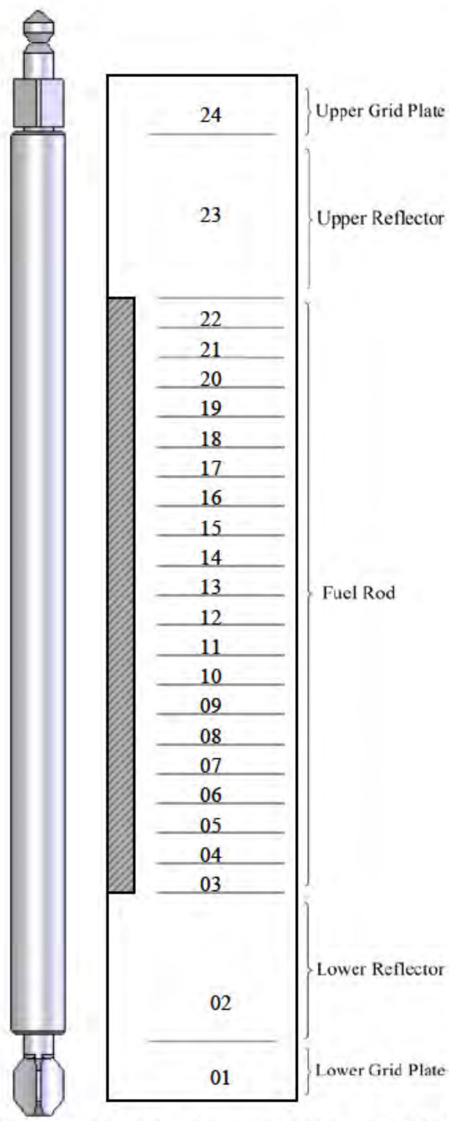

Figure 4.46: Comparison of MNRC fuel rod axial characteristics and RELAP5-3D subchannel

4.7.1.2 Coolant Source (100)

The coolant source models the coolant mass flux passing through the core hot channel in the MNRC. The coolant source is modeled in RELAP5-3D as a time dependent volume, which allows for temporal pressure and temperature boundary conditions to be imposed on the system. The coolant source is incorporated into the RELAP5-3D model for the sole purpose of implementing these two boundary conditions on the system.

By technical specifications the MNRC coolant may not exceed 45.0 °C (113 °F) in the reactor pool. In order to force the maximum potential conservatism in the solution a bounding inlet temperature of 45.0 °C (113 °F) is used.

A constant pressure of 1.01E5 Pa (14.7 psia) is assumed to exist at the top of the reactor pool. The MNRC's technical specification requires a minimum water column height above the top of the core to be 7.01 meters (23 feet), so this equivalent water column pressure boundary condition is used in the RELAP5-3D model for conservatism. RELAP5-3D requires that input pressure conditions be entered as absolute pressure, therefore the input RELAP5-3D pressure used in the model at the top of the core is 1.70068E5 Pa (24.6664 psia). These boundary input parameters summarized in Table 4.8. The geometric parameters of the coolant source are summarized in Table 4.9.

Table 4.8: Time dependent volume boundary conditions

Time Dependent Volume 100 & 104 Boundary Conditions	
Description	Value
Pressure [Pa] (psia)	1.70068E5 (24.6664)
Temperature °C (°F)	45.0 (113.0)

Table 4.9: Volume 100 & 104 geometric parameters

Volume 100 & 104 Geometric Parameters	
Description	Value
Flow Area [m^2] (in^2)	1.0 (1550.0)
Length [m] (in)	1.0 (39.37)
Surface Roughness [m] (in)	0.0 (0.0)
Orientation	Horizontal
Friction Calculation	Not included in volume

4.7.1.3 Cold Leg (101)

The cold leg is geometrically congruent to the hot channel; it connects the coolant source to the horizontal connector volume. This volume is included into the RELAP5-3D model in order to allow for the boundary conditions imposed by the coolant source to communicate correctly to the hot channel. The cold leg has the same height as the hot channel. This is important because it allows for an equilibration between the gravitational force imposed on the hot channel verses that imposed on the cold leg allowing for the sole method of cooling to be natural circulation. The geometric parameters of the cold leg volume are in summarized Table 4.10. For calculated justification of these values refer to Section 4.7.1.5.

Table 4.10: Volume 101 geometric parameters

Volume 101 Geometric Parameters	
Description	Value
Flow Area [m^2] (in^2)	3.304E-4 (5.12)
Length [m] (in)	0.711 (28.0)
Surface Roughness [m] (in)	0.0 (0.0)
Orientation	Vertical
Friction Calculation	Not included in volume

4.7.1.4 Horizontal Connector (102)

The horizontal connector is a non-physical component of the MNRC and is simply incorporated into the RELAP5-3D model to allow for computational communication between Volume 101 and 103.

The horizontal connector is horizontally oriented; therefore, no gravitational term is incorporated in the calculations within this volume, and thus it has no effect of the result produced in the hot channel. The geometric parameters for volume 102 are in Table 4.11. The flow area for the horizontal connector volume and cold leg volume do not affect the result produced in the hot channel, these volumes are input in the model to produce a pressure column of water that is equivalent to the hot channel pressure. This naturally imposes an equilibrium pressure boundary condition on the system. As a result of this boundary condition the only driving force for coolant flow through the hot channel is limited to natural circulation. All values in this volume are arbitrarily chosen.

Table 4.11: Volume 102 geometric parameters

Volume 102 Geometric Parameters	
Description	Value
Flow Area [m^2] (in^2)	1.0 (1550.0)
Length [m] (in)	1.0 (39.37)
Surface Roughness [m] (in)	0.0 (0.0)
Orientation	Horizontal
Friction Calculation	Not included in volume

4.7.1.5 Hot Channel (103)

Subchannel Flow Area: The B Ring pitch from fuel element centerline to centerline is 0.040568 meters, and it has a triangular subchannel geometry also containing the smallest subchannel flow area. It is for this reason that the subchannel flow area for the RELAP5-3D model is calculated with reference to the B Ring subchannel flow area.

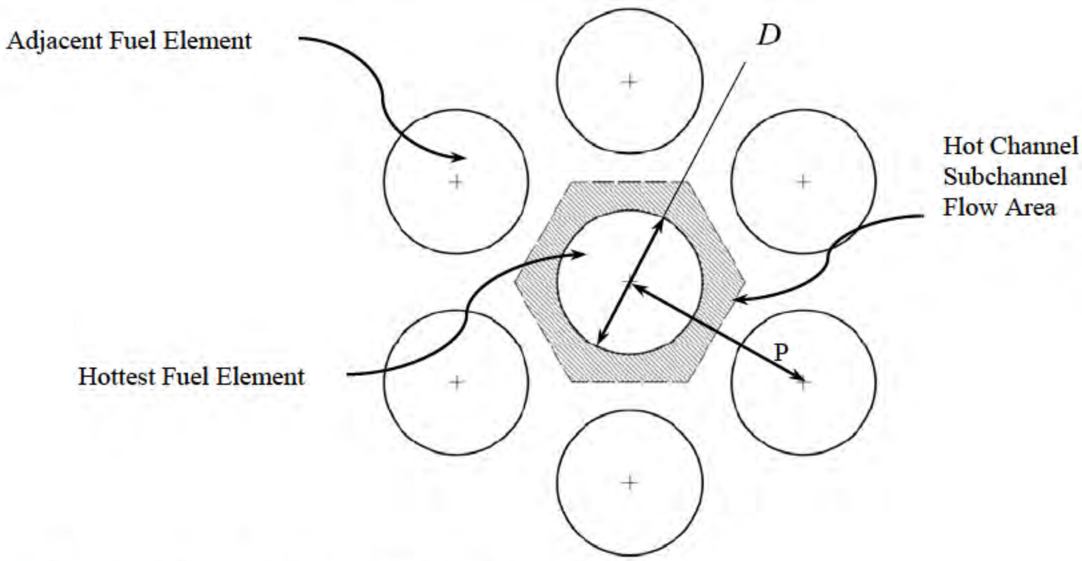

Figure 4.47: Hexagonal array axial average unit subchannel dimensions

The pitch for the B Ring subchannel is 0.041 meters (1.60 in) [4.46]. The fuel element outer diameter for all MNRC core fuel elements is defined as 0.037 meters (1.47 in). Equation (21) defines the subchannel flow area for a hexagonal array [4.51] as:

$$A_{fi} = \frac{\sqrt{3}}{4}P^2 - \frac{\pi D^2}{8} \quad (21)$$

where *P* represents fuel element pitch, and *D* represents the fuel element outer diameter as shown in Figure 4.47. From equation (21) the subchannel flow area is calculated to be $1.65 \cdot 10^{-4}$ meters^2 (0.26 in^2).

The wetted perimeter for the subchannel only occupies one half of an entire fuel element, therefore the total flow area for the subchannel input into the RELAP5-3D model is $3.30 \cdot 10^{-4}$ meters^2 (0.51 in^2).

Subchannel Length: Figure 4.46 compares an MNRC fuel element and the RELAP5-3D discretized subchannel volume. As previously discussed, Nodes 01 and 24 represent the lower and upper grid plates, both being 0.019 meters (0.75 in) thick; the upper grid plate is located exactly 0.67 meters (26.5 in) above the lower grid plate [4.52].

The total axial length in the fuel elements (lower graphite, fuel, upper graphite) is 0.56 meters (21.91 in). Because the sum of these defined segments is not equivalent to the required 0.673 meters (26.5 in) to span between the upper and lower grid plates and the lower graphite segment is approximately $7.62 \cdot 10^{-3}$ meters (0.03 in) longer than the upper graphite segment it is assumed that the fuel portion of the fuel element is skewed up from the axial center of the core by $7.62 \cdot 10^{-3}$ meters (0.03 in). Therefore, the equation to calculate the length of Node 02 (lower unheated node) is:

$$L_{02} = \frac{\left(0.6731 - L_{fuel} - L_{Upper,graphite} - L_{Lower,graphite}\right)}{2} + L_{Lower,graphite} \quad (22)$$

L_{02} is the length of Node 02 and $L_{Upper,graphite}$ and $L_{Lower,graphite}$ are the upper and lower graphite lengths of the fuel element.

The fuel nodal lengths must be discretized accordingly and can be done by referring to the following equation for Nodes 03 through 22:

$$L_{03 \to 22} = \frac{L_{fuel}}{n_u - \left(n_l - 1\right)} \quad (23)$$

$L_{03 \to 22}$ refers to the nodal length for Nodes 03 through 22, n_u is the larger nodal number, n_l is the smaller nodal number.

To calculate the nodal length for Node 23 (upper unheated node) the following equation is used:

$$L_{23} = \frac{\left(0.6731 - L_{fuel} - L_{Upper,graphite} - L_{Lower,graphite}\right)}{2} + L_{Upper,graphite} \quad (24)$$

From equations (22), (23), and (24), Table 4.12 is formulated. It is important to note that the nodal lengths stated in Table 4.12 apply to Volume 103 only.

Table 4.12: Core volume nodal lengths

Core Volume Axial Nodal Lengths		
Nodal Description	Node Number	Nodal Length [m] (in)
Upper Grid Plate	24	0.01905 (0.75000)
Upper Graphite	23	0.14567 (5.73504)
Fuel	22	0.01905 (0.75000)
	21	0.01905 (0.75000)
	20	0.01905 (0.75000)
	19	0.01905 (0.75000)
	18	0.01905 (0.75000)
	17	0.01905 (0.75000)

	16	0.01905 (0.75000)
	15	0.01905 (0.75000)
	14	0.01905 (0.75000)
	13	0.01905 (0.75000)
	12	0.01905 (0.75000)
	11	0.01905 (0.75000)
	10	0.01905 (0.75000)
	09	0.01905 (0.75000)
	08	0.01905 (0.75000)
	07	0.01905 (0.75000)
	06	0.01905 (0.75000)
	05	0.01905 (0.75000)
	04	0.01905 (0.75000)
	03	0.01905 (0.75000)
Lower Graphite	02	0.14643 (5.76504)
Lower Grid Plate	01	0.01905 (0.75000)

Material Roughness: A value of 2.134E-6 meters (8.4E-5 inches) has been estimated for the fuel clad roughness due to its wide range of possible fabrication methods, it is also the most likely roughness given the fabrication methods used for this particular application [4.53].

Form Loss: The inlet and outlet form loss coefficients represent the form losses of the rod bottom and top rod fixtures. The form loss coefficient is a localized geometric parameter that quantifies fluid flow resistance due to a local change in geometry. Form loss is a dimensionless parameter and is given as [10]:

$$K = \frac{2\Delta P}{\rho v^2} \quad (25)$$

Past studies have been performed toward quantifying TRIGA® core form losses of different lattice configurations, the results from these studies represented large variance in their results, and were not able to correlate a definite form loss value for each lattice configuration [4.55]. Previous studies performed by TRIGA® Reactors during license efforts developed and presented a methodology for calculating each effective subchannel form loss rather than local form losses within the core [4.48]. These coefficients as well as a summary of the thermal hydraulic parameters found in the MNRC core hot channel are presented in Table 4.. A brief description of how these form losses were calculated is presented herein.

Equation (24) is not easily quantified with reference to the MNRC lower and upper grid plate geometry, for this reason it is assumed that the form losses are comprised of either sudden expansions or sudden contractions equations (25) and (26).

$$K_{SE} = \left(1 - \frac{d^2}{D^2}\right)^2 \quad (26)$$

$$K_{SC} \approx 0.42\left(1-\frac{d^2}{D^2}\right) \quad (27)$$

Sudden expansion form losses typically result in a form loss less than a value of 1.0. For low flow, low pressure systems the resulting form loss is approximately 0.3. A visual of a sudden expansion is presented in figure 4.48(a).

Sudden contraction form losses are nominally ~1.3 at low pressures and low flow rates (i.e. natural circulation) [4.56]. A visual of a sudden contraction is presented in Figure 4.1 (b).

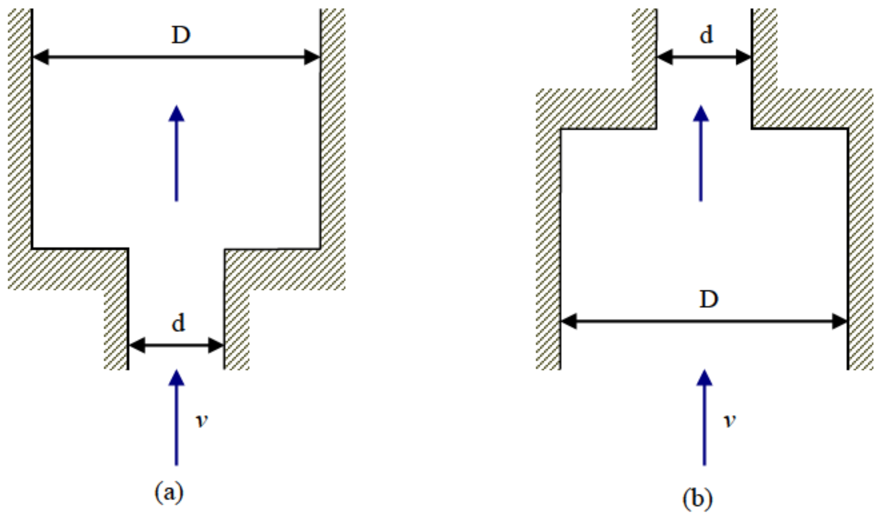

Figure 4.48: Sudden expansion (a) and sudden contraction (b)

Table 4.13: Single channel model geometric thermal hydraulic properties summary

Parameter	Value
Flow area [m^2]	3.80E-4
Fuel Element Pitch [m]	0.04064
Wetted perimeter [m]	0.117
Hydraulic diameter [m]	1.301E-2
Heated diameter [m]	3.734E-2
Fuel element heated length [m]	0.381
Fuel element surface area [m^2]	4.469E-2
Fuel element surface roughness [m]	2.134E-6
Inlet pressure loss coefficient	2.26
Exit pressure loss coefficient	0.63
Absolute pressure at the top of the core [Pa]	1.70068E5

4.7.1.6 Coolant Sink (104)

The coolant sink models the coolant mass flux leaving the hot channel in the MNRC. The coolant sink is modeled in RELAP5-3D as a time dependent volume and is identical in geometry and orientation to the coolant source.

The initial outlet coolant temperature and absolute pressure are defined respectively as 45.0 °C (113.0 °F) and 1.70068E5 Pa (24.6664 psia). These boundary input parameters can be seen in Table . The geometric parameters for volume 104 are presented in Table 4.. It is important to note that the boundary conditions in the coolant sink do not convect back into the solution domain.

General Volume Parameters: The RELAP5-3D model, excluding core volumes, contains filler volumes (i.e. non-physical volumes that are required for the model to function correctly). These volumes, including the coolant source, cold leg, horizontal connector, and coolant sink, contain parameters that are similar to one another throughout the system and do not influence the hot channel solution.

Material Roughness: All volumes, excluding the hot channel volume, neglect material roughness calculations entirely as they are non-physical geometries in the RELAP5-3D model.

Form Loss: All volumes, excluding the hot channel volume, neglect form loss calculations entirely as they are non-physical geometries in the RELAP5-3D model.

4.7.2 Hot Channel Heat Structure

All heat generation takes place within the heat structure. This heat structure is defined by material composition, heat transfer surface area, geometric orientation, and power density. A heat structure cannot be implemented in a system without tying it to a subsequent hydraulic volume; volumes produce the boundary conditions that allow heat structures to complete their calculations correctly. The MNRC RELAP5-3D model is comprised of a single heat structure volume (hot channel fuel element). This volume represents the core power generation of the hot channel in the MNRC.

The MCNP5 core neutronic analysis identified that the hot channel was located in the I6 fuel element location for the MNRC at 1.0 MW_{th} in the LCC core and I6 fuel element location in the OCC core. A power table was developed in the RELAP5-3D model to simulate the thermal output of the core.

All thermal hydraulic results are dependent on the power distribution found in the single channel heat structure (301). From the same neutronic analysis the axial fuel element power distribution and radial fuel element power distribution were calculated.

As mentioned, the hot channel contains a single heat structure (301). Figure 4.14 represents the hot channel heat structure and the subsequent nodes that it parallels. Two core configurations were analyzed during this project for the MNRC: OCC and LCC cores. The development of these different thermal models is described in the following sections.

4.7.2.1 Heat Structure Discretization

Heat structures are two dimensional elements in RELAP5-3D. Therefore, both a radial and axial power profile must be input into the RELAP5-3D model. The neutronic analysis conducted in MCNP5 resulted in an output of thermal power for each fuel element in the MNRC for a core thermal output of 1.0 MW_{th}.

A single fuel element is comprised of a pure zirconium pin located in the radial center of a fuel element, U-ZrH fuel press fit around the zirconium pin, a small hydrogen filled gap created by contact resistance, and finally a stainless steel cladding as the outer casing for the fuel element, this configuration are presented in Figure 4.49.

The fuel to clad contact gap that is created by material surface roughness is originally hydrided during manufacturing of TRIGA® fuel. As the U-ZrH fuel is burnt through its lifetime fission product gasses are released and migrate from the fuel lattice structure into the gap. These inert fission product gasses are an order of magnitude more heat resistant than the hydrogen that originally filled the gap. As a result of this increase in resistance the fuel temperature increases. It is therefore assumed that these fission product gasses are found in the fuel to clad gap during the study in order maintain a conservative approach to the thermal hydraulic safety analysis.

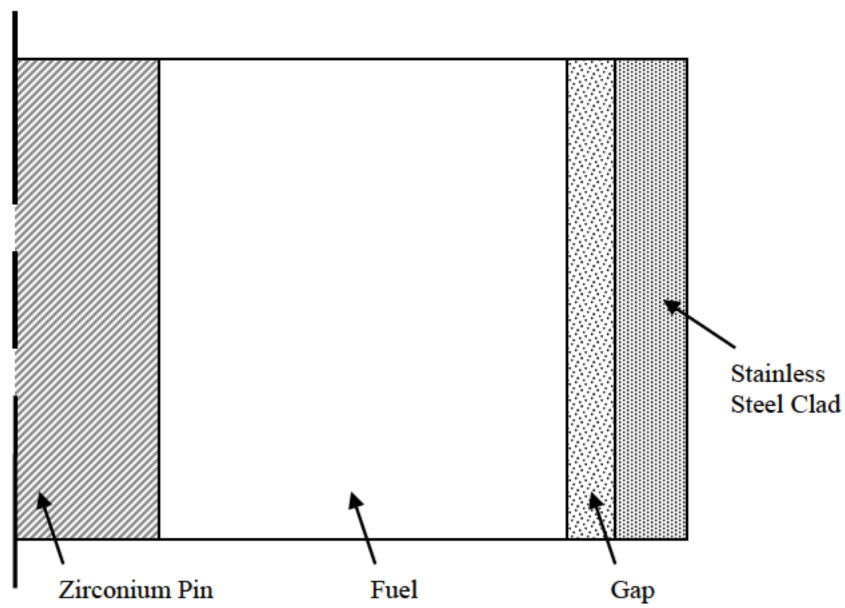

Figure 4.49: Cross sectional view of fuel element

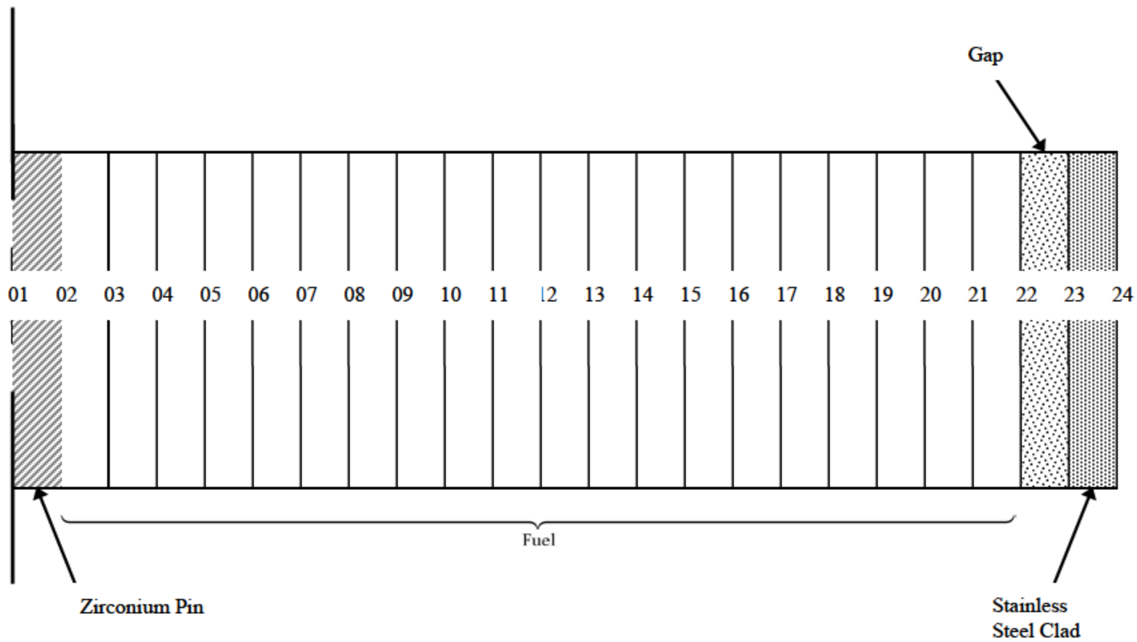

Figure 4.50: Radial nodal breakdown of fuel element

Because fission only occurs in the fuel region, is the only radial region where a power profile must be defined in the RELAP5-3D model. The mesh points within the fuel region used in the RELAP5-3D model correspond to the results produced at given coordinate points in the MCNP5 analysis. As seen in Table 4.50, the radial distance for each material is identified. The outer gap coordinate (Node 22) is varied during this study.

The fuel to clad gap is due to surface roughness, and this roughens has the potential to vary due to the fuel manufacturing process. Because the gap has the highest thermal resistance found in a TRIGA® fuel element and all heat removal is radially conducted through the gap into the cladding and finally out to its ultimate heat sink (bulk coolant) it is important to acquire the accurate thickness of the gap so that the correct thermal resistance is used. Several references for TRIGA® fuel identify that the fuel to clad gap can vary from 0.05 to 0.4 mils [4.58, 4.59]. The effect of varying the fuel to clad gap was investigated using a RELAP5-3D Model in previous studies [4.48].

Table 4.14: Heat structure radial node lengths

Heat Structure Radial Node Lengths		
Nodal Description	Node Number	Node Coordinate [m] (in)
Outer Zirconium Pin	01	0.00000 (0.00000)
	02	0.00318 (0.12500)
Fuel	03	0.00355 (0.13976)
	04	0.00430 (0.16929)
	05	0.00506 (0.19921)
	06	0.00581 (0.22874)
	07	0.00656 (0.25827)
	08	0.00731 (0.28779)
	09	0.00807 (0.31772)
	10	0.00882 (0.34724)

	11	0.00957 (0.37677)
	12	0.01032 (0.40630)
	13	0.01108 (0.43622)
	14	0.01183 (0.46575)
	15	0.01258 (0.49527)
	16	0.01333 (0.52480)
	17	0.01409 (0.55472)
	18	0.01484 (0.58425)
	19	0.01559 (0.61378)
	20	0.01634 (0.64331)
	21	0.01710 (0.67323)
	22	0.01785 (0.70275)
Outer Gap	23	0.01785-0.01786 (0.70285-0.70305)
Outer Stainless Steel Clad	24	0.01873 (0.73750)

4.7.2.2 Heat Structure Thermo-Physical Properties

Data from thermal diffusivity measurements taken by General Atomics along with the best available data for density and specific heat showed that the thermal conductivity is both independent of temperature and uranium content and can be seen below in equation (28) [4.58].

$$K(T) = 0.18 \pm 0.009 \ \ [\mathrm{W/cm\text{-}^{\circ}C}] \quad (28)$$

TRIGA® FLIP fuel has a defined volumetric heat capacity as presented in equation (29) [4.57, 4.59].

$$\rho C_p(T) = 2.04 + 4.17\mathrm{E\text{-}3}(T) \ \ [\mathrm{W\text{-}sec/cm^3\text{-}^{\circ}C}] \ (29)$$

4.7.3 Steady State Results

The TRIGA® system operating with cooling provided by natural convection water flow around the fuel elements was analysed. The predicted steady state thermal-hydraulic performance of the MNRC OCC and LCC core configurations is determined for the reactor operating at 1.0 MW_{th} with a water inlet temperature of 45°C. Per the Technical Specifications, the maximum pool temperature is 45°C. The maximum power fuel rod and maximum power heated subchannel were analysed under steady-state. The RELAP5-3D computer code [4.60] was used to determine the natural convection flow rate, fuel centerline temperature profile, clad temperature profile, axial temperature profile and radial fuel temperature distribution. The power in the hottest rod at which critical heat flux is predicted to occur was calculated with the aid of the RELAP5-3D code. The code was used to calculate coolant flow rate as a function of rod power. Groeneveld 1986, 1995, and 2006 [4.62] critical heat flux tables were used as the primary means for predicting margin to departure from nucleate boiling with the he Bernath correlation [4.61] provided as a qualitative reference for historic purposes. These two methods were used to provide independent and diverse approaches for predicting the transition from nuclear boiling to transition or film boiling over others as a result of numerous previous studies who have demonstrated the credible application of these correlations over others as well as the developed boundedness within the operating conditions of TRIGA® reactors [4.48, 4.63]. A recent study demonstrated the relevance of these methods through empirical comparison under representative TRIGA® conditions [4.64].

The predicted parameters produced from this code for steady state operation include: channel flow rate, axial fuel centerline temperature distribution, axial clad temperature distribution, axial bulk coolant temperature distribution and axial DNBR. To simplify the RELAP5-3D model, it was assumed that there is no cross flow between adjacent channels. This assumption is conservative since higher values of temperature and lower margins to DNB are predicted when cross flow between adjacent channels is ignored.

In previous studies led by the author one, two, and eight channel RELAP5-3D models were individually analyzed for a common configuration and compared against one another. The outcome of this exercise led the authors to objectively demonstrate the similar results with slightly conservative outcomes produced by the one-channel model [4.48, 4.49]. Therefore, as a result, the one-channel model was used herein to produce those thermal hydraulic results supporting the safety analysis. Cross flow was incorporated in the two and eight channel models through junctions connected at each individual axial nodal location between adjacent subchannels. The axial and radial fuel temperature distributions were assumed to be identical in each model.

4.7.3.1 OCC BOL Core

The OCC core contains fuel elements that are geometrically similar and therefore the hot channel geometric parameters (i.e. hydraulic diameter, length, etc.) do not change. The hot channel power summary in terms of parameters and results are given in Table 4.15 and Table 4.16, respectively.

Figure 4.52 through Figure 4.55 graphically illustrate the results of the analysis on the OCC core.

Each parameter in Figure 4.52 is taken at a different elevation in the hot subchannel. The fuel centerline temperature is shown at the axial nodal location which produces the maximum fuel centerline temperature (fuel axial center). The outer cladding temperature is shown at the axial location which produces the maximum outer cladding temperature (slightly above the axial fuel centerline). The bulk coolant temperature is shown at the location which produces the maximum bulk coolant temperature (highest vertical subchannel node). Coolant mass flux is shown as the mass flux which corresponds to the maximum bulk coolant temperature (highest vertical subchannel node). Each location was selected to show the most limiting value of the associated parameter.

Table 4.15: Steady state results for the OCC BOL Core at 1.0 MW_{th}

Parameter		Value
Flow rate for hottest rod [kg/s]		0.0729
Maximum wall heat flux [kW/m^2]		410.00
Maximum fuel centerline temperature [°C]		401.16
Maximum clad temperature [°C]		131.81
Exit clad temperature [°C]		129.14
Exit bulk coolant temperature [°C]		93.62
MDNBR	Groeneveld 1986	8.24
	Groeneveld 1995	6.95
	Groeneveld 2006	6.46
	Bernath	3.16

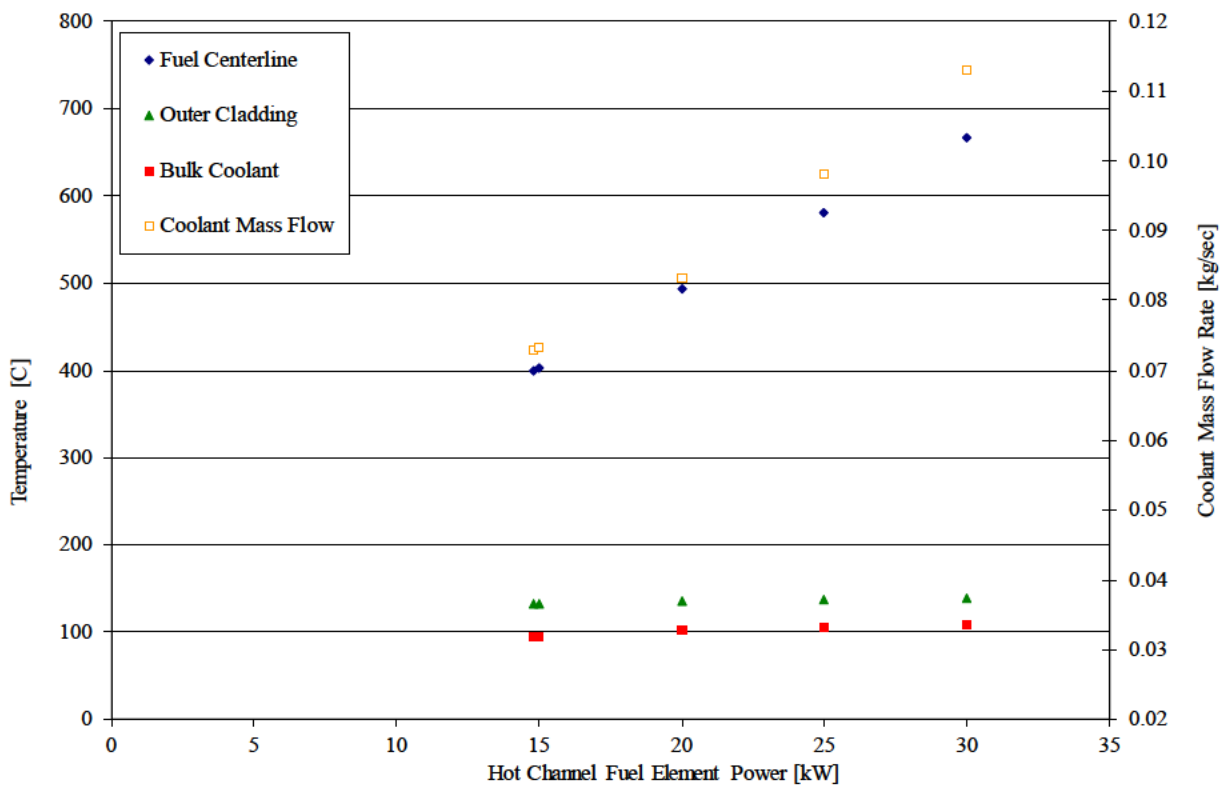

Figure 4.51: Hot channel characteristics (OCC BOL Core)

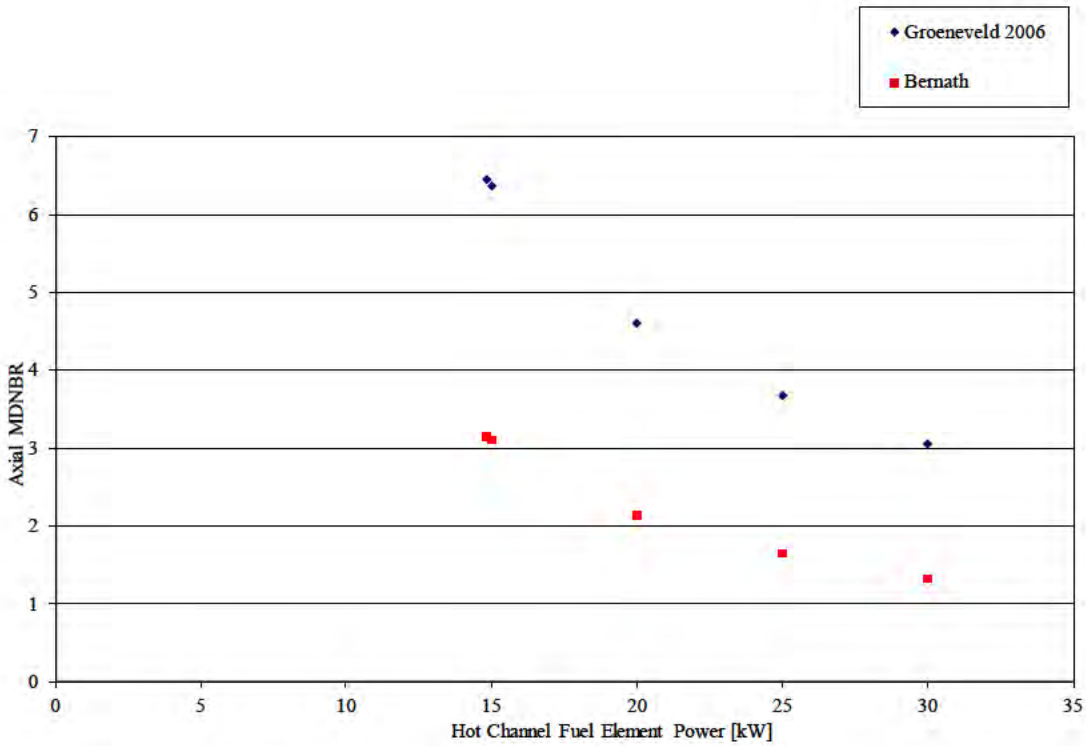

Figure 4.52 Hot channel MDNBR (OCC BOL core)

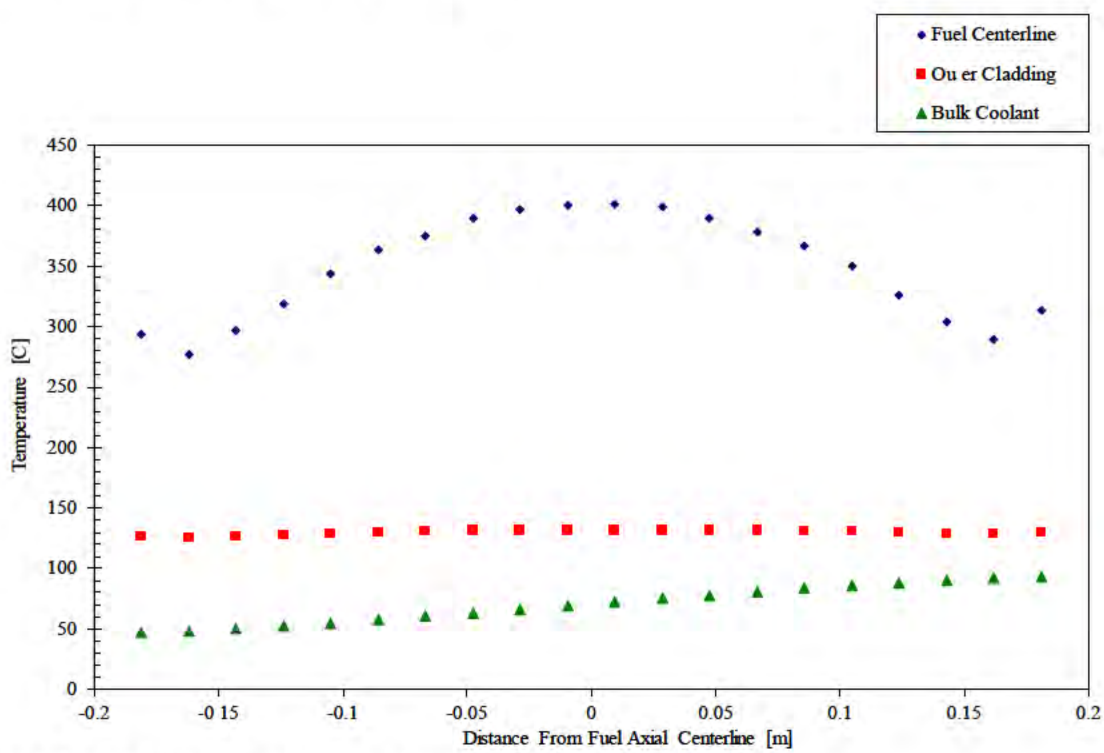

Figure 4.51: Axial temperature distribution at 14.81 kW_{th} (OCC BOL Core)

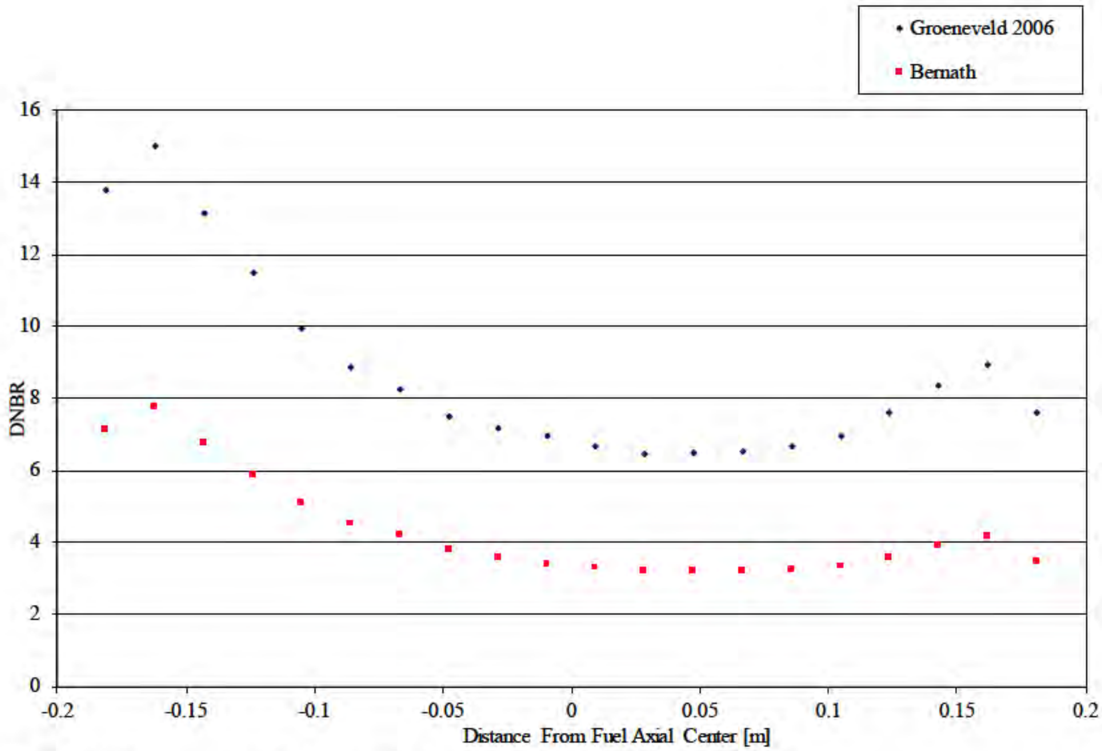

Figure 4.52: Hot channel axial DNBR at 14.81 kW$_{th}$ (OCC BOL Core)

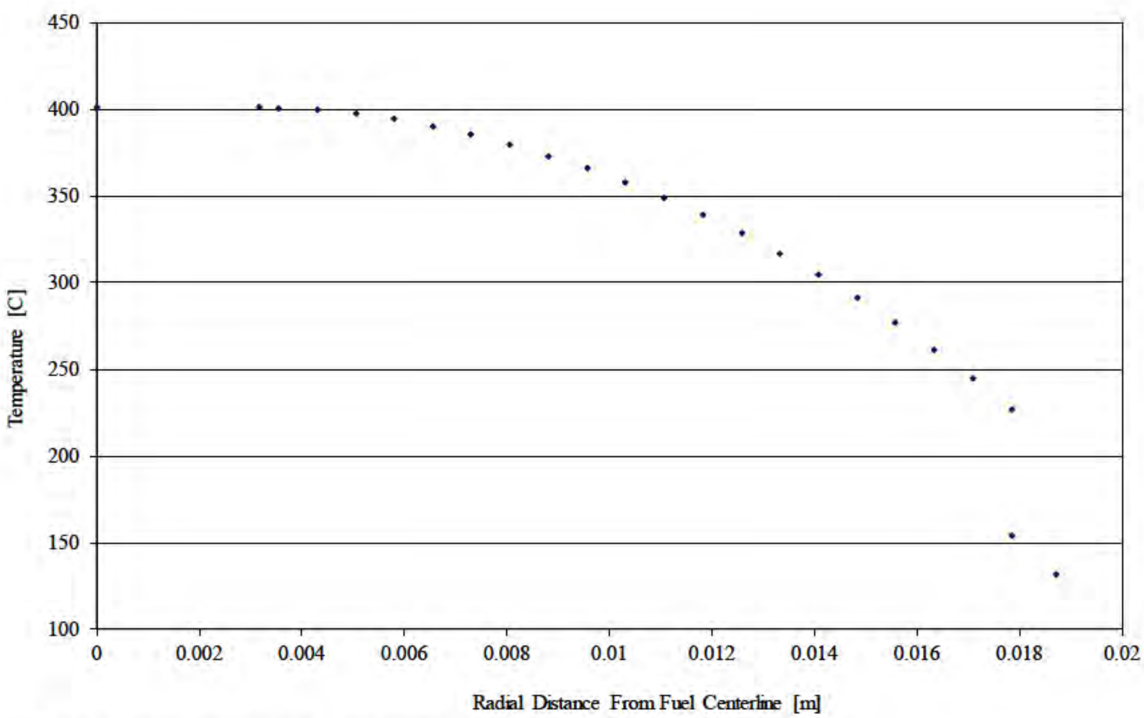

Figure 4.53: Radial temperature distribution at 14.81 kW$_{th}$ (OCC BOL Core)

Table 4.16: Calculated fuel temperatures for various channel powers (OCC BOL Core)

$P_{hot\text{-}channel}$ (kW)	Calculated [°C]			
	T_{max}	$T_{0.3}$	T_{clad}	$T_{coolant}$
14.81	399.97	384.38	131.66	93.62
15.00	403.43	387.63	131.79	94.00
20.00	493.11	472.05	134.70	102.34
25.00	580.91	554.58	137.00	105.69
30.00	667.43	635.84	139.05	108.15

The OCC BOL Core steady state results shown above provide a succinct summary of the thermal safety related characteristics while operating at the license limit power, lowest water level, and hottest water temperature in the hot-channel (most limiting location within the core). Figure 4.5 shows that the MDNBR in the hot channel will reach a value of 2.00 at approximately 20.5 kW_{th} hot channel steady state power. This is 138.4 % of the 14.81 KW_{th} produced in the hot channel of the OCC BOL Core operating at 1.0 MW_{th}. Using either the Bernath or the Groeneveld 2006 correlations, the OCC BOL Core is operating at power well below that required for departure from nucleate boiling.

4.7.3.2 LCC BOL Core

The LCC BOL Core contain fuel elements that are geometrically similar and therefore the hot channel geometric parameters (i.e. hydraulic diameter, length, etc.) do not change. The hot channel power summary in terms of parameters and results are given in Table 4.17 and Table 4.18, respectively.

Figure 4.56 through Figure 4.60 graphically illustrate the results of the analysis on the LCC BOL Core.

Each parameter in Figure 4.56 is taken at a different elevation in the hot subchannel. The fuel centerline temperature is shown at the axial nodal location which produces the maximum fuel centerline temperature (fuel axial center). The outer cladding temperature is shown at the axial location which produces the maximum outer cladding temperature (slightly above the axial fuel centerline). The bulk coolant temperature is shown at the location which produces the maximum bulk coolant temperature (highest vertical subchannel node). Coolant mass flux is shown as the mass flux which corresponds to the maximum bulk coolant temperature (highest vertical subchannel node). Each location was selected to show the most limiting value of the associated parameter.

Table 4.17: Steady state results for the LCC BOL Core at 1.0 MW_{th}

Parameter	Value
Flow rate for hottest rod [kg/s]	0.07862
Maximum wall heat flux [kW/m^2]	479.00
Maximum fuel centerline temperature [°C]	420.24
Maximum clad temperature [°C]	133.20
Exit clad temperature [°C]	132.61
Exit bulk coolant temperature [°C]	98.96

MDNBR	Groeneveld 1986	6.46
	Groeneveld 1995	5.15
	Groeneveld 2006	4.90
	Bernath	2.12

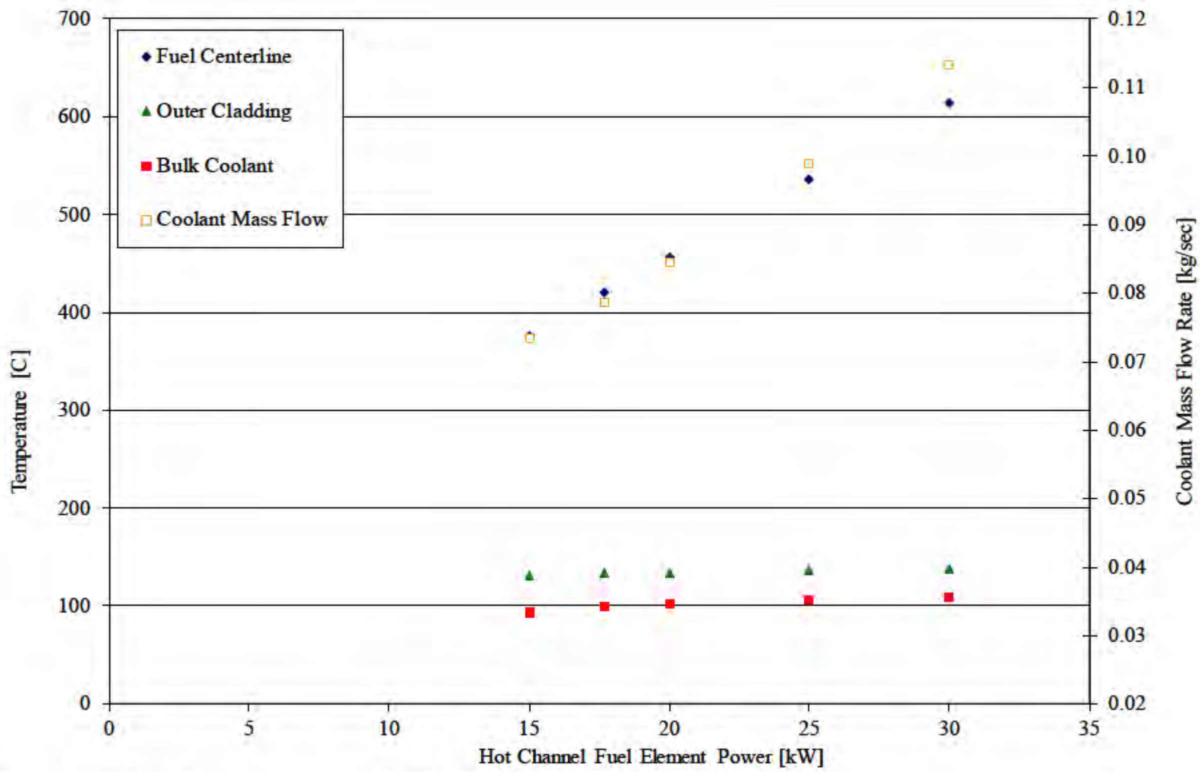

Figure 4.54: Hot channel properties (LCC BOL Core)

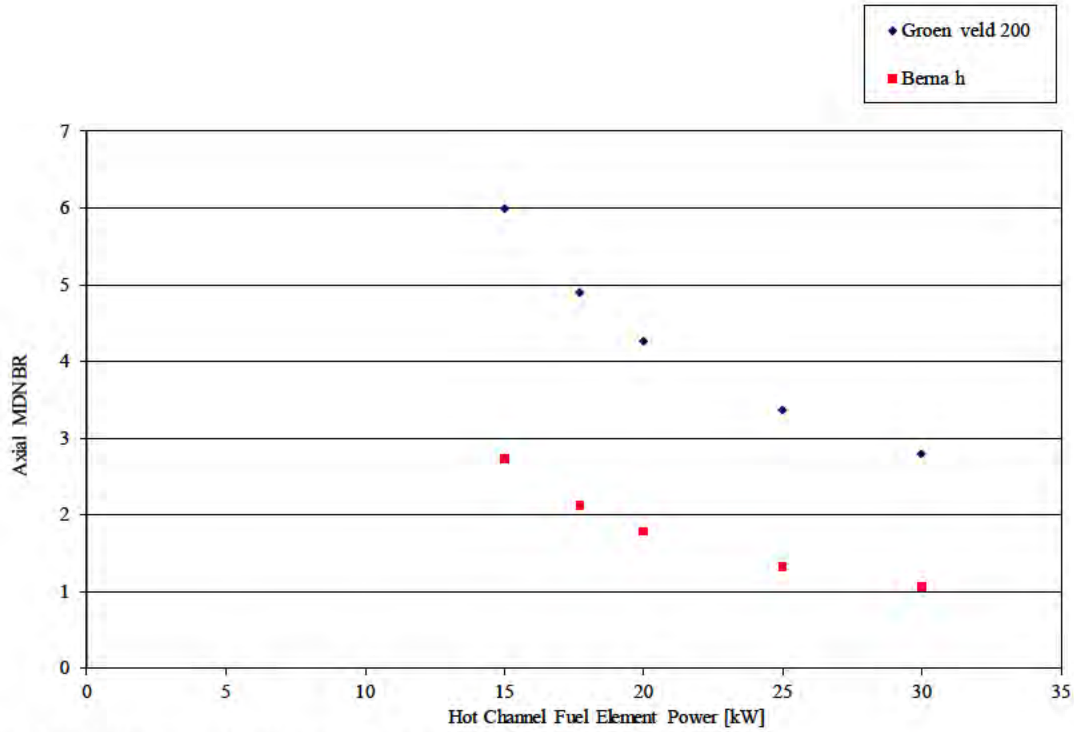

Figure 4.55: Hot channel MDNBR (LCC BOL Core)

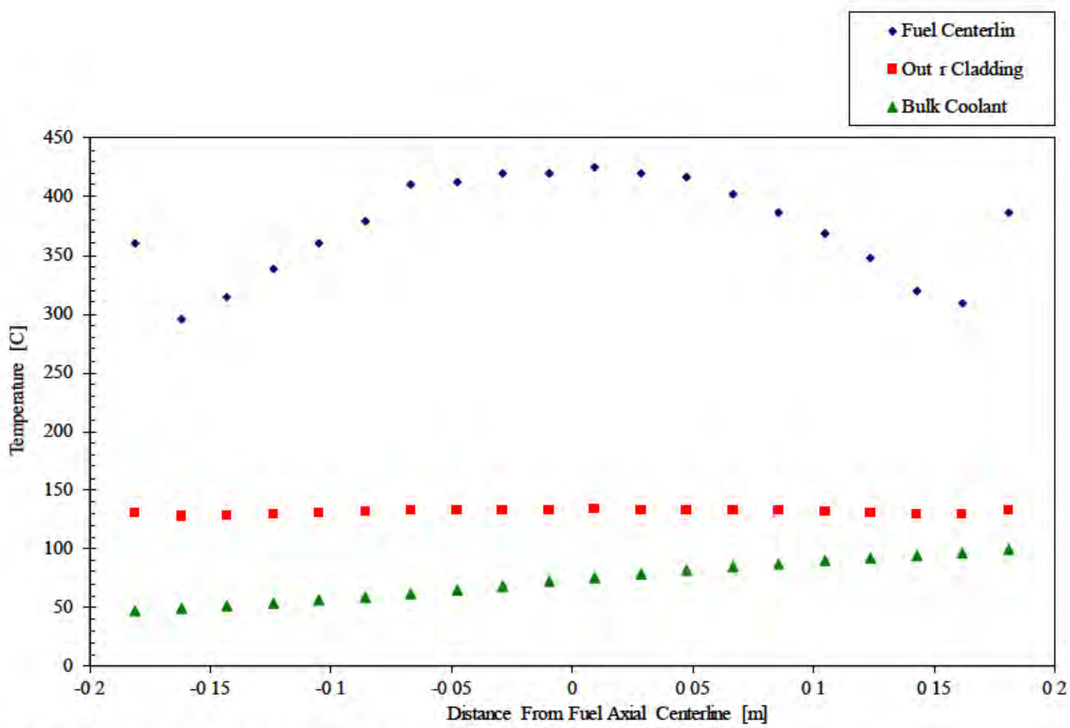

Figure 4.56: Axial temperature distribution at 17.69 kW_{th} (LCC BOL Core)

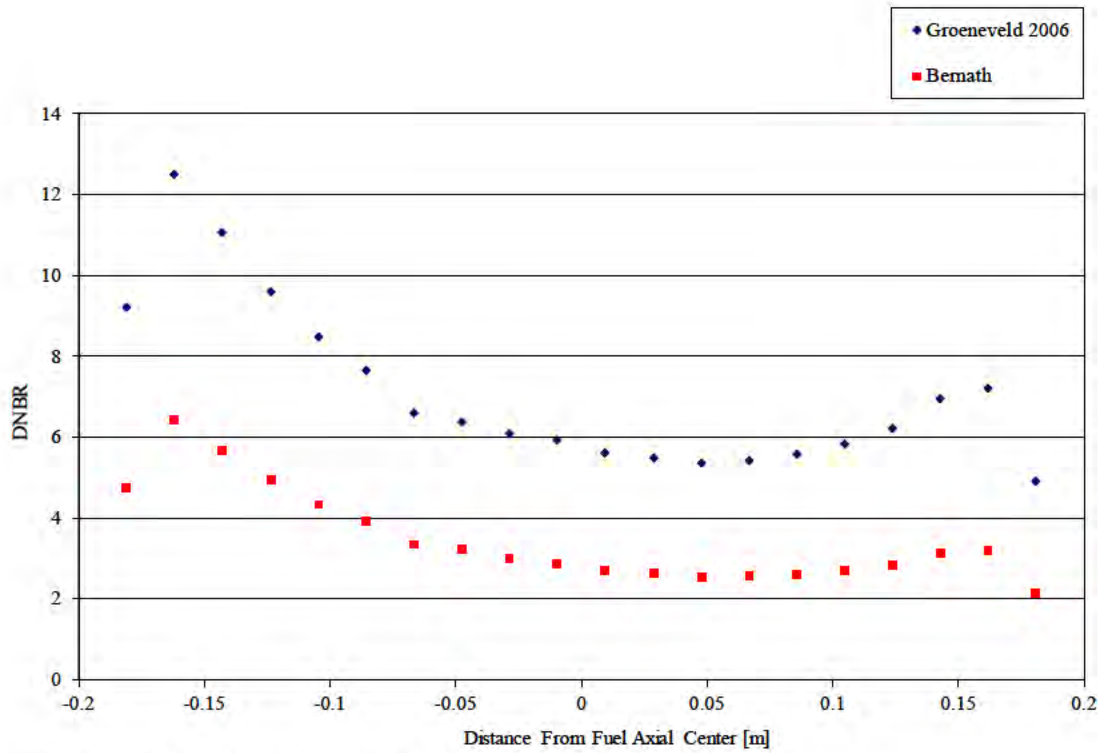

Figure 4.57: Hot channel axial DNBR at 17.69 kW$_{th}$ (LCC BOL Core)

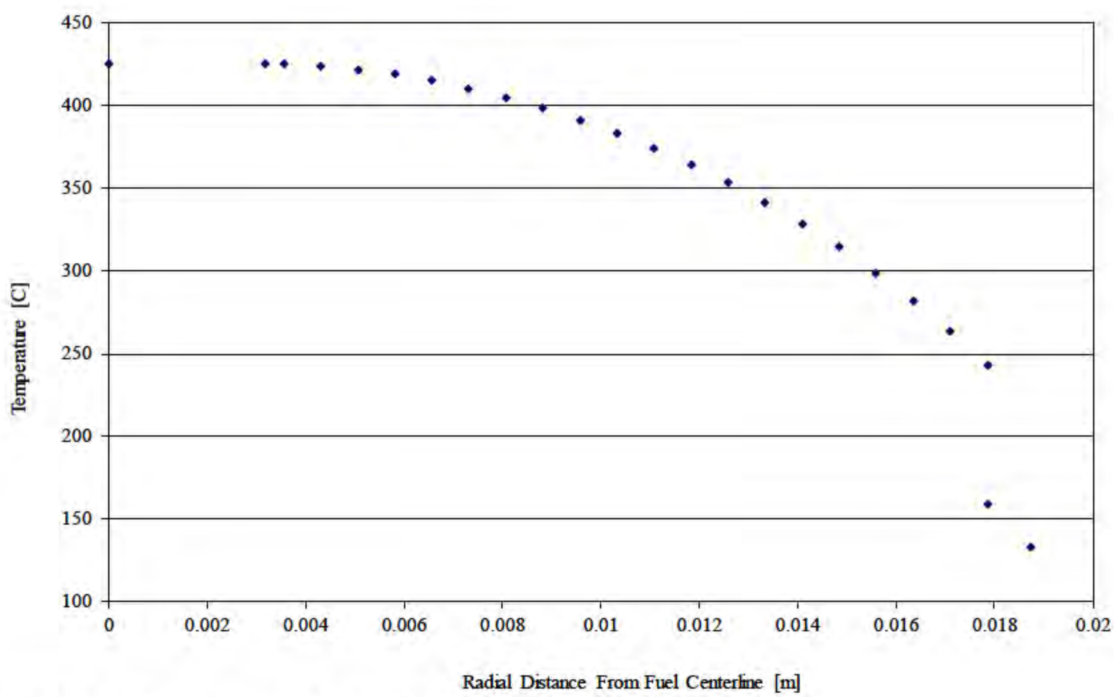

Figure 4.8: Radial temperature distribution at 17.69 kW$_{th}$ (LCC BOL Core)

Table 4.18: Calculated fuel temperatures for various channel powers (LCC BOL Core)

$P_{hot\text{-}channel}$ (kW)	Calculated [°C]			
	T_{max}	$T_{0.3}$	T_{clad}	$T_{coolant}$
15.00	376.42	364.01	131.45	94.096
17.69	420.24	405.61	133.10	98.96
20.00	457.35	440.81	134.29	101.664
25.00	536.49	515.82	136.58	105.498
30.00	614.42	589.61	138.61	108.167

The LCC BOL Core steady state results shown above provide a succinct summary of the thermal safety related characteristics while operating at the license limit power, lowest water level, and hottest water temperature in the hot-channel (most limiting location within the core). Figure 4.57 shows that the MDNBR in the hot channel will reach a value of 2.00 at approximately 18.25 kW_{th} hot channel steady state power. This is 103% of the 17.69 kW_{th} produced in the hot channel of the LCC core operating at 1.0 MW_{th}. Using either the Bernath or the Groeneveld 2006 correlations, the LCC core is operating at power well below that required for departure from nucleate boiling.

4.7.3.3 LCC EOL Core

The LCC EOL Core contain fuel elements that are geometrically similar and therefore the hot channel geometric parameters (i.e. hydraulic diameter, length, etc.) do not change. The hot channel power summary in terms of parameters and results are given in Table 4.19 and Table 4.20, respectively.

Figure 4.61 through Figure 4.65 graphically illustrate the results of the analysis on the LCC EOL Core.

Each parameter in Figure 4.11 is taken at a different elevation in the hot subchannel. The fuel centerline temperature is shown at the axial nodal location which produces the maximum fuel centerline temperature (fuel axial center). The outer cladding temperature is shown at the axial location which produces the maximum outer cladding temperature (slightly above the axial fuel centerline). The bulk coolant temperature is shown at the location which produces the maximum bulk coolant temperature (highest vertical subchannel node). Coolant mass flux is shown as the mass flux which corresponds to the maximum bulk coolant temperature (highest vertical subchannel node). Each location was selected to show the most limiting value of the associated parameter.

Table 4.19: Steady state results for the LCC EOL Core at 1.0 MW_{th}

Parameter		Value
Flow rate for hottest rod [kg/s]		0.07836
Maximum wall heat flux [kW/m^2]		476.00
Maximum fuel centerline temperature [°C]		423.48
Maximum clad temperature [°C]		133.30
Exit clad temperature [°C]		132.70
Exit bulk coolant temperature [°C]		98.68
MDNBR	Groeneveld 1986	6.50
	Groeneveld 1995	5.18

	Groeneveld 2006	4.93
	Bernath	2.14

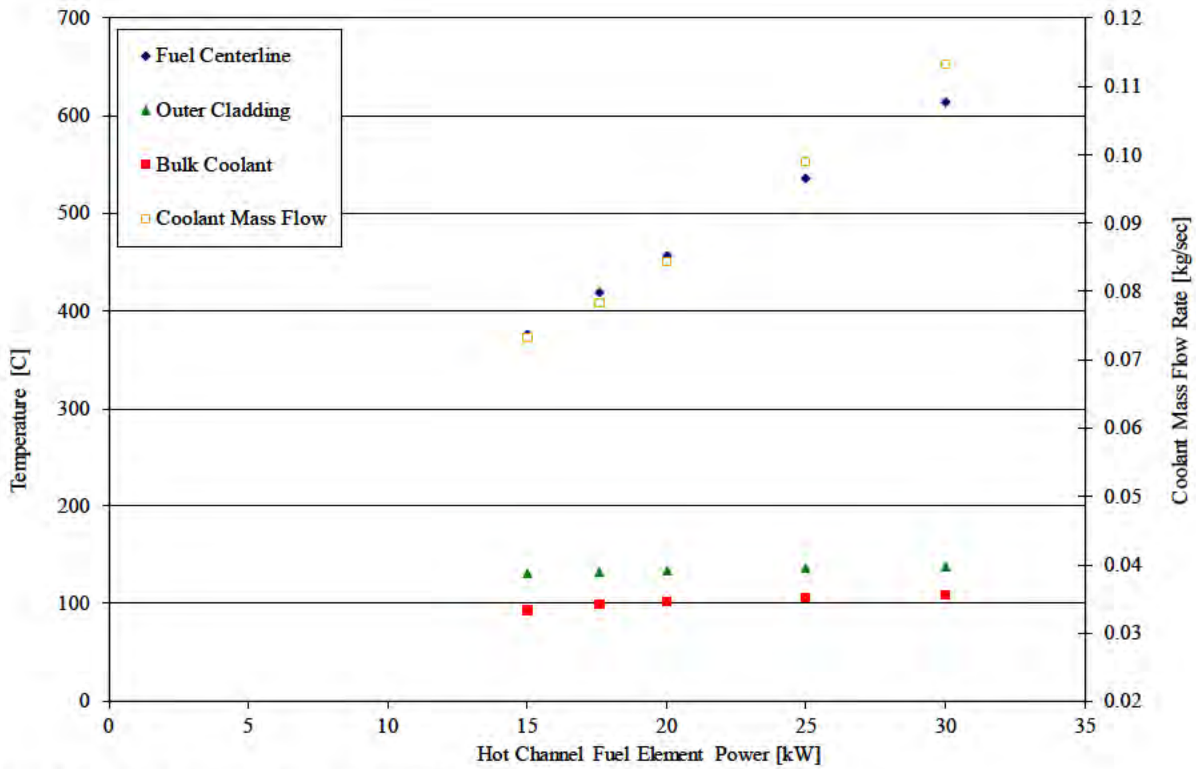

Figure 4.61: Hot channel properties (LCC EOL Core)

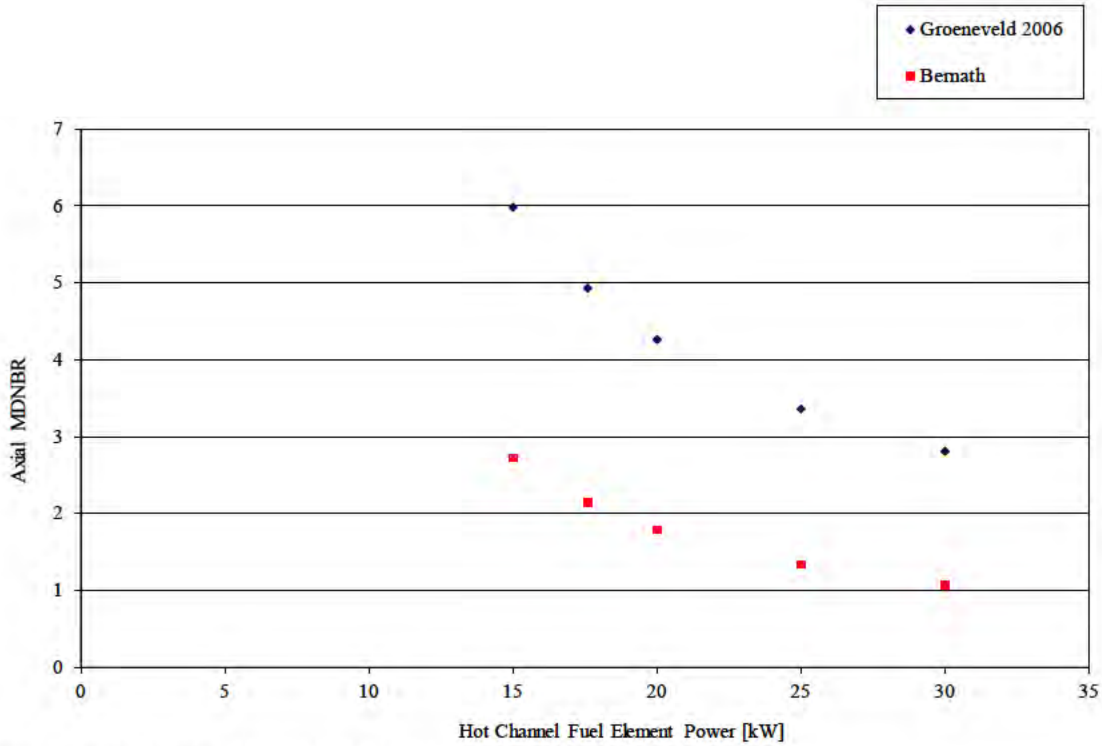

Figure 4.9: Hot channel MDNBR (LCC EOL Core)

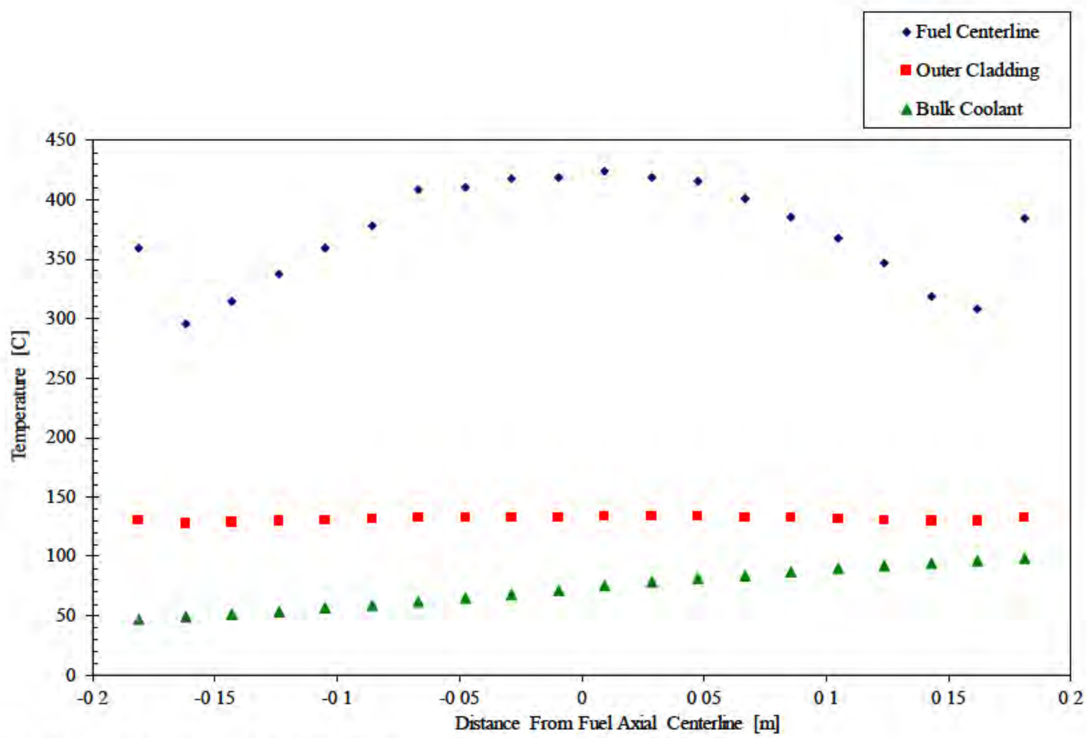

Figure 4.10: Axial temperature distribution at 17.59 kW_{th} (LCC EOL Core)

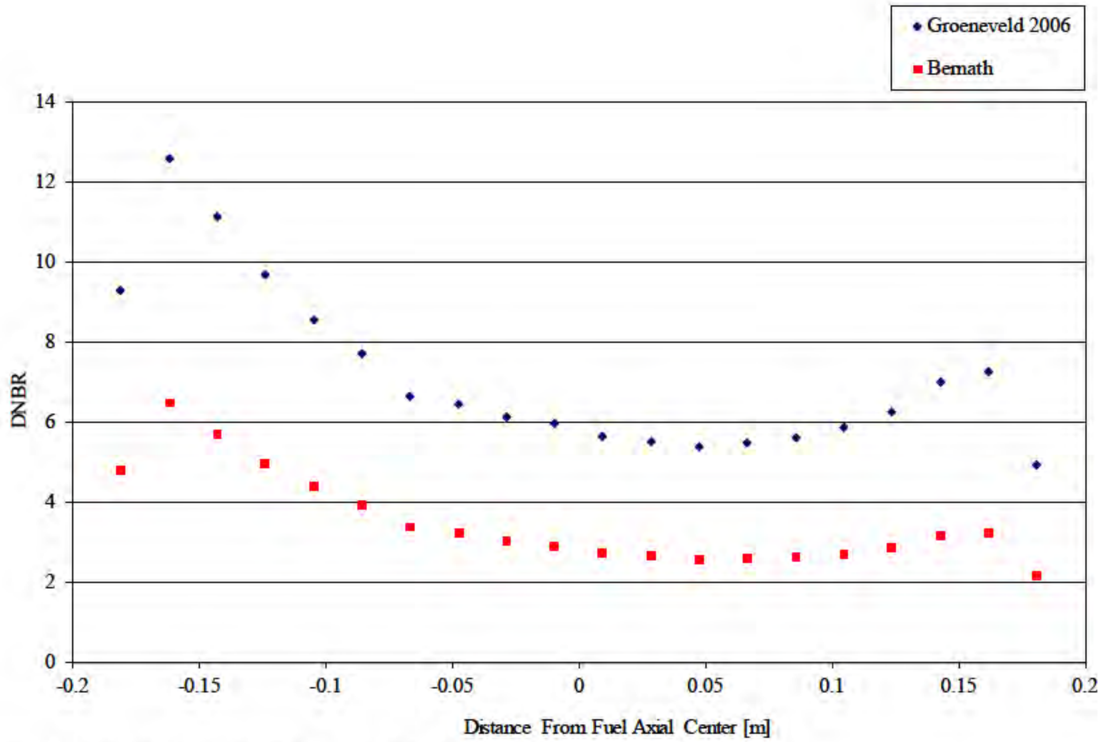

Figure 4.11: Hot channel axial DNBR at 17.59 kW_{th} (LCC EOL Core)

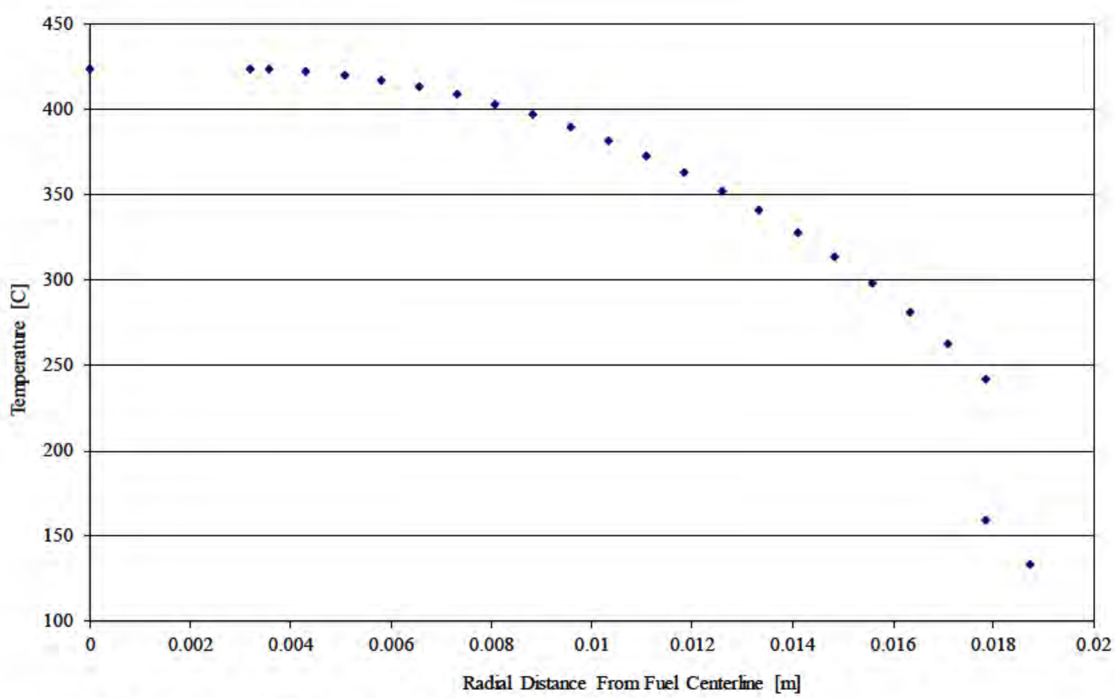

Figure 4.65: Radial temperature distribution at 17.59 kW_{th} (LCC EOL Core)

Table 4.20: Calculated fuel temperatures for various channel powers (LCC EOL Core)

$P_{hot\text{-}channel}$ (kW)	Calculated [°C]			
	T_{max}	$T_{0.3}$	T_{clad}	$T_{coolant}$
15.00	376.42	364.01	131.45	94.096
17.59	418.63	404.08	133.05	98.833
20.00	457.35	440.81	134.29	101.664
25.00	536.49	515.82	136.58	105.498
30.00	614.42	589.61	138.61	108.17

The LCC EOL Core steady state results shown above provide a succinct summary of the thermal safety related characteristics while operating at the license limit power, lowest water level, and hottest water temperature in the hot-channel (most limiting location within the core). Figure 4.62 shows that the MDNBR in the hot channel will reach a value of 2.00 at approximately 18.38 kW_{th} hot channel steady state power. This is 104.5% of the 17.59 kW_{th} produced in the hot channel of the LCC core operating at 1.0 MW_{th}. Using either the Bernath or the Groeneveld 2006 correlations, the LCC core is operating at power well below that required for departure from nucleate boiling.

4.8 Operating Limits

4.8.1 Operating Parameters

The main safety consideration is to maintain the fuel temperatures below the value that would result in fuel damage. The fuel temperature is controlled by setting limits on other operating parameters (i.e., limiting safety system settings). The operating parameters established for the UCD/MNRC reactor are:

a. Steady-state power level;

b. Fuel temperature measured by thermocouple;

c. Maximum core inlet coolant water temperature;

d. Maximum heat output per element.

4.8.2 Limiting Safety System Settings

The limiting safety system settings given in Table 4-21 are defined to assure that the safety limits in the design basis will not be exceeded for normal and abnormal operations.

TABLE 4-21 LIMITING SAFETY SYSTEM SETTINGS

Parameter Limited	Safety Setting	Function
Power level at steady-state	1.02 MW_t	Reactor Scram
Measured fuel temperature	750°C	Reactor Scram

In addition, Technical Specification limits are imposed for the transient rod and coolant water temperatures as follows:

- Reactor tank inlet water temperature of less than 45°C.
- No fuel elements will be placed in the core such that the total rod output is expected to exceed 17.69 kW during 1.0 MW_t steady state reactor operations.

The $1.75 reactivity limit for a single fixed experiment is justified by the analysis in Section 13.2.2, which shows that there will be no damage if insertions are less than $1.92. These safety settings are conservative in the sense that if they are adhered to the consequence of normal or abnormal operation would be fuel or cladding temperatures well below the safety limits indicated in the reactor design basis.

CHAPTER 5

REACTOR COOLANT SYSTEMS

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5.0 REACTOR COOLANT SYSTEMS

5.1. Reactor Tank

The reactor core is positioned near the bottom of an open 1/4 in. thick aluminum tank 7-1/2 ft in diameter by 24-1/2 ft high (Figure 5.1). The tank contains approximately 7,000 gallons of high-purity water so the core is clearly visible from the top. About 20 ft of water over the top of the reactor core provides biological shielding for personnel in the reactor room. The tank is imbedded in a massive concrete structure which provides biological shielding for personnel in surrounding areas.

Pipe assemblies welded to both the inside and outside of the tank wall (the tank wall is continuous), slightly above the reactor core, form one part of the beam tubes. Flanges have been welded to the pipe stubs on the inside of the tank and are used to attach the in-tank section of the beam tube (Section 9.2). Clearance has been provided between the pipe stubs outside the tank and the reactor bulk shielding to prevent structural loading of the tank wall from thermal expansion. An aluminum angle used for support of fuel storage racks, underwater lights, and other equipment is located around the tank top.

The exterior surface of the tank is coated with epoxy and tar-saturated roofing felt to prevent corrosion. The felt is applied in a double thickness using a bituminous material. In addition, a corrugated liner, approximately 1 in. in thickness, is located between the tank exterior and the concrete shield. The corrugated liner provides a path for water to drain to a collection point under the tank should the tank overflow or leak. A drain around the base of the tank is designed to collect any water from the corrugated section. The drain is installed so that it can be routinely monitored for evidence of leakage.

A bridge assembly provides support for the control rod drives and the tank covers. It is located above the top (8 feet above) of the reactor tank directly over the reactor core. The assembly consists of structural channels covered with plates.

The top of the reactor tank is closed by aluminum grating covers that are hinged. Lucite plastic is attached to the bottom of each grating section to prevent foreign matter from entering the tank while still permitting visual observation.

Tank materials, welding procedures, and welder qualifications were in accordance with the ASME code. The integrity of tank weld joints has been verified by radiography, dye penetrant checking, leak and hydrostatic testing.

[redacted]

FIGURE 5.1 REACTOR TANK

5.2. Primary Coolant System

The reactor core is cooled by natural circulation of the reactor tank water. The tank water temperature is maintained at approximately 90-110°F by the primary cooling system.

The primary cooling system, Figure 5.2, is designed to continually remove of up to 3 MW of heat from the reactor tank. It contains the necessary equipment and controls to circulate up to 1000 gpm of tank water and maintain the temperature of the water returning to the tank at about 32.2°C (90°F). Instrumentation is provided to monitor the system operation, water temperatures, pressure, flow, and tank level. Tank bulk water outlet and inlet temperatures are continuously recorded on the DAC.

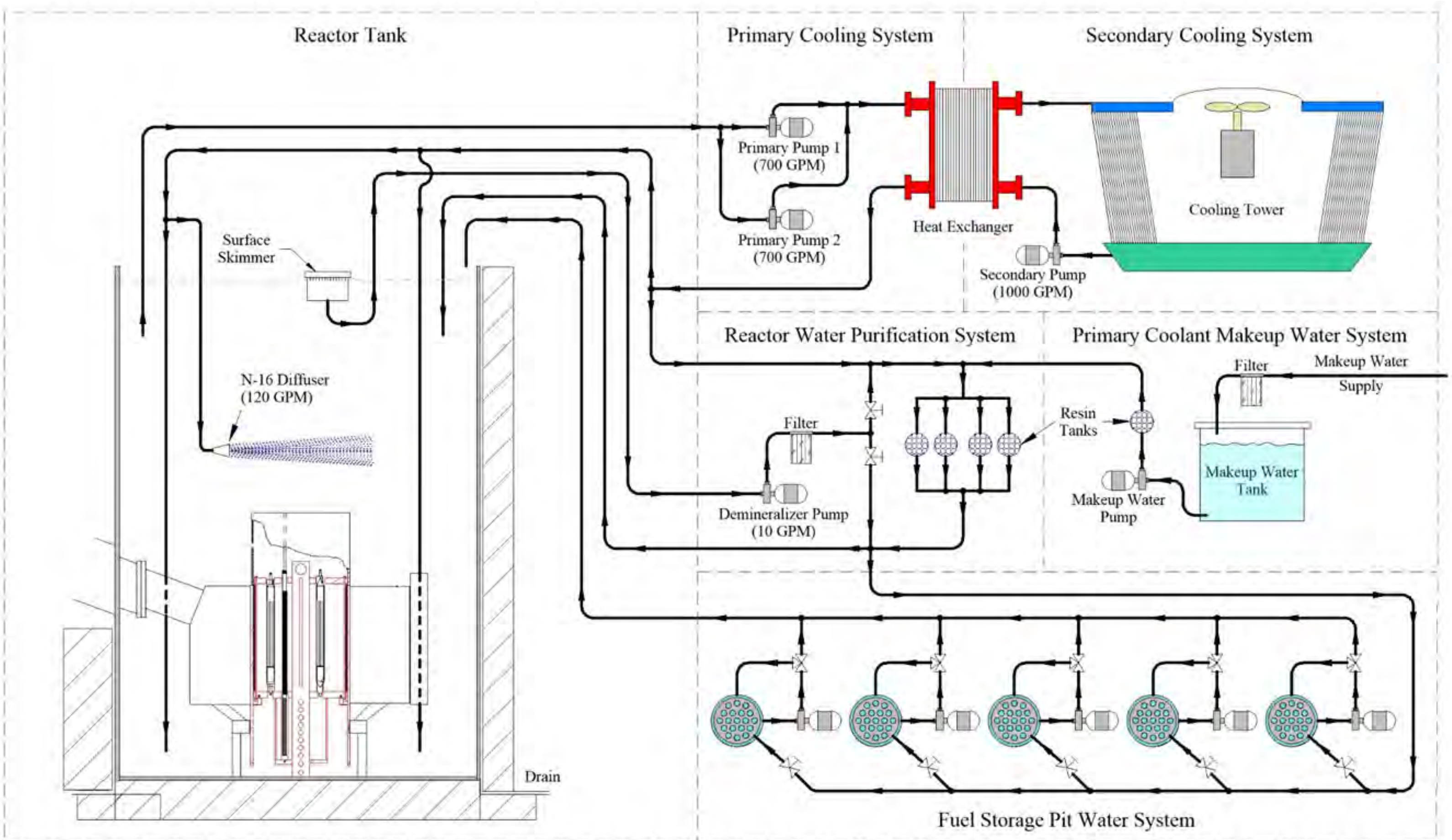
This system is operated and monitored from the reactor control room. The remote controls and monitoring instrumentation are located in the reactor room.

The system is regulated to maintain the primary water system pressure lower than the secondary system pressure. This pressure differential will prevent radioactivity from entering the secondary system, especially the cooling tower, should a leak develop between the two systems.

With the exception of pressure, system parameters are read out in the reactor control room. Alarms are provided on the reactor control console, if flow, tank bulk temperature, or tank water level exceeds preset limits. System pressure gages have local readouts. Tank water level can be monitored visually (via video camera) from the reactor room.

All system components that contact the primary water are normally made from either aluminum or stainless steel. The heat exchanger is a plate-type with the primary water flowing within the plates.

The entrance to the pump suction line is less than 3 ft below the normal tank water level. In addition, the line is perforated from about 8 in. below the normal tank water level to the entrance. Should a primary system component fail downstream of the pump, the tank water level would lower to the first perforation, about 8 in. At this point the pump should lose suction and quit pumping. However, in no case can the pump lower the water level beyond the entrance to the pump suction line, less than 3 ft. Even if the water level lowers to the entrance to the pump suction line there will be approximately 16-1/2 ft of water above fuel elements in the core. This feature prevents the loss of a significant amount of tank water should a leak develop in any of the primary system components when the pump is operating.



NOTE: This drawing only shows a conceptual overview of the reactor coolant systems. Not all flow paths and system components are shown. Valves (and {three-way valve}) are only shown where they are used to select between specific flow paths. All other valves and system components have been omitted to facilitate clear depiction of major flow paths and system interconnectivity.

Figure 5.2 MNRC Reactor Cooling System

5.3. Secondary Coolant System

The secondary cooling system is capable of continually removing up to 3 MW of heat from the primary system during normal weather conditions. The system circulates approximately 1000 gpm of water from a cooling tower through the primary-to-secondary heat exchanger and back to the cooling tower (Figure 5.3). The pressure of the secondary system is maintained higher than the primary system to prevent cross contamination of secondary water should a leak develop in the heat exchanger.

Water chemistry, conductivity, and pH are monitored and maintained by an automatic water conditioning system that adds chemicals as required.

5.4. Reactor Water Purification Systems

The reactor water purification systems maintain the primary water purity and optical clarity (Figure 5.4). There are two separate systems that can be operated independently or can be cross-connected to operate as one unit. One system is used to filter particulate matter from the surface of the reactor tank and the other system deionizes the water to maintain the purity.

The filtration system uses a drum surface skimmer that floats near the surface of the water in the reactor tank. A pump moves water from the surface skimmer to fiber cartridge filter elements. These filter elements remove any dirt or debris from the reactor tank water by mechanically filtering them from the water before returning the water to the reactor tank. The system can be used to return the filtered water directly to the reactor tank or, through a series of valve manipulations, it can send the filtered water through the deionizers (resin columns) and then back to the reactor tank. The system is used to supply the deionizing resin bed during extended shutdown periods when the primary cooling system is not operational.

A set of deionizing resin beds (four) are supplied from the primary cooling system (outlet of the heat exchanger) at a nominal flow rate of fifteen gallons per minute (11 gpm). The resin bed consists of four fiberglass canisters of mixed-bed resin. Two of the canisters are normally on-line and the other two canisters are in a stand-by condition. Two conductivity cells are used to measure the conductivity of water entering the resin beds and the conductivity of the water exiting the resin beds subsequent to entering the reactor tank. There are local readouts of the conductivity near the resin tanks and remote readouts and alarm functions located on the reactor console.

Pressure gauges are located within the systems to monitor the overall performance of the systems.

5.5. Primary Coolant Makeup Water System

A 300 gallon plastic tank of demineralized water is available to make up any primary cooling system water lost by evaporation or other means. The makeup system is equipped with a positive displacement pump and resin canister of the same type that are used in the purification system. The outlet flow of the makeup system discharges to the purification system. Water addition to the primary tank is recorded and tracked to detect non-evaporative primary water loss.

5.6. Nitrogen-16 Control System

A diffuser has been incorporated into the system to reduce the N-16 at the tank top. The diffuser discharge is located about 2 ft above the reactor and directs about 120 gpm through two nozzles designed to produce a laminar flow sheet across the entire top of the reactor (Figure 5.2). The diffuser operates anytime the primary pump is running.

5.7. Fuel Storage Pit Water System

The fuel storage pit water system is used when shielding of stored fuel elements is required (Figure 5.4). The system's water supply is from the demineralized system outlet and pit water level is controlled by a float actuated water supply valve. Each pit subsystem contains a pump and a three-way valve in the pump discharge line. This configuration allows for once-through, recirculation, or feed-and-bleed operation depending on fuel element shielding requirements. When operating in the once-through or feed-and-bleed modes, excess water is returned to the reactor tank. In the history of the facility these fuel storage pits have never been flooded. Spent fuel removed from service is stored in-tank and moved to the fuel storage pits only after sufficient decay time, so that water cooling is not required.

CHAPTER 6

ENGINEERED SAFETY FEATURES

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6.0 ENGINEERED SAFETY FEATURES

During the design of the UCD/MNRC and subsequent analysis for safety considerations for the UCD/MNRC, the only requirement identified for an Engineered Safety Feature was for an Emergency Core Cooling System (ECCS). This feature is required for operation of the UCD/MNRC at >1.5 MW. Previous analysis has shown that an ECCS was not required for the UCD/MNRC, since at 1 MW even an instantaneous loss of the entire tank water would not have resulted in fuel temperatures which would have threatened the fuel clad.

The subcomponent of 2.0 MW ECCS that is still required if the Exhaust Fan #1 (EF1). The uniquely small reactor room at MNRC results in a relatively small air volume to act as a thermal heat sink in the event of the complete instantaneous LOCA. Therefore, in the event of a complete instantaneous LOCA EF1 would be required to operate so that fresh cool air is introduced to the reactor room to provide a heat sink for the decay heat of the reactor.

CHAPTER 7

INSTRUMENTATION AND CONTROL

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REFERENCES

7.1 "Microprocessor Based Research Reactor Instrumentation and Control System," INS-24, GA Technologies, Inc., May, 1986.

7.0 INSTRUMENTATION AND CONTROL

7.1 Introduction

The Instrumentation and Control System (ICS) for the UCD/MNRC TRIGA® reactor is a computer-based system incorporating the use of a GA-developed, multifunction, NM-1000 microprocessor-based neutron monitoring channel and a NPP-1000 analog-type neutron monitoring channel (Figure 7.1). The NM-1000 system provides a safety channel (percent power with scram), a wide-range log percent power channel (below source level to full power), period indication, and a multirange linear power channel (source level to full power) (Reference 7.1). The NPP-1000 system provides a second safety channel for redundancy (percent power with scram). In the pulse mode of operation, the Data Acquisition Computer (DAC) makes a gain change in the NPP-1000 safety channel to provide NV and NVT indication along with a peak pulse power scram. The NM-1000 is essentially bypassed once a pulse has been initiated. *Note MNRC no longer utilizes the reactor's pulse mode.*

The NM-1000 digital neutron monitor system was developed for the nuclear power industry. The system is based on a special, GA-designed, fission chamber and low-noise ultra-fast pulse amplifier. The NPP-1000 safety channel was designed to the same high performance criteria as the NM-1000 channels.

The control system logic is contained in a separate Control System Computer (CSC) with a color graphics display. While information from the NM-1000, NPP-1000, and fuel temperature channels is processed and displayed by the CSC, each is direct wired to its own output display, and the safety channel connects directly to the protective system scram circuit. That is, signals to the scram circuits are not processed by the Data Acquisition Computer or the control computer. The nuclear information goes directly from the detectors to either the NM-1000 or NPP-1000 where it is processed. The processed signals connect directly to the scram circuit switches. Fuel temperature information goes directly to "action pack modules" for amplification and then to the scram circuit switches. The ability of this configuration to meet the intent of protection system requirements for reliability, redundancy, and independence for TRIGA®-type reactors has been accepted by the NRC.

The CSC manages all control rod movements, accounting for such things as interlocks and choice of particular operating modes. It processes and displays information on control rod positions, power level, fuel and water temperature, and pulse characteristics. The CSC performs many other functions, such as monitoring reactor usage and facility radiation instruments, and storing historical operating data for replay at a later time. A computer-based control system has many advantages over an analog system: speed, accuracy, reliability, the ability for self-calibration, improved diagnostics, graphic displays, and the logging of vital information.

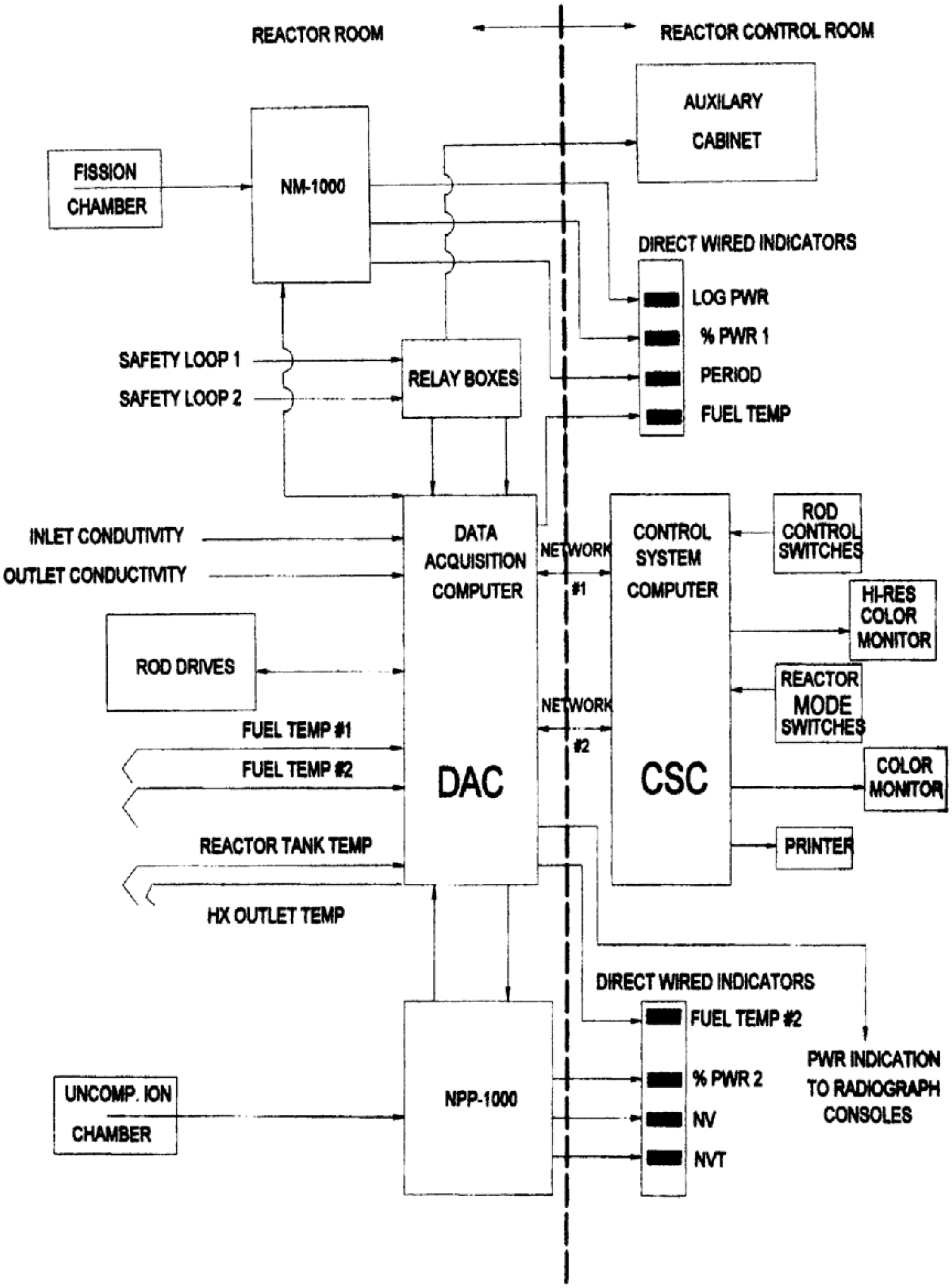


FIGURE 7.1 BLOCK DIAGRAM OF INSTRUMENTATION AND CONTROL SYSTEM

7.1.1 Design Basis

The ICS for the UCD/MNRC reactor is designed to perform the following functions:

- Provide the operator with information on the status of the reactor and facility;
- Provide the means for insertion or withdrawal of control rods;
- Provide for automatic control of the reactor power level;
- Provide for detecting overpower conditions and automatically scram the control rods to terminate the overpower condition;
- Provide for the storage of data for later retrieval.

A scram system is included as part of the instrumentation and control system. The scram system is designed to meet the single-failure criterion applied to power reactors and is independent of the normal reactivity-control system.

7.1.2 Instrumentation and Control System Design

7.1.2.1 NM-1000 Safety and Neutron Monitoring Channel

The NM-1000 nuclear channel has the multifunction capability to provide safety (scram) action as well as neutron monitoring over a wide power range from a single detector. The functions are the following:

1. Percent power with scram;
2. Wide-range log power;
3. Power rate of change;
4. Multirange linear power.

For the UCD/MNRC ICS, the NM-1000 system is designated to provide the wide-range log power function and the percent power safety channel with scram (linear power level from 1% to 120%). The wide-range log power function is a digital version of the patented GA 10-decade log power system to cover the reactor power range from below source level to 150% power and provide a period signal. For the log power function, the chamber signal from startup (pulse counting) range through the Campbelling [root mean square (RMS) signal processing] range covers in excess of 10-decades of power level. The self-contained microprocessor combines these signals and derives the power rate of change (period) through the full range of power. The microprocessor automatically tests the system to ensure that the upper decades are operable while the reactor is operating in the lower decades and vice versa when the reactor is at high power. The output signal from the microprocessor goes directly to the scram circuit switches and the direct reading bar graphs on the console.

For the multirange function, the NM-1000 uses the same signal source as for the log function. However, instead of the microprocessor converting the signal into a log function, it converts it into 10 linear power ranges. This feature provides for a more precise reading of linear power level over the entire range of reactor power. The same self-checking features are included as for the log function. The multirange function is auto-ranging.

The NM-1000 system is contained in two National Electrical Manufacturers Association (NEMA) enclosures located in the reactor room. The amplifier assembly contains modular plug-in subassemblies for pulse preamplifier electronics, bandpass filter and RMS electronics, signal conditioning circuits, low-voltage power supplies, detector high-voltage power supply, digital diagnostics, and communication electronics. The processor assembly is made up of modular plug-in subassemblies for communication electronics (between amplifier and processor), the microprocessor, a control/display module, low-voltage power supplies, isolated 4 to 20 mA outputs, and isolated alarm outputs. Outputs are Class IE as specified by IEEE 323-1974. Communication between the amplifier and processor assemblies is via twisted-pair shielded cables. The amplifier/microprocessor circuit design employs automatic on-line self-diagnostics and calibration verification. Detection of unacceptable circuit performance is automatically alarmed. The system can be automatically calibrated and checked (including the testing of trip levels) prior to operation. The checkout data is recorded for future use. The accuracy of the channels is equal to or better than ±3% of full scale, and trip settings are repeatable within 1% of full-scale input.

The neutron detector uses the standard 0.2 counts/s per nv fission chamber that has provided reliable service in the past. It has, however, been improved by additional shielding to provide a greater signal-to-noise ratio. The low noise construction of the chamber assembly allows the system to respond to a low reactor shutdown level which is subject to being masked by noise. An illustration of the neutron channel operating ranges is shown in Figure 7.2.

7.1.2.2 NPP-1000 Safety Channel

The NPP-1000 system provides the redundant percent power safety channel with scram. The amplified signal from this channel goes directly to the direct wired % power indicator and the scram circuit switches. In the pulse mode of operation, the DAC makes a gain change in the NPP-1000 safety channel to provide NV and NVT indication along with a peak pulse power scram. The NPP-1000 system is an upgrade of GA systems which have been in use in TRIGA® installations world-wide for many years. The nuclear detector for the NPP-1000 is an uncompensated ionization chamber. NPP-1000 systems are utilized at the Sandia National Laboratory, the AFRRI reactor at Bethesda, MD, the University of Texas, and at GA's facility at San Diego, CA.

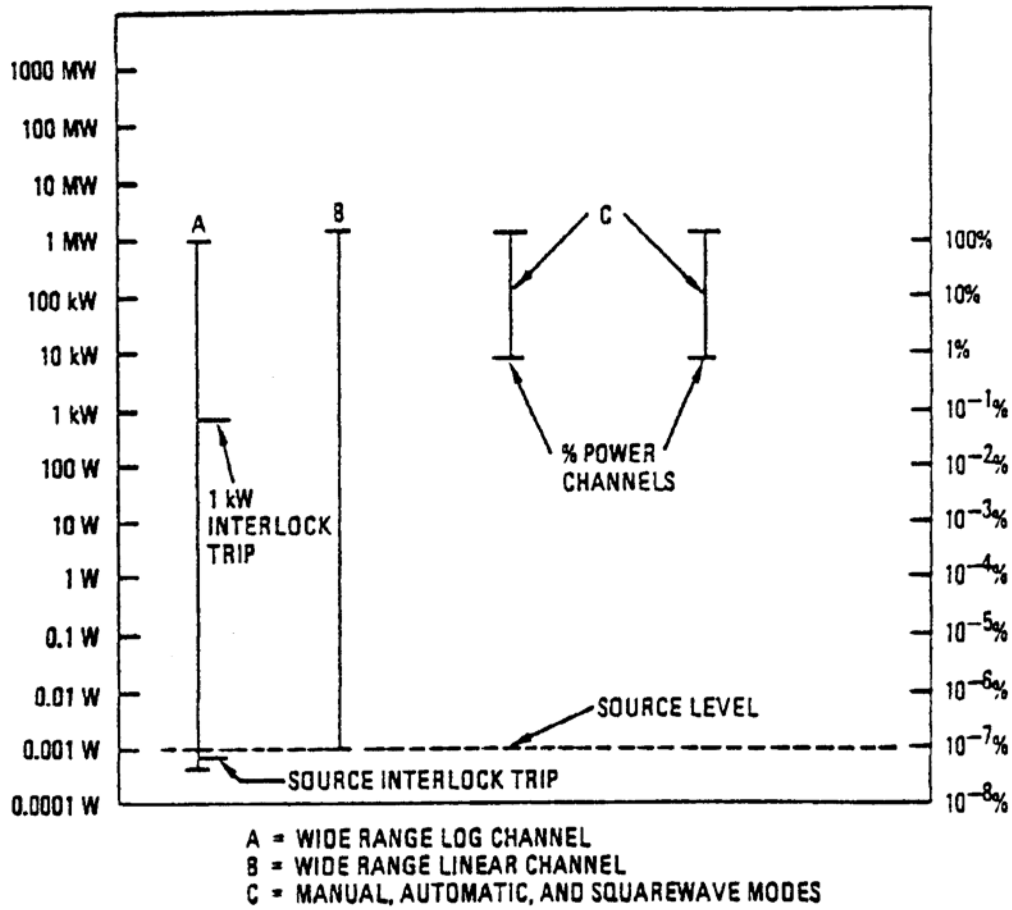


FIGURE 7.2 TYPICAL NEUTRON CHANNEL OPERATING RANGES

The NPP-1000 is located in the reactor room in the DAC assembly cabinet. The cabinet that houses the NPP-1000 has a heat detection and a halon fire suppression system. If the heat detector is activated, a “DAC HALON RELEASE” warning appears on the reactor control room console. After a short time delay, the electric power to the cabinet is turned off. The time delay is long enough for the operator to confirm reactor shutdown.

7.1.2.3 Data Acquisition Computer

As indicated in Figure 7.1, the Data Acquisition Computer (DAC) receives and processes, converts from analog-to-digital form or digital-to-analog form, information from the NM-1000 and NPP-1000 as well as from numerous other instruments associated with reactor and facility operations. The processed information is then transmitted, as appropriate, to the Control System Computer (CSC), the NM-1000, or the NPP-1000. Information transfer between the DAC and CSC is by high speed data transmitter.

In the pulse mode of operation, the DAC makes a gain change in the NPP-1000 channel to provide NV and NVT information along with a peak pulse power scram.

The control and transient rod drive control signals produced by the CSC are processed by the DAC prior to being sent to the devices.

The DAC is located in the reactor room and is housed in the same enclosure as the NPP-1000.

7.1.2.4 Control System Computer/Printer

The Control System Computer (CSC) provides all of the logic functions needed to control the reactor and augments the safety system by monitoring operating characteristics. Information from this computer is displayed on monitors for ease of comprehension. Essentially, all of the control system logic contained in previous TRIGA® reactor control systems is incorporated into the CSC.

However, instead of using electronic circuits and electrical relay circuits, the logic is programmed into the computer. The availability of the computer allows great versatility and flexibility in operationally-related activities aside from the direct control of rod movements. Many other functions are performed by the CSC, such as monitoring reactor usage, monitoring radiation instruments, storing data, and logging operator identity. A rod-drop timing circuit and a display, capable of time measurements in the 10 ms range, are provided within the CSC and displays.

The computer samples all operational data in the steady-state mode every 30 seconds and stores this data. The memory can hold 9,000 such samples or 75 hrs of operational data. In the pulse mode (no longer utilized), there is enough storage for 10 pulses with all parameters.

Operational data can be printed in the same format as displayed on the console CRTs. This includes all real time and archival data. The displays can be reproduced in graphic and print form only.

The computer is located in the reactor control console (Figure 7.3). The reactor control console and the reactor control room both contain halon fire suppression systems. While the reactor control console system is activated by a thermal detector, the reactor control room system located under the false floor is activated by a signal from at least two smoke detectors.

7.1.2.5 Reactor Operating Controls

The UCD/MNRC reactor can be operated in four modes: manual, automatic, square wave, and pulse. The operations are controlled from the mode control panel (Figure 7.4) and the rod control panel (Figure 7.5). *Note square wave and pulse mode are no longer utilized at MNRC.*

The manual and automatic modes are steady-state reactor conditions.

The manual and automatic reactor control modes are used for reactor operation from source level to 100% power. These two modes are used for manual reactor startup, change in power level, and steady-state operation.

A captive keyswitch located on the rod control panel controls the current to the control and transient rod magnets. This keyswitch must be in the "ON" position for any rod movement actions. Anytime the magnet current has been removed, this switch must be turned to the "RESET" position and then back to the "ON" position for the magnet current to be restored. This keyswitch causes "REACTOR ON" lights to be illuminated throughout the UCD/MNRC.

Manual rod control is accomplished through the use of pushbuttons on the rod control panel. The top row of pushbuttons (magnet) is used to interrupt the current to the rod drive magnet. If the rod is above the down limit, it will fall back into the core and the magnet will automatically drive to the down limit, where it will again contact the armature.

The middle row of pushbuttons (up) and the bottom row (down) are used to position the control rods. Depressing the pushbuttons causes the control rod to move in the direction indicated. Interlocks prevent the movement of the rods in the up direction under the following conditions:

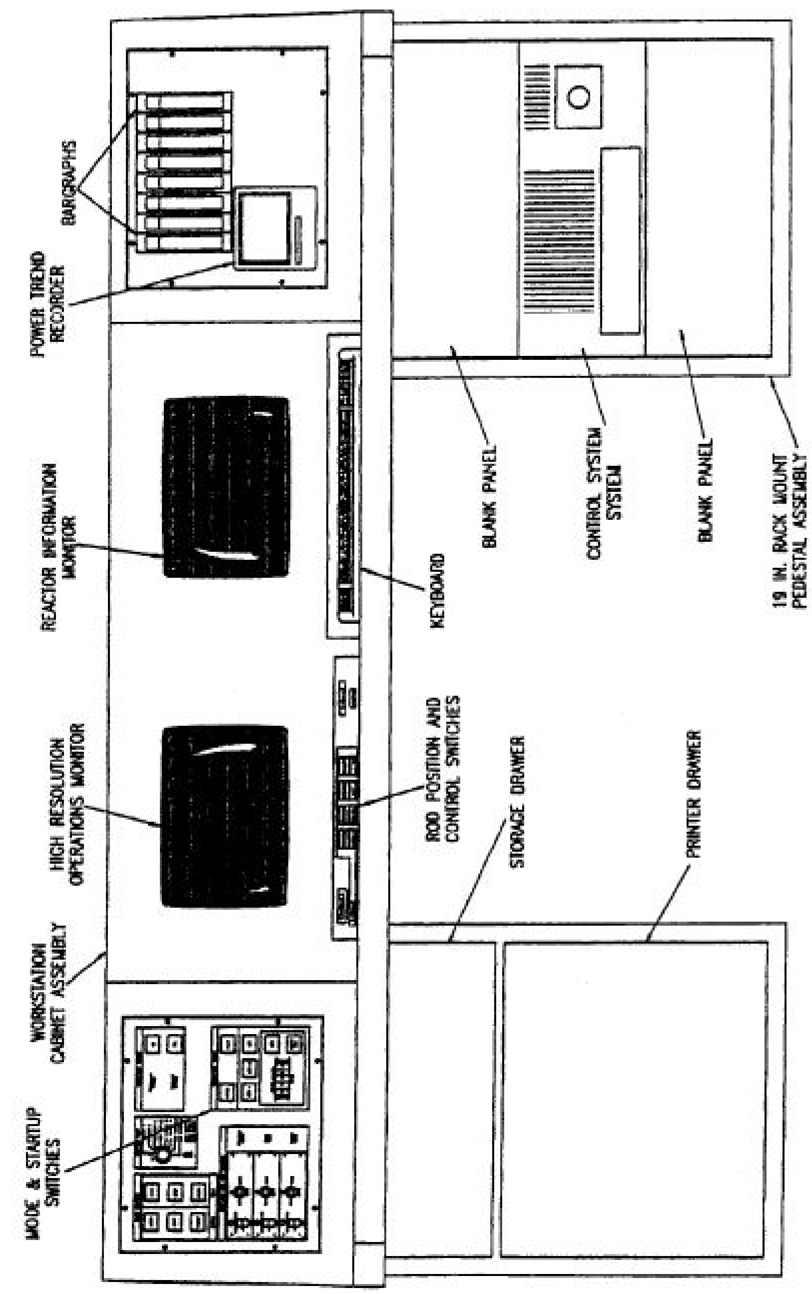


FIGURE 7.3 UCD/MNRC REACTOR CONTROL CONSOLE

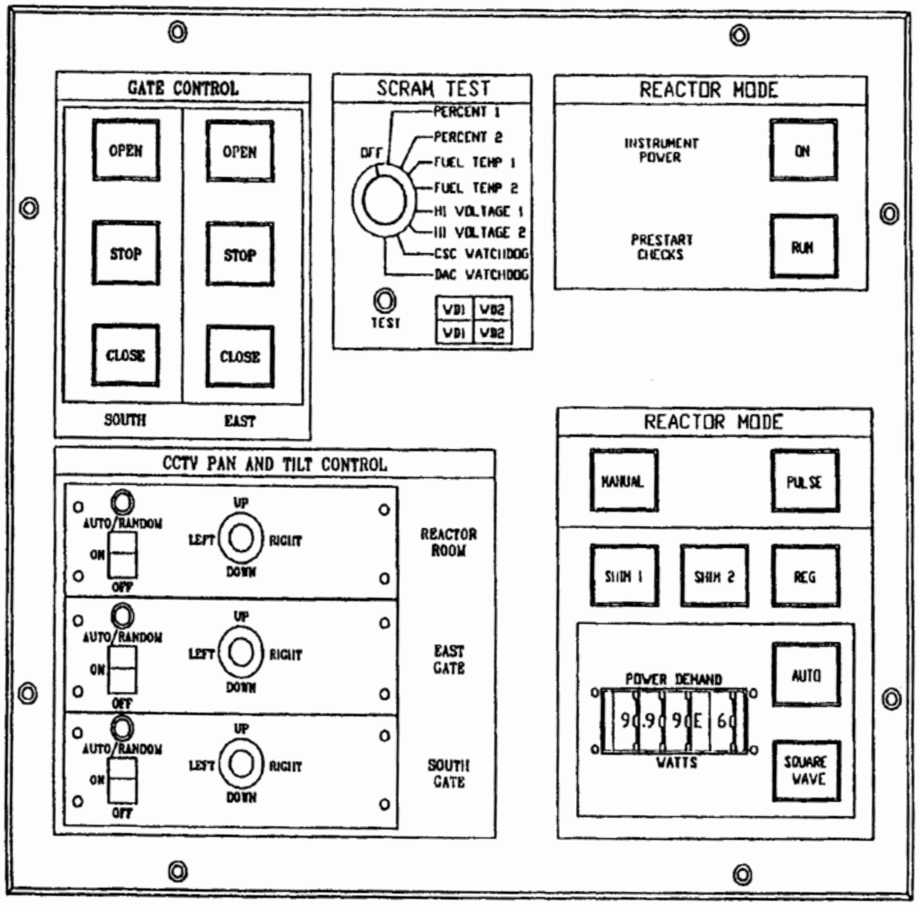


FIGURE 7.4 TYPICAL MODE CONTROL PANEL

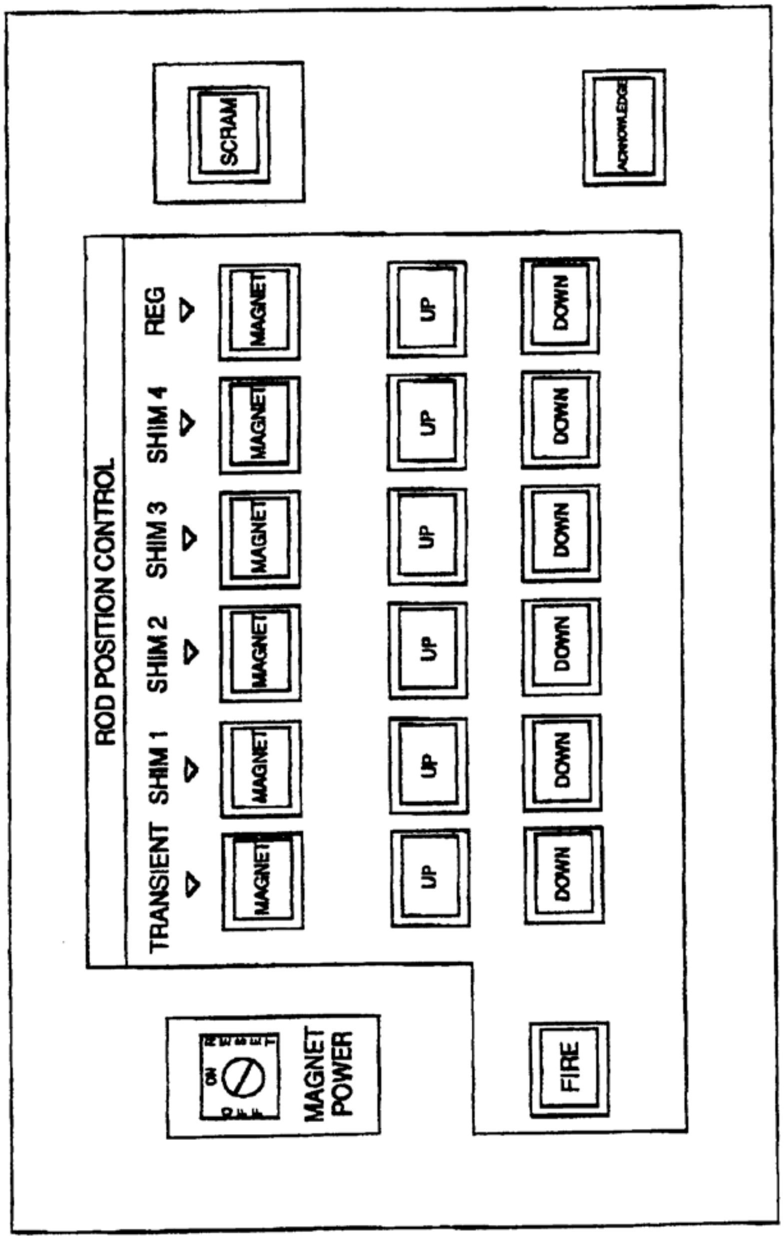


FIGURE 7.5 TYPICAL ROD CONTROL PANEL

1. Scrams not reset;
2. Source level below minimum count;
3. Two UP switches depressed at the same time;
4. Mode switch in the PULSE position;
5. Mode switch in the AUTOMATIC position [servocontrolled rod(s) only];
6. Square wave mode - switch depressed or lighted.

There is no interlock inhibiting the down direction of the control rods except in the case of the servocontrolled rod(s) while in the automatic mode.

Automatic power control can be obtained by switching from manual operation to automatic operation on the mode control panel. All the instrumentation, safety, and interlock circuitry described above applies in the operation of this mode. However, the servocontrolled rod(s) is (are) controlled automatically to a power level and period signal. The reactor power level is compared with the demand level set by the operator, on the mode control panel, and used to bring the reactor power to the demand level on a fixed preset period. The purpose of this feature is to maintain automatically the preset power level during long-term power runs.

The square-wave mode (no longer utilized) allows the reactor power to be quickly raised to a desired power level. In a square-wave operation, the reactor is first brought to criticality below one kW in the manual mode, leaving the transient rod partially in the core. The desired power level is set by the reactor operator using the power demand selector located on the mode control panel. All of the steady-state instrumentation is in operation. The transient rod is ejected from the core by means of the transient rod FIRE pushbutton located on the rod control panel. When the power level reaches the demand level, it is maintained in the automatic mode.

Reactor control in the pulsing mode (no longer utilized) consists of manually establishing criticality at a flux level below one kW in the steady-state mode. This is accomplished by the use of the control rods, leaving the transient rod either fully or partially inserted. The pulse mode selector switch located on the mode control panel is then depressed. The MODE SELECTOR switch automatically causes the DAC to make a gain change in the NPP-1000 safety channel to monitor and record peak flux (NV), energy release (NVT), and to provide a peak pulse power scram. The pulse is initiated by activating the FIRE pushbutton. Once a pulse has been initiated and it is detected by the DAC, the NM-1000 safety scram is bypassed. Pulsing can be initiated from either the critical or subcritical reactor state.

The rod control panel contains a manual scram switch and a switch to acknowledge warning information that appears on one of the reactor control room displays.

The mode control panel contains controls for instrument power, prestart checks, and reactor scram test. Controls for the UCD/MNRC entrance gates and the CCTV cameras for the reactor room are located on this panel.

7.1.2.6 Reactor and Facility Display Equipment

Reactor and facility operating and monitoring information is displayed on two color monitors and a bar graph indicator located on the reactor control console.

The high resolution monitor displays important reactor operating information (Figure 7.6). This monitor has a scram/warning window which indicates the cause of the scram/warning when a scram occurs or a predetermined limit is reached. This window is normally black, but changes to red when under a scram/warning condition. An audible alarm is sounded when a scram/warning condition exists. Tables 7-1 and 7-2 list the parameters that can appear on the window, one at a time. If more than one limit is reached, the first-in will be displayed. Once acknowledged and cleared, the next parameter will appear. The date, time of day, operating mode, and demanded power are also displayed. The reactor operating information, generated by the Control System Computer, is displayed on this monitor as follows:

- Linear power;
- Log power;
- Percent power from both safety channels;
- Rod position (resolution of $\leq$ 0.1 in.);
- Fuel temperature;
- Tank water temperature.

The second monitor is used to display reactor and facility information.

Three types of information are made available for reactor operator use: scram, warning, and status. The information available for display for each of these three categories is shown in Tables 7-1, 7-2, and 7-3, respectively. The console keyboard is used to select the category to be displayed. If the scram category is selected, the parameters in Table 7-1 that have exceeded the scram setpoints will be displayed in the order in which the setpoints were exceeded, first-in. As noted above, the first parameter to cause the scram is indicated in the scram/warning window on the high resolution monitor. The scram indication will remain on the display until it has been cleared.

If the warning category is selected, the parameters in Table 7-2 that have exceeded the warning setpoint will be displayed. The display on the high resolution monitor, the order, and clearing is the same as for the scram category.

The third category that may be selected is System Status. The parameters listed in Table 7-3, with the current reading, will be displayed.

The bar graph indicator panel displays information important to reactor operations. The information displayed on this panel is shown in Figure 7.7. The reactor power and period information displayed on this panel comes directly from the NM-1000 and NPP-1000 safety channels. It is hard-wired and does not come through the Control System Computer.

Included on the indicator panel is a single-pen recorder for wide-range-linear power.

FIGURE 7.6 TYPICAL CRT DISPLAY OF REACTOR OPERATIONAL INFORMATION

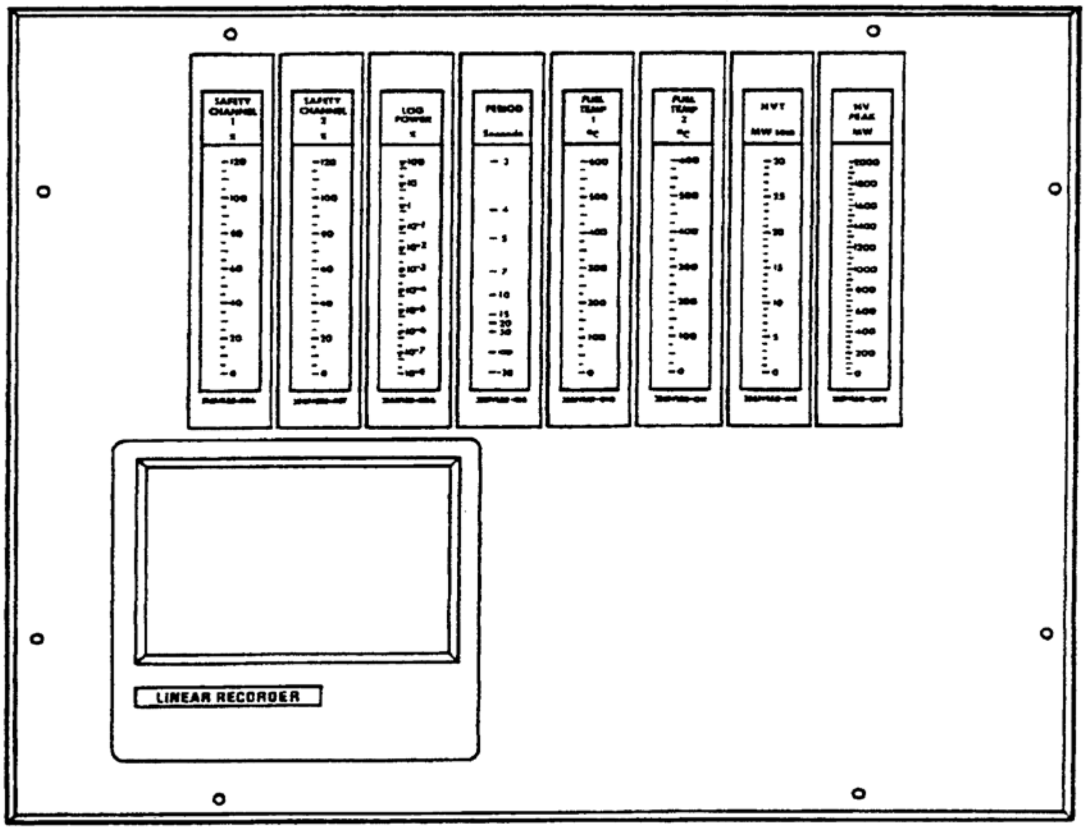


FIGURE 7.7 INDICATOR PANEL

7.2 Reactor Protective System

The reactor protective system SCRAM logic is shown in Figure 7.8. A reactor protective action interrupts the rod magnet current and results in the immediate insertion of all rods if any reactor protective action's triggering parameters are observed. These parameters are described in Table 7-1.

External SCRAM protective actions (items 6 and 7 in Table 7-1) are triggered if a radiography SCRAM interlock is activated. Radiography SCRAM interlocks are used to ensure that the reactor is either scrammed or cannot be operated if a bay's shutter and bay door are both open or a bay ripcord circuit has been activated (more information about radiography SCRAM interlocks can be found in the Auxiliary Systems chapter, Section 9.6). External SCRAM protective systems, External #1 and External #2, serve redundant protective action functions.

The majority of the reactor protective actions listed in Table 7-1 initiate SCRAM action in two ways. First, the in-series circuit elements (typically relays) associated with a specific reactor protective action (shown in Figure 7.8) will interrupt the rod drive magnet current, causing a SCRAM if that protective action's triggering parameters are observed. Second, the SCRAM signal is detected by the computer and the computer generates a redundant scram signal that opens the six parallel relays (RLY08-3 K2, K3, K4, and RLY 08-1 K5, K6, K7), interrupting rod magnet current.

Scram conditions are automatically indicated on the monitors and there is an audible annunciator. A manual scram may be used for a normal fast shutdown of the reactor.

TABLE 7-1 MONITOR SCRAM WINDOW DISPLAY AND ASSOCIATED REACTOR PROTECTIVE ACTIONS

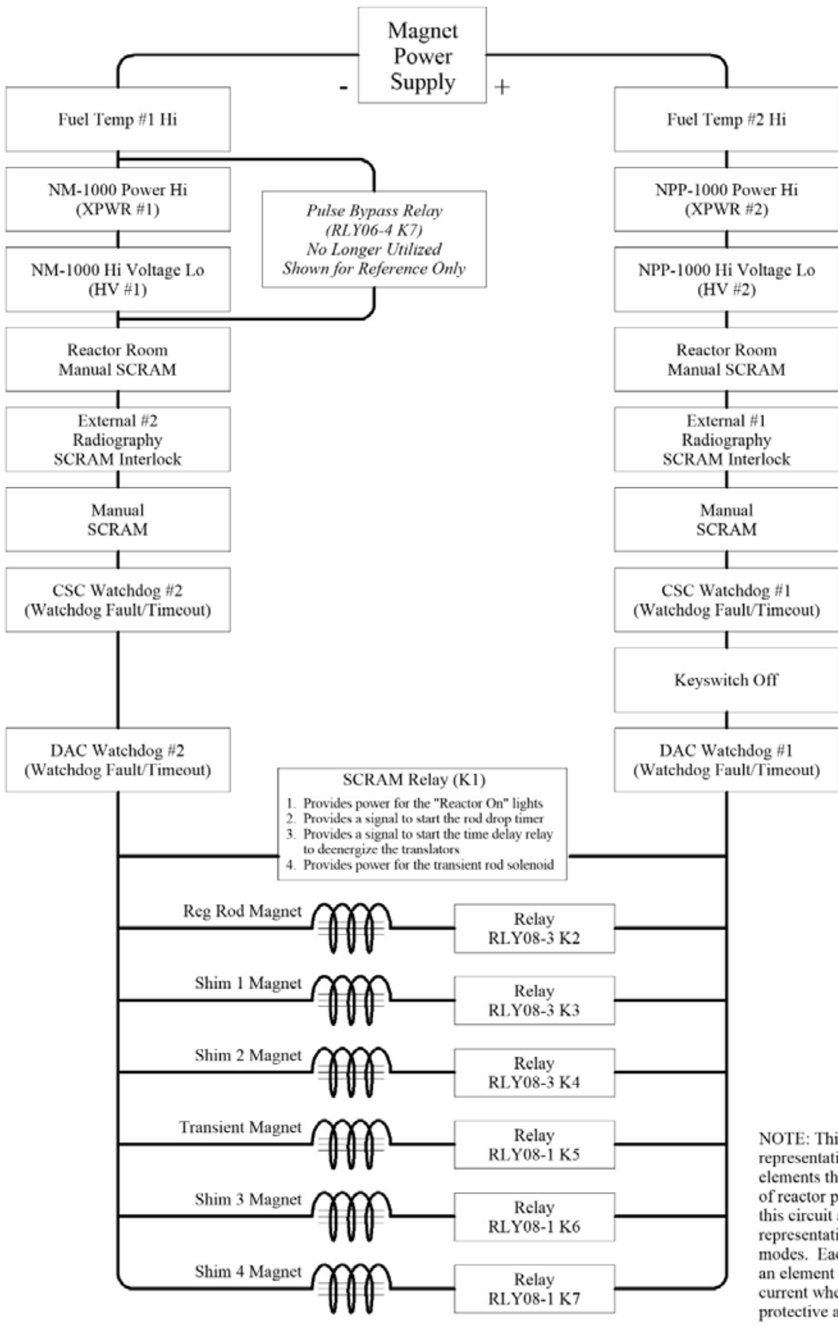
1. Scram – Console Manual
   Manually-initiated protective action caused by pressing the reactor control console scram button
2. Scram – Reactor Room Manual
   Manually-initiated protective action caused by pressing the reactor room scram button
3. Scram – Bay Rip Cord
   Manually-initiated protective action caused by pulling a rip cord in any of the radiography bays
4. Scram – Fuel Temp #1 Hi
   Automatic protective action that is triggered before an instrumented fuel element's temperature measurements (channel #1) reach 750°C
5. Scram – Fuel Temp #2 Hi
   Automatic protective action that is triggered before an instrumented fuel element's temperature measurements (channel #2) reach 750°C
6. Scram – External #1
   Automatic protective action that is triggered if a radiography scram interlock is activated

14. Scram – NPP-1000 Power Hi
    Automatic protective action triggered by an NPP-1000 measurement of reactor power level exceeding the operator-set threshold
15. Scram – NM-1000 Power Hi
    Automatic protective action triggered by an NM-1000 measurement of reactor power level exceeding the operator-set threshold
16. Scram – NM-1000 Hi Voltage Lo
    Automatic protective action triggered by low voltage measurements on the NM-1000
17. Scram – NPP-1000 Hi Voltage Lo
    Automatic protective action triggered by low voltage measurements on the NPP-1000
18. Scram – Keyswitch Off
    Manually-initiated protective action caused by the reactor control console's keyswitch being in the "Off" or "Reset" position
19. Scram – Please Log In
    Automatic protective action that occurs when an operator is not properly logged in to the reactor control console
20. Scram – Net Fault, Please Reboot

7. Scram – External #2
   Automatic protective action that is triggered if a radiography scram interlock is activated (Redundant functionality to External #1)
8. Scram – CSC DIS64 Timeout
   Automatic protective action triggered by a data transfer error with the CSC
9. Scram – DAC DIS64 Timeout
   Automatic protective action triggered by a data transfer error with the DAC
10. Scram – CSC Watchdog Fault
    Automatic protective action triggered by a data transfer error with the CSC
11. Scram – CSC Watchdog Timeout
    Automatic protective action triggered by a data transfer error with the CSC
12. Scram – DAC Watchdog Fault
    Automatic protective action triggered by a data transfer error with the DAC
13. Scram – DAC Watchdog Timeout
    Automatic protective action triggered by a data transfer error with the DAC

Automatic protective action that occurs when there is a problem with the computer network

21. Scram – Database Timeout
    Automatic protective action triggered by a data transfer error with the NM-1000
22. Scram – NM-1000 Comm Fault
    Automatic protective action triggered by a data transfer error with the NM-1000
23. Scram – NM-1000 Data Error
    Automatic protective action triggered by a data transfer error with the NM-1000
24. Scram – DOM32 Fault
    Automatic protective action triggered by a digital data error with the DOM32
25. Scram – AIO16 #1 Fault
    Automatic protective action triggered by an analog data error observed by the AIO16.
26. Scram – AIO16 #2 Fault
    Automatic protective action triggered by an analog data error observed by the AIO16.



NOTE: This drawing only shows a conceptual representation of control rod magnet circuit elements that are directly related to the execution of reactor protective actions. Other elements of this circuit are omitted to facilitate clear representation of the reactor's various SCRAM modes. Each Block within this diagram represents an element of the circuit that can interrupt magnet current when triggered by the specified reactor protective action.

Figure 7.8 PROTECTIVE SYSTEM SCRAM LOGIC

TABLE 7-1 CRT WARNING WINDOW DISPLAY

1.	Pulse Not Detected	22.	Rx Room RAM
2.	Demand Power Not Reached	23.	Demineralizer RAM
3.	High IC-Net Comm Fault	24.	Equipment Area RAM
4.	Low IC-Net Comm Fault	25.	Staging Area #1 RAM
5.	Power Too Hi to Pulse	26.	Staging Area #2 RAM
6.	Trans Rod Air Must Be Off	27.	Staging Area #4 RAM
7.	Period Too Short to Pulse	28.	Rx Rm Particulate
8.	Line Printer Not On Line	29.	Rx Rm Noble Gas
9.	Rod Withdrawal Prohibit	30.	Rx Rm Iodine
10.	Rx Tank Return Temp Hi	31.	Bay Particulate
11.	Magnet Supply Voltage	32.	Stack Particulate
	Grounded-Hi Side	33.	Stack Noble Gas
12.	Magnet Supply Voltage	34.	Stack Argon
	Grounded - Low Side	35.	Bay Argon
13.	Primary System Flow	36.	Rx Rm CAM Fault
14.	Demin System Flow	37.	Stack CAM Fault
15.	Secondary System Flow	38.	Bay CAM Fault
16.	Demin Inlet Condtvty	39.	Rx Rm CAM Alert
17.	Demin Outlet Condtvty	40.	Stack CAM Alert
18.	Rx Tank Water Level Hi	41.	Rx Rm CAM Alarm
19.	Rx Tank Water Level Lo	42.	Stack CAM Alarm
20.	Cooling Tower Water Level Hi	43.	Fire in DAC
21.	Cooling Tower Water Level Lo		

TABLE 7-2 CRT STATUS WINDOW DISPLAY

Primary System Flow	000.0 gpm	Staging Area #1 RAM	000 mR/hr
Secondary System Flow	000.0 gpm	Staging Area #2 RAM	000 mR/hr
Demin System Flow	00.0 gpm	Staging Area #4 RAM	000 mR/hr
Demin Inlet Condtvty	0.0 uMHOS	Rx Rm Particulate	0.0e+0 cpm
Demin Outlet Condtvty	0.0 uMHOS	Rx Rm Noble Gas	0.0e+0 cpm
Rx Tank Temp	00.0 C	Rx Rm Iodine	0.0e+0 cpm
Hx Outlet Temp	00.0 C	Stack Particulate	0.0e+0 cpm
Hx Inlet Temp	00.0 C	Stack Noble Gas	0.0e+0 cpm
Rod Drop Timer	0.00 sec		
Reactor Room RAM	000 mR/hr	Stack Argon	0.0e+0 cpm
Demineralizer RAM	000 mR/hr	Bay Particulate	0.0e+0 cpm
Equipment Area RAM	000 mR/hr	Bay Argon	0.0e+0 cpm
One Kilowatt Interlock	Yes		
Rod Withdrawal Prohibit	No		

7.3 Rod Control System

The reactivity of the UCD/MNRC reactor is controlled by six control rods. The control and transient rod drives are mounted on a bridge at the top of the reactor tank. The drives are connected to the control and transient rods through a connecting rod assembly. The following sections describe the control and transient rods and their respective drive assemblies.

7.3.1 Control Rods

Reactor core loadings utilize fuel-followed control rods, i.e., control rods that have a fuel section below the absorber section. The uppermost section is 6.5 inch-long air-filled void and the next 15 inches is a solid boron carbide neutron absorber section. Immediately below the absorber is the fuel section consisting of 15 inches of U-$ZrH_{1.7}$ whose uranium is enriched in ^{235}U to less than 19.7%. The weight percent of uranium in the fuel is 20.

The bottom section of the rod has an air-filled void approximately 6.5 inches long. The fuel and absorber sections are sealed in a Type 304 stainless steel tube approximately 43 inches long by 1.35 inches in diameter.

The fuel-followed control rods pass through and are guided by 1.5 in. diameter holes in the top and bottom grid plates. A typical control rod with fuel follower is shown in the withdrawn and inserted positions in Figure 7.10.

The transient rod is a sealed, 44.25 in. long by 1.25 in. diameter tube containing solid boron carbide as a neutron absorber and air as a follower. The absorber section is 21 in. long and the follower is approximately 23 in. long. The transient rod passes through the core in a perforated aluminum guide tube. The tube receives its support from the safety plate and its lateral positioning from both grid plates. It extends above the top grid plate. Water passage through the tube is provided by a large number of holes distributed evenly over its length.

7.3.2 Control Rod Drive Assemblies (For Transient Rod Assembly see Section 7.3.3)

The control rods are positioned by five standard TRIGA® electrically powered rack and pinion drives (Figure 7.11). One rod is designated as a regulating rod and used in conjunction with an automatic power control. All rods and rod drives are exactly the same and operate at a nominal rate of approximately 24 in. per minute.

The rod drives are connected to the control rods through a connecting rod assembly. These assemblies contain a bolted connection at each end to accept the control rod at one end and the control rod drive at the other. The grid plates provide guidance for all control rods during operation of the reactor. No control rods can be inserted or removed by their drives a sufficient distance to allow disengagement from the grid plates.

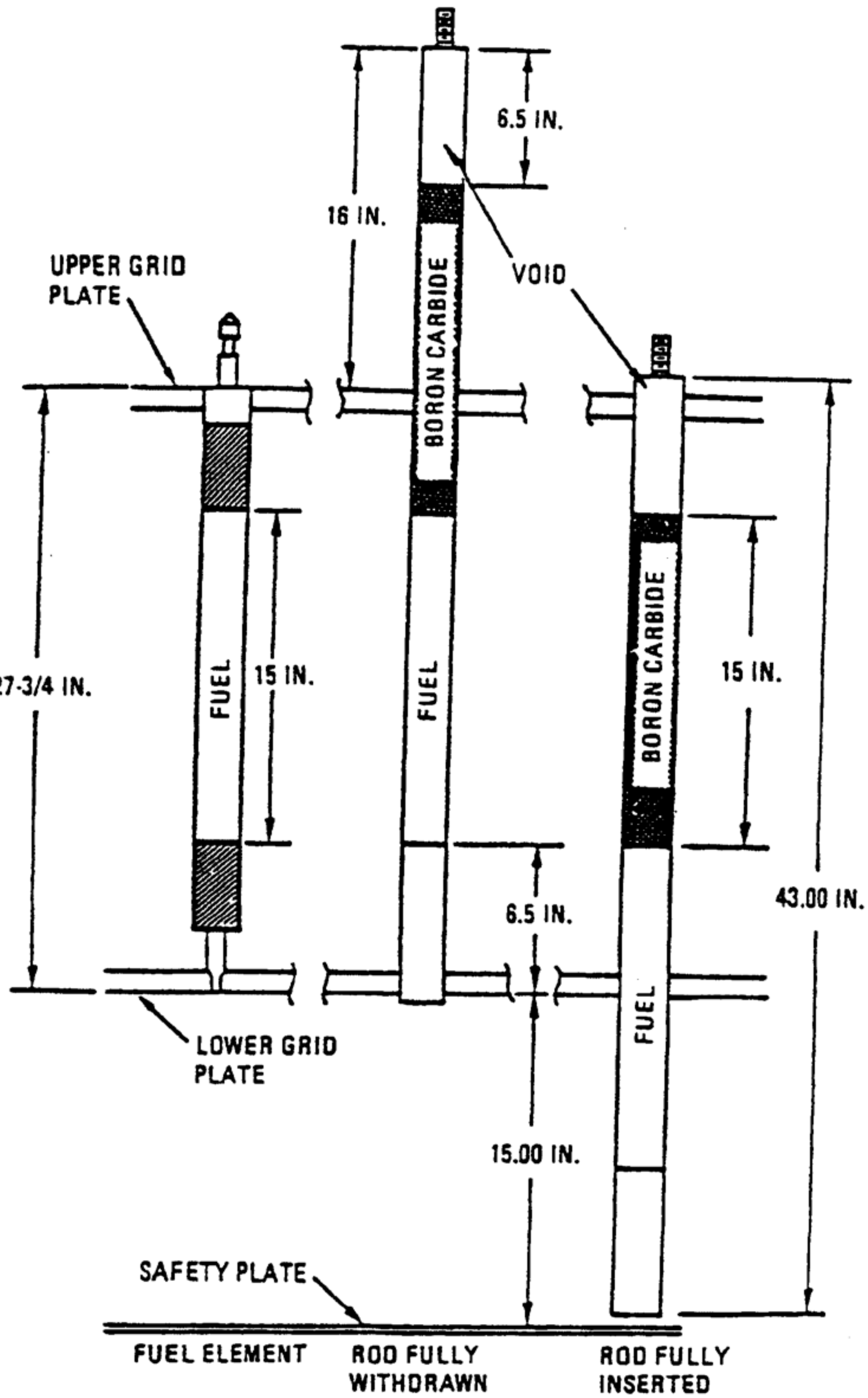


FIGURE 7.9 TYPICAL FUEL FOLLOWER CONTROL ROD SHOWN WITHDRAWN AND INSERTED

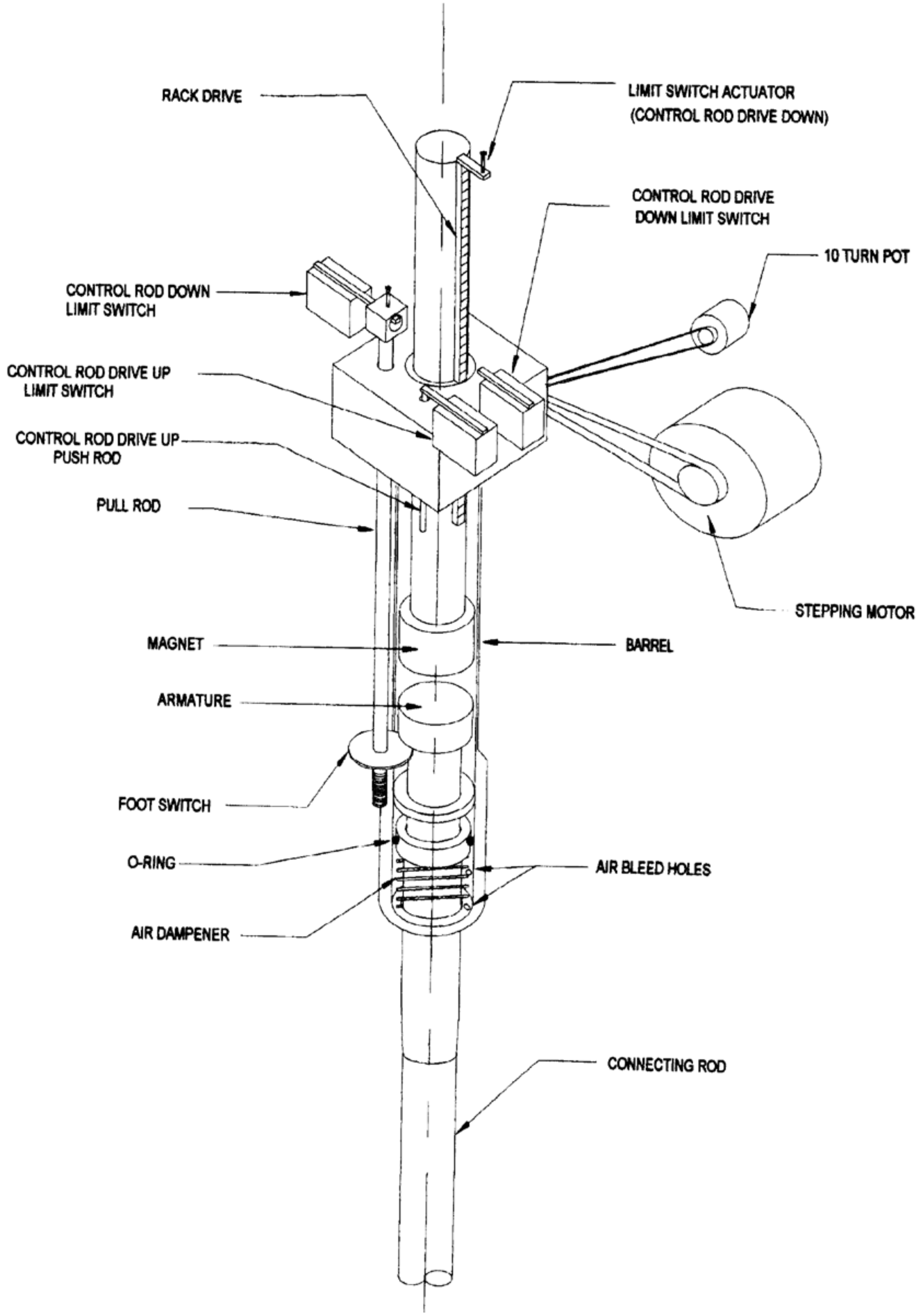


FIGURE 7.10 RACK-AND-PINION CONTROL ROD DRIVE (TYPICAL)

Each drive consists of a stepping motor, a magnet rod-coupler, a rack and pinion gear system, and a ten-turn potentiometer used to provide an indication of rod position. The pinion gear engages a rack attached to a draw tube which supports an electromagnet. The magnet engages a chrome-plated armature to the end of a connecting rod that fits into the connecting tube. The connecting tube extends down to the control rod. The magnet, its draw tube, the armature, and the upper portion of the connecting rod are housed in a tubular barrel. The barrel extends below the control rod drive mounting plate with the lower end of the barrel serving as a mechanical stop to limit the downward travel of the control rod drive assembly. The lower section of the barrel contains an air snubber to dampen the shock of the scrammed rod. In the snubber section, the control rods are decelerated through a length of 3 in. During this length, air is compressed under a piston attached to the connecting rod by the weight of the control rod and is slowly bled to atmosphere through an adjustable needle valve. The control rod can be withdrawn from the reactor core when the electromagnet is energized. When the reactor is scrammed, the electromagnet is de-energized and the armature is released.

The rod drive motors are stepping motors driven by a translator. The speed of the rods is adjustable and is normally set to insert or withdraw the control rods at a nominal rate of 24 in./min. The unique characteristics of a stepping motor/translator system are used to provide fast stops and to limit coasting or overtravel. The control rod drive speeds are administratively controlled. Access to the control rod drives is restricted to authorized personnel and the physical location is in a restricted area.

These rod drives have the capability of withdrawing the rods at a maximum rate of 42 in./min.. The system is fail-safe, that is, multiple system failures are required to get uncontrolled withdrawal of the rods at this maximum speed. In addition, reactivity insertion accident analyses, Chapter 13, have shown no significant effects.

Limit switches mounted on each drive assembly stop the rod drive motor at the top and bottom of travel and provide switching for console indication which shows:

1. When the magnet is in the up position;
2. When the magnet (and thus the control rod) is in the down position;
3. When the control rod is in the down position.

A key-locked switch on the reactor console power supply prevents unauthorized operation of all control rod drives.

These rod drives were first developed in 1959, and have been modified and improved a number of times. The design has proven to be reliable and has been used in more than 60 TRIGA® reactors containing more than 160 rod drives.

7.3.3 Transient Rod Drive Assembly

The UCD/MNRC adjustable fast transient rod drive (Figure 7.12) consists of a combination of a standard TRIGA® rack-and-pinion control drive, described in Section 7.3.2, and a

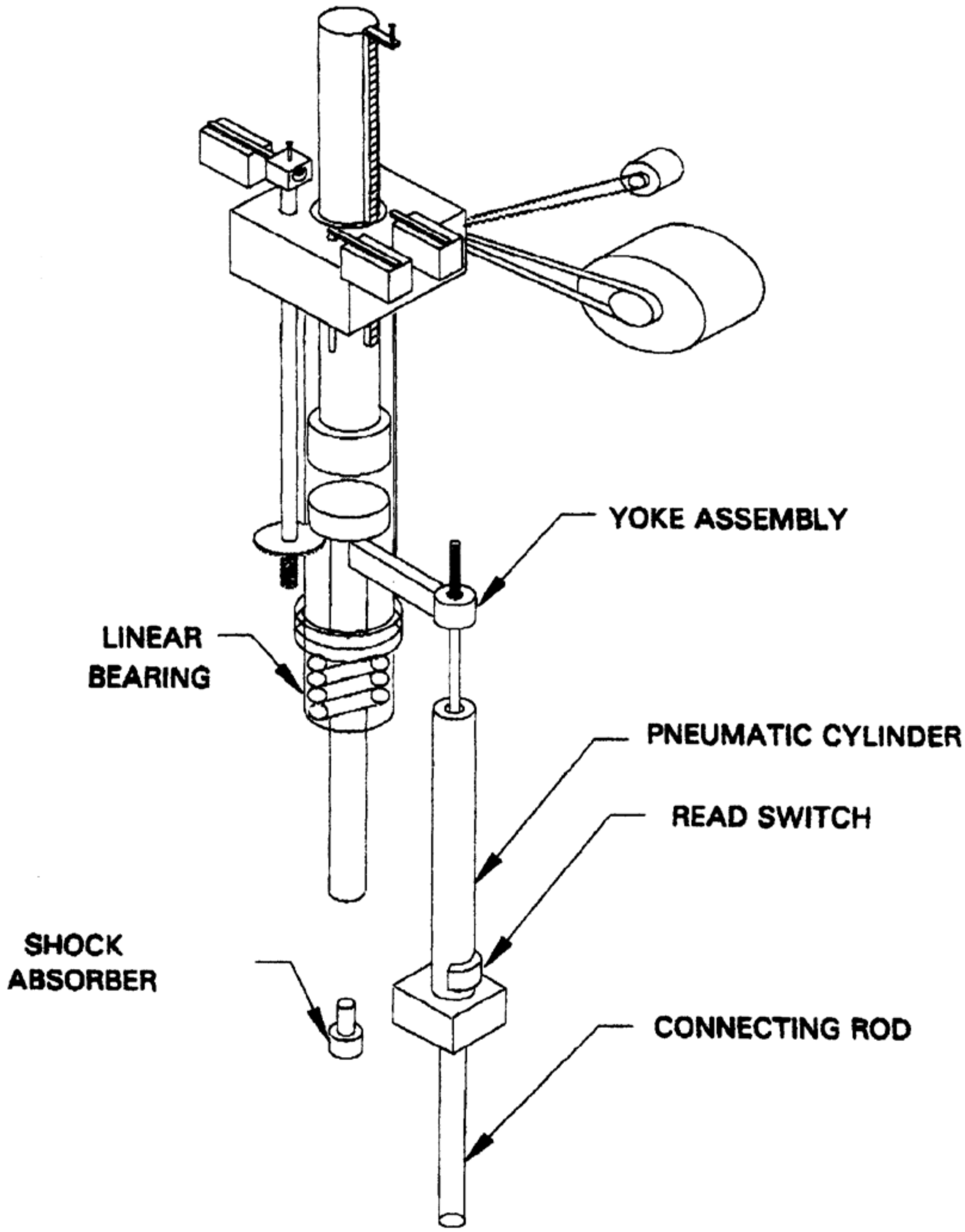


FIGURE 7.11 ADJUSTABLE FAST TRANSIENT ROD DRIVE ASSEMBLY (TYPICAL)

standard TRIGA® fast transient control rod drive, both of which have been modified. The MNRC no longer utilized the transient rod to pulse the reactor or for square wave mode. The transient rod is now only operated as essentially a standard control. Though unlike the other 5 control rods the transient rod is not fuel followed. The description of the transient rod given here is for reference purposes only. This combination transient rod drive can be used to fire low-level pulses and keep the pulse rod totally out of the core during the pulse. This combination drive unit was chosen to take advantage of the extensive operating experience gained on both the standard rack-and-pinion drive and on the standard fast transient rod drive. This combination drive unit has extensive operating experience at the Japan Atomic Energy Institute (JAERI) and Sandia National Laboratory.

The standard fast transient rod drive portion of the assembly consists of a pneumatic cylinder to drive the transient rod out of the core and a dashpot to decelerate the transient rod and drive system at the end of the stroke. The total length of the transient rod travel is 21 in. including a 6 in. deceleration length.

The pneumatic cylinder is single-acting and has a maximum stroke of 22 in. Clearance is provided to ensure that the dashpot will bottom out before the upper limit of the pneumatic cylinder. The cylinder is equipped with low friction seals so that the rod will drop freely back into the core after the transient. A piston position switch is provided to indicate when the piston, and therefore the control rod, is in the full down position. The dashpot is at the lower end of the cylinder assembly. This assembly has its own piston and bleed ports and is designed to decelerate the transient rod and eliminate any hard stops.

The standard fast transient rod drive is provided with its own pneumatic cylinder, accumulator tank, pressure regulator, and solenoid valve. The solenoid valve actuates the cylinder from the accumulator tank which may be pressurized up to 150 psi. Aluminum tubing 7/8 in. diameter is used as a connecting rod between the pneumatic cylinder, which is mounted on the rod drive bridge, and the transient rod.

The standard fast transient rod drive is thoroughly developed and tested. These drives have been installed on the TRIGA® reactors at GA Technologies Facility, University of California at Irvine, Sandia National Laboratory, University of Illinois, Japan Atomic Energy Institute (JAERI), and the dual-core TRIGA® research reactor in Romania.

The basic concept used in the UCD/MNRC drive is to modify the standard fast transient rod drive to have a portion of the piston assembly extend through the top of the drive. This extension of the transient rod piston engages a yoke mounted below the armature. The rack-and-pinion drive is mounted slightly above and to one side of the transient rod drive. When the rack-and-pinion drive is driven up, the yoke moves under the transient rod drive piston extension and moves the piston assembly, and therefore the transient rod upward with it. The rack-and-pinion drive position is read out on the console and the drive can be stopped at any position of its travel by the reactor console operator. Upon initiating a pulse, the transient rod will move until it is completely out of the reactor core. During this portion of travel, the piston rod extension will slide freely in the yoke mounted to the rack-and-pinion drive and no movement of the rack-and-pinion drive will be required.

Upon completion of a transient, both of the coupled drives will scram and the transient rod will fall completely back into the reactor core. The yoke mounted on the armature will fall about the same speed as the piston rod assembly attached to the transient drive. The rack-and-pinion drive is capable of moving and holding the transient rod at any position between full out and full in. The rack-and-pinion rod drive is capable of moving the rod approximately 15 inches of travel, the same as the travel of a standard control rod drive, and of scramming and dropping the transient rod from any position.

In order to combine the operation of the rack-and-pinion drive and the transient rod drive, the rack-and-pinion rod drive has been modified. The drive uses the same rack-and-pinion assembly, magnet and armature connection, and modified version of the lower barrel assembly as the standard control rod drive. The lower barrel assembly is shorter and contains a slot on one side for the yoke assembly. The lower barrel assembly, as modified, terminates in a large heavy flange. A bearing housing with a double set of ball bearings is bolted to the bottom of the lower barrel and an actuator shaft passes through the bearing housing. The top end of the actuator shaft contains the magnet armature, and the yoke assembly is bolted to the actuator shaft just below the magnet armature. The bearing housing provides a rigid and accurate parallel path for the entire rack-and-pinion rod drive assembly.

The entire assembly consisting of the standard control rod drive assembly, the modified lower barrel, and the bearing housing are rigidly bolted to a support which runs parallel to the transient rod air cylinder.

CHAPTER 8

ELECTRICAL DISTRIBUTION

Chapter 8 - Valid Pages
Rev. 6   06/10/20

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8.0 ELECTRICAL POWER

8.1 Introduction

The electrical power for the UCD/MNRC is supplied from a transformer located to the south of the facility. The interconnections between the transformer and the UCD/MNRC are designed in accordance with the following codes and standards:

National Electrical Code - NFPA-70;
National Electrical Safety Code;
NEMA Standards.

The design of the UCD/MNRC reactor does not require electrical power to safely shut down the reactor, nor does it require electrical power to maintain acceptable shutdown conditions.

8.2 UCD/MNRC Electrical Power System

The UCD/MNRC receives its electrical power through an underground primary 480/277 V, 3-phase, 3-wire distribution system from the nearby transformer.

As shown in Figure 8.1, the UCD/MNRC electrical power is channeled through a 480/277 V, 800 A, 4-wire, main breaker which incorporates a “UFER” ground system. This breaker feeds the facility main distribution panel, HD. The reactor system receives electrical power through the 50 kVA, 480 V, 3ϕ input 208/120 V transformer through panel 2A.

A Uninterruptible Power Supply (UPS) feeds the reactor instrumentation and control system and radiation monitoring equipment. This system is designed to provide power to the reactor console and the translator rack for approximately 15 minutes after loss of normal electrical power.

The UCD/MNRC UPS also provides power to the stack continuous air monitor (CAM), and the six facility remote area monitors (RAMs), for a minimum of four (4) hours after loss of normal electrical power.

The UCD/MNRC UPS is not needed for safe reactor shutdown or maintenance of safe shutdown conditions. It does, however, supply the necessary instrumentation so that the operator can initiate and affirm complete reactor shutdown, rod positions, and power level. More importantly, it supplies radiation monitoring equipment with power so that radiation levels are known.

The electrical power for the UCD/MNRC reactor’s primary, makeup, and purification water systems, as well as the pool and reactor “on” lights, is supplied from panel 2B.

The facility air handling and exhaust systems are fed through panels 2AC and 2A.

A propane generator provides backup power to the reactor room ventilation system (Chapter 9).

Two other UCD/MNRC systems, fire alarm and security, are equipped with their own UPSs. The battery packs for both of these systems are capable of maintaining normal operations for 24 hours after loss of normal power.

The reactor/radiation instruments receive their power from a regulated power supply that meets a commercial grade standard.

8.3 UCD/MNRC Raceway System

The UCD/MNRC raceway system consists of the conduit runs, cable trays, pull boxes, and fittings that contain all power, instrumentation, and control wiring associated with the reactor. Cabling originating in (detectors) or above the tank (control rod drives) is routed either along the tank wall or under the bridge to the reactor room cable trench. A raceway contains the cables between the cable trench and the NM-1000 and the NPP-1000 (Chapter 7). Separate conduit runs have been provided between the reactor room and the control room for reactor control and instrumentation wiring. The routing is such that there are two independent paths giving physical isolation. That is, the reactor-instrumentation wiring is designed so that one control and one safety-instrumentation channel takes one path. Additionally, the other control and safety-instrumentation channel is contained in the other path. The control wiring for control-rod drives is split in the same manner.

Since controls are not required for safe shutdown, no special fire-protection system is required for the raceway system.

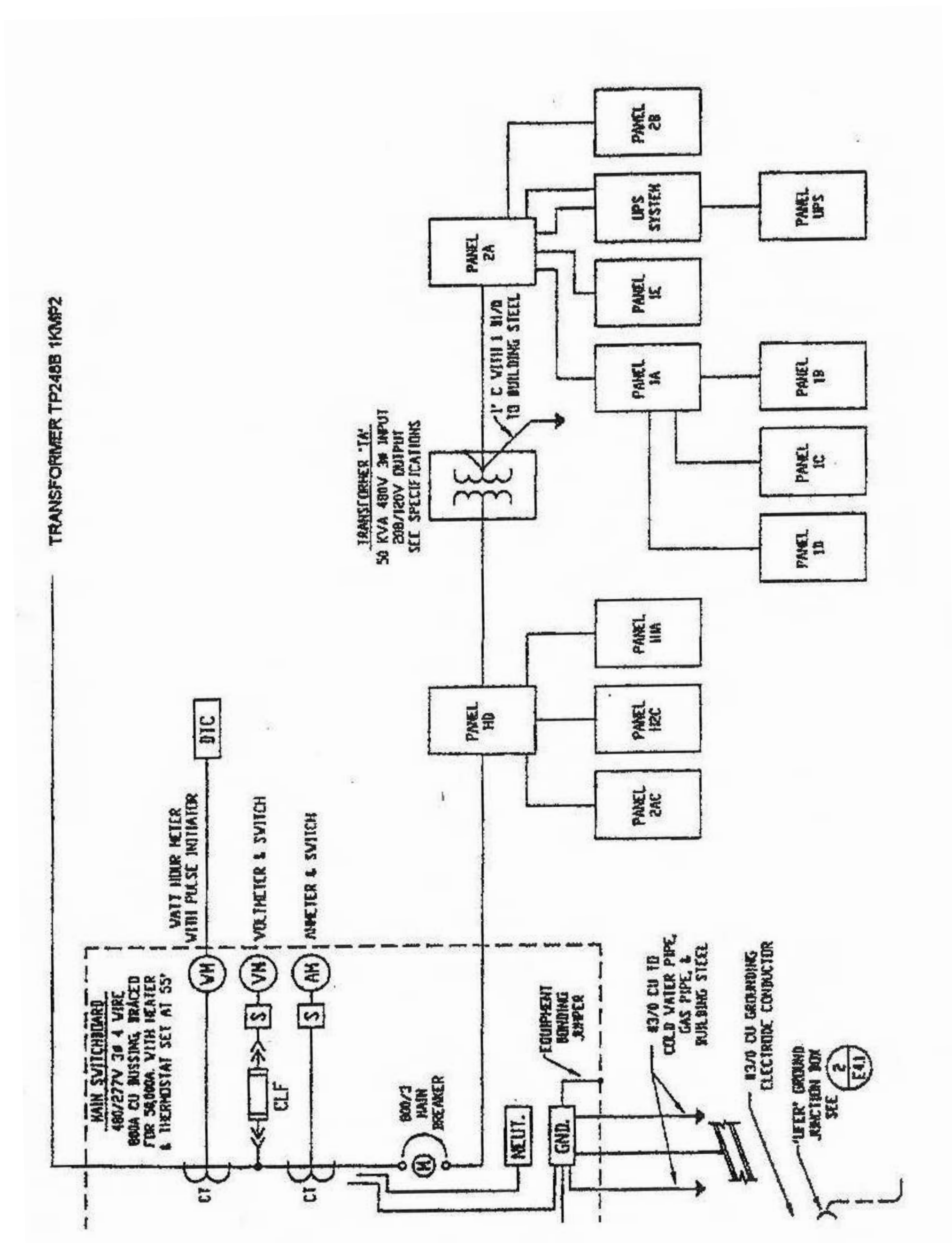

FIGURE 8.1 UCD/MNRC ELECTRICAL DISTRIBUTION SYSTEM - SINGLE LINE DIAGRAM

CHAPTER 9

AUXILIARY SYSTEMS

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9.0 AUXILIARY SYSTEMS

This chapter discusses the auxiliary systems that support the UCD/MNRC operation.

9.1 Fuel Storage and Handling

The fuel loading for the UCD/MNRC reactor will consist of approximately 100 fuel elements, including up to five control rods, one transient rod, and graphite elements. Fuel elements can be stored in the reactor tank and/or storage pits in the reactor room floor to facilitate burn-up management or, when spent, until such time that they can be shipped to a repository facility.

Basically, the fuel handling cycle within the UCD/MNRC consists of (1) receiving fresh fuel packaged in accordance with DOT transport requirements, (2) storing fuel in one or more of the three authorized fuel storage locations, (3) transferring the fuel elements into in-tank storage racks by use of the fuel element handling tool, (4) unloading spent fuel elements from the reactor grid into the in-tank storage racks, (5) loading the fuel elements from the in-tank storage racks into the reactor grid, (6) repositioning fuel elements within the reactor grid, (7) interchanging fuel elements between the reactor grid and the in-tank storage racks, (8) transferring irradiated fuel elements from the reactor in-tank storage rack by use of the fuel storage pits in the floor of the reactor room, and (9) transferring fuel from either the storage pits or in-tank storage racks to a shipping cask for removal.

9.1.1 Bay 2 Fuel Storage Area

9.1.1.1 Design Basis

a. All SNM stored in the Bay 2 Fuel Storage Area shall be in NRC/DOT shipping containers approved at the time of receipt for the contained SNM. All SNM stored in the Bay 2 Fuel Storage Area shall have a k-effective value less than 0.9 for any stored configuration.

b. The Bay 2 Fuel Storage Area shall comply with the applicable NRC security requirements.

9.1.1.2 Facility Description

The Bay 2 Fuel Storage Area shall be located in the lower portion of the Bay 2 escape trunk. Normal entry into Bay 2 is through one of two massive shield doors. In order to reduce the scattered thermal neutron flux in the Bay 2 fuel storage area, a one-inch thick borated polyethylene shield has been placed across the front of the storage area.

9.1.1.3 Safety Evaluation

All SNM stored in the Bay 2 Fuel Storage Area shall be in NRC/DOT shipping containers approved at the time of receipt for the contained SNM. The design basis 9.1.1.1 (a) may be derived from either an approved NRC/DOT certificate of compliance, an established transportation index, or calculation.

9.1.2 In-Tank Fuel Storage

9.1.2.1 Design Basis

a. The in-tank fuel storage racks are designed with sufficient spacing between fuel elements to ensure that the array, when fully loaded, will be substantially subcritical.

b. The in-tank fuel storage racks have a combined capacity for storage of nearly an entire core of irradiated fuel elements with one fuel element per storage hole.

c. The in-tank fuel storage racks are mounted on the inside of the reactor tank and deep enough below the water surface to provide adequate radiation shielding.

d. The in-tank fuel storage racks are designed and arranged to permit efficient handling of fuel elements during insertion, removal, or interchange of fuel elements.

9.1.2.2 Facilities Description

Three in-tank aluminum fuel storage racks, with a combined capacity to accommodate 60 irradiated fuel elements are provided (Figure 9.1). The in-tank fuel storage racks are located at the outer edge of the reactor tank (Figure 9.2). Each rack has two levels with storage space to accommodate 20 fuel elements.

The fuel elements are loaded into the in-tank fuel storage racks from above. Each storage hole has adequate clearance for inserting or withdrawing a fuel element without interference. The weight of the fuel elements is supported by the lower plates of the racks.

Each in-tank fuel storage rack is securely hung from the top of the reactor tank by two 3/4- in. diameter aluminum rods. These rods are secured to the tank flange and the rack by threaded fasteners. This mounting arrangement prevents the racks from tipping or being laterally displaced.

9.1.2.3 Safety Evaluation

Within a fuel storage rack, control of spacing is not actually required to limit the effective multiplication factor of the array (k_{eff}). The in-tank fuel storage racks are configured such that criticality is not possible (Reference 9.1).

Based on the fact that the storage racks are limited to 20 elements, there should be no effect on the criticality conditions, since even with the heaviest elements (i.e., all 30/20), 60 elements are required to go critical with optimal geometry and moderation.

Furthermore, Reference 9.1 shows that 2 racks of 8.5 wt% fuel stored back to back are subcritical (i.e., k_{eff} = 0.74 for twice the ^{235}U mass). While the 30/20 fuel increases the ^{235}U mass by ~4.00, it contains erbium, causing the 30/20 fuel to have a reactivity more similar to an 8.5 wt % fuel element. Therefore, there should be no effect on the criticality of the system. In the unlikely event of loss of reactor tank coolant water, the loss of the water moderator would increase the safety margin by reducing the k_{eff}. The in-tank fuel storage racks are made of aluminum and are designed to withstand a UBC Zone 3 earthquake with importance factor 1.5, when fully loaded.

The in-tank storage racks are bolted to the upper tank flange and will resist a limited pull-up force in the event that the fuel element or handling equipment becomes fouled during handling operations.

Seismic analysis performed on the in-tank storage racks could not substantiate conclusively the survival of the racks post a design basis seismic event. Therefore, analysis was performed to ascertain the survival of a fuel element dropped from the fuel rack location and impacting on the bottom of the tank (Reference 9.2). This analysis predicts survival of a fuel element under all impact conditions. Clumping of fuel elements from all three fuel storage racks simultaneously into a critical matrix after falling to the tank bottom is considered incredible.

9.1.2.4 Inspection and Testing

The in-tank fuel storage racks will be visually inspected during installation to check that they are not deformed and that all fasteners are tight and in place.

[redacted]

FIGURE 9.1 TYPICAL IN-TANK FUEL STORAGE RACK

[redacted]

FIGURE 9.2 POTENTIAL IN-TANK FUEL STORAGE RACK LOCATIONS

9.1.3 Spent Fuel Storage

9.1.3.1 Design Basis

The spent fuel storage pits are designed with sufficient spacing to ensure that the array, when fully loaded, will be substantially subcritical.

The spent fuel storage pits are designed to withstand earthquake loading to prevent damage and distortion of the pit arrangement.

The spent fuel storage pits have a combined capacity for storage of 190 irradiated fuel elements with 38 fuel elements per storage pit (Reference 9.5).

The spent fuel storage pits are fabricated from materials compatible with the fuel elements and provide adequate personnel shielding.

The spent fuel storage pits are designed and arranged to permit efficient handling of fuel elements during insertion or removal of fuel elements.

The spent fuel storage pits have shield plugs that can be locked in place.

9.1.3.2 Facilities Description

Five spent fuel storage pits, with a combined capacity to accommodate 190 (38 each) irradiated fuel elements, are located in the floor of the reactor room (Figure 9.3). Each pit has a liner and a lead-filled shield plug that will be locked in place when fuel is not being moved into or out of the pits. The pits have racks with holes for holding fuel elements (Figure 9.4). Each hole in the rack can only hold one fuel element. All storage pit material (liners, racks, plug casing, and pipes) that may contact either the fuel elements or the pit water are fabricated from aluminum or 304 stainless steel. This is the same type of material as used for the fuel element cladding and end fittings.

The fuel elements are loaded into the racks from above. Each hole in the rack has adequate clearance for inserting or withdrawing a fuel element without interference. However, with a fuel element in place in a hole, additional fuel elements cannot be inserted into the same hole. The weight of the fuel elements is supported by the lower plates of the racks which are, in turn, supported by the pit liners.

Each rack is designed so that it is constrained by the pit liner and cannot tip or become laterally displaced.

The storage pits are equipped with a cooling water system that will be used if required by the stored fuel elements. This system is described in Chapter 5.

[redacted]

FIGURE 9.3 FUEL STORAGE LOCATION

[redacted]

FIGURE 9.4 FUEL STORAGE PIT/RACK

Analysis shows that 19 fuel elements that have been in the core operating at 200 kW (one- tenth of the core power) can be removed from the reactor tank after one day and can be safely stored in a single fuel storage pit either with or without water (dry) (References 9.3 and 9.4). Since in a core operating at 1 MW power, the initial residual heat generation will be significantly higher, at least 5 days of decay must be allowed prior to transferring 19 fuel elements from the pool to the storage pit. This will allow for the residual thermal load to decrease to a level equivalent to that associated with the prior analysis conducted for a core operating at 200 kW and provides a margin of safety so that temperatures remain within analyzed safety limits. Elements are typically decayed in tank for months to years before moved to dry storage.

Reference 9.4 shows the expected radiation levels at the floor level with 38 irradiated elements in a single storage pit that have been in the core operating indefinitely at 1 MW, cooled for 24 hours, will be below one (1) µR/hr if the pit is filled with water or the lead shield plug is in place. Without water or the lead plug, the radiation level at floor level is predicted to be 15 rem/hr.

9.1.3.3 Safety Evaluation

Within a fuel storage pit filled with water, control of spacing is not required to limit the effective multiplication factor of the array (K_{eff}). An analysis shows the largest K_{eff} for a pit is approximately 0.93 when all five pits are loaded to capacity with fresh 8.5 wt% fuel elements (38 each) and are full of water (0.45 when dry) (Reference 9.5). Since both 20/20 and 30/20 fuel contains erbium, they are similar in reactivity to 8.5 wt %, and there should be no significant changes to the criticality of the storage pits. Besides, 38 elements are ~2/3 the number required for criticality under any conditions. Also, radiation levels at the reactor room floor level with either water in the storage pits or the lead plug in place are below two (2) µr/hr,(see above). Exposures during fuel handling are discussed in Section 9.1.4. The MNRC has never introduced water into these spent fuel storage tanks and will likely never do so. A more complete analysis of the resulting k_{eff} from an actual configuration (including fuel burn up) will be performed before any fuel pit is flooded with water to demonstrate a k_{eff} less than 0.90.

The spent fuel storage pits are designed to withstand horizontal and vertical accelerations due to earthquakes. Stresses in a fully loaded storage pit will not exceed stresses specified by the UBC Zone 3, importance factor 1.5, seismic criteria.

9.1.4 Fuel Handling System

9.1.4.1 Design Basis

The fuel handling system provides a safe, effective, and reliable means of transporting and handling reactor fuel from the time it enters the UCD/MNRC facility until it leaves.

9.1.4.2 Equipment Description

9.1.4.2.1 Fuel Handling Tools

Tools are provided for handling individual fuel elements and for manipulating other core components. Individual fuel elements are handled with a flexible or rigid handling tool (Figure 9.5). The fuel element handling tool utilizes a locking ball-detent grapple to attach to the top end fitting of a fuel element. [redacted]

9.1.4.2.2 Overhead Handling Systems

The reactor room has an electrically driven 5-ton overhead bridge crane. The crane is dual speed and pendant controlled and has provisions for locking the controls when it is not in use. The preparation area has an electrically driven 5-ton overhead monorail hoisting system. The monorail system is positioned directly above the preparation room floor access doors. These hoisting devices have been designed, fabricated, installed, and initially load tested in accordance with OSHA 1910.184.

The reactor room crane and the preparation room monorail system are operated in accordance with the ANSI B30.11, Monorail Systems and Underhung Cranes, prior to fuel cask handling. In addition, any slings required to transfer the fuel cask will be used in accordance with 29 CFR Part 1910.184, Slings.

9.1.4.2.3 Fuel Transfer Cask

A shielded fuel transfer cask is used to transfer irradiated fuel elements from the reactor tank to the spent fuel storage pits or to a shipping cask. The fuel transfer cask is both top and bottom loading and will hold either one fuel element, an instrumented fuel element, or a fuel-followed control rod (Figure 9.6). The structural components are fabricated from stainless steel with a lead filler. The radiation exposure to operating personnel is about 5 mr/hr (gamma) at the outer surface of the transfer cask when it is loaded with an irradiated fuel element that has been allowed a six-month cooling time after operating in the highest flux region of the core for one year at one megawatt power (Reference 9.6). The cask internals that contact the fuel are fabricated from stainless steel. Cask-lifting lugs have been designed using the ASME code for analysis guidelines. This analysis shows that the maximum load on the lifting lug to cask weld is less than 1000 lb/in. of weld when the entire weight of the cask is on one lug (Reference 9.7). The allowable load for this weld is 6360 lb/in. of weld, a margin of greater than six even with the conservative assumption that all weight is on one lug. The cask lugs have been load tested in accordance with NE F8-6T.

[redacted]

FIGURE 9.5 FUEL ELEMENT HANDLING TOOL

[redacted]

FIGURE 9.6 FUEL ELEMENT TRANSFER CASK

9.1.4.2.4 Cask Positioning Plate

The cask positioning adapter supports and locates the transfer cask above the fuel storage pit. This adapter can be indexed so that the center bore of the cask can be located directly above the hole in the fuel storage rack designated to receive the fuel element being transferred.

A closed circuit television camera and light are located on the lower side of the adapter plate. A television monitor is used to observe the lowering, or retrieval, of an element either into or from the storage pit rack.

9.1.4.3 Description of Fuel Transfer

The fuel handling system provides a safe and effective means for transporting and handling the reactor fuel from the time it enters the boundaries of the UCD/MNRC facility until it leaves.

Previous sections described and listed the major pieces of equipment and the methods that are used in fuel handling. The following paragraphs describe the steps during fuel handling.

Unirradiated fuel arrives at the facility in Nuclear Regulatory Commission/Department of Transportation approved shipping containers. The fresh fuel, [redacted] lbs per element, is removed from the shipping containers by hand and stored until needed.

All handling of fuel within the reactor tank is accomplished by use of the fuel element handling tool, [redacted]

The reactor room overhead crane and the fuel transfer cask are used to transfer irradiated fuel elements between the in-tank storage racks and the fuel storage pits.

The reactor room overhead crane is used to position the fuel transfer cask in the reactor tank such that the cask top is approximately 9 ft below the water level. With the lead plug removed, the fuel element handling tool is used to lift an irradiated fuel element from an in-tank fuel storage rack and place it into the cask. The reactor room crane is used to raise the cask out of the tank and transport it to a position over a fuel storage pit. Using the fuel element handling tool, the fuel element is raised from the bottom door of the transfer cask, allowing the bottom door to be opened. The fuel element is lowered out of the cask into its storage location. The TV camera is used to monitor the lowering operation and to confirm that the element has been placed in the proper rack location. The reverse operation is used to remove an irradiated element from the storage pits and place it in the fuel transfer cask. Appropriate radiation monitoring by health physics personnel will be conducted during the preceding operation in order to assure that doses are kept as low as reasonably achievable.

An approved shipping cask will be used to transport irradiated fuel elements from the UCD/MNRC to a reprocessing or long term storage location. Much of the same equipment, described above, used to transfer an irradiated fuel element from the in-tank storage racks to the UCD/MNRC storage pits, will be used to place an irradiated fuel element in the shipping cask.

The first step in this operation is to load an irradiated fuel element into the fuel transfer cask, as described above, from the storage pits or in-tank storage racks. The reactor room crane is used to move the cask to an area near the reactor room/preparation room door. The fuel transfer cask is then loaded on a pallet truck, rated capacity of 6,500 lbs. This pallet truck is pushed from the reactor room into the preparation room. The preparation room monorail system is used to position the cask over the opened preparation room floor access doors. The fuel element transfer cask is lowered through the access doors and mated to the top of the shipping cask. The shipping cask, mounted on a trailer, will be positioned in the staging area directly below the preparation room floor access doors.

The transfer of the irradiated fuel element from the fuel element transfer cask to the shipping cask is the same for the transfer of a fuel element from the transfer cask to the storage pits.

9.1.4.4 Safety Evaluation

All parts of the cask and crane system are rigorously maintained, including load tests and radiographic or dye penetrant inspections as appropriate. Therefore, the dropping of the transfer cask during fuel transfers is considered a highly unlikely event.

If, however, a cask drop accident should occur, the event is considered enveloped by the evaluation presented for the single fuel element dropped in air accident (MHA), since that evaluation conservatively assumed a ground level release of material.

9.2 Helium Supply System

A system to inert the beam tube sections with helium has been provided. Replacing the air with helium reduces the neutron beam attenuation, i.e., scattering, resulting in a more intense and purer beam for radiography.

The helium system is essentially a static system, i.e., once a helium environment is established, helium is only added to or exhausted from the system to compensate for temperature and barometric related pressure changes or to make-up helium lost by leakage (Figure 9.7).

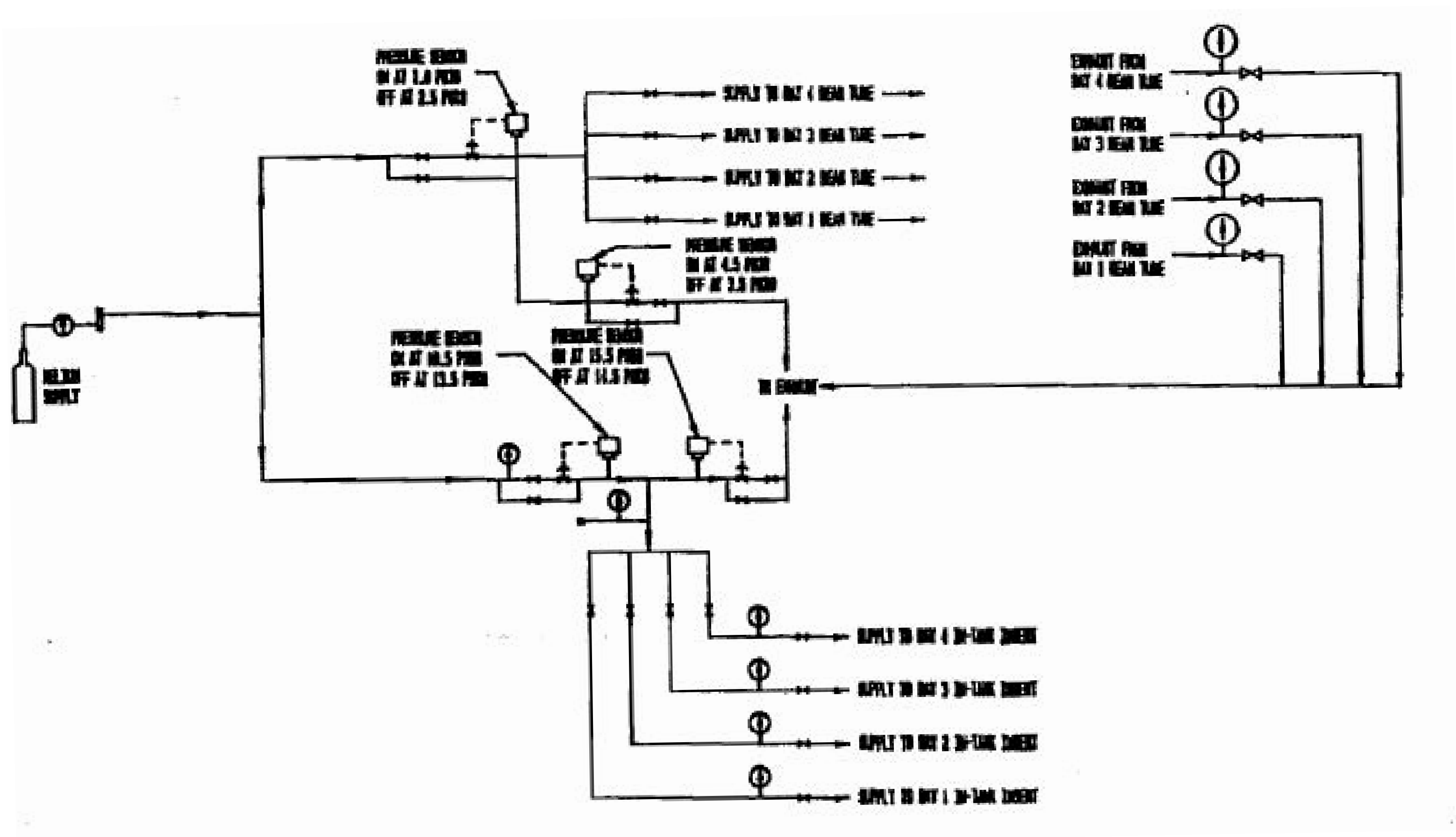

FIGURE 9.7 TYPICAL HELIUM SUPPLY

The supply for this system is from one standard helium bottle and regulator. Downstream of the regulator the system splits into two subsystems. One of these subsystems controls the pressure in the bulk shielding sections of the beam tubes. The other subsystem controls the pressure in the in-tank and tank wall sections of the beam tubes to prevent entrance of water. The venting from these subsystems is to the pneumatic transfer system exhaust duct.

The method of pressure control for both subsystems is identical. Both subsystems contain helium makeup and vent valves that are activated by pressure sensors. The valves are sequenced to maintain a positive pressure within the subsystems.

There must be multiple equipment failures to overpressure the system. To overpressure the in-tank and tank-wall sections, both the pressure regulator and supply valves must fail open so that the supply capacity exceeds the vent capacity. For the in-tank and tank-wall sections, both the supply valve and pressure relief valve must fail.

9.3 Building Water Systems

The water supply for the UCD/MNRC is connected to the on-site 12 in. combination fire and domestic water main. The UCD/MNRC water supply system consists of a water meter, main shutoff valve, and facility distribution piping.

The potable water requirements for the UCD/MNRC are minimum. The main users are the wash/change rooms, heating and ventilating units, and the cooling tower.

Water from the radiography bays and the decontamination shower and sink is pumped to a 2000 gallon tank. Water collected in the tank will be sampled for radioactivity and disposed of in accordance with 10 CFR 20.

9.3.1 Auxiliary Make-Up Water System (AMUWS)

The auxiliary make-up water system (AMUWS) can supply water to the reactor core from a source external to the domestic water supply (Figure 9.8). Water is supplied from two storage tanks located below the secondary cooling tower. Each tank contains approximately twenty-three hundred (2300) gallons of deionized water. The water storage tanks have enough capacity, if needed, to supply water to the reactor core area for approximately four hours at twenty gallons per minute as a backup supply to the ECCS. Water purity is maintained by a set of resin columns located next to the storage tanks.

A control switch, located on the temperature control panel (TCP), enables the reactor operator to start a three horsepower pump from the reactor control room. The pump can supply water to the reactor tank at a flow rate of twenty gallons per minute. A flow proof light illuminates on the TCP when flow has been initiated through the AMUWS.

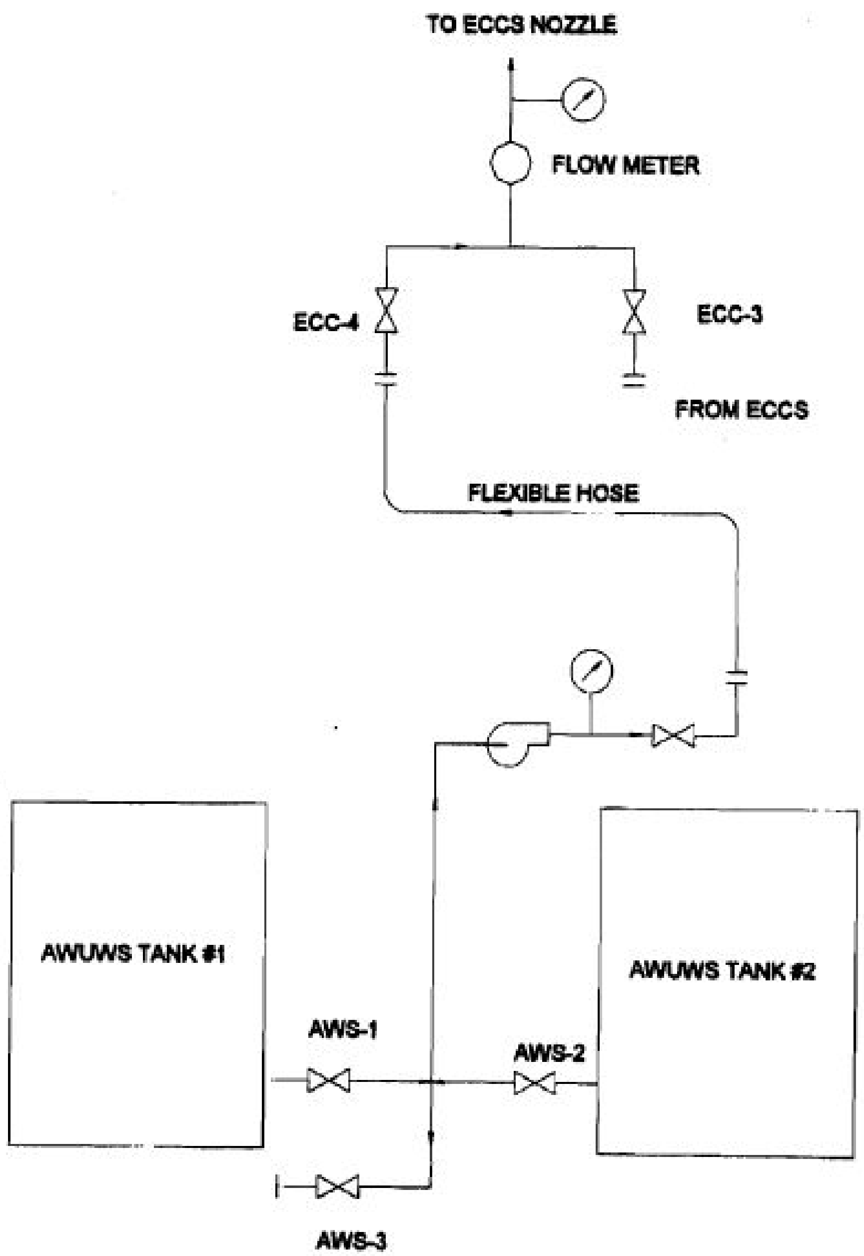

FIGURE 9.8 AUXILIARY MAKEUP WATER SYSTEM (AMUWS)

Normally the AMUWS piping is dry and will only be filled with water when the pump is started by the reactor operator from the control room. Check valves located in the reactor room prevent water from siphoning from the reactor tank back into the storage tanks when the system is in the stand-by mode of operation.

A propane electrical generator supplies back-up electrical power to the AMUWS pump, the TCP, the reactor room ventilation fan (EF-1), and the damper controls for the reactor room in the event that normal electrical power is lost. A light on the TCP indicates if the generator is operational.

The AMUWS contains pressure and flow gauges to verify sufficient water flow is maintained for the duration of its use (Section 13.2.3.2.2).

Note the AMUWS is not considered an engineered safety feature as it is not required to prevent fuel from exceeding its safety limit in the event of an instantaneous at power loss of coolant accident.

9.4 Fire Protection

9.4.1 Design Basis

The design basis for the UCD/MNRC fire protection system is to provide a detection and suppression capability which will mitigate any losses to property should a fire develop. It should be noted that fire protection is not required to accomplish a safe shutdown of the reactor or to maintain a safe shutdown condition.

9.4.2 Description

Both detection and suppression systems installed in accordance with National Fire Protection Code are utilized in the UCD/MNRC.

A dry-pipe, pre-action fire sprinkler system provides fire suppression for the UCD/MNRC as shown in Figures 9.9 and 9.10. This system receives its water supply from the existing on-site 12-in. combination fire and domestic water main. Also, a fire hydrant is located approximately 150 ft from the UCD/MNRC.

In addition to the dry-pipe system, the DAC in the reactor room, and the instrument cabinets and control consoles in the reactor and radiography control rooms contain fire detection and halon suppression systems, i.e., units located within the enclosures.

The entire UCD/MNRC has either thermal or ionization-type fire detection devices as well as manual pull boxes. Thermal detectors located in select air handling system ducts shut down the system when activated (Section 9.5).

The UCD/MNRC fire detection/suppression system is automatic, zoned, and is supervised with hardwired signal connections. The system has a self-contained 24-hr battery backup. There are two master panels: one is located in the control room; the other panel is outside near the vehicle gate. The master panel provides local alarm information and transmits signals to a 24-hour monitored location.

Whenever one of the fire detection devices activates, visual and audible warning devices alarm throughout the facility.

9.4.3 Evaluation

The UCD/MNRC fire protection system has been designed to meet the design basis. The dry-pipe suppression provides coverage of the critical areas and the detection system covers the entire structure. Special halon systems have been provided to protect instrumentation and control cabinets/consoles.

[redacted]

FIGURE 9.9 UCD/MNRC FIRE SUPPRESION SYSTEM, MAIN FLOOR

[redacted]

FIGURE 9. 10 UCD/MNRC FIRE SUPPRESION SYSTEM, SECOND FLOOR

9.5 Air Handling System

9.5.1 Design Basis

The UCD/MNRC air handling system provides heating and cooling for personnel comfort and serves the following important roles for radiological control:

- Maintenance of pressure differentials throughout the facility to limit spread of radioactive contamination;
- Provision of air changes in the reactor room and other areas throughout the facility to maintain Ar-41 and N-16 concentrations within the limits defined in 10 CFR Part 20;
- Provision of a means to isolate, recirculate, and filter the air in the reactor room, should there be a release of fission products or other abnormal airborne radionuclides.

9.5.2 Description

Ventilation throughout the UCD/MNRC facility has been designed and balanced so that the reactor room and radiography bays are at a slightly negative pressure with respect to their surrounding areas. The facility's air handling system can be broken down into four subsystems that handle air management within the following areas of the building:

1. General building spaces
2. Radiography bays
3. Reactor equipment room
4. Reactor room

The air handling system for the general UCD/MNRC building spaces services all areas except the reactor room, reactor equipment room, and four radiography bays. Ventilation in these general building areas is designed to control air temperature and maintain positive building air pressure with respect to the reactor room and radiography bays to prevent the spread of any radioactive contamination. A summary of the UCD/MNRC air handling systems for the reactor room, reactor equipment room, and radiography bays is shown in Figure 9.11, and these systems are described further in the following paragraphs.

Ventilation in the radiography bays is designed to provide the following:

- Adequate ventilation in the radiography bays to maintain the Ar-41 concentrations within the limits set forth by 10 CFR Part 20;
- Maintenance of negative air pressure with respect to adjacent staging areas to prevent the spread of radioactive contamination;
- Isolation of the radiography bays upon detection of radioactive contamination;
- Control of individual radiography bay air temperatures.

During radiographic operations, the bay doors are closed, and the bays are maintained at a negative air pressure in relation to their exterior staging areas by exhausting the air within.

Combination heating and cooling air conditioning units (AC-3, AC-4, AC-5, AC-6, and AC-7) provide air circulation within each of the bays, and an exhaust fan (EF-2) draws air through a vent located near the floor in each bay. The combined exhaust air from all four radiography bays is filtered through a standard particulate filter before being combined with filtered exhaust from the fume hoods in the reactor equipment room and sent on to the facility's 60-foot high effluent exhaust stack. Air from the bays is monitored for radioactivity by argon, particulate, and noble gas detection channels in the stack's continuous air monitor (stack CAM), and each radiography bay's exhaust line contains a motor-operated damper (D-9, D-10, D-11, and D-12) that can be closed for isolation.

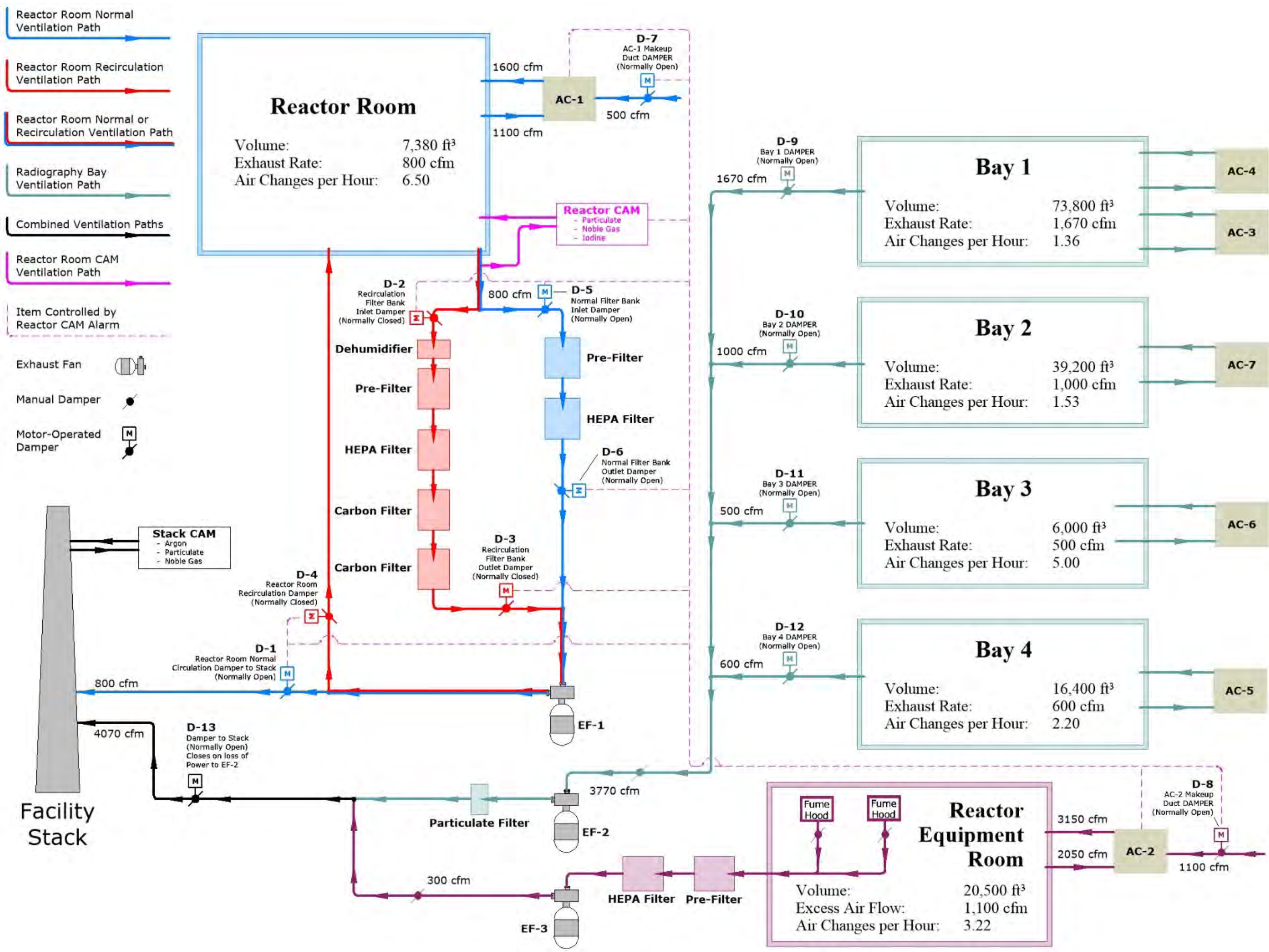

FIGURE 9.11 UCD/MNRC AIR HANDLING SYSTEM

The radiography bay air handling system will normally operate whenever the reactor is operating. However, it is permissible to operate the reactor without running this part of the air handling system, because the impact of not running this ventilation system while the reactor is operating is not significant relative to occupational or offsite doses. This subject is discussed in detail in Chapter 11 and Appendix A.

Ventilation in the reactor equipment room is designed to provide the following:

- Maintenance of positive air pressure with respect to the reactor room to prevent the spread of radioactive contamination;
- Fume hood ventilation for facility operations and pneumatic transfer system utilization;
- Control of reactor equipment room air temperature.

Reactor equipment room supply air is provided by a combination heating and cooling air conditioning unit (AC-2), which supplies more air to the room than it removes. This results in a positive pressure in the equipment room with respect to surrounding spaces, including the reactor room, minimizing air flow from the reactor room into the equipment room, even if the reactor room loses negative pressure during isolation and recirculation or from exhaust fan failure.

The reactor equipment room also houses two fume hoods. One fume hood contains the receiver for the UCD/MNRC Pneumatic Transfer System (described in Chapter 10, section 10.4.4), and the other is used for general facility operations. An exhaust fan (EF-3) is used to draw air through the fume hoods and then through a pre-filter and HEPA filter. This exhaust fan keeps the hoods at a negative pressure with respect to the surrounding room and maintains hood face air velocity at approximately 150 feet per minute when is in use. After filtration, exhaust air from the fume hoods is combined with the exhaust air from the radiography bays and routed to the facility's effluent stack, where it is monitored by the stack CAM. The air drawn from the reactor equipment room by the fume hoods is less than the excess air flow provided to the room by AC-2, so positive air pressure with respect to the reactor room is not compromised by fume hood operation.

The reactor room ventilation system is designed to provide the following:

- Adequate exhaust and/or dilution to maintain Ar-41 concentrations in the reactor room and outside of the MNRC facility at levels less than those specified in 10 CFR Part 20;
- Adequate exhaust to maintain the reactor room air pressure negative with respect to adjacent spaces to prevent the release of airborne radioactive contamination during normal operating conditions;
- Isolation of the reactor room, with dehumidification, filtration, and recirculation of reactor room air in the event of airborne radioactive contamination;
- Control of reactor room air temperature.

The reactor room ventilation system has two modes of operation: normal and recirculation.

During normal operation, reactor room supply air is provided by a combination heating and cooling air conditioning unit (AC-1), and reactor room air is filtered with a pre-filter and HEPA filter before being discharged to the effluent stack by an exhaust fan (EF-1). The exhaust fan maintains the reactor room air pressure negative with respect to the reactor equipment room and other exterior areas during normal operations by providing an exhaust flow rate that is greater than the air conditioning unit supply rate (sourced through AC-1's makeup duct).

During reactor operation, the reactor room air is monitored for radioactive airborne contaminants by particulate, noble gas, and iodine detection channels in the reactor room's continuous air monitor (reactor CAM), which samples the exhaust effluent from the reactor room prior to filtration. Upon detection of airborne radioactivity above a preset level, the system automatically isolates the reactor room and enters recirculation mode. In this mode, reactor room air is drawn through a separate filtration system (still using exhaust fan EF-1) before being returned to the reactor room in a continuous loop that persists until the reason for alarm is resolved and the reactor CAM alarm has been reset. The recirculation mode's filtration system includes a dehumidifier, pre-filter, HEPA filter, and two carbon filters that are all connected in series. The pre-filter and HEPA filter are similar to those used for normal operations, and if the airborne radioactive contamination is accompanied by steam or moisture, the dehumidifier will dry the air to protect the HEPA and carbon filters.

During recirculation, the reactor room's air conditioning unit (AC-1) is shut down and the damper in its makeup duct (D-7) is closed. This action prevents the reactor room from being pressurized by the unit. In addition, the reactor equipment room air recirculation and conditioning system (AC-2) is prevented from being shut down so that the area adjacent to the reactor room is maintained at a slightly positive pressure with respect to the reactor room, reducing the potential for contamination spread.

A total of seven dampers and two air conditioning unit interlocks operate automatically to achieve reactor room isolation. The damper in the exhaust duct to the stack (D-1) closes. The normally-closed dampers that typically isolate the reactor room's recirculation ducting and filtration system from the reactor room air flow path (D-2, D-3, and D-4) are opened. The normally-opened dampers that typically allow air to flow through the reactor room's normal air filters (D-5 and D-6) are closed. The normally-opened damper in the AC-1 makeup duct (D-7) is closed and AC-1 is shut down, and AC-2 is prevented from being shut down. The net effect of these actions is the ventilation path shown in red in Figure 9.11, where reactor room air is drawn through a dehumidifier, pre-filter, HEPA filter, and two carbon filters by EF-1 before being returned to the reactor room, and the adjacent reactor equipment room air pressure is maintained slightly positive with respect to the isolated reactor room.

To return the reactor room ventilation system to normal operation, the reactor CAM alarm must be cleared, and the reactor CAM must be reset via the CAM RESET button on the temperature control panel (TCP) in the reactor control room. The CAM RESET button restores the interlocked dampers to their normal configuration, allowing filtered reactor room exhaust air to discharge to the atmosphere through the stack. The CAM RESET also restarts AC-1 and enables AC-2 to be controlled at the TCP.

During a loss of coolant accident (LOCA), the radiation levels in the reactor room could cause the reactor CAM to alarm and force the reactor room ventilation system into recirculation mode. Chapter 13 discusses specific situations in which reactor room ventilation must remain in the normal operating mode during a LOCA. A ventilation damper control switch located on the TCP in the reactor control room enables the reactor operator to override the damper controls for recirculation and continue exhausting air from the reactor room through the normal exhaust path.

9.5.3 Evaluation

The UCD/MNRC air handling system has been designed to maintain the reactor room consistently negative with respect to the air pressure in the surrounding areas. It provides the necessary air changes in the reactor room to maintain routine radioactive gas concentrations at a level where the 10 CFR Part 20 dose limits will be easily met. It also provides a means for isolating the reactor room and recirculating the room air through HEPA and carbon filters, should there be a release of fission products or other abnormal airborne radionuclides.

The air handling system will also maintain the radiography bays at a negative pressure relative to surrounding areas when the radiography bays ventilation system is operating, which is the normal operating mode for the UCD/MNRC facility.

9.6 Interlocks/Controls - Bay Shutters/Doors

Each of the UCD/MNRC radiography bay shutters (bulk shield) and the bay doors are equipped with controls incorporating interlocks to prevent personnel from entering the bays anytime the reactor is on and shutters are not closed. In addition to the shutter and door interlocks, there are reactor shutdown devices that will either scram the reactor or prevent it from being operated if an unsafe condition exists. The following sections describe the controls, interlocks, and reactor shutdown devices in detail.

9.6.1 Shutter (Bulk Shield) Control/Interlocks

Figure 9.12 is the Bay 2 shutter control/interlock schematic and Figure 9.13 shows the corresponding limit switches. The controls/interlocks for all four shutters are identical except for the number of bay doors. Bays 1, 3, and 4 have one door each while Bay 2 has two doors.

The shutter can be controlled from three locations, the radiography control room and two locations in the bay. One of the bay shutter control stations is located on the parapet next to the shutter, and the other is on the bay floor in the area of the motor control center.

The logic diagrams for shutter operation from the radiography control room and from the bay are shown in Figures 9.14 and 9.15, respectively.

The key features of this control/interlock system are as follows:

1. The shutter movement can be stopped at any time from any of the three shutter control stations;

2. The shutter can be closed at any time (except when a "stop" switch is depressed) from either of the two control stations located in the radiography bay;

3. There is a keyswitch associated with the radiography control room and the bay floor control stations. These keyswitches use the same key. The key must be removed from one location and taken to the other location before the controls can be activated;

4. The shutter cannot be closed from the radiography control room without the keyswitch being activated. This prevents closing of the shutter from the radiography control room if personnel are on the parapet;

5. The shutter can only be opened from the radiography control room if the keyswitch (S2-2) is in place and the bay doors (K2-7X and K2-11X) are closed;

6. The shutter can only be opened from the bay floor station if the keyswitch (S2-l) is activated and the reactor is scrammed (K2-SR).

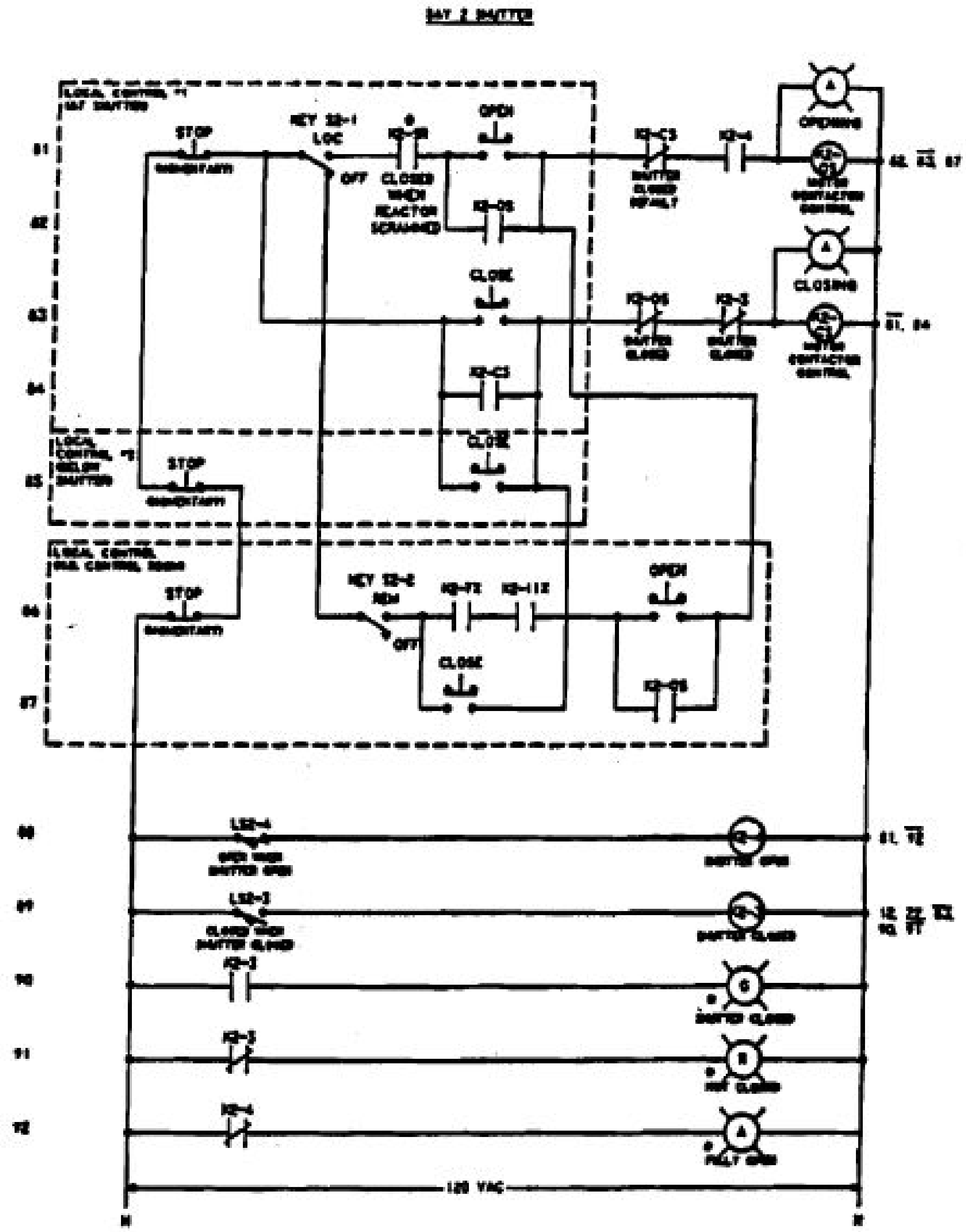

FIGURE 9.12 UCD/MNRC SHUTTER CONTROL SCHEMATIC (BAY 2)

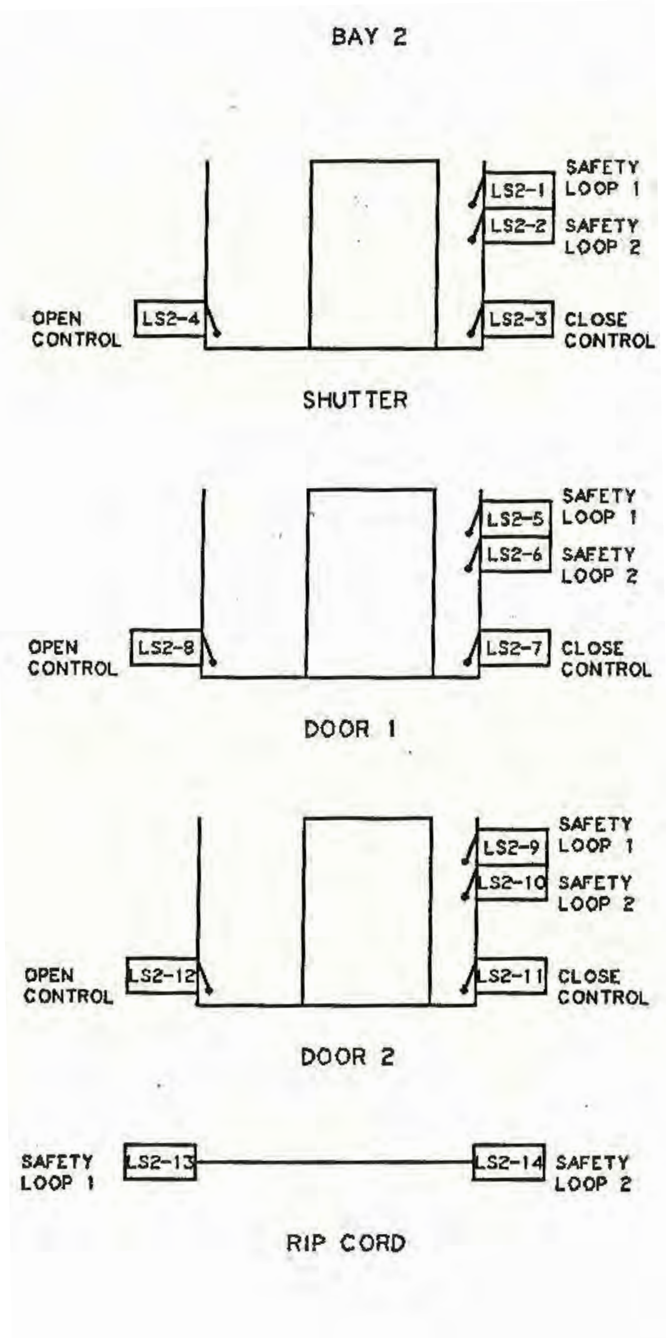

FIGURE 9.13 UCD/MNRC SHUTTER BAY DOOR AND RIP CORD LIMIT SWITCHES

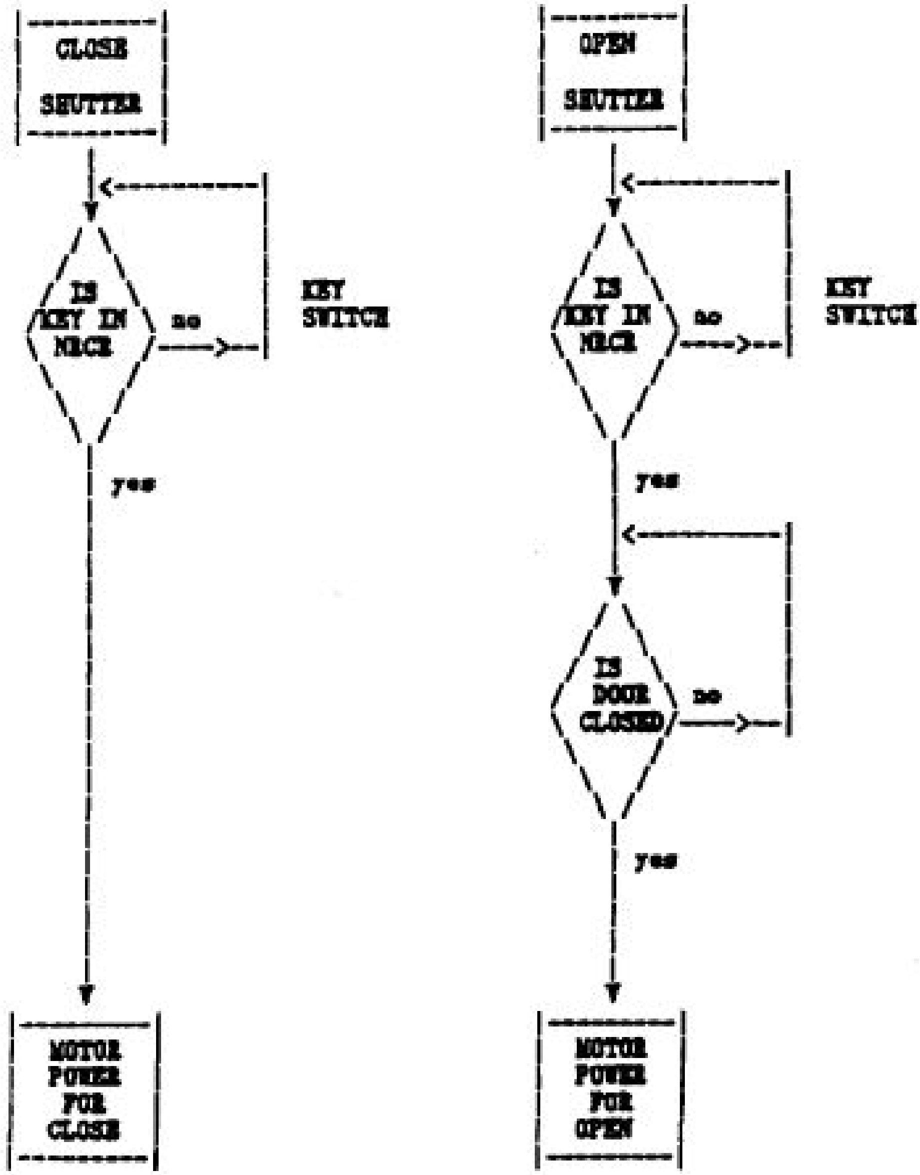

FIGURE 9.14 UCD/MNRC SHUTTER CONTROL LOGIC (RADIOGRAPHY CONTROL ROOM)

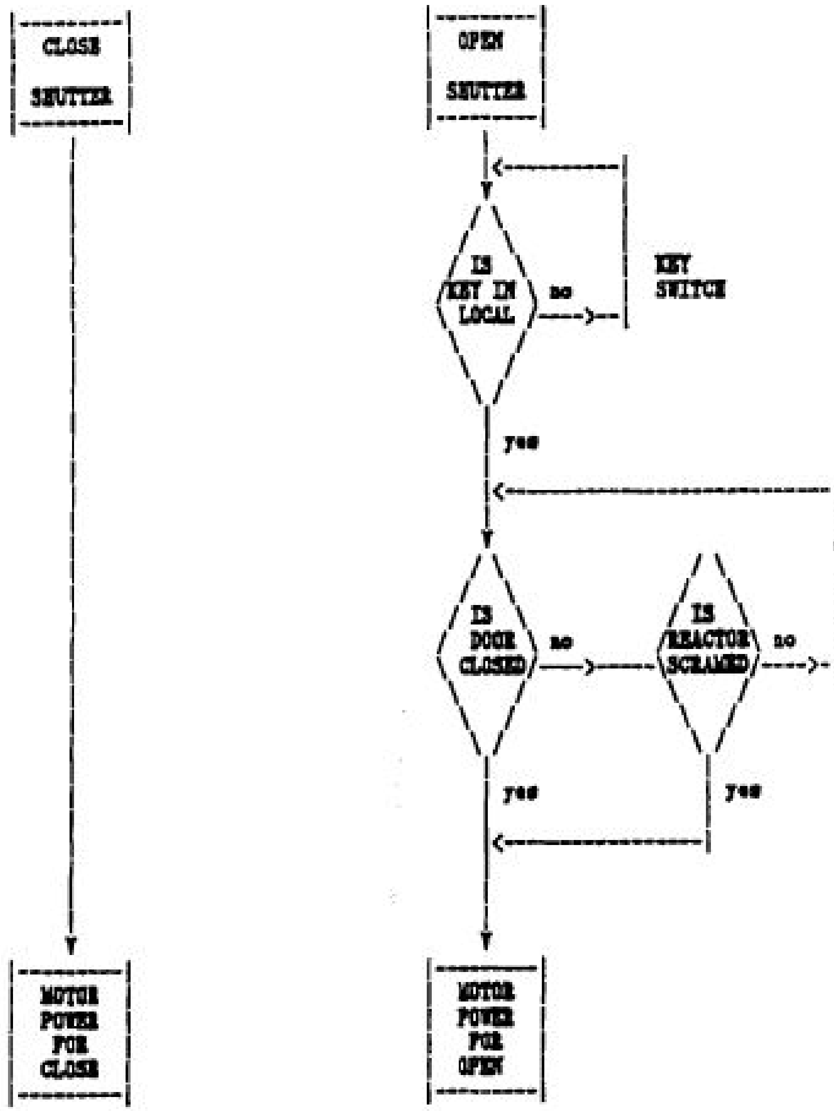

FIGURE 9.15 UCD/MNRC SHUTTER CONTROL LOGIC (INSIDE BAY)

The controls and indicator lights at each of the three control stations are as follows:

Radiography Control Room	
Controls	Indicator Lights
Keyswitch	Shutter closed - green
Closed	Shutter not closed - red
	Shutter fully open – amber & red
Stop	Shutter travelling open - red
Open	Shutter travelling closed - green

Control Station on Parapet	
Controls	Indicator Lights
Close	n/a
Stop	n/a

Control Station – Bay Floor	
Controls	Indicator Lights
Keyswitch	None
Closed	None
Stop	None
Open	None

Reactor Control Room	
Controls	Indicator Lights
None	Shutter closed - green
None	Shutter not closed - red

9.6.2 Bay Door Controls/Interlocks

Figure 9.16 is the Bay 2 door/interlock schematic and Figure 9.13 shows the corresponding limit switches. The controls/interlocks for the doors in all four bays are functionally the same. However, there are two doors in Bay 2 and only one door each in Bays 1,3, and 4.

The open-closed controls for the bay doors are mounted on the bay door. Both switches are momentary contact-type (deadman) and must be held in position to move the door.

In series with the door control power is a multiple keyswitch (11 captive keys). The door cannot be operated unless all 11 keys are in place. Whenever a person enters the bay, they will be required (administrative control) to remove a key and keep this key in their possession while in the bay. Upon leaving the bay, the key is reinserted and the controls can be activated. This key device is intended to prevent the bay doors from being closed with personnel in the bay.

The door position indicator lights are as follows:

Radiography Control Room	Reactor Control Room
Door closed - green	Door closed - green
Door not closed - red	Door not closed - red

The control/interlock logic for the bay doors is shown in Figure 9.17.
The key features of this control/interlock system are:

- All keys must be in the multiple keyswitch before power can be applied to the door;
- The control switches are the momentary contact-type so personnel must be in attendance anytime the door is operated;
- The door can be closed anytime all of the keys are in place;
- The door can only be opened if the shutter is closed (K2-3) or the reactor is scrammed

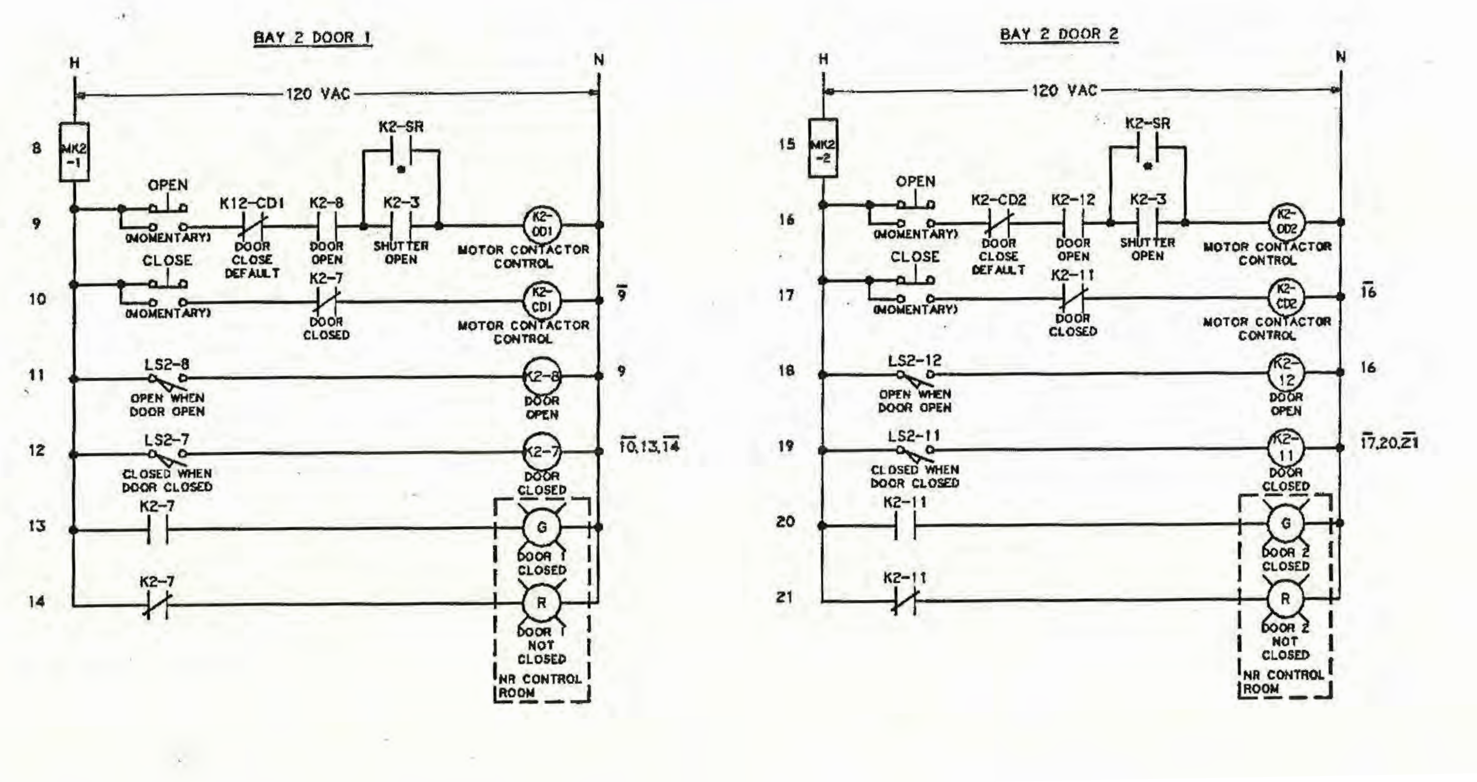

FIGURE 9.16 UCD/MNRC BAY DOOR CONTROL SCHEMATIC (BAY 2)

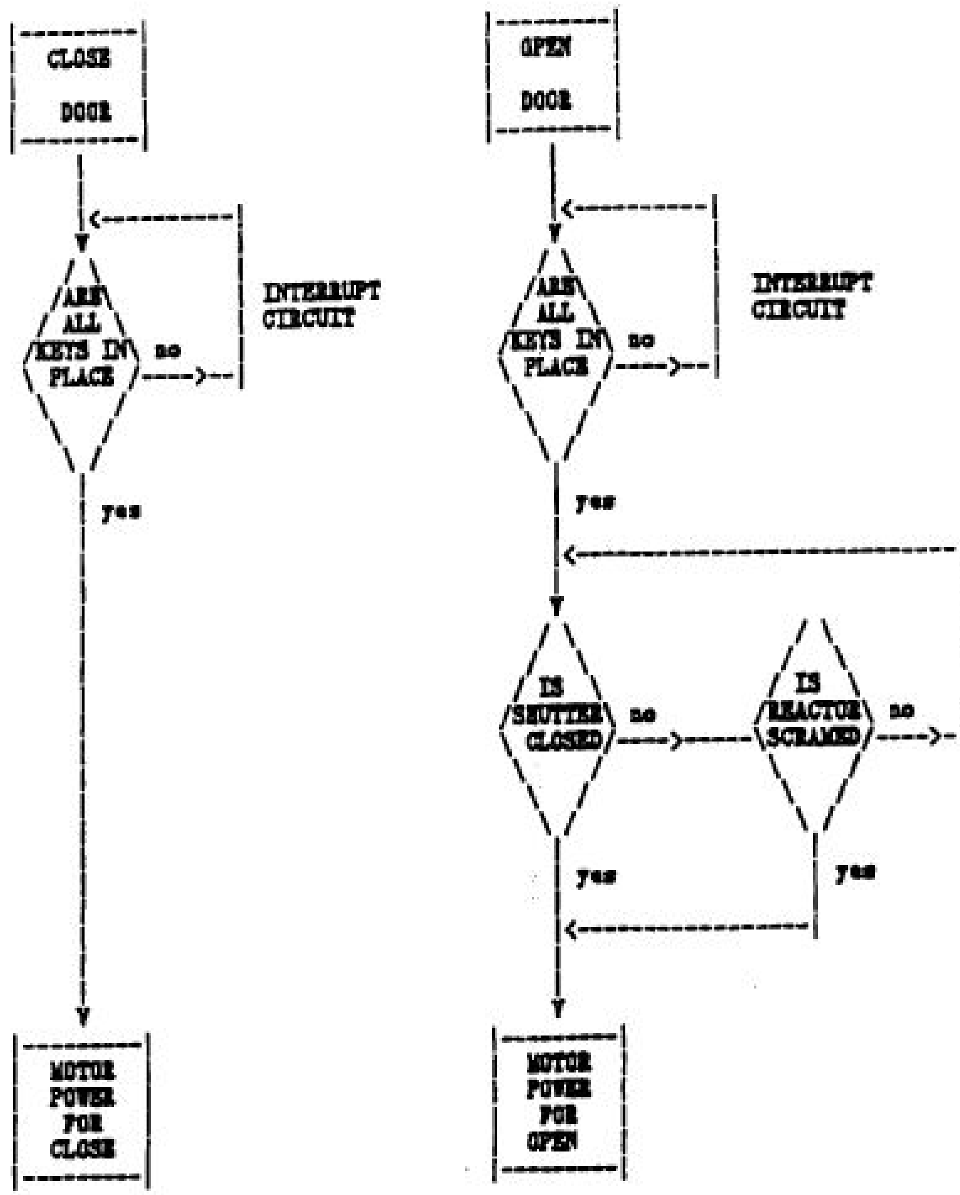

FIGURE 9.17 UCD/MNRC BAY DOOR CONTROL LOGIC

9.6.3 Reactor Interlocks

In addition to the interlocks described above that prevent access to the bays when radiation levels are high (i.e., reactor on and bay shutters not closed), there are three types of inputs from bay safety devices to the reactor scram chain. Figure 9.18 shows the schematic and Figure 9.16 shows the limit switches for the corresponding relays in the Bay 2 circuitry. All other bays are the same except for the number of doors.

The three types of scram chain inputs are from limit switches located on the shutters, the bay doors, and from switches located at the ends of rip cords located in each bay. The rip cord locations are as shown in Figures 9.19, 9.20, and 9.21. As shown in Figure 9.13, each shutter, door, and rip cord have two independent signal devices. These devices provide independent input signals to the reactor's external scram inputs. These devices and their installation is in accordance with requirements of the reactor safety system.

The key features of these reactor scram devices are as follows:

- The reactor is either scrammed or cannot be operated if the shutter and the bay door are open;
- The reactor is either scrammed or cannot be operated when the rip cord circuits have been activated;
- Once activated, the rip cord circuit can only be reset from inside the bay.

9.7 Communication and CCTV Systems

The UCD/MNRC contains telephone, intercom and closed circuit TV (CCTV) systems. The telephone system has been extended to a terminal board in the UCD/MNRC. Distribution within the UCD/MNRC is from this terminal board.

An intercom system has been provided between the reactor room, reactor control room, radiography control rooms, radiography bays, and equipment room. The master intercom stations are located in the reactor and radiography control rooms.

An emergency evacuation system has been installed in the UCD/MNRC. This system can be activated from the reactor control room and the reactor room. When energized, a number of evacuation horns in the facility are sounded.

There are CCTV cameras located in the UCD/MNRC facility. These cameras are positioned so that key areas can be monitored. The typical locations and types of cameras are shown in Figures 9.22 and 9.23. Monitors for the TV cameras are located in both the reactor control room and the radiography control rooms.

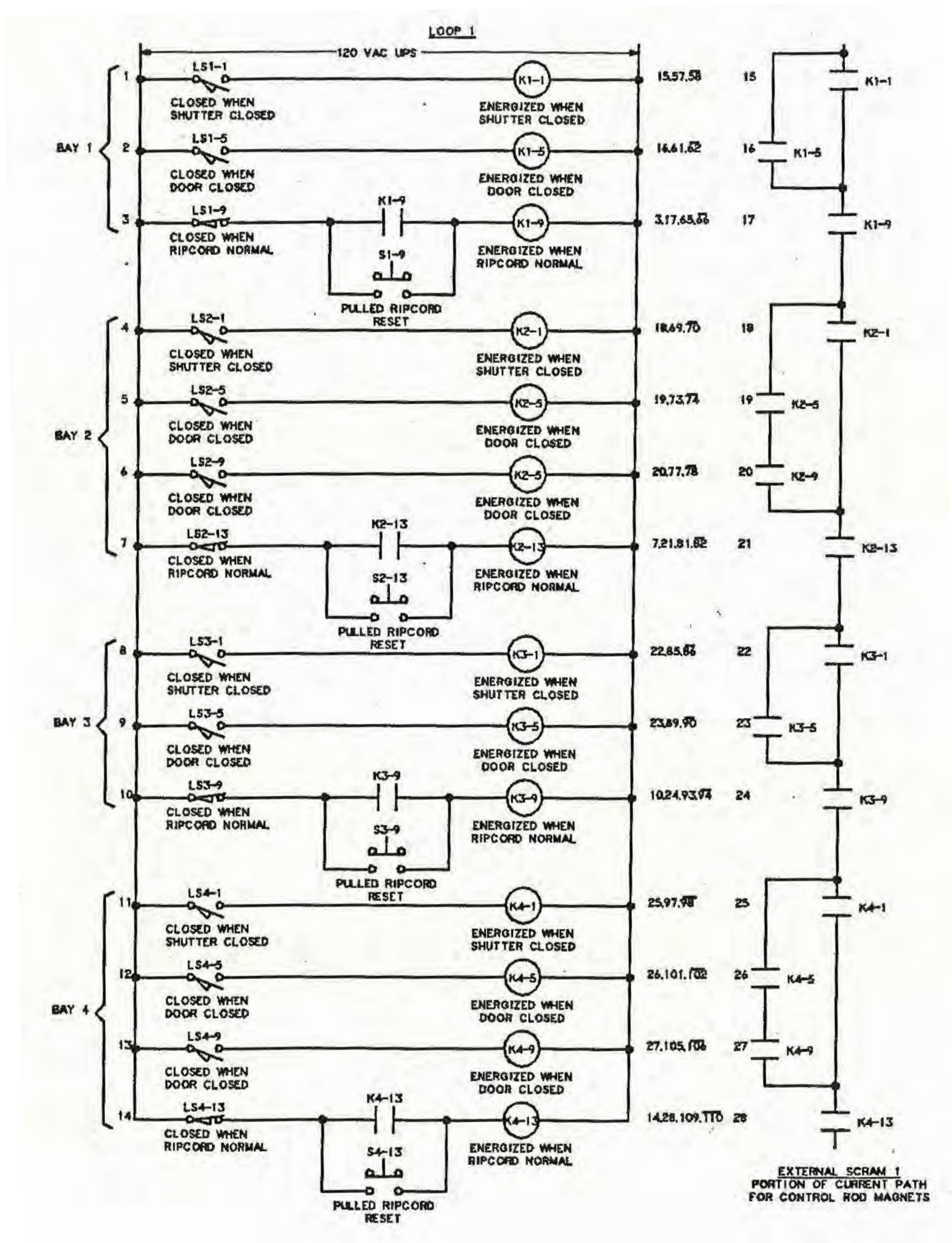

FIGURE 9.18 UCD/MNRC BAY DOOR, SHUTTER, AND RIP CORD CONTROL SCHEMATIC – REACTOR SCRAM CHAIN INPUT (LOOP NO. 1)

[redacted]

FIGURE 9.19 RIP CORD LOCATION – BAYS 1 AND 2

[redacted]

FIGURE 9.20 RIP CORD LOCATION – BAYS 3 AND 4

[redacted]

FIGURE 9.21 RIP CORD LOCATION – PLAN VIEW

[redacted]

[redacted]

FIGURE 9.23 TYPICAL UCD/MNRC CLOSED CIRCUIT TELEVISION – SECOND FLOOR

[redacted]

CHAPTER 10

EXPERIMENTAL FACILITIES
AND
UTILIZATION

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10.0 EXPERIMENTAL FACILITIES AND UTILIZATION

10.1 Summary Description

The UCD/MNRC provides a broad range of radiographic and irradiation services to the military and non-military sector. The facility presently provides four radiography bays and consequently four beams of neutrons for radiography purposes. In addition to the radiography bays, the UCD/MNRC reactor core and associated experiment facilities are completely accessible for the irradiation of material. These irradiation services include: silicon doping, isotope production (both medical and industrial) and neutron activation analysis (e.g., geological samples). Although all four radiography bays are capable of using radiography film techniques, Bays 1, 2, and 3 are equipped with electronic imaging devices. All bays contain the equipment required to position parts for inspection as well as the radiography equipment. To meet facility use requirements, the reactor system and associated experiment facilities are designed to operate three shifts per day, though 1 shift operation is the typical mode of operation.

10.2 Beam Tubes and Beam Tube Shutter/Bulk Shield

10.2.1 Beam Tubes

10.2.1.1 Design Basis

The design basis for the beam tubes is to provide a path for primary neutrons with minimum scattering and attenuation between the reflector and the radiography bays.

10.2.1.2 Description

Four beam tubes spaced at 90° intervals around the base of the reactor tank penetrate the reactor graphite reflector and provide a direct path for neutrons to each of the radiography bays. The beam tubes are positioned tangentially with respect to the reactor core and are inclined (20° and 30°) with respect to the horizontal plane (Figures 10.1, 10.2, 10.3, and 10.4). Each of the four beam tubes is made up of three major sections: the in-tank section, the tank wall section, and the reactor bulk shielding section.

The in-tank section of the beam tube (a replaceable aperture, made from neutron absorbing material and graphite housed in a water-tight aluminum container) is shown in Figure 10.4. This section is the most important part of the beam tube since it is part of the reactor core reflector, provides a source of neutrons, and purifies and shapes the beam. It consists of a large graphite block with a 6 in. diameter hole bored along the beam centerline. The key elements within the bore are a graphite end plug which serves as a source of neutrons, a bismuth crystal which attenuates gamma rays and a boron-carbide aperture which shapes the beam. An aluminum spacer and lead-cadmium sleeve and shield are also located in the bore.

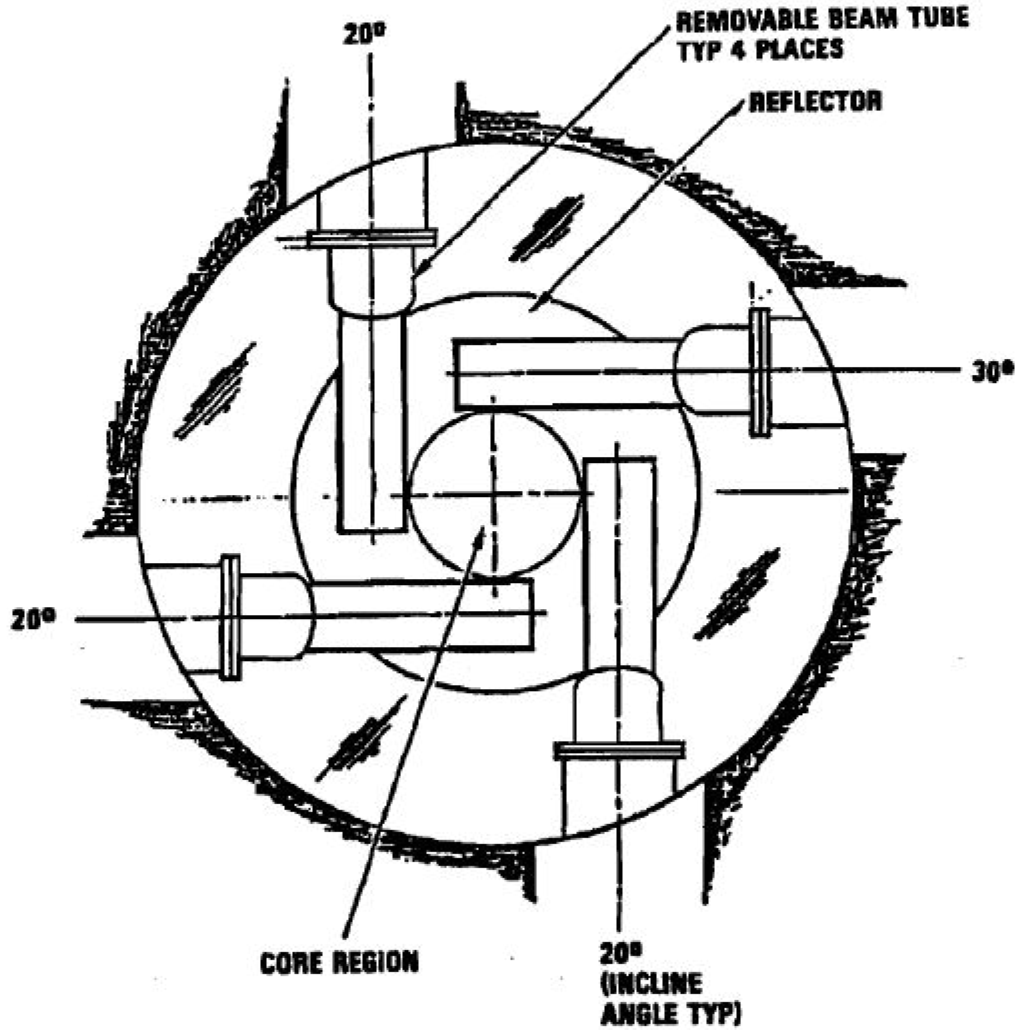

FIGURE 10.1 REFLECTOR BEAM TUBE LOCATION

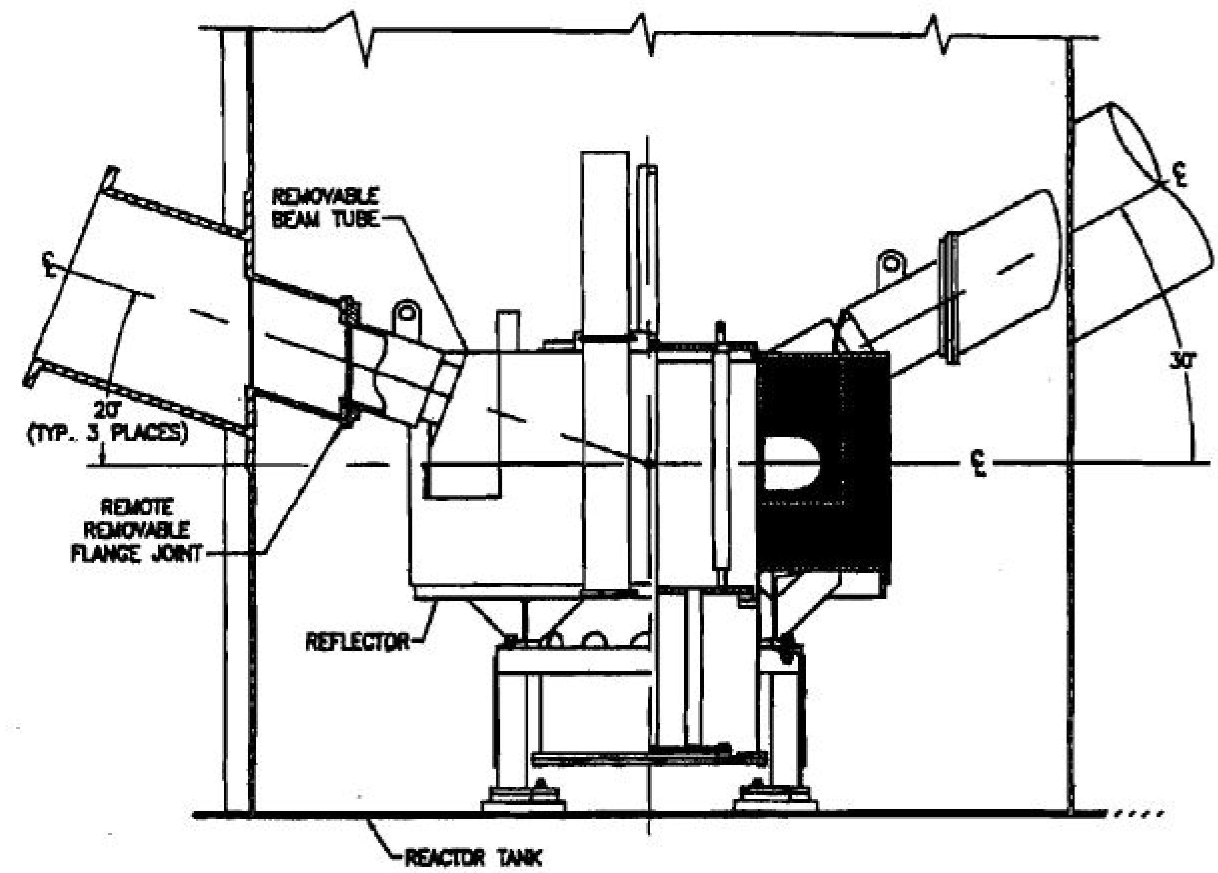

FIGURE 10.2 SIDE VIEW OF REACTOR AND BEAMTUBES

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FIGURE 10.3 MNRC IN-TANK EXPERIMENTAL FACILITIES AND BEAMLINES INSERTS

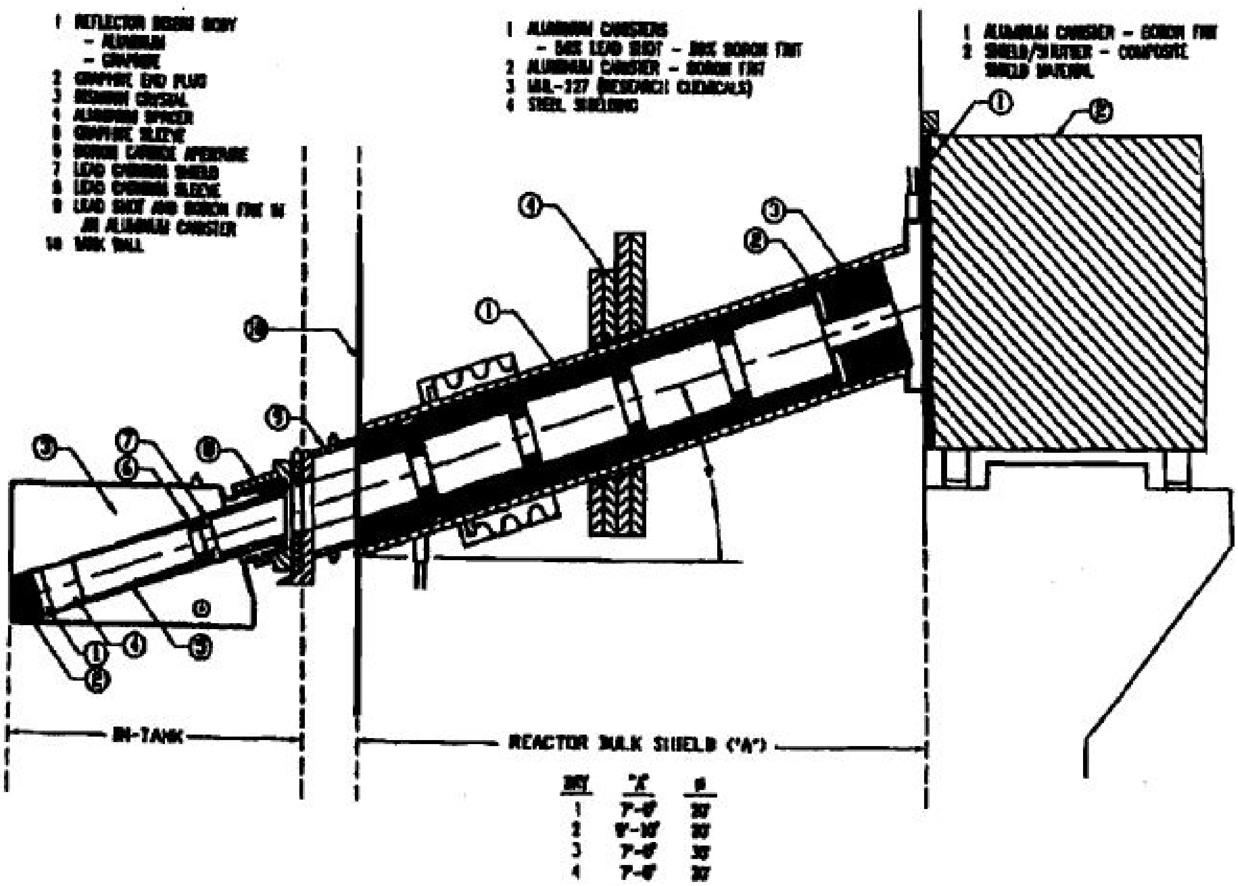

FIGURE 10.4 UCD/MNRC BEAM TUBE AND BIOLOGICAL SHIELD

The sleeve and shield serve as neutron and gamma ray shields. All of these components are contained in an aluminum housing that transitions into a 12-1/2 in. diameter circular cross section with a bellows assembly and flange with a bolt-on faceplate. A lead coated metal O- ring forms the seal between the flange and the faceplate. The faceplate and the in-tank assembly have two tube fittings that connect to a helium supply or vacuum system. The entire unit is watertight and can be remotely removed and replaced from the tank top. The assembly mates with the tank-wall section of the beam tube to provide a water free path within the reactor tank for the neutron beam. Removal and replacement of the in-tank section of the beam tube has a small effect on the reactor core reactivity. Although the entire in-tank section of the beam tube is watertight, none of the components will react with water nor will they degrade should water enter the assembly.

The tank-wall section of the beam tube consists of a 12-1/2 in. diameter pipe welded to the tank wall and a special flange welded to the core end (Figure 10.4). An aluminum container filled with 50 vol % boron frit and 50 vol % #9 lead shot by volume is located within the pipe section. The internal surface of the aluminum container is coated with gadolinium. The tank wall section does not penetrate the tank wall and serves as a watertight container when assembled as well as both a neutron and gamma shield. The gadolinium helps prevent scattered neutrons from reentering the beam. This section contains tube fittings that are attached to a helium supply or evacuation system.

The in-tank and tank-wall section flanges are held together by a two-piece bolted clamp. The clamp bolts can be remotely removed and replaced from the tank top.

The bulk shielding section of the beam tube extends from the outside of the tank wall to the radiography bays (Figure 10.4). The housing for this section is a 20 in. diameter steel pipe and bellows assembly imbedded into the concrete. The bellows assembly provides flexibility for expansion and contraction. The pipe is in close proximity, but is not physically attached to the tank wall. Within the housing are a number of annular shaped aluminum containers filled with 50 vol % boron frit and 50 vol % #9 lead shot. The primary function of these materials and their design is to provide neutron and gamma ray shielding, help shape the beam, and prevent scattered neutrons from reentering the beam. The annular section next to the tank wall is permanently installed. The remaining boron/lead filler sections can be removed and replaced with units of different internal diameters if the beam size (cross section) needs to be changed. The two annular containers at the exit of the beam tube into the radiography bay contain 100% boron frit and a Research Chemicals MHL-277, respectively. These elements are the final beam shapers and both are excellent neutron shields. Both assemblies can be replaced from the radiography bay. The inner surfaces of all containers in this section are also coated with gadolinium.

The ends of the beam tubes are closed with aluminum plates. These plates are 0.60 inches for the beam tubes in radiography Bays 1, 2, and 4, and 0.75 inches for the beam tube in radiography Bay 3.

The amount of explosive material allowed in radiography Bays 1, 2, 3, and 4 is 3 pounds of TNT equivalent per bay. This explosives limit is supported by safety analyses performed by Southwest Research Institute (Reference 13.11) and by the UCD/MNRC (Reference 13.10). Actual results of these analyses show that the four radiography bays could each safely contain up to 6 pounds of TNT equivalent, provided the door tracks and suspension on Bays 2 and 3 were strengthened in the early 2000s. However, by establishing a limit of 3 pounds of TNT equivalent for each bay, only the beam tube cover plates specified in the previous paragraph are required.

All three sections of the beam tube are equipped with gas lines. These lines are attached to the helium supply or evacuation system and can be used to either evacuate or fill the tubes with helium to prevent degradation of the neutron beam. The helium supply and vacuum system has venting and/or pressure controls to prevent over-pressurization of the beam tube (Section 9.2). There is very little, if any, Ar-41 formed in these beam tubes because of the absence of air.

10.2.1.3 Evaluation

The beam tubes, by use of shaped rings and being sealed and void of air, provide a neutron path with minimum neutron scattering.

The beam tubes do not penetrate the reactor tank wall, and therefore, do not increase the probability of tank leakage.

The beam tube cover plates on the ends of the beam tubes, where they exit into the bays, provide closure and prevent pressure waves from reaching the reactor core and damaging the fuel should the maximum allowable amount of explosives being radiographed detonate.

It should be noted that supplemental shielding has been placed in the reactor bulk shield to compensate for the void volumes created by the beam tubes.

10.2.2 Beam Tube Shutter/Bulk Shield

10.2.2.1 Design Basis

The design basis for the beam tube shutter/shield is:

(a) To attenuate the neutron radiation beam at the location where it exits into the radiography bay such that radiation levels in the radiography bay are as-low-as-reasonably- achievable;

(b) To provide a fast-acting thermal neutron shutter so that radiography film exposure times and real-time imaging can be controlled.

10.2.2.2 Description

Each of the beam tubes has a bulk shield and shutter. These units are located adjacent to the radiography bay end of the beam tubes as shown in Figure 10.4, and serve two basic functions. First, they provide the biological shielding from reactor core neutrons and gamma rays when the beam is not being used and the radiography bays are occupied. Second, they provide a means to start and stop the flow of thermal neutrons during radiography operations.

The shield/shutter unit is motor-driven and can be positioned so that the bulk shield covers the beam tube or so that only the thermal neutron shutter is in the beam path. The bulk shield is a massive composite structure containing materials to thermalize fast neutrons, capture thermal neutrons, and shield against both direct and capture gamma rays.

The bulk shield has an average density of 4.7 gm/cm^3 and is made up of cement, boron carbide, limonite, and steel shot. Boron frit, approximately 1 in. thick contained in aluminum, is placed in front of the composite shield to attenuate thermal neutrons. This shield has been designed so that the surface radiation level on the radiography bay side where personnel will be working during reactor operation at 1 MW will be less than 1 mR/hr. The motor drive on the shield is controlled from the radiography control room or in the radiography bay. Indicator lights in the radiography and reactor control room show the shutter position. There is an interlock system that prevents the shield from being moved from the closed position any time the radiography bay door is opened and the reactor is operating. Sections 9.6 and 11.1.5.1 contain a complete description of the shield, shield controls and interlocks.

The thermal neutron shutter is a rectangular aluminum can approximately 1 in. thick filled with boron frit. The shutter is air actuator-driven, and remotely controlled from the radiography control room. As far as radiation protection is concerned, it is not considered an integral part of the bulk shield.

10.2.2.3 Evaluation

The beam tube shutter/shield provides the necessary biological shielding to protect personnel working in the radiography bay from the intense source of neutrons in the radiography beams. These shields limit the radiation levels, within the radiography bays, to less than 1 mr/hr at 1 MW (Chapter 11).

The boron frit shutter provides an effective means of controlling the flow of thermal neutrons.

10.3 Component Positioning Equipment

The UCD/MNRC has three automated component positioning systems. The automated systems are located in Bays 1, 2 and 3. Bay 4 is provided with an inspection table and fixtures.

10.3.1 Bay 1 Component Handling System

Following are specific design features which have been included in the component handling system of Bay 1. Figure 10.5 shows an elevation layout of the component handling system. This system is used to position large components. The maximum size component which can be inspected measures 32.5 ft long x 12.5 ft high and weighs 3800 lbs.

The system consists of one cart with fixtures to hold the components. The cart is latched to the positioning system which provides five axes of motion. Large components are held with special fixtures which provide positive location of the component on the cart. This fixturing has been designed to hold the components at each end to eliminate support structures at the center of the component which would interfere with the radiograph.

10.3.2 Bay 2 Component Handling System

Following are specific design features which have been included in the component handling system of Bay 2. Figure 10.5 shows an elevation layout of this system. The system is sized to handle parts weighing up to 1500 lbs and measuring up to 18 x 9 ft.

The system in Bay 2 includes two carts which hold the components to be inspected. The carts have been designed to accept large part fixturing which is used on the Bay 1 cart. The carts are also equipped with adjustable fixturing to hold smaller parts. This fixture can accommodate four components at one time. This system provides the same degrees of freedom as provided in Bay 1.

10.3.3 Bay 3 Component Handling System

Following are specific design features which are included in the component handling system of Bay 3. Figure 10.6 shows an elevation layout of this system. This system is sized to handle small parts up to 5 ft x 5 ft and curved parts with curvatures up to 160 deg. For inspecting curved parts, more yaw motion is required in this bay than in the other bays. For this reason, this positioner is designed differently than the ones in Bays 1 and 2.

This system does not use a cart. Instead, operators load the components onto the positioner in the inspection bay. To facilitate this, an adjustable frame has been provided which can be adjusted to support small parts. After the fixture is loaded, the system will position the component in the beam path.

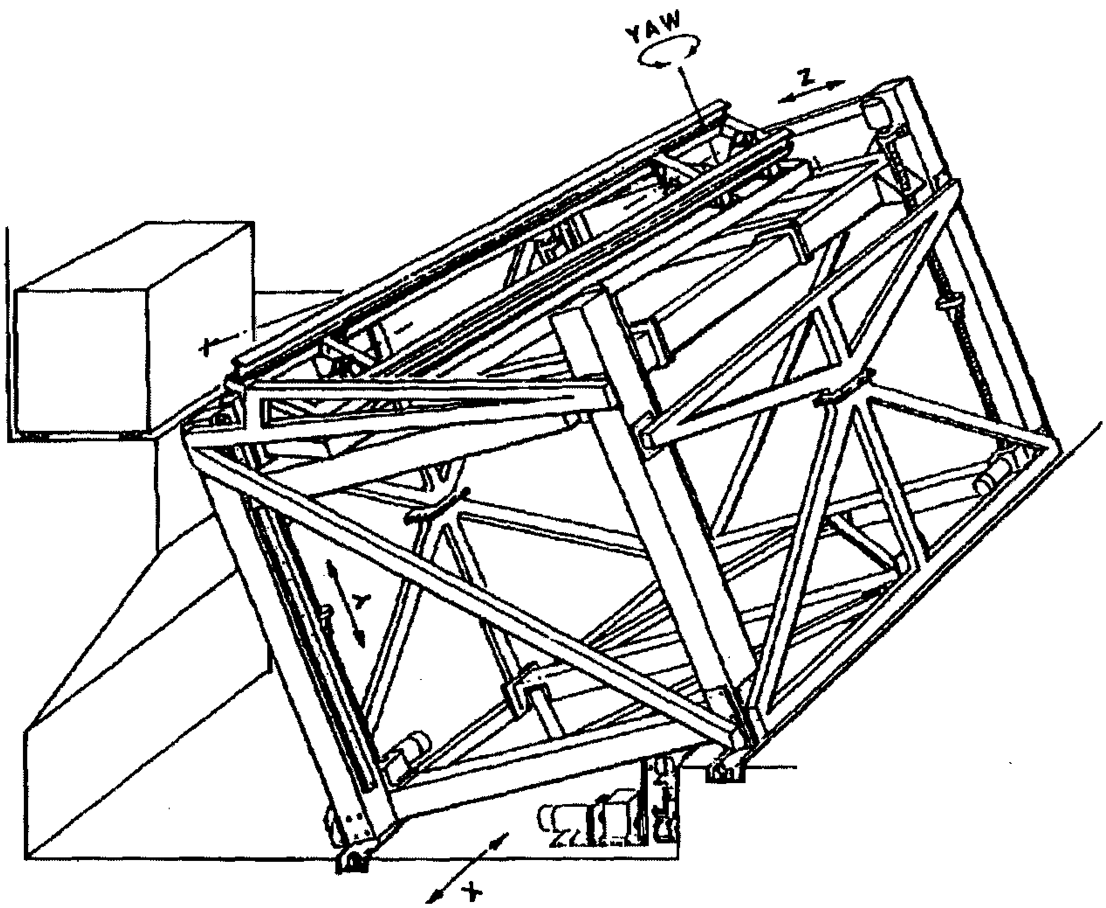

FIGURE 10.5 BAYS 1 AND 2 COMPONENT POSITIONING SYSTEM

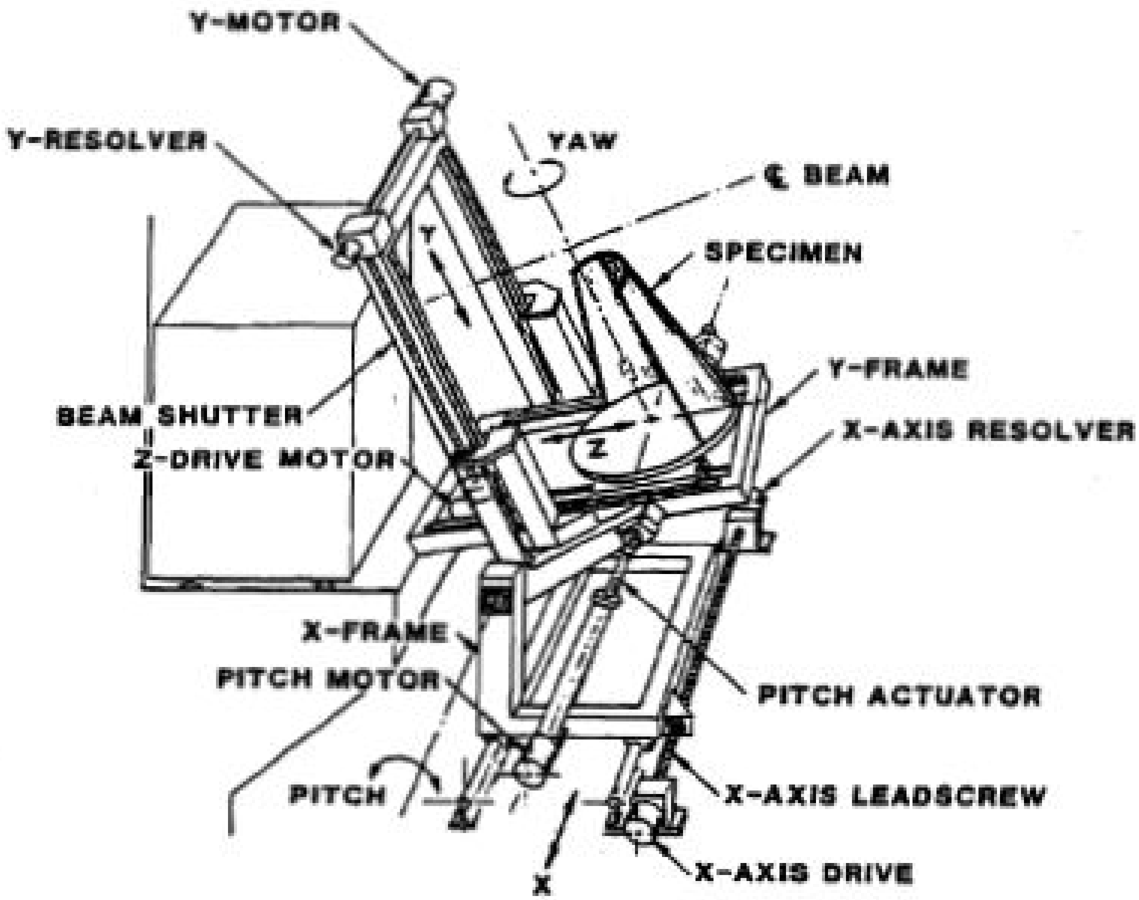

FIGURE 10.6 BAY 3 COMPONENT POSITIONING SYSTEM

10.4 In-core Irradiation Facilities

The UCD/MNRC reactor is designed with several locations for in-core irradiation facilities. These in-core locations are indicated in Figure 10.7 and include a central cavity, four experiment tube locations, a location for a pneumatic transfer tube, and individual fuel element locations. The irradiation facilities which may be installed in these in-core locations are described below.

10.4.1 Central Irradiation Facility

The central irradiation facility is formed by the installation of a central thimble (Figure 10.8) into the central cavity in the reactor core. Once installed in the central cavity, the central thimble shall not be removed from the reactor core unless it is to be replaced with another facility of similar dimensions that has been analyzed to show how it affects the overall operation of the reactor (See Section 10.4.1.1).

The central thimble is approximately 55 inches in length and 4.22 inches in diameter with an inside dimension of approximately 4.0 inches. The central thimble once in place passes through the upper grid plate, the lower grid plate and the safety plate. The bottom of the central thimble sits on the bottom of the reactor tank. An aluminum ring located approximately 24.5 inches from the bottom of the central thimble aligns with the bottom grid plate and prevents samples or fixtures from dropping below the lower grid plate. There is a 1.5 inch hole in the center of the aluminum ring and twenty-four 1.0 inch holes in the lower 24 inches to allow cooling flow throughout the central thimble. Aluminum shims have been added to the outer periphery of the central thimble in the fuel region. These shims align the central thimble and displace the water from the scallops of the fuel element locations in the B hex ring 4.25 inch hole. Samples or fixtures can be inserted into or removed from the central irradiation facility using underwater tools.

The central irradiation facility is most often used where a high flux is required for a specific experiment. Typically, these experiments are for micro Curie level isotope production, material radiation damage studies, and geochronology experiments. These experiments are double encapsulated and placed in a “wet-tube” which allow water to flow around the experiment to provide cooling. The “wet-tube” itself is placed into the CIF after the aluminum slug is removed.

10.4.1.1 Central Irradiation Facility (CIF-1)

The central irradiation fixture (CIF-1) consists of a graphite thimble plug and associated removable aluminum thimble plug insert positioned in the central irradiation facility (Figure 10.8).

The graphite thimble plug is a graphite-filled sealed aluminum can having dimensions of 26.88 inches long and 3.95 inches in diameter with a central through hole of 2.25 inch. A 6 inch long aluminum pipe welded to the top of the graphite thimble plug allows the removal or installation into the central thimble of this plug.

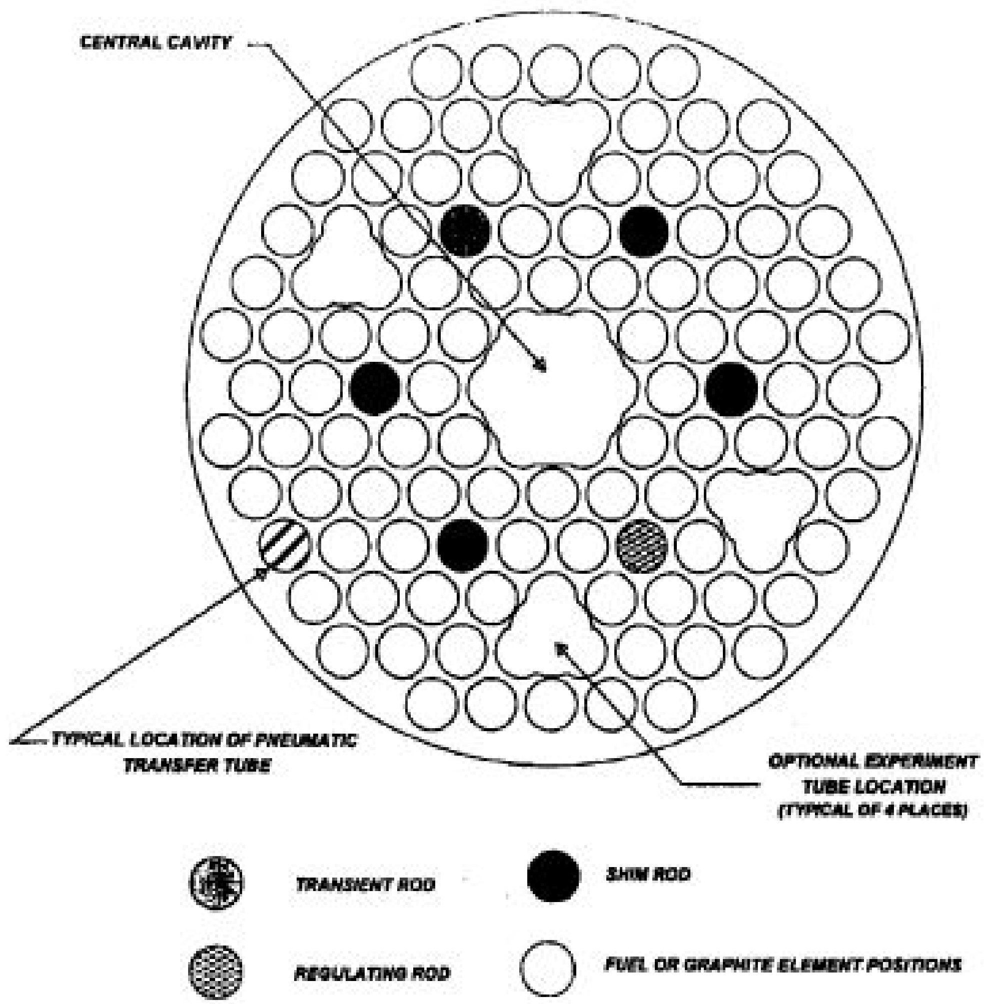

FIGURE 10.7 UCD/MNRC TYPICAL IN-CORE FACILITIES

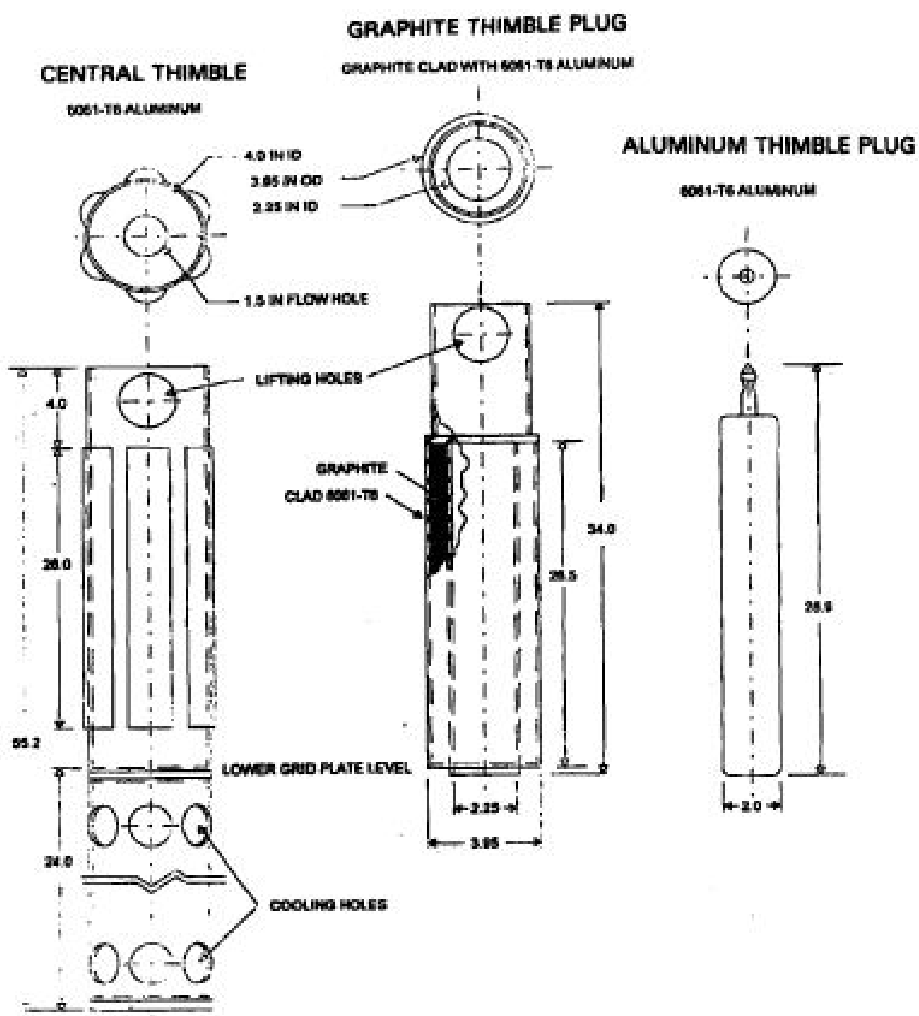

FIGURE 10.8 UCD/MNRC CENTRAL IRRADIATION FACILITY (THIMBLE) AND CENTRAL IRRADIATION FIXTURE-1 (CIF-1)

The removable aluminum thimble plug is a 2 inch aluminum bar approximately 29 inches in length. The upper end of the plug has been machined so the fuel element handling tool can be used to insert or remove the plug. Removal of the aluminum thimble plug provides a water filled region for the irradiation of experiments.

10.4.2 Experiment Tubes in Upper Grid Plate Cutout Positions

Four triangular shaped cutout sections, described in Section 4.2.3, have been provided in the upper grid plate to allow for removal of groupings of three fuel elements and the insertion of tubes up to 2.4 in. outside diameter for experiment placement. In the history of the MNRC facility these position have never been utilized in this way. Furthermore, it is unlike they will ever be utilized in this way.

10.4.3 Pneumatic Transfer System

The UCD/MNRC Pneumatic Transfer System, shown in Figure 10.9, is designed to quickly transfer individual specimens into and out of the reactor core. The specimens are placed in a small polyethylene holder, "rabbit," which in turn is placed into the receiver. The rabbit travels through aluminum tubing to the terminus at reactor core centerline, then returns along the same path to the receiver. Directional air flow moves the rabbit between receiver and terminus. A blower assembly moves air through the system, and a solenoid valve directs air flow. Controls to operate the blower and solenoid valve are wall-mounted adjacent to a hood which contains the receiver. The air flow design is such that the rabbit is never pushed but rather pulled from place to place, minimizing the possibility of fragments from a shattered rabbit becoming trapped in the terminus. The key system elements and their functions are described below.

The "rabbit" is an enclosed polyethylene holder. Experiments are inserted into the rabbit and contained by a screw cap on one end. Available space inside the rabbit is approximately 0.625 in. in diameter and 4.5 in. in length.

The receiver positions the rabbit for transfer to the terminus and receives the rabbit after irradiation. An aluminum door retains the rabbit in the receiver during transfer operations. Two transfer lines connect the receiver to the terminus: one allows the rabbit to travel between the receiver and terminus, the other controls the air flow direction.

The receiver is located in a stainless steel hood which encloses the area around the receiver and prevents uncontrolled release of airborne radioactivity. The hood's exhaust fan maintains the hood at a negative pressure with respect to the surrounding room and maintains a hood face air velocity of approximately 100 ft/min when the sash is open. The air to the fan passes through a prefilter, an absolute filter and exhausts to the facility stack. The hood is located in the preparation area and provides working space around the receiver for handling rabbits before and after irradiation.

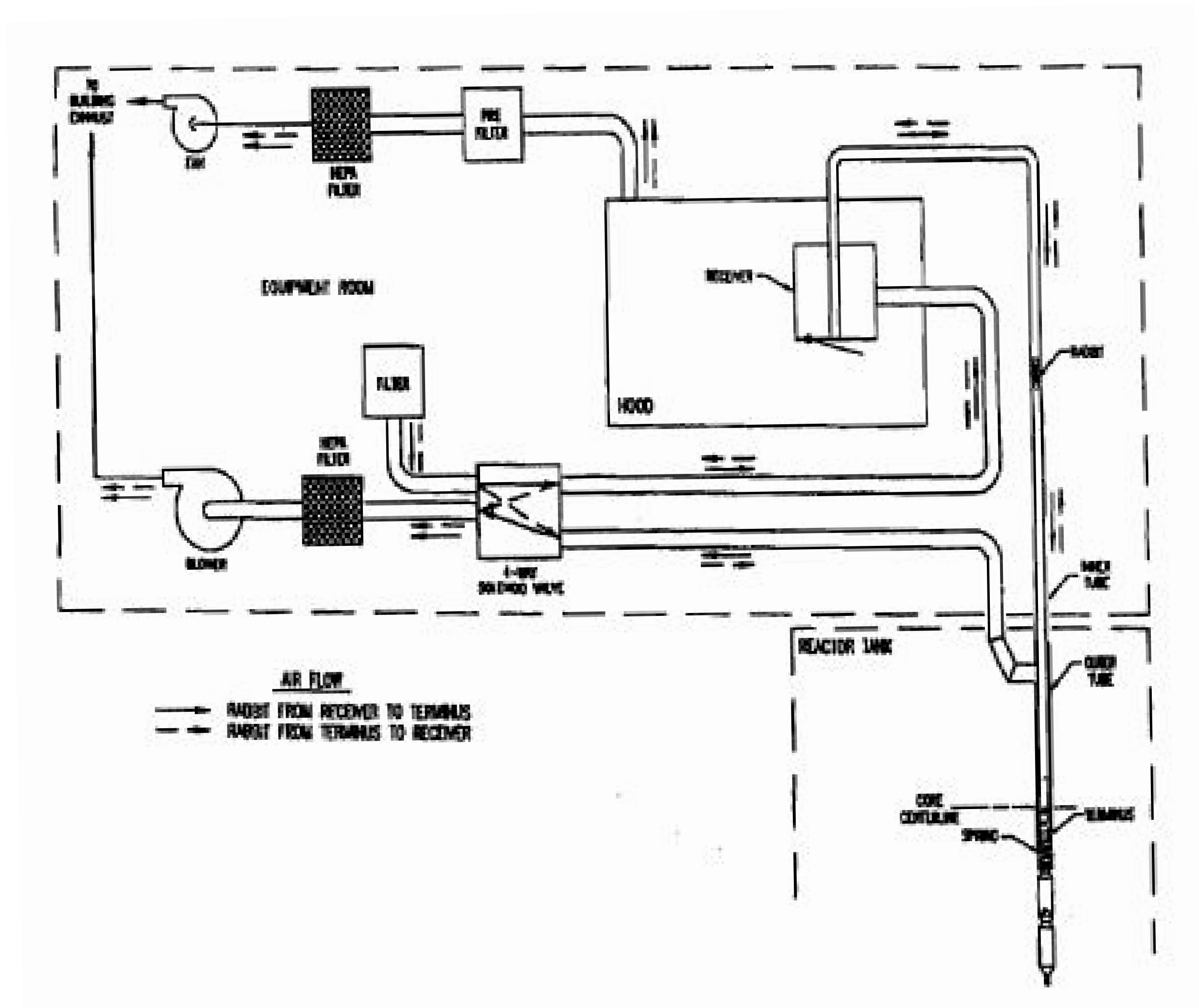

FIGURE 10.9 UCD/MNRC PNEUMATIC TRANSFER SYSTEM

The terminus consists of two concentric tubes which extend into the reactor core. The inner tube is perforated with holes (which are smaller than the rabbit diameter). The bottom of the inner tube contains an aluminum spring shock absorber to lessen the impact of the rabbit when it reaches this end of the transfer line, which is approximately at the mid-plane of the core. When air flows to the terminus, the capsule rests in the bottom of the inner tube; when air flows to the receiver, the capsule moves out of the inner tube by air flowing through the tube's holes. The outer tube supports the inner tube and provides a path for the air to flow through.

The outer tube bottom support is shaped like the bottom of a fuel element and can fit into any fuel location in the core lattice. Both tubes, which extend to the top of the reactor tank, are offset to reduce radiation streaming. A weight has been installed to counteract the buoyancy of the air-filled tubes and keep the terminus firmly positioned in the core. The terminus can be removed from the core by releasing two couplings.

Two 1.25 in. aluminum transfer lines form a loop with receiver and terminus. The "rabbit" transfer line provides a path for rabbit travel between the receiver and terminus while the "air" transfer line directs air flow between receiver and terminus. Tubing bends are a minimum 2 ft radius, allowing clearance for the rabbit.

A solenoid valve directs flow through the transfer-line-loop sending the rabbit either to the terminus or to the receiver depending on valve position. When the solenoid valve is deenergized, rabbit transfer line air flows from terminus to receiver; when the solenoid valve is energized, rabbit transfer line air flows from receiver to terminus. Solenoid status (energized or deenergized) is indicated by red markings on the solenoid alignment rod.

A two horsepower blower circulates air through the transfer lines. The blower draws filtered room air through the solenoid valve, transfer lines, and a High Efficiency Particulate Air (HEPA) filter. The blower outlet goes to the facility exhaust system.

The transfer systems' controls allow operations in either manual or automatic modes. In manual mode, the solenoid valve is activated by the operator; in the automatic mode, the solenoid valve is activated by the timer mechanism, sending the rabbit into the core-when the timer starts and retrieving the rabbit after a predetermined time period. The blower is manually operated in either mode. The controls for the system are located in a box next to the hood.

An interlock switch in the reactor control room provides the reactor operator with overall control of operation. The switch is interlocked to the power supply for the blower such that the switch must be "ON" for the blower to operate.

10.4.4 Individual Grid Plate Fuel Element Positions

Reactor grid positions vacant of fuel elements may be utilized for the irradiation of materials. These in-core irradiation facilities involve placement of an experiment in a fuel element grid position and use of these locations shall meet all the applicable requirements of the UCD/MNRC Technical Specifications.

10.5 Ex-Core In-Tank Facilities

Ex-core in-tank facilities have been established as shown in Figure 10.10. These facilities include the neutron irradiator facility, multiple silicon doping fixtures, and the Argon-41 production facility.

10.5.1 Neutron Irradiator Facility

The Neutron Irradiator Facility is used to expose experiments to a high energy neutron environment with minimal thermal neutron and gamma radiation (Figures 10.11 and 10.12). The Neutron Irradiator has four main components: a Conditioning Well, an Exposure Vessel, a Motor Drive Unit, and a Computer. The Conditioning Well is installed inside the reactor tank adjacent to the reflector and consists of boron nitride and lead (for shielding thermal neutrons and gammas respectively) encased in aluminum. The Exposure Vessel (EV) is lowered into the Conditioning Well for irradiation. The EV houses the experiment(s) and contains temperature probes for monitoring the EV internal temperature during irradiation. A 5-piece lead and boron nitride shield assembly placed on top of an assembled EV completes the shielding around the experiment(s). The Motor Drive Unit is mounted at the top of the reactor tank and rotates the exposure vessel to provide a uniform neutron flux distribution. The Computer is connected to the EV and the Motor Drive Unit to monitor temperature and control rotation respectively. The Conditioning Well and Exposure Vessel are described in further detail below. The facility is overwhelmingly used for seed mutagenesis studies and electronic hardness testing.

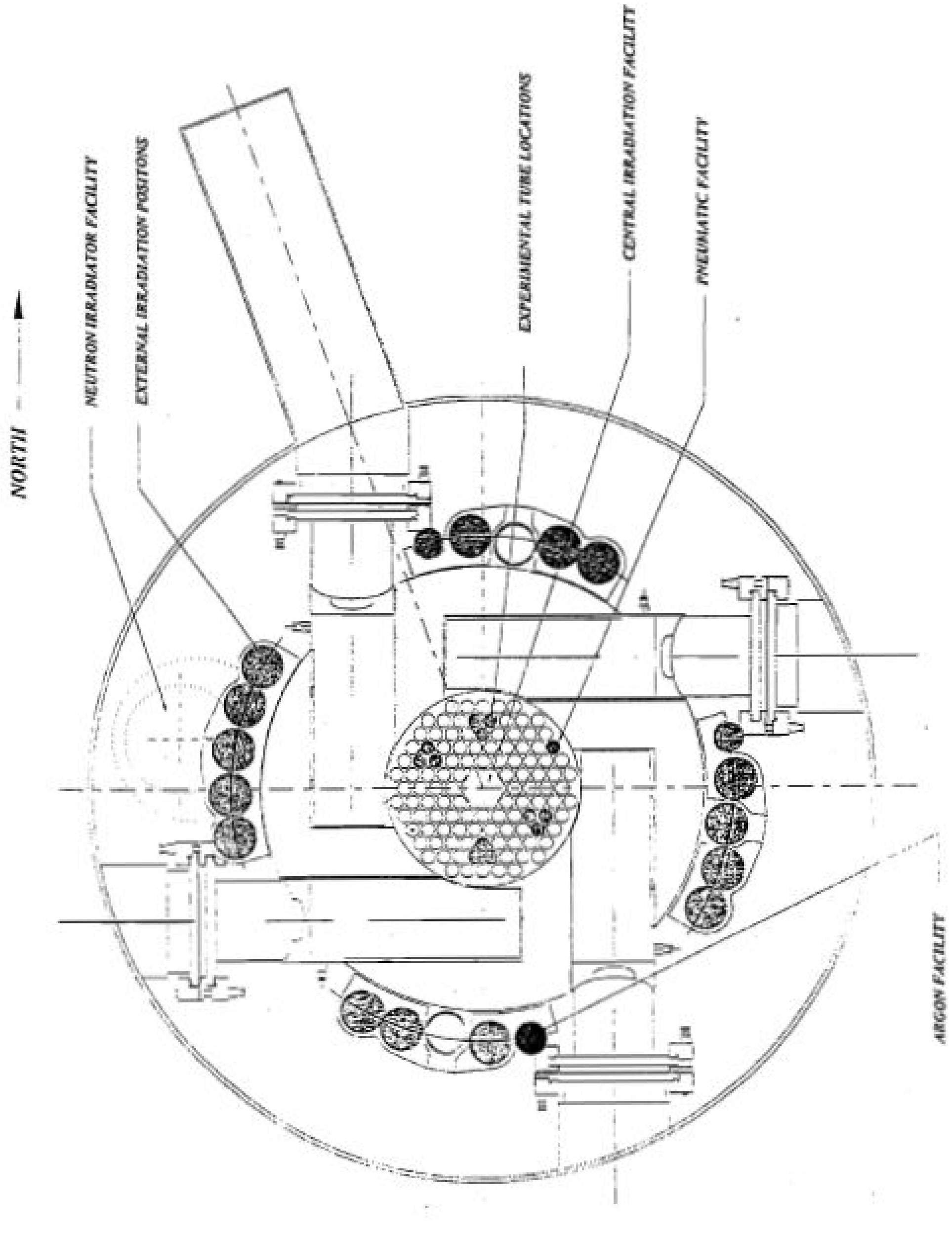

FIGURE 10.10 UCD/MNRC IN-CORE AND EX-CORE IN-TANK FACILITIES

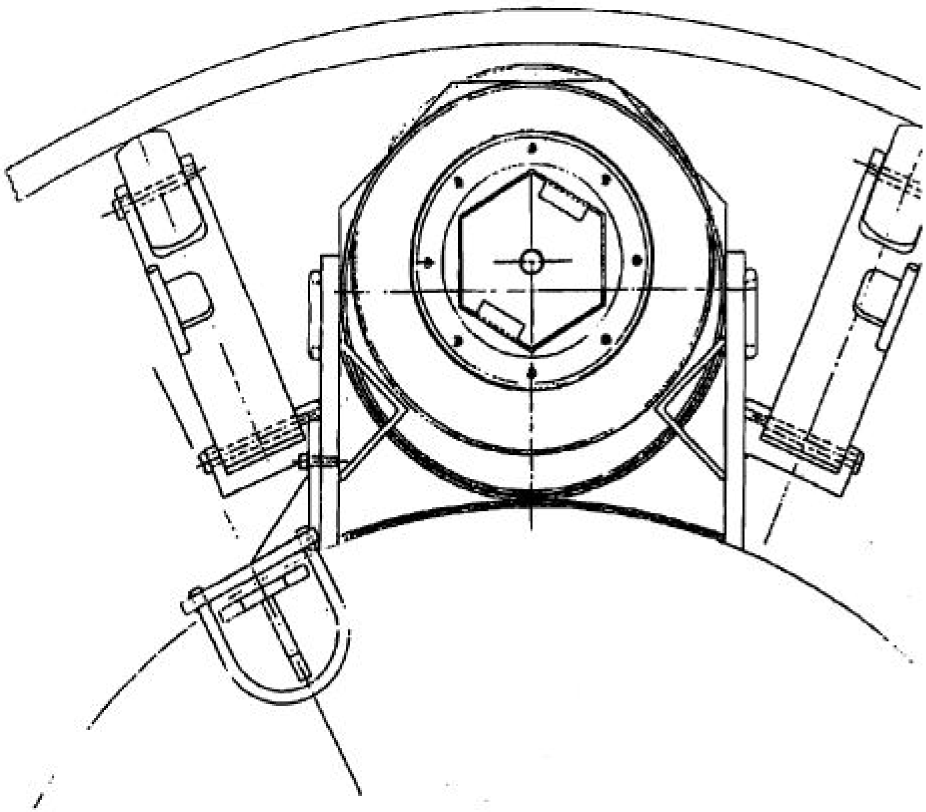

FIGURE 10.11 NEUTRON IRRADIATOR FACILITY - PLAN VIEW

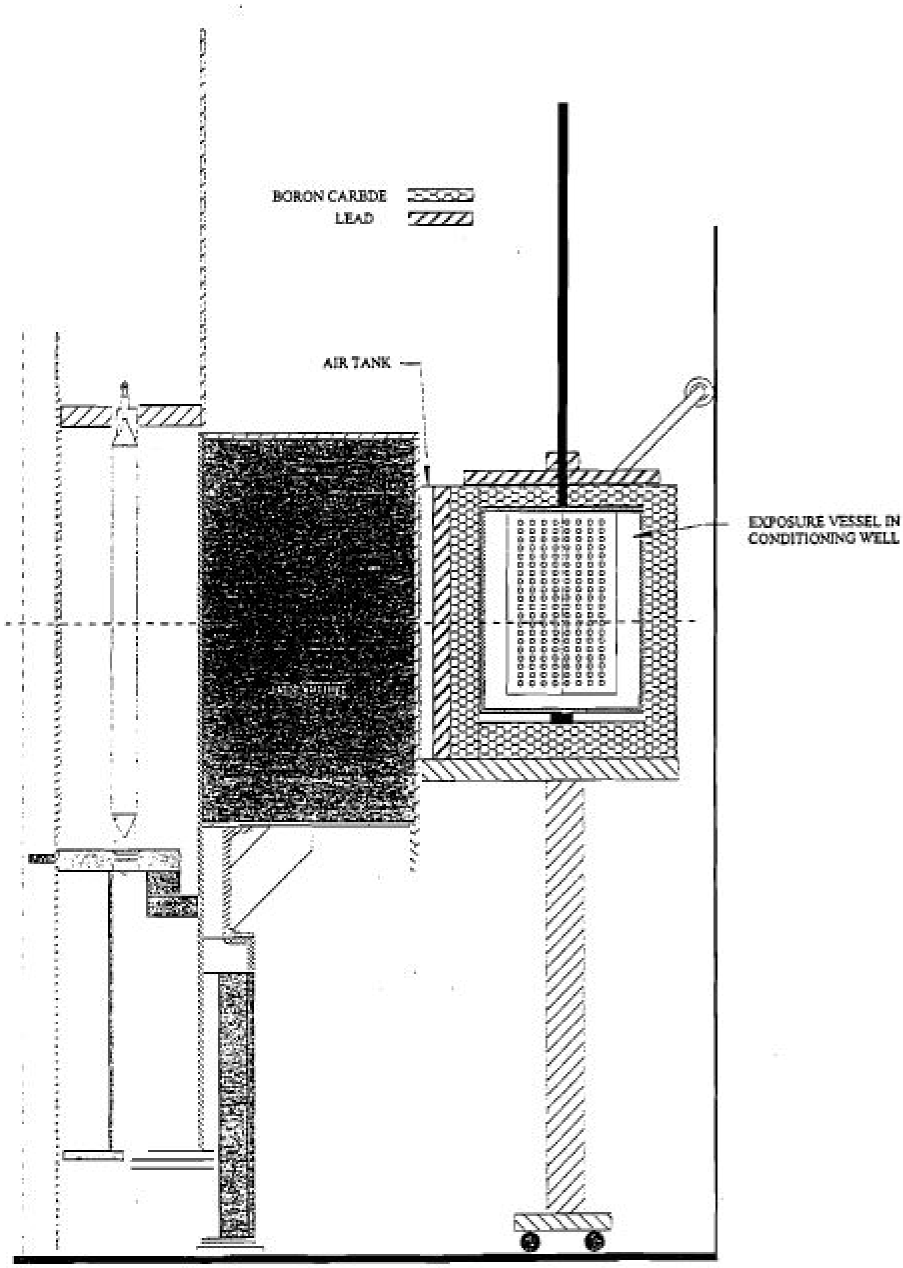

FIGURE 10.12 NEUTRON IRRADIATOR - VERTICAL VIEW

10.5.1.1 Conditioning Well

The Conditioning Well is installed adjacent to the core's graphite reflector in the reactor tank and is held vertically in place by a three wheeled stand which rests on the bottom of the tank. It is held laterally by the levering action of two arms with steel rollers lightly pressing against the reactor tank wall. No fasteners, nuts, or bolts are required to secure the well in place. The inner sleeve is approximately 9.5 in. in diameter and 12 in. deep and both the inner sleeve and outer casing are made of aluminum. The lead and boron nitride are completely enclosed between the inner sleeve and the outer casing.

10.5.1.2 Exposure Vessel

The Exposure Vessel (EV) consists of three major components: the Main Body, the Cylindrical Cup, and the 5-piece Detachable Upper Shield. The Main Body consists of a titanium top plate welded to a 48 in. titanium tube with a multi-pin electrical connector at the top. Attached to the bottom of the top plate are six titanium plates arranged in a hexagon; each plate is approximately 4 in. high and 3.5 in. wide with threaded holes for attaching experiments using aluminum screws and straps. The Cylindrical Cup is constructed of aluminum and covers the hexagonal plates enclosing and sealing the experiments. The cup is approximately 9 in. in diameter and 10 in. high. The cup's inside surface is lined with a gadolinium coating to absorb thermal neutrons. A thin sheet of aluminum protects the gadolinium coating and shields secondary radiation resulting from the neutron absorption in the gadolinium. The 5-piece Detachable Upper Shield is constructed using lead and boron nitride for shielding and is completely encased in aluminum. The shield assembly is placed on top of an assembled EV ensuring the seams are overlapping by at least 45° and then anchored in place by a collar to completely enclose the EV.

10.5.2 Silicon Doping Facility (Neutron Transmutation Doping)

A typical silicon doping facility consists of 5 individual motor drive assemblies mounted as a group to the tank top and positioned over an assembly at the bottom of the tank that positions irradiation canisters in locations adjacent to the reflector (Figure 10.10). The irradiation canisters containing silicon ingots have a recessed bottom section that fit over bayonets for positioning and have drive shafts extending vertically to the motor drives. The shaft assembly has a cross pin that is positioned in a yoke attached to the motor shaft. The weight of the drive shaft and irradiation canister is carried by the yoke assembly.

Each gear reduction motor and drive shaft assembly rotates the irradiation canister· at a slow rotational speed for uniform irradiation of the silicon ingot. The motor drive and shaft assembly is protected from damage by a clutch mechanism in the event the shaft or irradiation canister binds or locks in position. A vertical view of the silicon irradiation facility installation is shown in Figure 10.13.

The silicon irradiation canisters provide a water environment for the silicon ingot and are designed to accommodate the removal underwater of an irradiated ingot. This handling procedure reduces irradiation exposures to individuals handling the ingots in the interest of the ALARA program. An underwater table having an adjustable work platform for vertical positioning in the tank is also utilized in the handling and loading of the irradiation canisters.

It should be noted that these silicon doping positions have not been used for silicon doping in over 10 years and are now used most often for neutron activation analysis of longer-lived isotopes and radiation hardness testing of electronics.

[redacted]

FIGURE 10.13 SILICON IRRADIATION FACILITY – VERTICAL VIEW

10.5.3 Argon Production Facility

The Argon-41 Production Facility can produce up to 4 curies of ^{41}Ar, but normally will only produce 1-2 curies of ^{41}Ar for research and commercial use. The ^{41}Ar will be produced by introducing argon gas into a 6061-T aluminum container located on one of the silicon irradiation positions (adjacent to the graphite reflector and external to the reactor core - Figure 10.14). All the components containing activated ^{41}Ar are located in the reactor room. Argon gas from a commercial argon gas cylinder will supply the irradiation container. After the irradiation container is pressurized (approximately 500 psig) to the desired level, the gas cylinder will be isolated from the irradiation container. To produce the desired activity level of ^{41}Ar the sample will be irradiated for approximately 24 hours.

After irradiation, liquid nitrogen is added to a Dewar. A motor operated valve is opened to pressurize the cooling coils above the liquid nitrogen bath. The Dewar is then raised to cover the cooling coils and ^{41}Ar is cryogenically extricated from the irradiation container. After extrication is completed, the valve from the irradiation container is shut and another motor operated valve is opened. This allows diffusion of ^{41}Ar gas to the sample containers. The liquid nitrogen Dewar is lowered thus exposing the cooling coils. Remote heaters are energized to raise the cooling coil temperature. When that portion of the system between the cooling coils and the sample containers has reached equilibrium, the sample containers will be isolated and removed from the system. The coil is surrounded with a lead shield to minimize the radiation exposure to personnel.

A catch tank surrounds the Dewar to contain any liquid nitrogen escaping from the Dewar or in the unlikely event of a total failure of the Dewar.

Over pressure protection of the overall system is provided by several relief valves that vent to an over pressure tank. The over pressure tank is protected by its own relief valve which vents to the reactor room. The tank is located as high as possible in the reactor room.

All piping and valves in the system are stainless steel. Compression fittings or double-ended shut-off quick connectors are used for all connections normally in contact with the ^{41}Ar.

[redacted]

FIGURE 10.2 FLOOR LAY-OUT FOR ARGON-41 PRODUCTION FACILITY

The Argon-41 Production Facility consists of several different components with the major components listed below:

COMPONENT	MATERIAL	DESCRIPTION
Irradiation Chamber	6061-T aluminum	The irradiation container is a 1000-ml sample cylinder. The container is constructed of 6061-T aluminum with a working pressure of 600 psig and a maximum rated pressure of 1000 psig. It conforms to the "Shipping Container Specifications" from the U.S. Code of Federal Regulations, Title 49 or Bureau of Explosives Tariff No. BOE 6000
Over Pressure Relief Valves	304 stainless steel	The adjustable proportional pressure relief valves have a working pressure up to 6000 psig. When upstream pressure overcomes the force exerted by the spring, the popper opens, allowing flow through the valve. As the upstream pressure increases, flow through the valve increases proportionately. Cracking pressure is only sensitive to inlet pressure and is not affected by outlet pressure
Over Pressure Relief Tank	Carbon steel	30 gallon tank
Valves	304 Stainless steel	Bellows sealed valves
Tubing	304 Stainless steel	1/4-inch and 1/2-inch

10.6 Experiment Review

The UCD/MNRC experiment review and authorization process is described in MNRC-0027-DOC, “Utilization of the University of California- Davis/McClellan Nuclear Research Center Research Facility," and in more detail in MNRC-0033-DOC, “University of California-Davis/McClellan Nuclear Research Center Reactor Facility Experiment Review and Authorization Process." This process requires that any individual wishing to utilize the UCD/MNRC reactor experiment facilities submit an Experimenter Approval Request Form to the UCD/MNRC Director's Office, where it will be reviewed and processed according to procedures in the above documents.

10.6.1 UCD/MNRC Experiment Coordinator (EC)

The UCD/MNRC Experiment Coordinator (EC) is the primary point of contact between the experimenter and the use of the UCD/MNRC experimental facilities. The EC reviews all forms submitted and ensures that all required information has been supplied and validated. The EC then forwards completed requests to the UCD/MNRC Director.

10.6.2 UCD/MNRC Director

The UCD/MNRC Director reviews new submitted experiment requests and takes one of the following actions:

- If the newly proposed experiment, in the judgment of the UCD/MNRC Director falls within the scope of a currently approved Facility Use Authorization and does not qualify as a change, tests, or experiments that is not permitted based on NRC regulations in 10 CFR Part 50.59, or based on facility experience or similar experiments, then the UCD/MNRC Director may approve the proposed experiment; or

- If the proposed experiment does not fall within the scope of one of the currently approved Facility Use Authorizations, the UCD/MNRC Director shall, after any necessary consultation and review by the Experiment Review Board (ERB), submit the experiment and a new or amended Facility Use Authorization to the Nuclear Safety Committee (NSC). The UCD/MNRC Director shall not approve the experiment until an appropriate Facility Use Authorization is approved by the NSC that will allow performance of the newly proposed experiment.

- If the proposed experiment, in the judgment of the UCD/MNRC Director, requires a change to the license or the Technical Specifications per 10 CFR 50.59 the UCD/MNRC Director shall, after any necessary consultation and review by the Experiment Review Board (ERB), submit the proposed change to the license and/or the Technical Specifications and an applicable safety analysis to the NSC for approval prior to submission to the Nuclear Regulatory Commission (NRC). The UCD/MNRC Director shall not approve the experiment until licensing authorization is received from the NRC and an appropriate Facility Use Authorization has been approved by the NSC.

10.6.3 Experiment Review Board

The Reactor Supervisor serves as the ERB Chairman and conducts the ERB meetings in accordance with a written charter. The ERB is assembled as a working group that performs a technical evaluation of proposed UCD/MNRC experiments sent to them by the UCD/MNRC Director. As a result of their technical evaluation, the ERB Chairman makes a recommendation to the UCD/MNRC Director concerning the approval or disapproval of the experiment.

10.6.4 Nuclear Safety Committee (NSC)

The Nuclear Safety Committee (NSC) is responsible for oversight of radiation safety and nuclear operations at the UCD/MNRC facility and operates in accordance with a written charter. Once the UCD/MNRC Director submits a new or amended Facility Use

Authorization for approval, the NSC reviews the new or amended Facility Use Authorization and takes one of the following actions:

- The NSC may approve the new or amended Facility Use Authorization and return it to the UCD/MNRC Director for implementation, or

- The NSC may disapprove the new or amended Facility Use Authorization and send it back to the UCD/MNRC Director for resolution of NSC concerns.

CHAPTER 11

RADIATION PROTECTION AND WASTE MANAGEMENT PROGRAM

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11.0 RADIATION PROTECTION AND WASTE MANAGEMENT

This chapter deals with the overall MNRC radiation protection program and the corresponding program for management of radioactive waste. The chapter is focused on identifying the radiation sources which will be present during normal operation of the reactor and upon the many different types of facility radiation protection programs carried out to monitor and control these sources. This chapter also identifies expected radiation exposures due to normal operation and use of the reactor. Many of the detailed calculations supporting this chapter are contained in Appendix A.

11.1 Radiation Protection

The purpose of the MNRC radiation protection program is to allow the maximum beneficial use of radiation sources with minimum radiation exposure to personnel. Requirements and procedures set forth in this program are designed to meet the following fundamental principles of radiation protection:

- Justification - No practice shall be adopted unless its introduction produces a net positive benefit;

- Optimization - All exposures shall be kept as low as reasonably achievable, economic and social factors being taken into account;

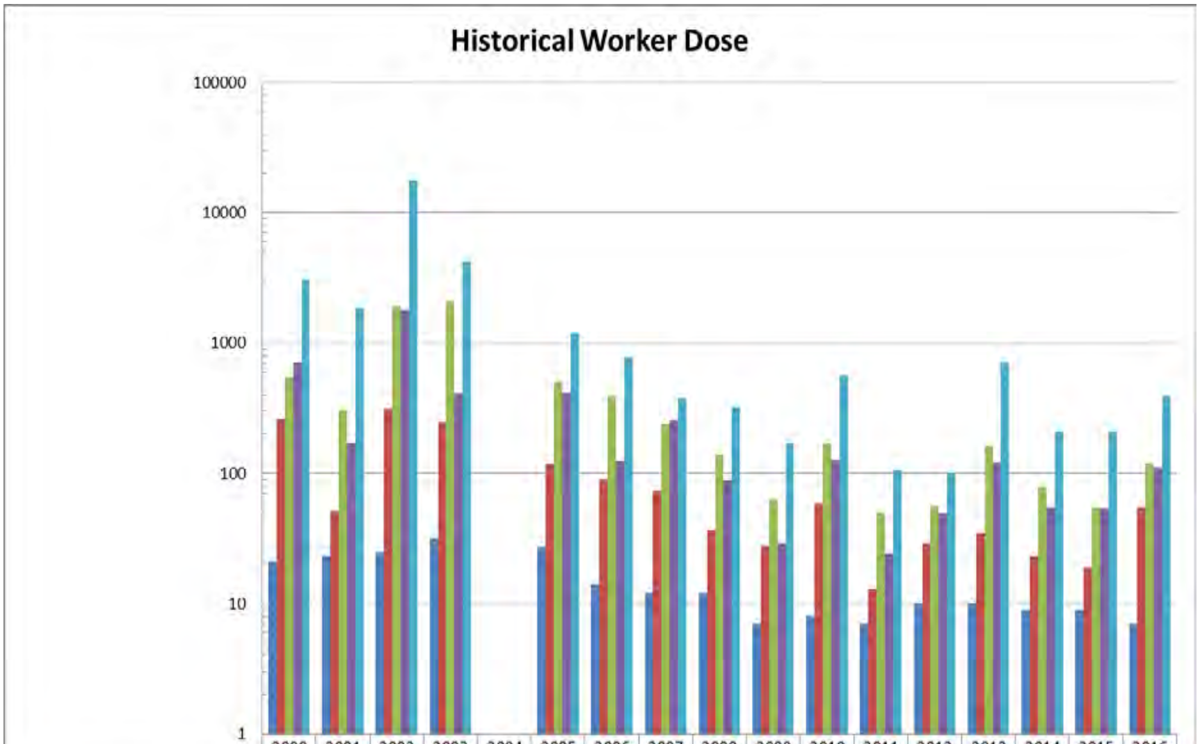
- Limitation - The dose equivalent to individuals shall not exceed limits established by appropriate state and federal agencies. These limits shall include, but not be limited to, those set forth in Title 10, Code of Federal Regulations, Part 20 (10 CFR 20) (Reference 11.1).

The radiation protection measures used at the MNRC are patterned after other TRIGA® reactor facilities where the radiation sources are much the same as well as the ANSI 15.11 standard. Facility organization charts, actual radiation measurements and operating data from around the MNRC, and a description of radiation protection program components will be used to characterize the features of the different programs used to maintain occupational doses and releases of radioactivity to the unrestricted environment as low as reasonably achievable (ALARA).

11.1.1 Historical Health Physics Data from MNRC

This section is provided to give quantitative data on the relevant health physics information for the MNRC from the year 2000 to the year 2016. It is important to note data from the year 2004 could not be located at MNRC or on the NRC database. It is reasonable to assume that based on the relatively high number of MW hours that year, all health physics data was near the highest annual values recorded (2002, 2003, 2005, and 2006).

The most significant historical trend that can be observed is the reduction in all health physics doses and effluence beginning in the year 2006. This drop in doses and effluence can be directly attributed to the facility moving from two-shift to one-shift operation. Further reduction in doses and effluence in subsequent years can be explained by the fact the reactor was operated less often above 1.0 MW.



	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016
# OF INDIVIDUALS	21	23	25	32		27	14	12	12	7	8	7	10	10	9	9	7
AVG TEDE mrem	262	51	312	248		118	90	74	37	28	59	13	29	34.8	23	19	55
GREATEST TEDE mrem	552	308	1892	2072		507	390	242	140	63	169	50	56	162	79	55	120
AVG EXTREMITY mrem	709	170	1796	410		414	126	257	89	29	128	24	49	122	55	53	111
GREATEST EXTREMITY mrem	3074	1840	17757	4218		1194	772	375	323	168	562	105	99	705	209	210	398

FIGURE 11.1 HISTORiCAL MNRC WORK DOSE DATA.

Personnel doses, as seen in figure 11.1, have been well below 10 CFR 20 limits. Since the year 2008 average radiation work TEDE have been approximately 1% of the 10 CFR 20 limits.

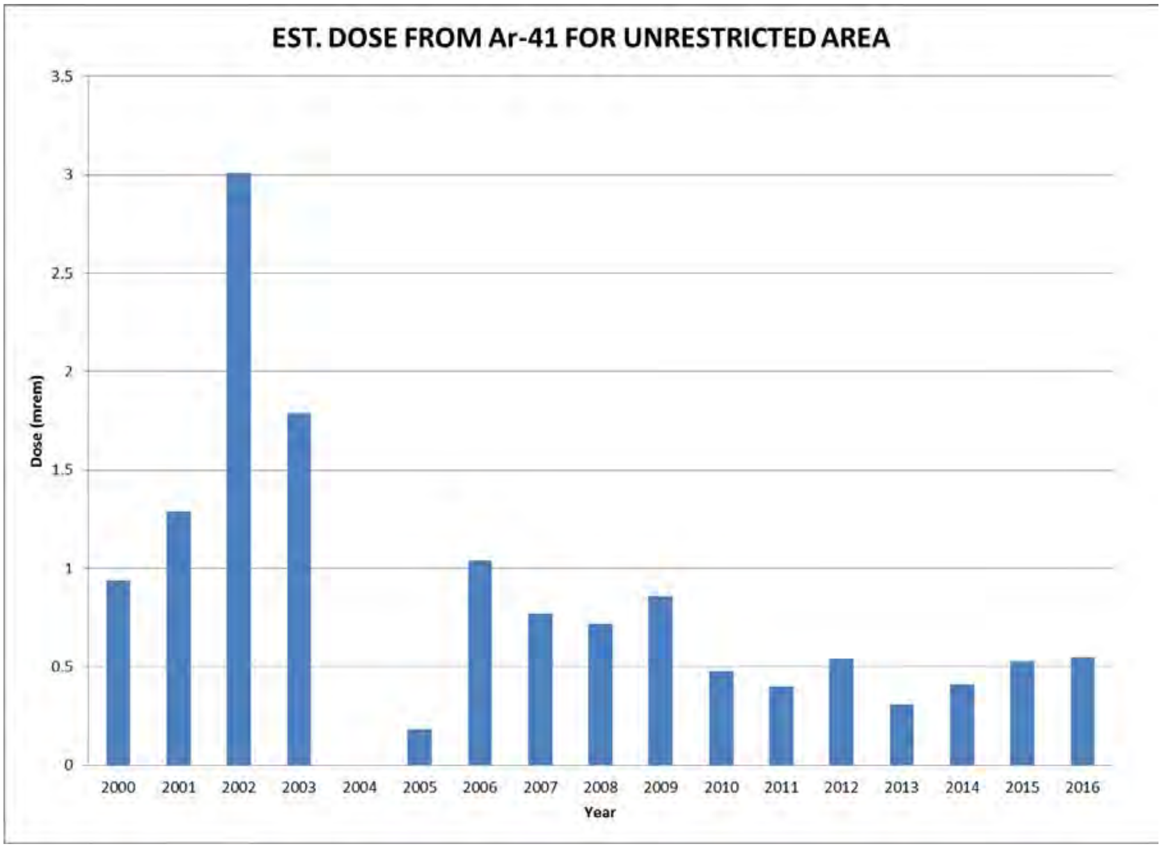


FIGURE 11.2 MAXIMUM EXPOSED INDIVIDUAL DOSE FROM MNRC AR-41 EFFLUENCE

All doses in unrestricted areas from Ar-41 effluences, as seen in figure 11.2, are below the 10 CFR 20 limits. Just as with radiation worker doses a decrease in unrestricted area doses can be seen in 2006 and again in 2010. This can be explained by the fact that the facility moved to single-shift operation in 2006 and rarely operated above 1.0 MW after 2009. These doses are for the maximum exposed individual standing at the highest dose rate position caused by Ar-41 effluence for the entire year. The CAP88 model provides a much more realistic estimation of public dose from radioactive effluence. These CAP88 values are significantly lower than the values given in figure 11.2.

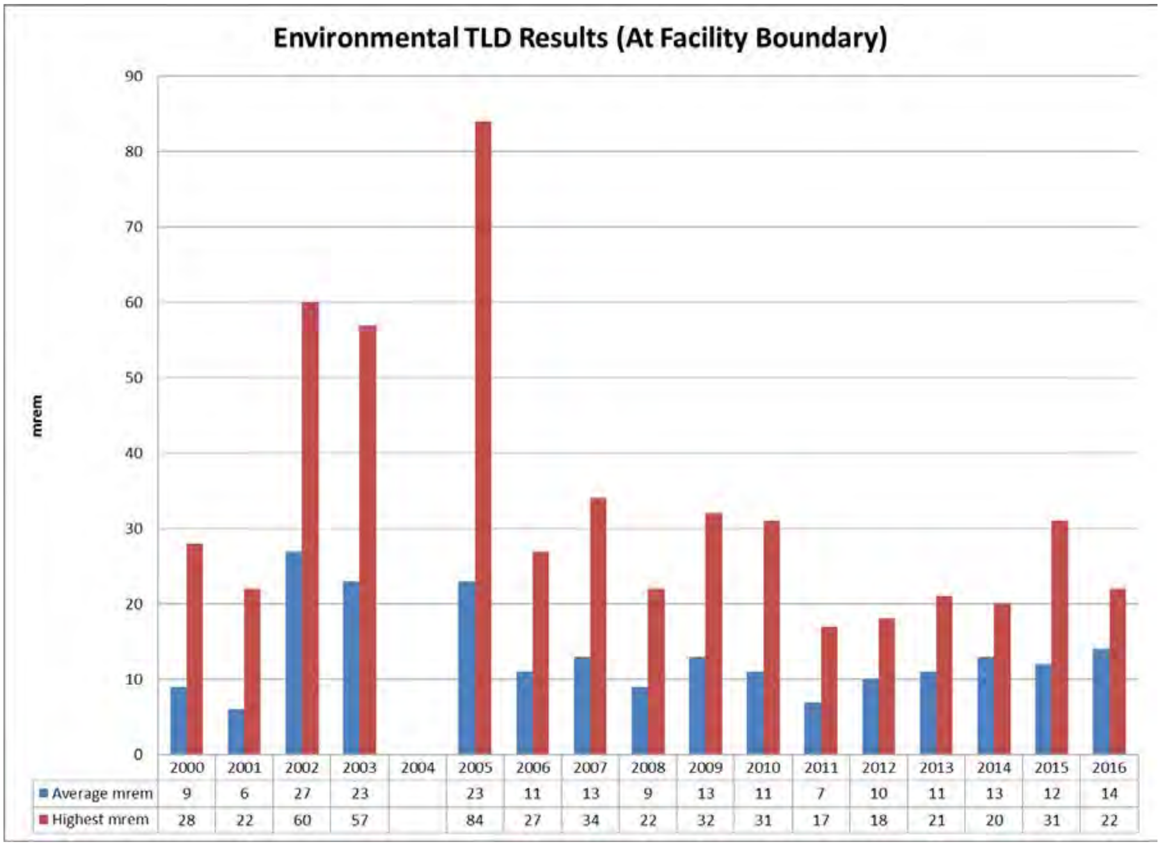


FIGURE 11.3 ANNUAL DOSE AT MNRC FACILITY BOUNDARY

Elevated radiation levels from reactor operations have been measured at the MNRC's boundaries. These values have been below the applicable 10 CFR 20 limits. As with personnel dose and Ar-41 effluence, facility boundary dose has fallen as annual reactor MW hours have decreased.

As MNRC continues its mission of neutron radiography, in-tank irradiations, education, and outreach it is likely these objectives can be completed during single-shift 1 MW operation. Therefore, during the foreseeable future all doses and effluences should be very comparable to the historical values observed from approximately 2006 until 2016. Any new experiments or operation that may cause a change in doses or effluence shall be evaluated as part of the MNRC's ALARA program.

11.1.2 Radiation Sources

The radiation sources present at the MNRC can be categorized as airborne, liquid, or solid. While each of these categories will be discussed individually in Sections 11.1.2.1 through 11.1.2.3, the major contributors to each category can be summarized as follows: Airborne sources consist mainly of Argon-41 (Ar-41, half-life 1.8 hrs.), due largely to neutron activation of air in the radiography bays and air dissolved in the reactor's primary coolant, and Nitrogen-16 (N-16, half-life 7.1 sec.), due to neutron interactions with oxygen in the primary coolant.

Liquid sources are quite limited at the MNRC and include mainly the reactor primary coolant. No routine liquid effluent or liquid waste is anticipated. Any non-routine liquid effluent or liquid waste will be discharged in accordance with 10 CFR 20.2003 requirements. Solid sources are a bit more diverse, but for the most part are very typical of a TRIGA® reactor facility. Such sources include the fuel in use in the current 1 MW core, irradiated fuel from the former 8.5 wt% fuel core, and unirradiated fuel. In addition, other solid sources are present such as the neutron startup source (AmBe), small fission chambers (NM-1000) for use with nuclear instrumentation, irradiated components subjected to neutron radiography, other items irradiated as part of normal reactor use, and small instrument check and calibration sources. Solid waste is yet another solid source, but is expected to be very limited in volume and curie content.

11.1.2.1 Airborne Radiation Sources

During normal operation of the MNRC reactor, there are two sources of airborne radioactivity, namely Ar-41 and N-16. The assumptions and calculations used to assess the production and radiological impact of these airborne sources during normal operations are detailed in Appendix A. Therefore, that information will only be summarized in this section.

Fuel element failure, although not expected, could occur while the reactor is operating normally. Such a failure would usually occur due to a manufacturing defect or corrosion of the cladding and would result in a small penetration of the cladding through which fission products would be slowly released into the primary coolant. Some of these fission products, primarily the noble gases, would migrate from the cooling water into the air of the reactor room. Although this type of failure could occur during normal operation, its occurrence is <u>not</u> normal and no normal operation would take place after such an event until the situation had been eliminated (i.e., the failed element located and removed from the core). As a result, the failure of a single element (for any reason) is evaluated in Appendix B as an abnormal situation or an accident, and is discussed further in Chapter 13.

11.1.2.1.1 Argon-41 in the Radiography Bays

Given the low neutron beam intensity in the neutron radiography bays the amount of Ar-41 produced is difficult to accurately measure. Therefore, the saturation Ar-41 activity is calculated by taking the average volume of each of the 4 MNRC neutron beamline and using the average beamline neutron radiography assumed to be at the MNRC radiography plane. The results of these analyses are given below.

TABLE 11-1: MNRC NEUTRON BEAMLINE CHARACTERISTICS

	Equivalent Diameter (cm) @ Film Plane	Area cm^2	Length* cm	Beam Volume cm^3	Effective Target Mass of Argon (g)
Radiography Bay 1	58.4	2680	941	2.52×10^6	28.3
Radiography Bay 2	58.4	2680	762	2.04×10^6	23.0
Radiography Bay 3	25.4	510	581	2.96×10^5	3.3
Radiography Bay 4	58.4	2680	762	2.04×10^6	23.0

TABLE 11-2: RESULTING Ar-41 PRODUCTION IN MNRC NEUTRON BEAMLINES

	Flux (n/cm^2 s)	Ar-41 produced (uCi/s)	Bay Volume (ml)	Maximum Ar-41 concentration (uCi/ml)
Radiography Bay 1	3.0×10^6	2.39×10^{-3}	2.09×10^9	1.08×10^{-8}
Radiography Bay 2	3.0×10^6	1.94×10^{-3}	1.11×10^9	1.67×10^{-8}
Radiography Bay 3	4.3×10^6	4.00×10^{-4}	1.7×10^8	2.24×10^{-8}
Radiography Bay 4	3.0×10^5	1.94×10^{-4}	4.63×10^8	4.00×10^{-9}

The results above show that Ar-41 production and maximum concentrations show the radiological consequences of Ar-41 production in the MNRC beamline are very modest. The maximum concentration of Ar-41 on the radiography bays is over 100 times less than the DAC value for Ar-41. In reality the concentration in these radiography bays will be significantly less due to the air changes in the room and the fact the beamlines are not "on" all of the time the reactor is operating. These low Ar-41 levels are further confirmed by the fact that neutron radiographer annual whole body dose is essentially zero despite spending hundreds of hours per year in the radiography bays.

11.1.2.1.2 Production and Evolution of Ar-41 in the Reactor Room

Though the concentration of Ar-41 in the reactor primary water and in the air in the reactor room can be calculated, it is more prudent to provide measured Ar-41 concentrations.

Actual measurements of Ar-41 in the reactor room after the reactor had operated for about 9 hours at 1 MW (reactor room exhaust system on) showed Ar-41 concentrations averaging about 1.5×10^{-6} µCi/ml for areas which are occupied during normal work in the room. Using the semi-infinite cloud, the corresponding dose would be 1.25 mRem/hr. However, given the small size of the reactor room the semi-infinite cloud model will greatly over predict dose. In actuality the Ar-41 concentration in the reactor room contributes only slightly to the typical measured reactor room dose rate of 3-4 mrem/hr.

Actual measurements of Ar-41 in the primary coolant water (after 4 MW hours of operation) average 1.0 x 10-3 µCi/ml. This provides strong indication that the vast majority of the Ar-41 effluences to the environment is from Ar-41 activation in the primary coolant and not the neutron beamlines.

11.1.2.1.3 Ar-41 from the Pneumatic Transfer System

Ar-41 will also be produced in the section of the pneumatic transfer system that is located in the reactor core. During operation of the transfer system, air containing very small amounts of Ar-41 is exhausted from the system through a HEPA filter to the facility stack. There has not been a significant increase in Ar-41 releases, as measured by the stack monitor, from numerous operations of this system. Therefore, the Ar-41 from the pneumatic transfer system is not considered to be a measurable contributor to the Ar-41 doses associated with MNRC operations.

11.1.2.1.4 Ar-41 Release to the Unrestricted Area

The Ar-41 will be discharged from the MNRC through the exhaust stack, which is 60 feet above ground level. Dilution with other building ventilation air and atmospheric dilution will reduce the Ar-41 concentration considerably before the exhaust plume returns to ground level locations which could be occupied by personnel or the general public.

It is important to note that only a modest amount of dilution is required to reduce the Ar-41 concentration to a level that is well below the 10 CFR Part 20 limit of 1×10^{-8} µCi/ml for unrestricted areas. Based on 2019 effluence data the MNRC operated for 1,430 hours at 1 MW and produced 27.6 Ci of Ar-41. This corresponds to an emission rate of 5.4×10^{-6} Ci/s. Based on a typical stack flow rate of 5678 CFM a concentration of 2.0×10^{-6} µCi/ml will be effluence during one MW operations. Though it is very unlikely MNRC will be operated again 24/7 the subsequent calculation will be based on a continuous effluence rate of 2.0×10^{-6} µCi/ml.

The radiography bay ventilation system provides both a significant dilution effect and increases the effective stack height by increasing the effluence exiting velocity. Both of which are taken credit for in this analysis. Therefore, the radiography bay ventilation system must be operated on a regular basis while the reactor is on. However, the radiography bay ventilation system does not need to be operated when the reactor is shutdown when no Ar-41 is being produced.

The Lawrence Livermore National Laboratory program "HotSpot" was used to determine the worst case radiation dose impact to the public for a variety of atmospheric stability class. Over the past several years the program has become the industry standard for relatively simple Gaussian plume modeling. The program was also used to provide the down field centerline maximum concentration position. Based on this distance the maximum concentration at ground level was calculated using the established Gaussian plume model equation found below. Dispersion coefficients were calculated based on the equations provided in the HotSpot user manual. Ground level maximum concentrations were selected as the most appropriate to evaluate compliance with 10 CFR 20 appendix B effluence limit concentration of Ar-41 as the prevailing wind direction will take MNRC effluence the majority of the time (chapter 2) into an area of the McClellan air field that is flat, controlled access, and largely free of any elevated buildings.

For the modeling in HotSpot the following inputs and assumptions were made:

-Average wind speed is assumed to be 3.4 m/s (chapter 2).
-MNRC Stack height is 18.2 m.
-Emission rate of Ar-41 of 5.4 x 10^{-6} Ci/s.
-Effluence exit velocity of 16 m/s based on 5678 cfm effluence rate and stack exit diameter of 0.5 meters which produces an effective stack height of 25 m in all scenarios.
-The "sample time" was kept at the default value of 10 minutes.
-Standard/rural terrain was used (conservative).

Hotspot unfortunately does not provide maximum centerline effluence concentration which is a value of concern. Hotspot was used to determine the ground level downwind distance at which the maximum concentration is expected to occur. Then the following established Gaussian Plume model equation was used to determine the maximum Ar-41 concentration in order to compare the results to regulatory limits of 1 x 10^{-8} uCi/ml. Sigma y and sigma z values were determined based on the formulas provided in the HopSpot user manual.

$$\chi_{(max,0)} = \frac{Q}{\pi u \sigma_z \sigma_y} exp\left(-\frac{1}{2}\frac{H^2}{\sigma_z^2}\right) = \frac{Ci}{m^3} = \frac{\mu Ci}{cm^3};$$

where:

Q = Emission Rate
H = 25m effective stack height (18.2 m physical stack height);
u = Mean Wind Speed (m/s);
σ_y = diffusion coefficient in the y-axis
σ_z = diffusion coefficient in the z-axis

TABLE 11-3: MNRC OFFSITE AR-41 RADIOLOGICAL CONSQUENCES FOR CONTINUOUS 1 MW OPERATIONS (w/ PLUME RISE)

Atmospheric Stability Classification	*Sigma y* (m)	Sigma z (m)	χ_{max} ($\mu Ci/cm^3$)	Distance to Max Dose (m)	Maximum Dose per Year (mrem)
Very Unstable (A)	19.3	17.6	5.4×10^{-10}	88	3.7
Moderately Unstable (B)	23.8	18	4.5×10^{-10}	150	3.1
Slightly Unstable (C)	23.9	17.3	4.3×10^{-10}	220	2.9
Neutral (D)	27.1	16.6	3.6×10^{-10}	340	2.4
Slightly Stable (E)	40.2	17.1	2.5×10^{-10}	690	1.2
Moderately Stable (F)	52.3	15.8	1.7×10^{-10}	1400	0.8

The results given above show that even under 24/7 operations at the maximum licensed steady state power Ar-41 effluence will not exceed 10 mRem per year for the maximum exposed individual located at the highest concentration year round. The most likely average annual atmospheric condition is class "C" slightly unstable which results in a maximum dose of 2.9 mrem per year which corresponds to a delusion factor of 4640 from the stack effluence to the maximum ground level concentration. The worst case atmospheric condition is class "A" very unstable which results in a maximum dose of 3.7 mrem per year which corresponds to a dilution factor of 3703 from the stack effluence to the maximum ground level concentration.

Determination of radiation dose to the general public from airborne effluents may also be carried out using several other computer codes recognized by regulatory authorities. One such method involves use of the Clean Air Assessment Package - 1988 (CAP88-PC) (Reference 11.3). Application of this code (V1.0) to the projected Ar-41 releases from the MNRC predicts a dose to the general public of less than 0.1 mrem per year.

11.1.2.1.5 Production and Evolution of N-16 in the Reactor Room

In addition to Ar-41, the other source of airborne radioactivity during normal operation of the MNRC reactor is Nitrogen-16 (N-16). N-16 is generated by the reaction of fast neutrons with Oxygen-16 (O-16) in water passing through the core. The amount of oxygen present in air, either in a beam path or entrained in the water near the reactor core, is insignificant compared to the amount of oxygen in the water molecule in the liquid state. Production of N-16 resulting from neutron interactions with the oxygen in air and air entrained in the cooling water can therefore be neglected.

The cross-section energy threshold for the O-16 (n,p) N-16 reaction is 9.4 MeV; however, the minimum energy of the incident neutrons must be about 10 MeV because of center of mass corrections. This high energy threshold limits the production of N-16, since only about 0.1% of all fission neutrons have an energy in excess of 10 MeV. Moreover, a single hydrogen scattering event will reduce the energy of these high-energy neutrons to below the necessary threshold.

After N-16 is produced in the core region, it rises to the tank surface and forms a disc source which creates a direct radiation field near the top of the tank. Some of the N-16 is subsequently released into the reactor room. Calculations for the production and mixing of N-16 in the primary coolant and for the evolution of N-16 from the reactor tank into the reactor room air are presented in Appendix A. Radiation levels associated with the N-16 in the tank and in the reactor room air are also addressed as part of Appendix A. Without exception, the calculated N-16 concentrations and dose rates are very conservative because they do not assume use of the conventional in-tank N-16 diffuser system, which is present in the MNRC primary water circulation system. Since this diffuser system is used during all normal operation of the reactor, and is designed to significantly delay the N-16 transit time to the upper regions of the tank, the 7.14 second N-16 half-life brings about considerable decay and a corresponding reduction in N-16 radiation levels at the tank surface and in the reactor room itself.

The escape of N-16 into the reactor room air will also deliver a radiation dose to workers in the room based on the N-16 concentration, which will be influenced by dilution in room air, by decay of this short-lived radionuclide and by room ventilation. This makes calculating the concentration of N-16 at various locations in the reactor very challenging and somewhat academic in nature. It is known that Ar-41 and radioactive contamination in the primary water are relatively minor contributors to the dose rate in the reactor room. Therefore, the majority of the dose rate in the reactor room must come from N-16 production. The N-16 concentrations during 1 MW operations produce dose levels of approximately 30 mRem/hr one foot above the reactor pool, 4-8 mRem/hr at the reactor room RAM 5 feet above the side of the reactor tank, and 1-2 mRem/hr in the rest of the reactor room. These dose rates are mitigated by minimizing the amount of time workers and visitors are allowed to stay in the reactor room and by closing monitoring recorded worker and visitor dose.

11.1.2.1.6 Ar-41 from the Ar-41 Production Facility

Ar-41 will be produced by the Ar-41 Production Facility (see Chapter 10) as needed. The Ar-41 that is produced by the Ar-41 Production Facility will be contained in the system so there should be no increase in the Ar-41 levels in the reactor room or the Ar-41 that is released to the unrestricted area. Catastrophic failure of the system will not result in any 10 CFR 20 limit being exceeded and is further discussed in Chapter 13.

11.1.2.2 Liquid Radioactive Sources

Liquid radioactive material routinely produced as part of the normal operation of the UCD/MNRC will be miscellaneous neutron activation product impurities in the primary coolant, most of which are deposited in the mechanical filter and the demineralizer resins. Therefore, these materials are dealt with as solid waste (Section 11.1.2.3, Table 11-5). Non-routine liquid radioactive waste could result from decontamination or maintenance activities (i.e., filter or resin changes). The amount of this type of liquid waste is expected to remain small, especially based on past experience. Because of this, the liquid will be processed to a solid waste form on site and will be disposed of with other solid wastes.

11.1.2.2.1 Radioactivity in the Primary Coolant

As mentioned above, the only significant liquid radioactive source at the MNRC is the reactor primary coolant. Radioactivity in this liquid source occurs due to neutron activation of Argon-40 in entrained air (creating Ar-41); neutron interactions with oxygen in the water molecule (creating N-16); and neutron interactions with tank and structural components with subsequent transfer of the radioactivity into the primary coolant. Radionuclides such as 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 activation of natural deuterium in water and from tritium migrating out of the fuel from the fission process.

As noted, other sources of liquid radioactivity are not currently projected for the MNRC reactor operations.

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 purification system) remain constant. The typical primary coolant radionuclide concentrations are given below (~30-35 MW hours per week of operation) along with the projected equilibrium concentrations at 1 MW.

TABLE 11-4 PREDOMINANT RADIONUCLIDES AND THEIR EQUILIBRIUM CONCENTRATION IN MNRC REACTOR PRIMARY COOLANT AT 1 MW

Radionuclide	Half Life	Typical Concentration at 1MW (uCi/mL)
Aluminum-28	2.3 min	3.0×10^{-3}
Argon-41	1.8 hr	1.0×10^{-3}
Hydrogen-3	12 yr	2.0×10^{-2}
Magnesium-27	9.46 min	2.0×10^{-4}
Molybdenum-99	2.75 d	1.0×10^{-6}
Cobalt-60	5.27 y	1.0×10^{-6}
Manganese-56	2.58 hr	5×10^{-5}
Sodium-24	14.96 hr	2.0×10^{-4}

As mentioned, it is MNRC policy not to release liquid radioactivity as an effluent or as liquid waste. Therefore, the primary coolant does not represent a source of exposure to the general public during normal operations. Furthermore, occupational exposure from liquid sources is also limited because there are few operations which require contact with the primary coolant. In cases where contact is a potential, such as in certain maintenance operations, the primary coolant could be allowed to decay for several days or more to significantly reduce radioactivity concentrations. Because of the short half-lives of most of the predominant radionuclides in the primary coolant, most radionuclides would be essentially gone after 48 hours, sodium-24 would be reduced by about a factor of 10, and experience at other TRIGA® reactors indicates that Hydrogen-3 would not be a source of significant occupational dose.

11.1.2.2.2 N-16 Radiation Dose Rates from Primary Cooling System Components

N-16 has been addressed previously in Section 11.1.2.1, however, the potential for N-16 radiation dose rates from primary coolant piping and from the heat exchanger were not included in that discussion. Measurements of gamma dose rates at contact with these cooling system components after extended operation at 1 MW is 1 to 2 millirem per hour. These radiation levels are not considered abnormal and do not represent a radiation protection problem since they were expected and they occur inside the posted radiation area on the second floor of the reactor building.

11.1.2.3 Solid Radioactive Sources

The solid radioactive sources associated with the MNRC program are summarized in Table 11-5. Because the actual inventory of reactor fuel and other radioactive sources continuously changes as part of the normal operation, the information in Table 11-5 is to be considered representative rather than an exact inventory.

TABLE 11-5 REPRESENTATIVE RADIOACTIVE SOURCES FOR THE UCD/MNRC

[redacted]

Although solid waste is included in the preceding table, more information on waste classification, storage, packaging and shipment is included in Section 11.2. In an effort to elaborate somewhat on the waste entry in Table 11-5, it can be stated that routinely produced solid waste includes water purification system demineralizer resin bottles, mechanical filters, rags, paper towels, plastic bags, rubber gloves, and other materials used for contamination control or decontamination. The radioactivity level of this material is normally in the microcurie range, and it is anticipated that approximately one (or two) regular 55 gallon drums of this type of material and 2 resin bottles will be generated each year. Typically, only one "B-25" box of radioactive waste is shipped from the facility every 5 years.

11.1.2.3.1 Shielding Logic

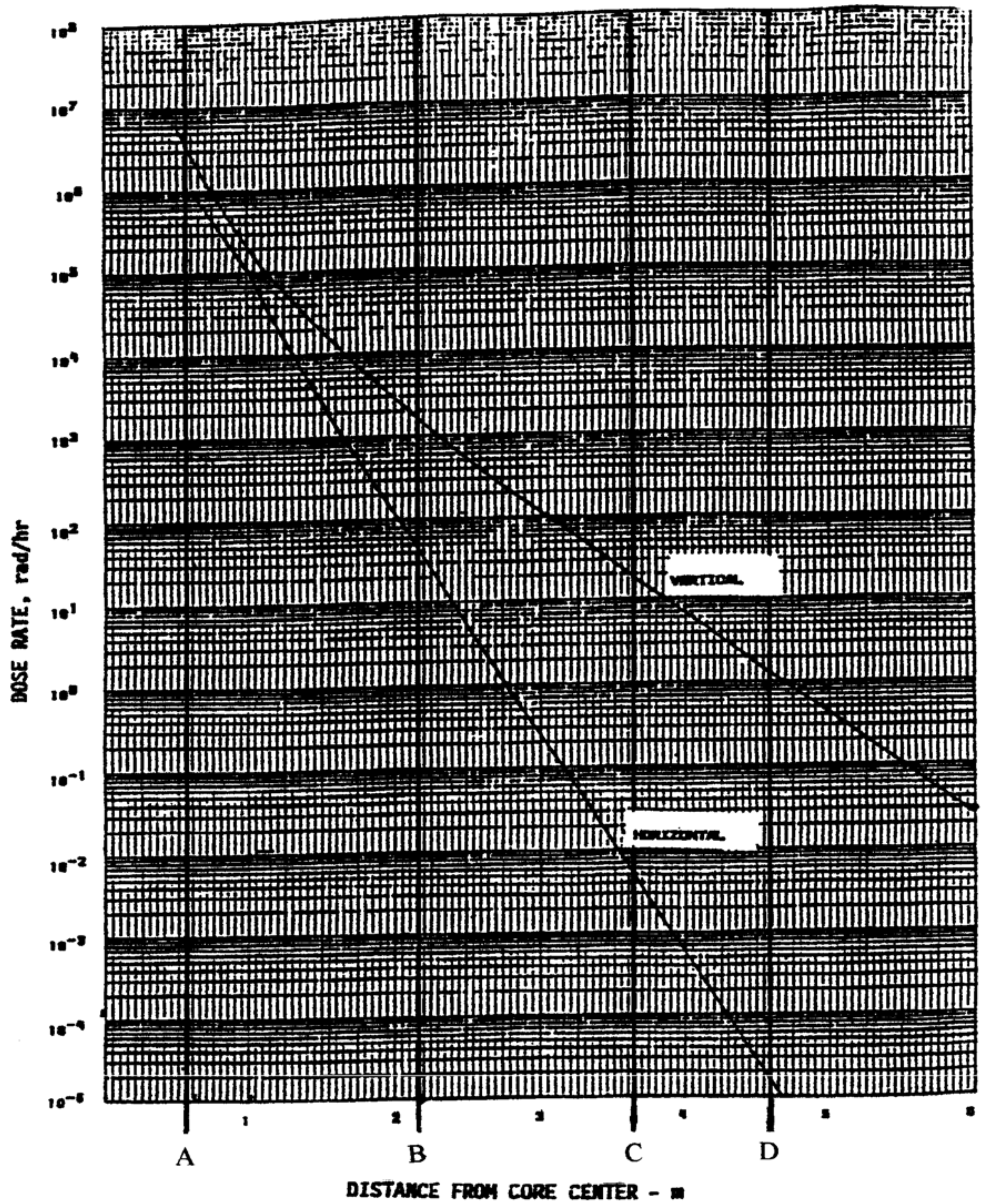
The MNRC reactor bulk shield is very similar, in material type and thickness, to other proven TRIGA® shields. Two significant differences are the beam tube penetrations. Where the basic shielding configuration has been penetrated by beam tubes, supplemental shielding was added. This supplemental shielding has been designed to provide the same attenuation to both neutrons and gammas as the basic unpenetrated shield. The second is the Bay 5 cavity. The radiation levels at the surface of the biological shield as a result of the Bay 5 cavity cut out are 0.35 mR/hr γ and < 1 mrem/hr neutron on contact.

11.1.3 Radiation Protection Program

The health physics program for the UCD/MNRC reactor is located organizationally within the UCD/MNRC. The organizational structure and reporting pathways relating to the UCD/MNRC radiation protection program are shown in Figures 11.5 and 11.6.

11.1.3.1 Organization of the Health Physics Branch

The Health Physics Branch within UCD/MNRC is the organization that administers the radiation protection program for the reactor.



A: Core Boundary

B: Tank Boundary

C&D: Biological Shield

FIGURE 11.4 REACTOR BULK SHIELD DIRECT DOSE RATE - 1 MW OPERATION

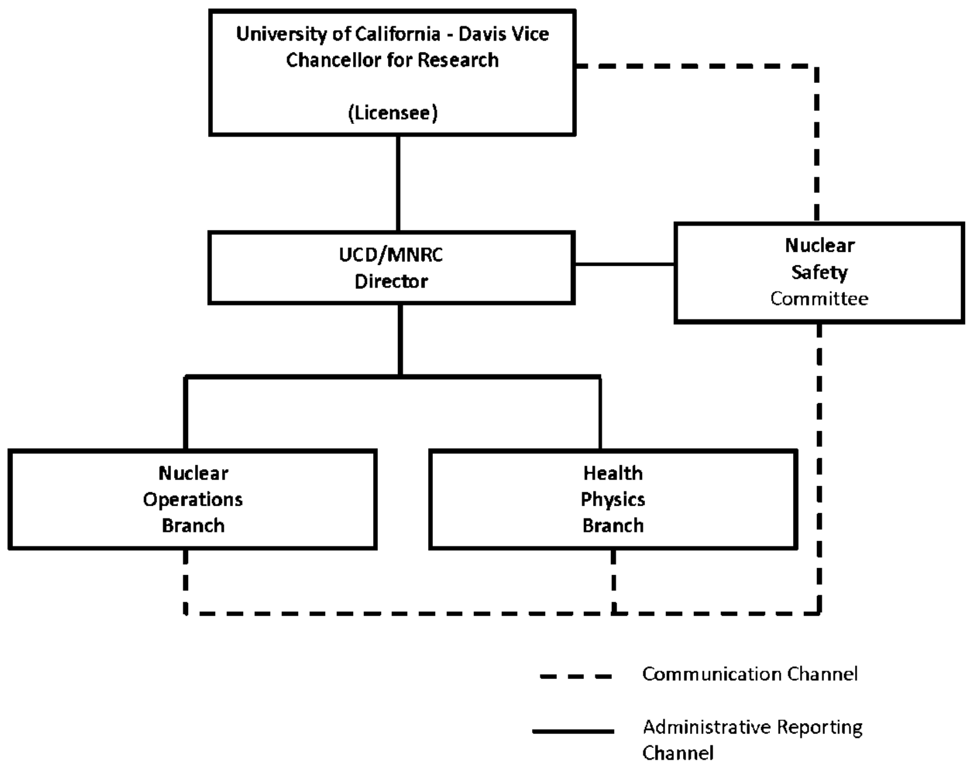


FIGURE 11.5 ORGANIZATIONAL STRUCTURE OF THE MNRC RADIATION PROTECTION PROGRAM

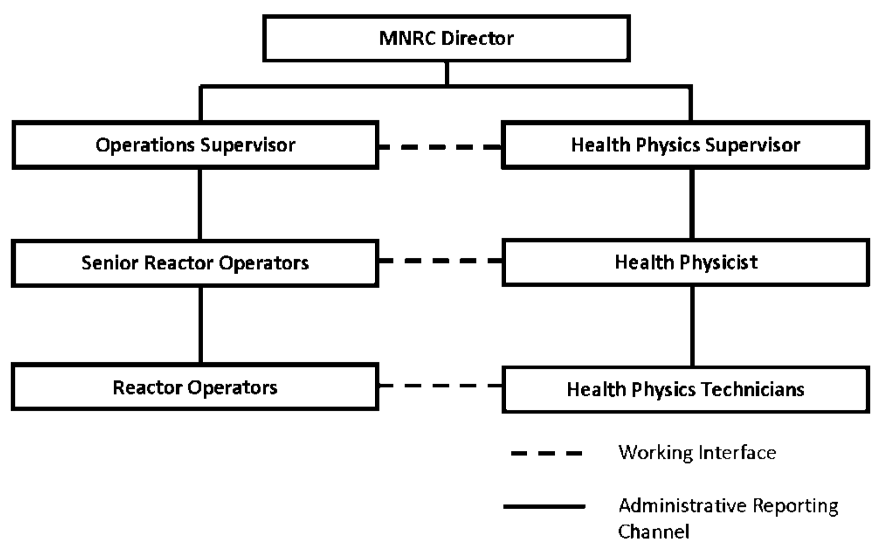


FIGURE 11.6 ORGANIZATIONAL STRUCTURE SHOWING THE RADIATION PROTECTION PROGRAM WITHIN THE UCD/MNRC

The positions of authority and responsibility within the Health Physics Branch are as follows:

- Health Physics Supervisor (Radiation Safety Officer) - The Health Physics Supervisor reports directly to the MNRC Facility Director. The Health Physics Supervisor is responsible for directing the activities of the Health Physics Branch including the development and implementation of the MNRC Radiation Protection Program;

- Health Physicist - Health Physicists report to the Health Physics Supervisor. Health Physicists are responsible for implementing the MNRC Radiation Protection Program policies and procedures, and directing the activities of the Health Physics Technicians.

- Health Physics Technicians - Health Physics Technicians report directly to the Health Physicist on-duty. Health Physics Technicians are responsible for providing radiological control during reactor operations and maintenance. This includes radiological monitoring, surveillance checks on radiological monitoring equipment and radiological control oversight of operations involving radiation and/or contamination. The position description for the health physics technician specifies the authority to interdict perceived unsafe practices. Typically, ROs and SROs are trained to perform some of the tasks described here. ROs and SROs trained in this manner are to report health physics issue to the RSO and other operational issue to the reactor supervisor.

The qualifications for the preceding positions are as follows:

- Health Physics Supervisor (Radiation Safety Officer)- The Health Physics Supervisor shall have a minimum of six years of health physics experience. The individual shall have a recognized baccalaureate or higher degree in health physics or related scientific field. The degree may fulfill four of the six years of the health physics experience requirement on a one-for-one basis;

- Health Physicist – The Health Physicist shall have a recognized baccalaureate or higher degree in health physics or related scientific field (work experience may be substituted for a degree on a case by case basis). At least two years of health physics experience is desired;

- Health Physics Technician - The Health Physics Technician shall have received sufficient training at the facility or elsewhere to satisfy the job requirements. Individuals shall have a high school diploma or have successfully completed the General Education Development (GED) test. Previous job-related experience or education shall be considered highly desirable.

11.1.3.2 Working Interface Between Health Physics and Reactor Operations

The working relationship of the health physics program relative to reactor operations is shown in Figure 11.6. As shown in this figure, there is a clear separation of responsibilities for the two groups, each with a clear reporting line to the Facility Director.

11.1.3.3 Health Physics Procedures and Radiation Work Permits (RWP)

Operation of the health physics program is carried out under the direction of the Health Physics Supervisor using written health physics procedures and written radiation work permits. These procedures and RWPs are reviewed for adequacy by the Health Physics Supervisor and the operations supervisor (reactor manager), and are approved by the UCD/MNRC Director. They are also audited, normally on an annual basis, by the Nuclear Safety Committee. The health physics procedures are reviewed annually by the MNRC staff and changes are made as necessary.

While not intended to be all inclusive, the following list provides an indication of typical radiation protection procedures and RWPs used in the MNRC program:

a) Testing and calibration of area radiation monitors, facility air monitors, laboratory radiation detection systems, and portable radiation monitoring instrumentation;

b) Working in laboratories and other areas where radioactive materials are used;

c) Facility radiation monitoring program including routine and special surveys, personnel monitoring, monitoring and handling of radioactive waste, and sampling and analysis of solid and liquid waste and gaseous effluents released from the facility;

d) Monitoring radioactivity in the environment surrounding the facility;

e) Administrative guidelines for the facility radiation protection program to include personnel orientation and training;

f) Receipt of radioactive materials at the facility, and unrestricted release of materials and items from the facility;

g) Leak testing of sealed sources containing radioactive materials;

h) Special nuclear material accountability;

i) Transportation of radioactive materials;

j) General decontamination procedures;

k) Personnel decontamination procedures;

l) Personnel exposure investigation procedures;

m) Personnel access procedures for radiography bays and the reactor room;

n) Spill procedures;

o) Radiation work permit procedures;

p) Pneumatic transfer system procedures;

q) In-core and in-tank irradiation facility procedures;

r) ALARA procedures.

11.1.3.4 Radiation Protection Training

The radiation protection training is conducted by the Health Physics Branch. It is structured at different levels in order to meet the needs of different categories of facility staff and researchers using the reactor. All personnel and visitors entering the MNRC facility shall receive training in radiation protection sufficient for the work/visit, or shall be escorted by an individual who has received such training. The general levels of training are as follows:

- Initial Training - All personnel permitted unescorted access in the MNRC facility shall receive training in radiation protection as required by 10 CFR19.12. Initial training shall cover the following areas in sufficient depth for the work being done:

  a) Storage, transfer, and use of radiation and/or radioactive material in portions of the restricted area, including radioactive waste management and disposal;

  b) Health protection problems and health risks (including prenatal risks) associated with exposure to radiation and/or radioactive materials

  c) Precautions and procedures to minimize radiation exposure (ALARA);

  d) Purposes and functions of protective devices;

  e) Applicable regulations and license requirements for the protection of personnel from exposure to radiation and/or radioactive materials;

  f) Responsibility to report potential regulatory and license violations or unnecessary exposure to radiation or radioactive materials;

  g) Appropriate response to warnings in the event of an unusual occurrence or malfunction that involves radiation or radioactive materials;

  h) Radiation exposure reports which workers will receive or may request.

- Specialized Training - Certain personnel (e.g., reactor operators) require more in-depth training than that described above. Such individuals shall successfully complete training over the following outlined topics in sufficient depth for the work being done and pass a written examination with a minimum grade of 70%:

  a) Principles of Atomic Structure;
  b) Radiation Characteristics;
  c) Sources of Radiation;
  d) Interaction of Radiation with Matter;
  e) Radiation Measurements;
  f) Biological Effects of Radiation;
  g) Radiation Detection;
  h) Radiation Protection Practices;
  i) ALARA;
  j) Radioactive Waste Management and Disposal.

- Annual Refresher Training - All personnel permitted unescorted access in the MNRC facility shall receive annual radiation safety refresher training. The annual training shall cover the following areas in sufficient depth for the work being done:

    a) Review of proper radiation safety practices, including radioactive waste management and disposal;
    b) Occurrences at the MNRC facility over the past year;
    c) ALARA summary;
    d) Notable changes in procedures, equipment, facility, etc.

11.1.3.5 Audits of the Health Physics Program

The Nuclear Safety Committee (NSC) provides timely, objective, and independent reviews, audits, recommendations and approvals on matters affecting nuclear safety at the UCD/MNRC. The NSC charter requires that membership shall consist of individuals who have the extensive experience necessary to evaluate the safety of the UCD/MNRC.

The chairman of the NSC is appointed by the UCD/MNRC license holder. Voting membership on the NSC is specified in the NSC Charter. The independent members are voting members and are selected based on their technical qualifications.

NSC meetings are held at least semi-annually (the period between meetings cannot exceed 7.5 months).

The NSC is chartered to conduct an annual on-site audit/inspection of the UCD/MNRC health physics and reactor operations programs and associated records. The annual health physics inspection is performed by an independent member of the NSC and normally covers all aspects of the radiation protection program.

The audit typically covers areas such as actions on NSC recommendations from previous audits, health physics staffing, the interface between health physics and reactor operations, health physics training for MNRC staff and MNRC users, health physics procedures, personnel monitoring, environmental monitoring, effluent monitoring, operational radiological surveys, instrument calibration, radioactive waste management and disposal, radioactive material transportation, SNM accountability, and a review of unusual occurrences.

The audit reports are sent to the chairman of the NSC, who in turn presents a report of the audit findings to the full NSC at the next NSC meeting. Copies of the audit findings are provided to the MNRC Facility Director who is responsible for ensuring that corrective actions are taken.

11.1.3.6 Health Physics Records and Record Keeping

Radiation protection program records such as radiological survey data sheets, personnel exposure reports, training records, inventories of radioactive materials, environmental monitoring results, waste disposal records, instrument calibration records and many more, are maintained by the Health Physics Branch. The records will be retained for the life of the facility either in hard copy, or on photographic or electronic storage media. Records for the current and previous year are retained in the health physicist's office in binders or file cabinets. Other records are retained in long-term storage. Radiation protection records are required to be reviewed and signed by a health physicist prior to filing.

Radiation protection records are used for developing trend analysis, particularly in the personnel dosimetry area, for keeping management informed regarding radiation protection matters, and for reporting to regulatory agencies, e.g., the ALARA dose trend analysis charts. In addition, they are also used for planning radiation-protection-related actions, e.g., radiological surveys to preplan work or to evaluate the effectiveness of decontamination or temporary shielding efforts.

11.1.4 ALARA Program

An ALARA program for the MNRC has been established in accordance with 10 CFR 20.1101. The bases for this program are the guidelines found in ANSI/ANS 15.11 (Reference 11.4). The licensee (UCD) has the ultimate responsibility for the ALARA program, but has delegated this responsibility to the Health Physics Supervisor. The ALARA program incorporates a review of all MNRC operations with emphasis on operational procedures and practices that might reduce MNRC staff exposures to radiation and lower potential radioactive effluent releases to unrestricted areas.

Personnel radiation doses at the MNRC are minimized by considering use of the following ALARA actions when performing work with radiation or radioactive materials:

- Reviewing records of similar work previously performed;
- Eliminating unnecessary work;
- Preparing written procedures;
- Using special tools;
- Installing temporary shielding;
- Performing as much work as possible outside of radiation areas;
- Performing mockup training;
- Conducting prework briefings and postwork critiques;
- Keeping unnecessary personnel out of areas where radiation exposure may occur.

In addition to the above actions, the MNRC ALARA program also contains the following elements which are designed to enhance the effectiveness of the overall program:

- Exposure investigations are conducted when an individual receives greater than 100 millirem in one month or 300 millirem in one quarter. The investigation is focused on determining the cause of the exposure so that appropriate ALARA actions, if any, can be applied;

- ALARA dose trend analysis charts are prepared quarterly and posted for review by all MNRC personnel;

- An annual inspection of the UCD/MNRC ALARA program; and

- A health physicist is required to be involved during planning, design approval, and construction of new MNRC facilities; during planning and implementation of new MNRC reactor use; during maintenance activities; and during the management and disposal of radioactive waste. In addition, written procedures pertaining to the preceding operational facilities are required to be reviewed by the Health Physics Supervisor for ALARA considerations prior to implementation.

11.1.5 Radiation Monitoring and Surveying

The radiation monitoring program for the MNRC reactor is structured to ensure that all three categories of radiation sources (air, liquid and solid) are detected and assessed in a timely manner. To achieve this, the monitoring program is organized such that two major types of radiation surveys are carried out: namely, routine radiation level and contamination level surveys of specific areas and activities within the facility, and special radiation surveys necessary to support non-routine facility operations.

11.1.5.1 Monitoring for Radiation Levels and Contamination

The routine monitoring program is structured to make sure that adequate radiation measurements of both radiation fields and contamination are made on a regular basis. This program includes but is not limited to the following:

Typical surveys for radiation fields as follows:

1. Surveys whenever operations are performed that might significantly change radiation levels in occupied areas;

2. Daily surveys at temporary boundaries (e.g., rope barriers);

3. Monthly surveys in accessible radiation areas and high radiation areas, and in all other occupied areas of the MNRC facility;

4. Annual surveys outside of the MNRC facility, but within the facility fence;

5. Annual surveys in radioactive material storage areas;

6. Annual surveys of potentially contaminated ventilation ducting outside of the MNRC facility;

7. Surveys upon initial entry into a radiography bay after the shutter is closed or upon entry into the demineralizer cubicle;

8. Surveys in surrounding areas where personnel could potentially be exposed when radioactive material is moved;

9. Surveys when performing operations that could result in personnel being exposed to small intense beams of radiation (e.g., when transferring irradiated fuel, when removing shielding, or when opening shipping/storage containers);

10. Surveys of packages received from another organization;

11. Surveys when irradiated parts or equipment are removed from a radiography bay, or from the reactor core, from a fuel storage pit, from the pneumatic transfer system terminal, or from the reactor room;

12. Surveys as necessary to control personnel exposure. Such surveys may include the following:

    a) Gamma surveys of potentially contaminated exhaust ventilation filters when work is performed on these filters;

    b) Gamma and neutron surveys on loaded irradiated fuel containers;

    c) Gamma and neutron surveys when handling an unshielded neutron source.

Typical surveys for contamination as follows:

1. Surveys at the exits to the MNRC facility once per shift;

2. Daily surveys in accessible contaminated areas and occupied areas surrounding contaminated areas;

3. Monthly surveys in occupied non-contaminated areas of the MNRC;

4. Annually surveys in areas outside of the MNRC facility, but within the facility fence;

5. Annually surveys in radioactive material storage areas;

6. Surveys as necessary to control the spread of contamination whenever operations are performed that are known to result in, or expected to result in, the spread of contamination;

7. Surveys prior to removal of paint from areas where contaminated paint is possible;

8. Surveys as part of the following operations:

    a) Decontamination of equipment;

    b) Removal of irradiated parts or equipment from a radiography bay, from the reactor core, from a fuel storage pit, from the pneumatic transfer system terminal, from the reactor room, or from the MNRC facility;

    c) Inspection, maintenance, or repair of the primary cooling system;

    d) Initial opening of the secondary cooling system (heat exchanger) for inspection, maintenance, or repair;

    e) When working in or entering areas where radioactive leaks or airborne radioactivity has occurred previously;

f) Upon initial entry into potentially contaminated exhaust ventilation ducting;

g) Prior to replacing filters or ducting in potentially contaminated exhaust ventilation systems.

11.1.5.2 Radiation Monitoring Equipment

Radiation monitoring equipment used in the MNRC reactor program is summarized in Table 11-6. The locations of many of the pieces of equipment are shown in Figures 11.7 and 11.8. Because equipment is updated and replaced as technology and performance requires, the equipment in Table 11-6 should be considered representative rather than an exact listing. The function this equipment performs will remain the same.

TABLE 11-6 RADIATION MONITORING AND RELATED EQUIPMENT USED IN THE MNRC RADIATION PROTECTION PROGRAM

ITEM	LOCATION	FUNCTION
Continuous Air Monitors (3) • Stack Effluent Monitor • Reactor Room Air • Radiography Bays Air	CAM Room CAM Room Sample Preparation Area	Measure radioactivity in stack effluent Measure reactor room airborne radioactivity Measure radiography bay airborne radioactivity (All monitors measure particulate)
Radiation Area Monitors (6)	Staging Area No. 1 Staging Area No. 2 Staging Area No. 4 Equipment Room Demineralizer Area Reactor Room	Measure gamma radiation fields in occupied or accessible areas of the MNRC facility
Portable Ionization Chamber Survey Meters (3)	Staging Area No. 1 Staging Area No. 4 Sample Preparation Area	Measure beta-gamma radiation dose rates
Portable Neutron Survey Meters (2)	Staging Area No. 1 Sample Preparation Area	Measure neutron radiation dose rates
Portable MicroR Survey Meters (2)	Staging Area No. 1	Measure low level and environmental gamma radiation dose rates
Portable G-M Survey Meters (4)	Staging Area No. 1 Staging Area No. 4 Sample Preparation Area Health Physics Lab	Measure beta/gamma contamination levels
Portable Alpha Survey Meters (2)	Staging Area No. 1	Measure alpha contamination levels
Lab Swipe Counter (1)	Health Physics Lab	Measure alpha/beta contamination on swipes
Gamma Spectroscopy Systems (HPGe) (4)	Health Physics Lab	Gamma Spectroscopy

Hand and Foot Monitor (1)	Equipment Room Exit	Measure potential contamination on hands and feet prior to leaving radiation restricted areas
Direct Reading Pocket Dosimeters (20)	Staging Area No. 1	Measure personnel gamma dose
Environmental TLDs	Various on-site, on-base, and off-base locations	Measure environmental gamma radiation doses
Portable Air Sampler (1)	Staging Area No. 1	Collect grab air samples
Air Flow Velometer (1)	Sample Preparation Area	Measure ventilation flow rates
Air Flow Calibrator (1)	Health Physics Lab	Calibrate CAM air flows

[redacted]

FIGURE 11.7 RADIATION MONITORING EQUIPMENT - MAIN FLOOR FIGURE

[redacted]

FIGURE 11.8 RADIATION MONITORING EQUIPMENT - SECOND FLOOR

11.1.5.3 Instrument Calibration

Radiation monitoring instrumentation is calibrated according to written procedures. It is the policy of the MNRC to use NIST traceable sources for instrument calibrations whenever possible. The following instrumentation is normally calibrated at the MNRC by health physics personnel:

- Continuous Air Monitors;
- Radiation Area Monitors;
- Swipe Counter;
- Gamma Spectroscopy Systems;
- Portable G-M Survey Meters;
- Hand and Foot Monitor;
- Portable Air Sampler.

The following instrumentation is normally calibrated at a contractor calibration facility:

- Portable Ionization Chamber Survey Meters;
- Alpha Survey Meters;
- Direct Reading Dosimeters;
- Air Flow Velometer;
- MicroR Survey Meters;
- Portable Neutron Survey Meters;
- Air Flow Calibrator.

Instrument calibrations are tracked by a computer-based tracking system. Instrument calibration records are maintained by the Health Physics Branch and calibration stickers showing pertinent calibration information (e.g., counting efficiency, the most recent calibration date, and the date the next calibration is due) is attached to all instruments.

11.1.6 Radiation Exposure Control and Dosimetry

Radiation exposure control depends on many different factors including facility design features, operating procedures, training, proper equipment, etc. Training and procedures have been discussed previously under the section dealing with the MNRC's radiation protection program (Section 11.1.3). Therefore, this section will focus on design features such as shielding, ventilation, containment and entry control devices for high radiation areas, and will also include protective equipment, personnel dosimetry, and estimates of annual radiation exposure for specific locations within the facility. A description of the dosimetry records used to document facility exposures and a summary of exposure trends at the MNRC will also be presented.

11.1.6.1 Shielding

The biological shielding around the MNRC reactor is the single biggest design feature in controlling radiation exposure during operation of the facility. The shielding is based on TRIGA® shield designs used successfully at many other similar reactors, but has been modified to accommodate the beam tubes and radiography bays unique to this reactor.

- The MNRC Reactor bulk shield is very similar, in material type and thickness, to other proven TRIGA® shields. The one significant difference is the beam tube penetrations. Where the basic shielding configuration has been penetrated by beam tubes, supplemental shielding has been added. This supplemental shielding has been designed to provide the same attenuation to both neutrons and gammas as the basic unpenetrated shield.

The MNRC has eight areas with specially designed shielding: the reactor bulk shield, the four radiography bays, the demineralizer resin cubicle, the CAM room, and the second floor hand and foot monitor. Included in the radiography bays' shielding are the shutter biological shields, the beam stops, and the walls and roof of the individual bays. Shielding has been designed so that radiation levels in areas occupied by personnel are as-low-as- reasonably-achievable.

- Reactor Bulk Shield The reactor shield is essentially the same as that which has been used for other above ground TRIGA® reactors. The shield consists of approximately 20 ft of water above the core to protect personnel in the reactor room (Figure 11.9). The radial shielding, which protects personnel in the adjoining radiography bays, is provided by the graphite reflector and pool water to a radius of 3.5 ft and by standard reinforced concrete extending to a radius of 10.5 ft (7 ft thick in Bay 1). This basic shield has been augmented in the areas of beam tube penetration with shadow shields of steel. Actual measured radiation levels at the surface of this shield at 1 MW show 1 millirem per hour. The reinforced concrete pad below the tank is approximately 10 ft thick and prevents soil and ground water activation.

  The 20 ft of water above the core provides the bulk shielding for personnel in the reactor room. The results of surveys of two similar 1 MW TRIGA® facilities and of the MNRC reactor operating at 1 MW showed the following radiation levels above the center of the reactor tank. (NOTE: These levels drop off rapidly at the edge of the tank).

[redacted]

FIGURE 11.9 REACTOR BULK SHIELD

TABLE 11-7 Radiation Levels (millirem/Hr)

	Tank Size		millirem/hr Diffuser "ON"	millirem/hr Diffuser "OFF"
Reactor	Dia. (Ft)	Height (Ft)	Water Surface	Water Surface
USGS	8.0	24.8	16	600
Malaysia	6.5	20.7	240	750
MNRC	7.5	24.5	80	600

- Neutron Beam Shutter, Biological Shield, Beam Stops, Radiography Bay Walls, and Roof The neutron beam shutter/biological shield, the beam stop, and the radiography bay walls and roof must protect personnel from both gamma and neutron radiation contained in the radiography beam. The beam tube shutter/biological shield and the radiography bay interior walls were designed to reduce radiation levels to less than 5 millirem/hr in areas in the radiography bays that are routinely occupied when the reactor is operating. The actual radiation levels were less than 5 millirem/hr for 1 MW operations.

  Two sources of radiation were considered when designing the radiography bay shields. First, the direct beam which is attenuated by the shutter/biological shield. Second, the neutrons that scatter from adjoining radiography bays and are attenuated by the interior walls of the radiography bays. The shutter shield orientation is shown in Figure 11.7. The face of the shield contains a section filled with boron frit approximately 1 in. thick to attenuate thermal neutrons. The remaining shield is a composite made up of the materials shown in Table 11-8.

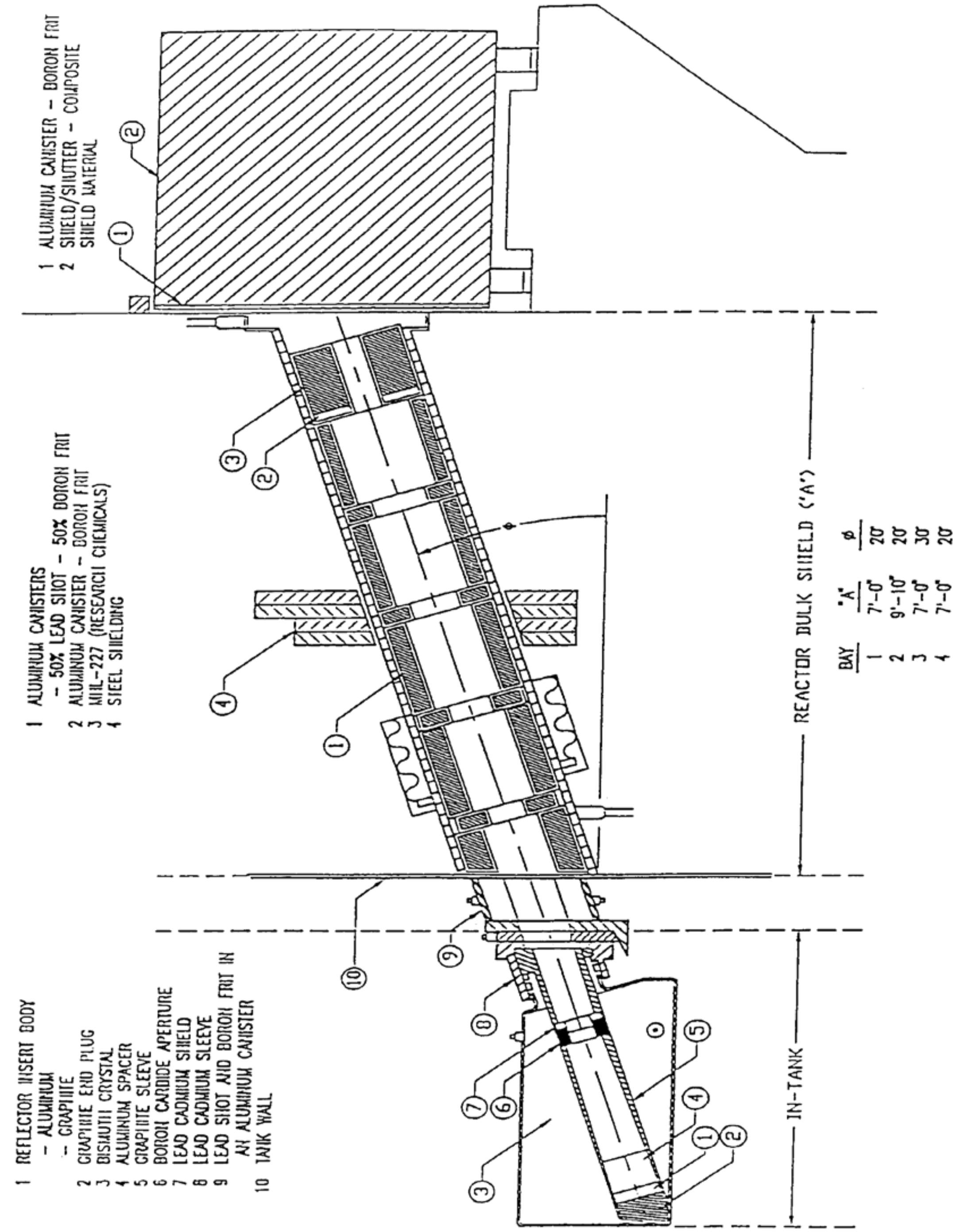


FIGURE 11.10 MNRC BEAM TUBE AND BIOLOGICAL SHIELD

TABLE 11-8 Composite Shield Makeup

Material	Wt %
Cement	11.7
Water	3.4
Boron Carbide	0.5
Limonite	18.7
Steel Shot	65.8

The shutter shield is designed to thermalize fast neutrons, to capture thermal neutrons, and to attenuate direct gamma radiation from the core as well as the capture gammas produced in the shield. A MORSE and ANISN analysis (References 11.5 and 11.6, respectively) predicted radiation levels on the bay side of this shield of approximately 0.1 millirem/hr. The actual measured radiation level on the bay side of the shield with the reactor operating at 1 MW was 0.3 millirem/hr.

The interior walls of the radiography bays are made from [redacted] ft thick standard reinforced concrete. Calculations, similar to those discussed below for the exterior walls and roofs, predict maximum radiation levels of less than 5 millirem/hr at the closest point on the wall of an adjacent bay when the maximum scattering target is directly in the beam. Survey results indicate these calculations were overestimates.

The exterior walls and doors of the radiography bays, shown in Figures 11.11 and 11.12, are made of standard reinforced concrete with thicknesses that range from 2 to 3 ft. They have been designed to reduce the radiation levels from scattered gammas and neutrons, especially in the staging and preparation areas. Beam stops have been incorporated into the bay walls to attenuate the direct neutron and gamma radiation. These stops not only reduce the direct radiation levels but also reduce scattering.

The roofs of the radiography bays are constructed from standard reinforced concrete [redacted] ft. thick at the outer edge of the building. The thickness of the roofs increases at approximately 0.25 in./ft so they are about 32 in. thick at the building center.

To determine the required thickness of the walls and roofs of the radiography bays, it was assumed that a typical aircraft component placed in the beam represented a point source of neutrons arising from scattering collisions in the target.

[redacted]

FIGURE 11.11 MNRC SHIELDING - BAYS 1 AND 2

[redacted]

FIGURE 11.12 MNRC SHIELDING - BAYS 3 AND 4

For these calculations a neutron beam consisting of one-half thermal and one-half fast neutrons was used. The largest components are handled in Bay 1, so the target was assumed to be a 38.5 kg piece of B6AC stainless steel plate with a surface area, normal to the beam, of 0.223 m^3. The 38.5 kg piece of material is representative of an aircraft wing hinge and has a composition as shown in Table 11-9. The size of targets for the other bays were somewhat smaller and representative of the components being examined.

TABLE 11-9 Composition Of B6AC Stainless Steel

Element	Wt-%
Carbon	0.47
Manganese	0.75
Sulfur	0.22
Chromium	1.05
Nickel	0.55
Molybdenum	1.00
Vanadium	0.11
Iron	95.80

Macroscopic thermal capture cross sections, and thermal and fast scattering cross- sections were calculated from cross-sections given in Etherington (Reference 11.7). By this method it was found that about 19% of all the neutrons were captured and about 75% were scattered when the beam was one-half thermal and one-half fast neutrons. A Monte Carlo calculation also indicated that about 26% of the source neutrons are captured in the target and 68% are scattered out of the beam. In this calculation a 7-energy group distribution was used. The Monte Carlo calculation also showed that neutron scattering is essentially isotropic as would be expected.

A FORTRAN program using removal cross-sections was written to calculate either the concrete thickness for a given dose rate or the dose rate for a given concrete thickness, where the inside dimensions of the room, the target location and composition, and the beam strength and horizontal angle are specified as input. The last was included to account for any anisotropy in neutron scattering.

It was assumed that all fast neutrons that were scattered were available to penetrate the walls, and that all thermal neutrons that were not captured in the target would be captured ultimately within the room.

In each of the four radiography bays the wall calculations were done in a horizontal plane containing the target and a vertical plane containing the beam.

Additional calculations were made to assess the contribution to the dose rate outside the room from capture gammas in the room or in the walls themselves. Three different assumptions were made:

1. Every neutron entering a spherical room was captured uniformly in the room (The room volume was that of radiography Bay 3);

2. Every neutron entering the room was captured uniformly over the inside surface of the room;

3. Every neutron entering the room was subject to removal (using a removal cross-section for concrete) and that removal produced a 1.5 MeV photon.

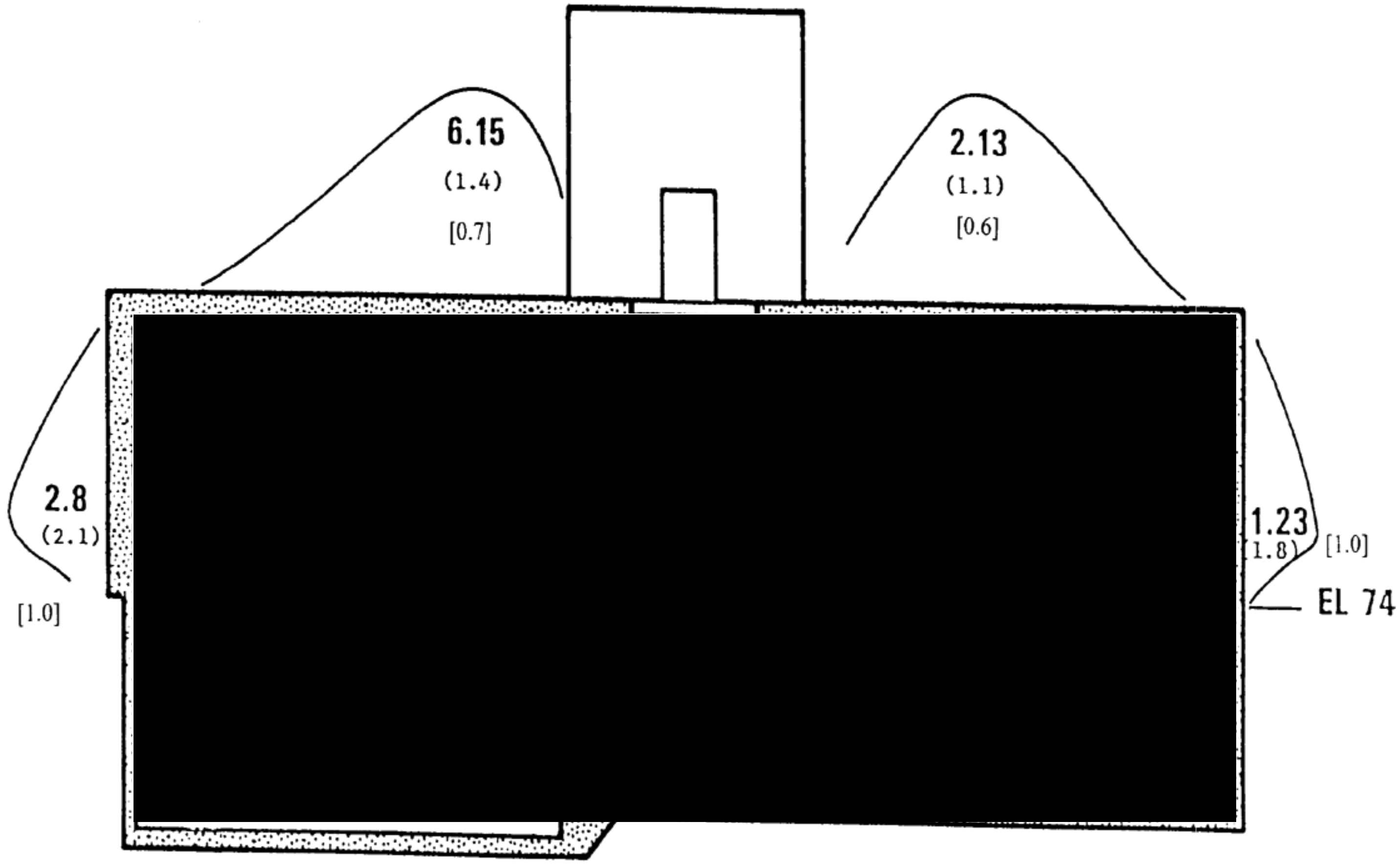
Only the third assumption yielded dose rates of any significance and these were only about 3% of the neutron dose rates from the detailed calculations and are comparable to the gamma doses in those calculations.

The results of this analysis were used to establish the concrete thickness shown in Figures 11.11 and 11.12. With these thicknesses, the predicted radiation dose rates at 1 MW are shown in Figures 11.13, 11.14, and 11.15. Actual measured dose rates at 1 MW and projected dose rates at 2 MW (no longer applicable) outside the radiography bays and on the roofs are also shown in these three figures.

- Auxiliary Systems Shielding In addition to the primary biological shielding for the reactor and radiography bays, certain auxiliary systems require shielding. An additional 1 foot of concrete was installed around the demineralizer system in order to keep the radiation levels on the second floor of the reactor building as low as reasonably achievable during 2 MW operations, and to maintain an acceptable radiation background for health physics instrumentation in the general area. In order to achieve a background reduction sufficient to maintain adequate counting sensitivity, in addition to the demineralizer resins, the east wall of the CAM room (containing the reactor room and the stack CAMs) is also shielded with one foot of concrete in an “L” configuration. Figure 11.16 shows the locations of the shielding for the demineralizer and radiation monitoring systems.

- Bay 4 Structural Modification Shielding (Bay 5) The lower level in Bay 4 has a large cavity cut up to the reactor tank wall for planned neutron cancer therapy (NCT) research. The cavity is approximately 10' x 10' x 10' and is currently filled with concrete blocks stacked in overlapping layers (to prevent radiation streaming). The radiation levels at 1 MW are less than 0.5 mR/hr gamma and less than 0.1 mrem/hr neutron on the outside of the concrete blocks.

FIGURE 11.13 RADIATION DOSE RATES BAYS 1 AND 2 AT 1 MW PROJECTED DOSE AT 2 MW (FOR REFERENCE)

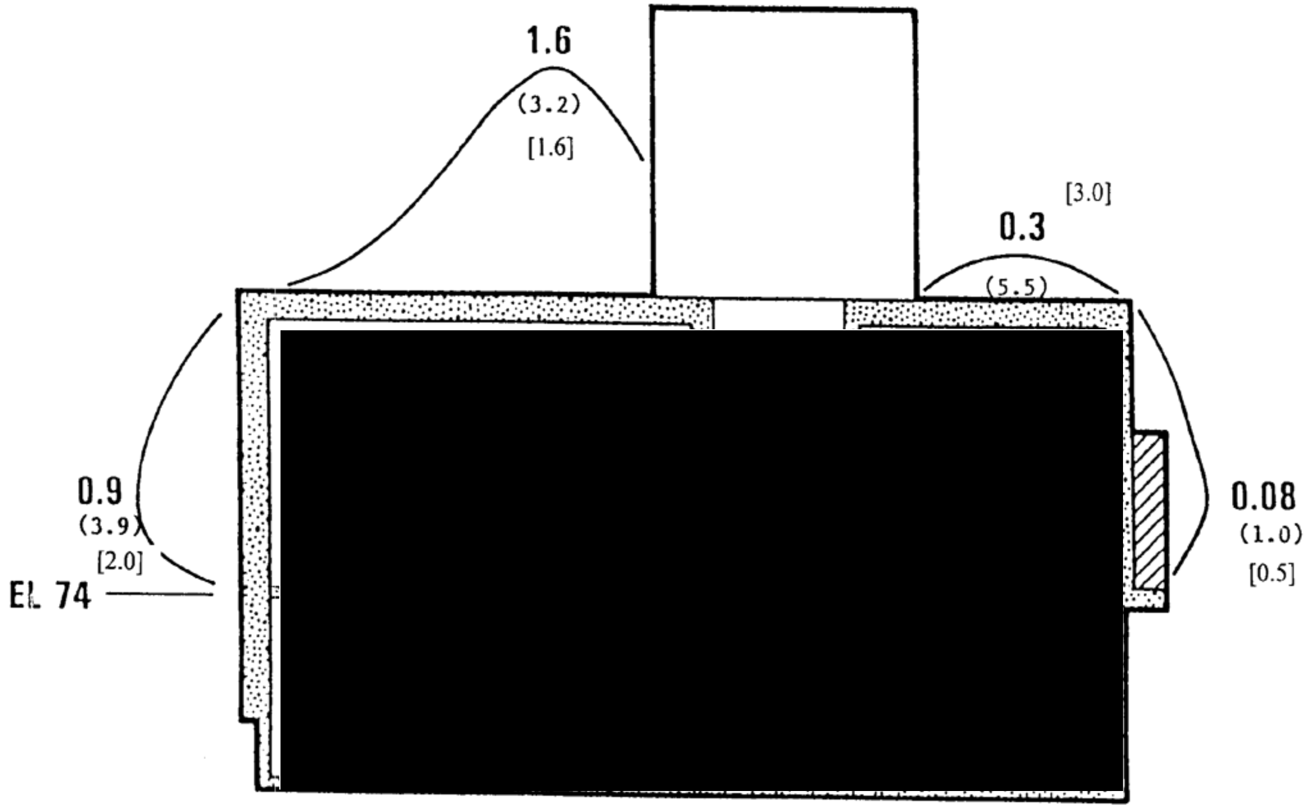


All Dose Rates in millirem/hr

Bold No's - Projected dose rates at 1 MW

( ) Actual measurements during load and test. Reactor at 1 MW.

[ ] - Projected dose rates at 2 MW based on reduced beam intensities (see text)



All Dose Rates in millirem/hr

Bold No's - Projected dose rates at 1 MW

( ) Actual measurements during load and test Reactor at 1 MW.

[ ] - Projected dose rates at 2 MW based on reduced beam intensities (see text)

FIGURE 11.14 RADIATION DOSE RATES BAYS 3 AND 4 AT 1 MW PROJECTED DOSE AT 2 MW (FOR REFERENCE)

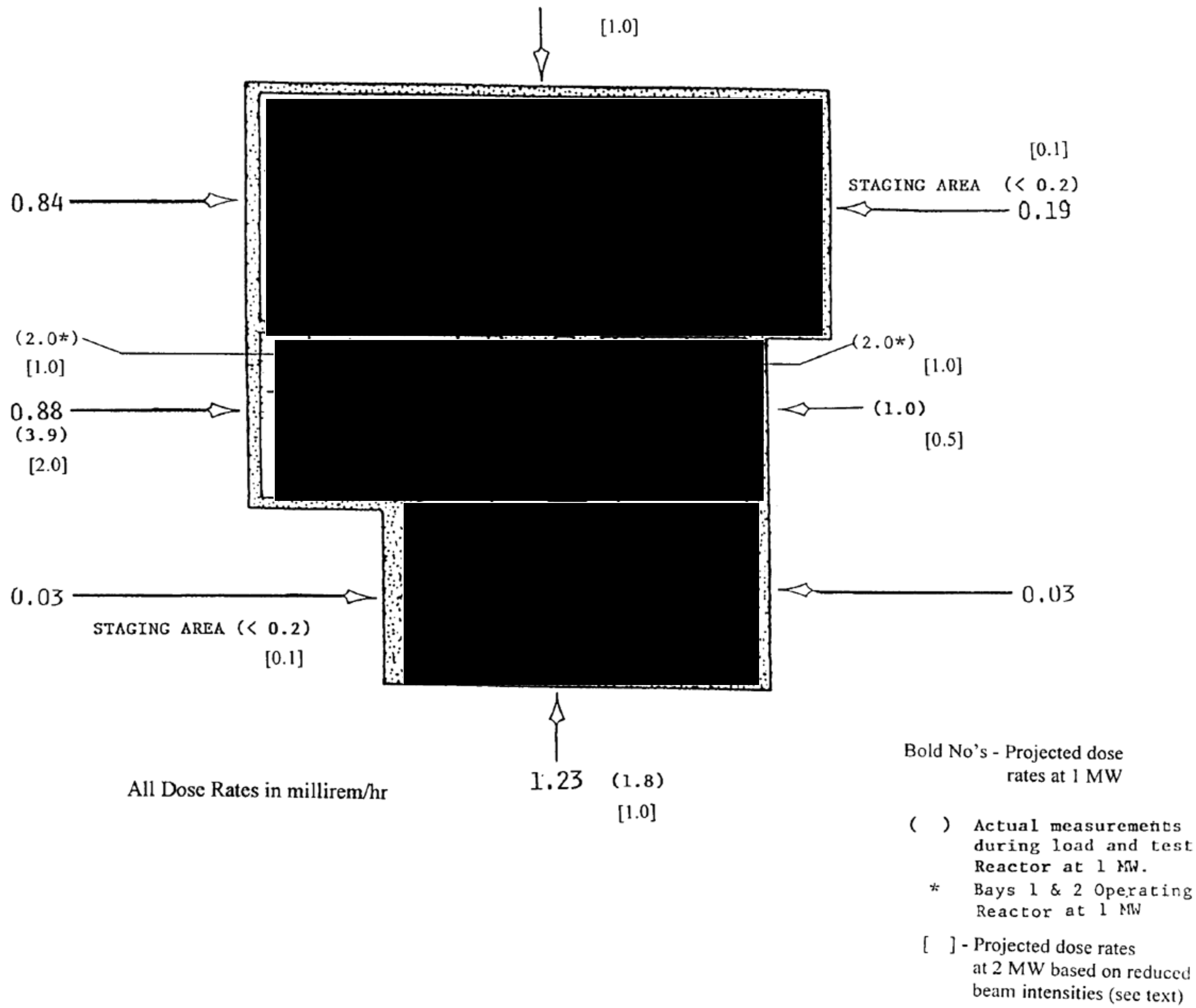


FIGURE 11.15 RADIATION DOSE RATES PLANE VIEW AT 1 MW PROJECTED DOSE AT 2 MW (FOR REFERENCE)

11.1.6.2 Ventilation System

Control of radiation exposure due to airborne sources is discussed in Section 11.1.2.1 and in Appendix A. In addition, details of the ventilation system (an integral part of the control process for airborne emitters) are provided in Section 9.5. This section discusses only those ventilation design features that have been incorporated for radiation protection.

- First and most important, the design of the radiography bays and reactor room exhaust systems will maintain Ar-41 and N-16 levels in the reactor room and Ar- 41 levels in the radiography bays at concentrations consistent with keeping occupational doses well below the limits in 10 CFR Part 20. However, even when the radiography bays exhaust system is not operational, Ar-41 concentrations in the bays and subsequent occupational doses will still be below 10 CFR Part 20 limits) (See Section 11.1.2.1.1).

- Second, the ventilation systems are balanced so that the differential air pressure in the reactor room, the equipment room and the sample preparation area is negative with respect to surrounding areas. The radiography bays will also have a negative air pressure relative to surrounding areas when the radiography bays exhaust system is operative, which will be the normal mode of operation.

- Third, the reactor room exhaust system contains a high efficiency filter (99.95% for 0.3 micron sized particles) to remove any radioactive particulates.

- Fourth, the reactor room exhaust system recirculates the air exhaust back into the reactor room should the reactor room CAM exceed preset limits. (Reactor room air can then be recirculated through HEPA and charcoal filters to remove radionuclides.)

- In this mode, no reactor room air is exhausted through the stack.

- Fifth, the hood in the sample preparation/pneumatic transfer area exhausts through a HEPA filter. It also maintains an in-flow of air through the hood to prevent the release of radioactivity into the surrounding area.

[redacted]

FIGURE 11.15 SHIELDING FOR DEMINERALIZER RESINS AND RADIATION MONITORING EQUIPMENT - SECOND FLOOR

11.1.6.3 Containment

Containment of radioactivity within the MNRC is primarily a concern with respect to experiments being irradiated in the various irradiation facilities and with the reactor fuel. Containment of fission products within the fuel elements is achieved by maintaining the integrity of the fuel's stainless steel cladding, which is accomplished by maintaining the fuel and cladding temperatures below specified levels. This matter is discussed in detail in Chapter 4. Containment of other radionuclides generated during use of the irradiation facilities is achieved through strict encapsulation procedures for samples and strict limits on what materials will be irradiated, as specified in "Utilization of the McClellan Nuclear Research Center Reactor Facility" (Reference 11.8) and in Chapter 10.

To further improve containment and minimize the potential release of radioactivity from experiments irradiated in the in-core pneumatic transfer system, the terminal where samples are loaded and unloaded is located inside a fume hood. The hood, which exhausts through a HEPA filter, maintains an in-flow of air to prevent the release of radioactivity to the surrounding area (Figure 9.10).

11.1.6.4 Entry Control - Radiography Bays and Demineralizer Cubicle

There are five main areas within the MNRC facility which will require entry control in order to meet the 10 CFR 20 requirements for limiting access into high radiation areas. Specifically, these are presently the four radiography bays and the small cubicle containing the demineralizer resins.

11.1.6.4.1 Entry Control for Radiography Bays

Access into the radiography bays is controlled by a system of interlocks and warning devices incorporated into the facility design and described in Section 9.6.

Operation of the neutron beam shutters within the radiography bays is normally controlled remotely from the respective radiography bay control room; however, during maintenance activities, etc., these shutters can be controlled from within the respective bays. The bay doors are locally operated only by "dead man" switches mounted to the door. An interlock and warning system has been incorporated into each beam shutter control and bay door control to do the following:

- Prevent the radiography bay doors from opening when the beam shutter is open and the reactor is operating;

- Scram the reactor if both the beam shutter and the bay door are open;

- Sound an audible alarm and activate a red flashing light within the bay when the beam shutter starts to open;

- Show the operational status of the reactor throughout the facility.

The logic for this interlock system is shown in Figures 9.14 and 9.15.

Another feature of the interlock system is a series key-lock subsystem. A key box mounted near the closure side of each radiography bay shield door contains 10 captive keys and a master key that is controlled by a qualified UCD/MNRC staff member. When any one of the radiography bay keys is removed, electrical power cannot be applied to the respective bay door drive mechanism. Use of the master key assures the required attendance of a qualified UCD/MNRC during opening and closing of radiography bay doors. Each individual entering a radiography bay is required to remove one of the ten keys and to maintain the key in his or her possession while in the bay. When an individual leaves the bay, that person's key is reinserted into the key box. When all 10 individual keys and the master key have been inserted and are captive in the key box, the bay door can be closed. Following the door closure, the master key is removed returning the control and interlock system to its secured mode.

Rip cords have also been located in each of the radiography bays. Figures 9.19, 9.20, and 9.21 show their locations in the different bays. Activation of any rip cord will scram the reactor or will not allow it to start if shutdown. To reset a tripped rip cord, personnel must enter the bay and depress the reset button (i.e., determine if personnel in the bay activated the system). In addition, a scram button is located in the reactor room and can be used to shut the reactor down.

Reactor "ON" lights are located throughout the facility. These lights illuminate any time the control rod drive magnets have power. Also, anytime a radiography bay shutter "open" command is given an audible horn is sounded for 15 sec and a red flashing light is illuminated in the bay. Figures 9.22 and 9.23 show the locations of reactor "ON" lights and the shutter "opening" warning lights.

11.1.6.4.2 Entry Control for the Demineralizer Cubicle

Access control for the demineralizer cubicle will be based on the fact that it is a high radiation area when the reactor is operating at 1 MW. This is due primarily to the expected buildup of primary coolant activation products (mainly Na-24) in the resins. Additional radiation shielding around the cubicle is in place, and the access is controlled by a locked barrier at the point of entry into the area. The locked barrier will be opened only under controlled conditions commensurate with the fact that the area is considered a high radiation area. Entry procedures will incorporate all 10 CFR 20 access requirements for entering a high radiation area.

11.1.6.5 Protective Equipment

Typical protective equipment and related materials used in the MNRC radiation protection program are summarized in Table 11-10.

TABLE 11-10 SUMMARY OF TYPICAL PROTECTIVE EQUIPMENT USED IN THE MNRC RADIATION PROTECTION PROGRAM

Item	Use - Routine (R) - Emergency (E)
Lab Coat	R & E
Rubber Gloves	R & E
Latex Examination Gloves	R & E
Safety Glasses	R & E
Face Shields	R & E
Coveralls	R & E
Hoods/Caps	R & E
Plastic Shoe Covers	R & E
Rubber Over Shoes	E
Small Spill Kits	E
Large Spill Kits	E
Decontamination Locker	E
Decontamination Shower	E
Decontamination Sink	E

11.1.6.5.1 Respiratory Protection Equipment

Other than Ar-41 and N-16, no airborne radioactivity is expected to occur at the MNRC as part of normal operations. Consequently, respiratory protection equipment is not part of the protective equipment typically used at the MNRC. Should the situation change and respiratory protection become necessary in order to meet ALARA objectives, the MNRC will implement a respiratory protection program in accordance with Subpart H of 10 CFR 20.

11.1.6.5.2 Personnel Dosimetry Devices

Personnel dosimetry devices in use at the MNRC have been selected to provide monitoring of all radiation categories likely to be encountered. Personnel dosimetry shall be provided and measured by a NVLAP certified provider. Table 11-11 summarizes the devices typically used. Personnel dosimeters are changed monthly. An administrative action level of 100 millirem in one month or 300 millirem in one quarter has been established. An exposure investigation is required if any action level is exceeded in order to determine the source of the exposure. This is part of the MNRC ALARA program described previously (Section 11.1.4).

There are no routine operations at the UCD/MNRC which present a potential for internal deposition of radionuclides. The requirement for annual whole body counting was eliminated in 2005 when I-125 production was terminated. (per UCD/MNRC RSO memo dated 20Jan2005 "Changes to the Health Physics Program")

Personnel exposure reports are maintained by the Health Physics Branch and are retained for the life of the facility. In addition, radiological survey data sheets which document worksite radiological conditions are maintained by the Health Physics Branch and are retained for the life of the facility.

TABLE 11-11 TYPICAL PERSONNEL MONITORING DEVICES USED AT THE UCD/MNRC

Type	Dose	Radiation Measured
OSL	Deep Dose Equivalent Eye Dose Equivalent Shallow Dose Equivalent	Beta, Gamma
Albedo OSL	Deep Dose Equivalent	Thermal Neutrons
OSL Finger Ring	Extremity Dose Equivalent	Beta, Gamma
CR-39 Track Etch	Deep Dose Equivalent	Fast Neutrons

The average annual occupational whole body exposure (Deep Dose Equivalent) for 2 MW operations for 1998 was 217 millirem. The average annual extremity and eye dose for 2 MW operations for 1998 was 181 and 195 millirem, respectively. In contrast the average annual whole body, extremity, and eye exposure for more recent single shift 1 MW operation has been approximately 25, 50, and 50 millirem respectively. These doses are not expected to change significantly in the future and are well below 10 CFR 20 limits.

11.1.6.6 Estimated Annual Radiation Exposure

The guidelines for radiation doses and for airborne concentrations of radionuclides during normal operations of the MNRC are contained in 10 CFR 20. These guidelines establish levels for both “restricted” and “unrestricted” areas. With respect to the MNRC, the “restricted” area is considered to be all locations within the operations boundary (within the UCD/MNRC perimeter fence). The “unrestricted” area includes all locations and the personnel outside the operations boundary. The following sections contain an estimate of annual radiation exposure in these two areas.

11.1.6.6.1 Estimated Annual Doses in the Restricted Area

It is assumed that an individual working at the UCD/MNRC will be in the facility only one shift/day (40 hrs/wk). Further, it is assumed that an occupationally exposed individual will only spend a fraction of the time in areas where there is a potential for significant radiation levels (within the radiography bays, the demineralizer cubicle, or in the reactor 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.

Radiation levels outside the radiography bays in the staging areas are typically less than 0.2 millirem/hr with the reactor operating at 1 MW. If personnel were exposed to these levels for 20 hr/wk for 50 weeks during the year the annual Total Effective Dose Equivalent (TEDE) would be 200 millirem, which is well below the 10 CFR 20 annual occupational dose limit. Radiation levels are higher (1-3 millirem/hr) immediately outside the radiography bay walls which contain the beam stops. However, personnel doses from these areas are still expected to be very low (less than 2 millirem/wk) because personnel spend less than 1 hr/wk in these areas. Historical worker dosimetry show these estimates to be conservative.

Radiation levels are high in the radiography bays when the neutron shutters and gamma shields are open and the reactor is operating. However, personnel are restricted from these areas anytime the shutters are open and the reactor is operating. Radiation levels in bays adjacent to an operating bay are approximately 1 millirem/hr during 2 MW operations.

A prediction of the dose rates from typical aircraft materials activated in the neutron beams was made in Appendix A. The predicted dose rate from an aluminum plate being radiographed using film techniques or from an entire wing scanned for 8 hrs using electronic imaging devices is less than 1 millirem/hr at five feet if a 5-10 min period is allowed for the aluminum to decay. The radiation levels from these components when compared to those discussed above will be insignificant since exposure times will be short. These components may need to be stored in an isolated area for a few days for all activity to decay.

The radiation exposures from activation products in the shutter bulk shield will be less than 1 mrem since nearly all of the activity is in the first 12 inches of the shield leaving 36 inches of high density material for attenuation. However, during decommissioning, the shield will have to be handled as low-level radioactive waste due to induced gamma emitting radionuclides, and more importantly, due to the long-lived non-gamma emitters, such as ^{55}Fe, which has a 2.7 year half-life.
The total effective dose equivalent from Ar-41 in the radiography bays was predicted in Appendix A. Using the highest Ar-41 concentration for 2000 hours of annual exposure will result in an annual TEDE of only 0.5 millirem. Nevertheless, the exposure of personnel working in radiography bays will be closely monitored so that guideline levels are not exceeded and exposure to all individuals is kept as-low-as-reasonably-achievable.

The radiation level (due primarily to N-16) is approximately 60 millirem per hour at one foot over the tank and about 10 millirem per hour at 3 feet above the tank, but these levels drop rapidly at the tank’s edge.

Radiation doses in the reactor room away from the tank will be mainly from airborne N-16 and Ar-41. General area dose rates in the reactor room vary from 2-8 mRem/hr from these two radioactive noble gases. Worker and visitor dose is limited by limiting the amount of time workers and visitors may spend in the reactor room while the reactor is operating.

Maintenance of equipment located in the reactor room, such as control rod drives, instrumentation, and primary water system components, will not be allowed when the reactor is operating. Therefore, it is estimated that personnel exposures from this type of activity will be insignificant.

Handling and inspection of MNRC fuel is accomplished in the reactor tank (with the reactor shut down). Removing or replacing fuel elements, either in the core or in the in- tank storage racks, requires that the element be raised in the vertical direction far enough to clear the grid plate/reflector or storage racks. However, with the fuel element at its highest point, it is still covered by about 15 ft of water and the radiation level at the tank surface is insignificant.

Should it be necessary to remove a fuel element from the tank after operation, it will normally be moved from the core and placed in the in-tank storage rack. It is anticipated that removal of most irradiated fuel elements from the tank will be carried out using the fuel element transfer cask. Therefore, the next step will normally be to lower the transfer cask into the tank, remove the element from the storage rack and then place it in the transfer cask. For this operation, there will be about 6-1/2 ft of water between the operator and the fuel element. Although the radiation level could be as much as 50 millirem per hour, the radiation dose to the operator will be insignificant since the time required for the operation is estimated to be less than one minute.

As discussed in Section 13.2.5, fission products would be released into the tank water should the cladding on a fuel element fail. Although not expected, if such a failure did occur, the noble gases, krypton and xenon, would escape from the water and into the reactor room. Most of the halogens, bromine and iodine, would be retained in the primary cooling water and would eventually end up in the water purification system resins or mechanical filter.

A prediction of the radiation doses in the reactor room due to a single fuel element failure is provided in Appendix B and in Section 13.2.5.

Other categories of individuals who might receive exposure at the MNRC include research and service personnel and visitors. Past exposure history on these two groups shows little or no recorded dose and there does not appear to be any reason to expect this situation to change. In addition, the MNRC has an administrative dose limit of 50 millirem per year TEDE for embryos, fetuses, declared pregnant women, minors and students, although the occupational exposure history at the facility would certainly indicate that it is very unlikely that this exposure would be received by anyone in these groups. Most visitors to MNRC receive a reactor room tour that is typically 30 minutes in duration. Most of the visitors only visit the MNRC once for a one-time dose of approximately 1.0 mrem.

11.1.6.6.2 Estimated Annual Dose in the Unrestricted Area

A detailed discussion of the expected annual TEDE in the unrestricted area from Ar-41 production during normal operation of the MNRC reactor is contained in Section 11.1.2.1.4 and in Appendix A. The annual dose values for the unrestricted area shown in both of the preceding parts of this SAR indicate a maximum TEDE (primarily from Ar-41) ranging between 0.1 and 1.4 millirem per year, depending upon which atmospheric dispersion model is used. The maximum historical TEDE measured at the MNRC fence line has been 61 millirem gross, 45 millirem background, and 26 millirem net.

11.1.7 Contamination Control

Radioactive contamination is controlled at the MNRC by using written procedures for radioactive material handling, by using trained personnel, and by operating a monitoring program designed to detect contamination in a timely manner. The program for routine monitoring to detect and identify fixed and loose contamination is described in Section 11.1.5.1. In addition to this monitoring program, the following items are also part of the program for contamination control at the MNRC:

- Two areas are known to be contaminated in the MNRC facility. These are the reactor tank and the pneumatic transfer system (PTS) hood. The MNRC Health Physics Procedures, MNRC-0029-DOC, contains specific procedures for working with radioactive material and for working with experiments that originate from in- tank or from the PTS hood. For other work where contamination is considered likely, a detailed written procedure or a radiation work permit (RWP) will provide the necessary contamination controls. All such work requires coverage by a qualified health physics technician and all material which must be removed from a contaminated area with suspected loose contamination is appropriately monitored and contained to minimize potential spread, or is decontaminated;

- After working in contaminated areas, personnel are required to perform surveys to ensure that no contamination is present on clothing, shoes, etc., before leaving the work location. Additionally, personnel exiting controlled areas surrounding a contaminated area are required to use a hand and foot monitor located at the exit. MNRC personnel are not exposed to sources of radioactivity likely to result in internal exposure.

- Anti-contamination (Anti-C) clothing designed to protect personnel against contamination is used as appropriate. Normally, Anti-C clothing will be specified in a written procedure o r in an RWP. Anti-C clothing is monitored after each use;

- The MNRC Health Physics Procedures, MNRC-0029-DOC, contains procedures for monitoring and handling contaminated equipment and components;

- Procedures for classifying contaminated material, equipment and working areas and managing, controlling, storing, and disposing of identified contaminated material are contained in the MNRC Health Physics Procedures, MNRC-029-DOC.

- Staff and visitors are trained on the risks of contamination and on the techniques for avoiding, limiting, and controlling contamination as specified in the MNRC Health Physics Procedures, MNRC-0029-DOC;

- Contamination events are documented in a radiological investigation report (RIP). These reports help avoid repeating events which caused unplanned contamination. RIPs are maintained by the Health Physics Branch and are retained for the life of the facility;

- Encapsulation requirements for items likely to cause contamination during or after irradiation are contained in the document entitled, Utilization of the University of California - Davis/McClellan Nuclear Radiation Center (UCD/MNRC) Research Reactor Facility, MNRC-0027-DOC (Reference 11.8).

11.1.8 Environmental Monitoring

The MNRC has carried out an environmental radiation monitoring program since 1988. For about two years, the program collected preoperational data, but since 1990 the program has monitored the facility during operation. While many different types of samples have been collected and analyzed, to date there has been no indication that MNRC operations have impacted the environment except at the facility fence line which shows a slight increase in ambient radiation levels above background at several specific locations, and there are no trends in environmental data which indicate that additional impacts will occur. This result is consistent with expectations for a facility of this type.

On an annual basis, the Nuclear Safety Committee audits the MNRC environmental monitoring program and the environmental data generated by the program. As a result of these audits, modifications have been made to improve the quality of the program.

The procedures for carrying out the environmental monitoring program are contained in the MNRC Health Physics Procedures (MNRC-0029-DOC). The procedures are focused on ensuring a comprehensive monitoring program which incorporates an adequate number of sample types, collected at the appropriate frequencies, analyzed with sufficient sensitivity, and reported in a timely manner to provide an early indication of any environmental impacts. Document Control measures for these procedures have already been described in Section 11.1.3.3.

With the exception of Ar-41, which has been thoroughly discussed in Section 11.1.2.1.4 and Appendix A, and in view of the MNRC policy of not discharging liquid radioactive materials down the sewer or as liquid effluents, there are virtually no pathways for radioactive materials from the MNRC to enter the unrestricted environment during normal facility operations. However, the MNRC environmental monitoring program has been structured to provide surveillance over a broad range of environmental media even though there is no credible way the facility could be impacting these portions of the environment.

The current environmental monitoring program consists of the following basic components which may change from time to time to meet program objectives; environmental monitoring locations and the types of measurements made or samples collected are summarized in Table 11-12:

- Integrated gamma dose measurements using optically stimulated luminescent dosimeters (OSL) which are exchanged quarterly. Currently OSLs are located at 37 on- industrial park sites (Sites 1-20, 50-62, and 64-71) and 7 off-industrial park sites (Sites 27, 28, 31, 38-40, and 42) (Typical sensitivity ~ 1 mrem/quarter);

- Water sample obtained quarterly. Currently this water sample is obtained at Well 54 (Site 42) (Typical sensitivity based on average minimum detectable activity for gamma emitters ~ 7 pCi/l).

- Water samples are submitted to a contractor's laboratory for analysis. Water samples are normally analyzed for gross alpha, gross beta, and tritium, and also normally undergo gamma spectroscopy. Soil and vegetation samples are normally analyzed for gross beta and normally undergo gamma spectroscopy. OSLs are processed by a contractor. All of the results are returned to the Health Physics Branch for review and compilation.

- Environmental procedures for direct radiation measurements, airborne, soil and vegetation sampling eliminated per UCD/MNRC RSO memo dated 20Jan2005 "Changes to the Health Physics Program". Memo also changed location of well water sample location.

TABLE 11-12 ENVIRONMENTAL MONITORING AND SAMPLING PROGRAM

	SITE	LOCATION	TYPE
ON-IP:	1	Control Tower Access Rd. / by control tower	O
	2	Control Tower Access Rd. / by Bldg. 1020 on fence	O
	3	A St. / by Bldg. 514	O
	4	A St./end of dorm parking lot	O
	5	Price Ave. / by Bldg. 1028	O
	6	Price Ave. / north end by perimeter fence	O
	7	Patrol Rd. / by creek	O
	8	Patrol Rd. / by red & white radar tower	O
	9	Patrol Rd. / by shooting range	O
	10	Patrol Rd. / by Magpie Creek montoring site	O
	11	North of fire station / near Bldg. 721	O
	12	Dudley & Kilzer/ E. side Bldg. 663	O
	13	Perimeter Fence / N.E. of Bldg. 475A	O
	14	Perimeter Fence / CE yard	O
	15	Parking Lot / E. of Bldg. 10	O
	16	Palm St. / across from Air Museum	O
	17	Main Water Tower / underneath tower on N.W. side	O
	18	Well #10 / by Gate 3	O
	19	Grass Area / west of Bldg. 258	O
	20	Price Ave. / Bldg. 878 roof	O
OFF-IP:	27	Rio Linda Well / Elkhorn Blvd. & 24th St. (N.W. of Base)	O
	28	Rio Linda Well / 20th St. by Vineland School (N.W. of	O
	31	Capehart Water Tower / south side of fence (N.E.	O
	38	City Well / Orange Grove Rd. (S. of Base)	O
	39	City Well / Elkhorn Blvd. & Butterball (N.E. of Base)	O
	40	City Well / Walnut Ave. water tower (E. of Base)	O
	42	City Well / Dry Creek & Ascot Ave. (W. of Base)	O, W
	50-61	MNRC Facility perimeter fence	O
	62	MNRC Staging Area (facility area monitor)	O
ON-IP:	64	Bldg. 243G / roof (west end)	O
	65-69	MNRC Facility perimeter fence	O
	71	MNRC Facility perimeter fence	O

Legend: O = OSL; W = Water Sample; IP = Industrial Park

11.2 Radioactive Waste Management

The MNRC reactor program generates very modest quantities of radioactive waste, as previously noted in Sections 11.1.2.2 and 11.1.2.3. This is due to the type of program carried out at the facility and to the fact that a conscious effort is made to keep waste volumes to a minimum.

11.2.1 Radioactive Waste Management Program

The objective of the radioactive waste management program is to ensure that radioactive waste is minimized, and that it is properly handled, stored and disposed of.

The Health Physics Branch is responsible for administering the radioactive waste management program. The organization and staffing levels, the authorities and responsibilities, and the position descriptions for the Health Physics Branch are discussed in Section 11.1.3.1. The working relationships between the health physics staff and the operations staff are discussed in Section 11.1.3.2.

The MNRC Health Physics Procedures, MNRC-0029-DOC, addresses the specific procedures for handling, storing and disposing of radioactive waste. Document control measures relating to these procedures and to other waste management documents are described in Section 11.1.3.3.

The radioactive waste management program is audited as part of the oversight function of the Nuclear Safety Committee (NSC). The NSC charter, responsibilities, meeting frequency, audit and review responsibilities, scope of audits and reviews, and qualifications and requirements for committee members are described in Section 11.1.3.5.

Waste management training is part of both the initial radiation protection training and the specialized training. It is also included in the annual refresher training. This training program and the topics covered were previously described in Section 11.1.3.4.

Radioactive waste management records are maintained by the Health Physics Branch. Radioactive waste packages in storage are tracked by a computer based radioactive material accountability system until shipment for disposal or transfer to an authorized broker. Radioactive material shipment and transfer records are also maintained by the Health Physics Branch. All records are retained for the life of the facility.

11.2.2 Radioactive Waste Controls

At the MNRC, 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 MNRC waste volumes are small and the nature of the waste items is limited and reasonably repetitive, there is usually little question about what is or is not radioactive waste. Equipment and components are categorized as waste by the reactor operations staff, while standard consumable supplies like plastic bags, gloves, absorbent material, disposable lab coats, etc., automatically become radioactive waste if detectable radioactivity above background is found to be present.

When possible, radioactive waste is initially segregated at the point of origin from items that will not be considered waste. Screening is based on the presence of detectable radioactivity using appropriate monitoring and detection techniques and on the projected future need for the items and materials involved. All items and materials initially categorized as radioactive waste are monitored a second time before packaging for disposal to confirm data needed for waste records, and to provide a final opportunity for decontamination/reclamation of an item. This helps reduce the volume of radioactive waste by eliminating disposal of items that can still be used.

11.2.2.1 Gaseous Waste

Although Ar-41 is released from the MNRC stack in the facility ventilation exhaust, this release is not considered to be waste in the same sense as the solid waste which is collected and disposed of by the facility. The Ar-41 is usually classified as an effluent which is a routine part of the normal operation of the MNRC reactor. In the MNRC facility, as in many non-power reactors, there are no special off-gas collection systems for the Ar-41. Typically, this gas simply mixes with reactor room and other facility air and is discharged along with the normal ventilation exhaust.

A complete description of Ar-41 production, evolution from the reactor tank and discharge into the unrestricted environment is contained in Sections 11.1.2.1.1 through 11.1.2.1.4 and in Appendix A. Furthermore, a description of MNRC ventilation system features which minimize releases of airborne radioactivity is contained in Section 11.1.6.2.

11.2.2.2 Liquid Waste

It is MNRC policy to minimize the release of radioactive liquid waste. Because normal MNRC operations create only small volumes of liquid which contain radioactivity, it has been possible to convert the liquids to a solid waste form and thus adhere to facility policy. In special cases, the MNRC may generate a large volume of radioactive liquid waste which cannot be converted to a solid waste. In these cases, disposal by the sanitary sewer in accordance with 10 CFR 20 may be required.

Section 11.1.2.2 describes the liquid radioactive sources associated with the MNRC reactor program . As indicated in Section 11.1.2.2, the reactor primary coolant is the only significant source. Since the primary coolant is by design contained to the maximum extent possible, there are no routine releases of this liquid and thus no significant volumes of liquid which require management as liquid waste. Certain maintenance operations, such as replacement of demineralizer resin bottles, result in very small amounts of primary coolant being drained from the water purification loop, but this liquid is easily collected at the point of origin and converted into an approved solid waste form. Other liquid radioactive waste sources such as laboratory wastes, decontamination solutions, and liquid spills have been very rare and easily within the capability of the health physics staff to convert to a solid.

Certain maintenance operations may generate a large volume of liquid waste, e.g., heat exchanger cleaning or activated concrete removal. In these cases, sewer disposal in accordance with 10 CFR 20 may be the only viable option for disposal. These cases are rare and still are not considered the norm.

11.2.2.3 Solid Waste

The procedures for managing solid waste are specified in Section 11.2.1. As with most non-power reactors, solid waste is generated from reactor maintenance operations and irradiations of various experiments. A general idea of where solid waste enters the waste control program can be obtained from the preceding information. No solid radioactive waste is intended to be retained or permanently stored on site.

Appropriate radiation monitoring instrumentation will be used for identifying and segregating solid radioactive waste. Radioactive waste is packaged in metal drums or boxes within the restricted area of the MNRC and is temporarily stored in a weatherproof enclosure within the MNRC site boundary until shipment for disposal or transfer to a waste broker. Typically a single routine “B-25 box” shipment is made every 5 years.

As stated previously, minimization of radioactive waste is a policy of the MNRC. Although there are no numerical volume goals set due to the small volume of waste generated at the MNRC, the health physics supervisor and the reactor operations supervisor periodically assess operations for the purpose of identifying opportunities or new technologies that will reduce or eliminate the generation of radioactive waste. The NSC also conducts an annual audit of the waste minimization programs as described in Section 11.1.3.5.

11.2.3 Release of Radioactive Waste

The MNRC releases Ar-41 in the ventilation exhaust as a radioactive effluent. All of the details relating to the release and potential impact of Ar-41 have been discussed previously in Sections 11.1.2.1.1 through 11.1.2.1.4 and in Appendix A. Aside from the release of this radionuclide, which may or may not qualify as a “controlled release of radioactive waste,” and infrequent releases of liquid waste as described in Section 11.2.2.2, the MNRC does not plan any routine controlled releases of radioactive waste to the environment.

Normally, the only transfer of solid radioactive waste is to an authorized solid waste broker. However, the MNRC may opt to ship solid radioactive waste directly to a low- level radioactive waste disposal site without using a broker.

CHAPTER 12

CONDUCT OF OPERATIONS

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12.6 Environmental Assessment for the McClellan Nuclear Radiation Center Reactor Operation at 2 MW, July 1995.

12.0 CONDUCT OF OPERATIONS

This chapter describes and discusses the Conduct of Operations at the University of California-Davis/ McClellan Nuclear Research Center (UCD/MNRC). The Conduct of Operations involves the administrative aspects of facility operations, the facility emergency plan, the security plan, the quality assurance plan, the reactor operator selection and re-qualification plan, the startup plan, and environmental reports. This chapter of the Safety Analysis Report (SAR) forms the basis of Section 6 of the Technical Specifications (Reference 12.1).

12.1 Organization

The UCD/MNRC Director reports directly to the UCD Vice Chancellor for Research. The UCD/MNRC is organized and administratively controlled as shown in Figures 12.1 and 12.2.

12.1.1 Structure

The organizational structures in Figures 12.1 and 12.2 show the UCD/MNRC licensee as the UCD Vice Chancellor for Research. The UCD/MNRC facility is under the direct control of the UCD/MNRC Director. The Director reports to the UCD Vice Chancellor for Research for all nuclear safety and licensing issues.

Both the Reactor Supervisor and Radiation Safety Officer report to the UCD/MNRC Director. Both the Reactor Supervisor and the Radiation Safety Officer can go directly to the Nuclear Safety Committee (NSC) with nuclear or radiation safety concerns if they cannot resolve the issue with the UCD/MNRC Director.

The UCD/MNRC license to operate is issued by the United States Nuclear Regulatory Commission (USNRC). Licensing and reporting information goes from the UCD/MNRC Director through the UCD Vice Chancellor for Research to the USNRC.

The Vice Chancellor for Research has a Nuclear Safety Committee (NSC) that meets at least semi- annually. This committee performs the review and audit of nuclear operations for the Vice Chancellor and Director, and in some cases issues approvals of various specified activities. The committee also issues an annual audit report to the UCD/MNRC Director concerning the regulatory compliance and operation of the UCD/MNRC. The UCD/MNRC Director shall review the annual audit report with the licensee once each year.

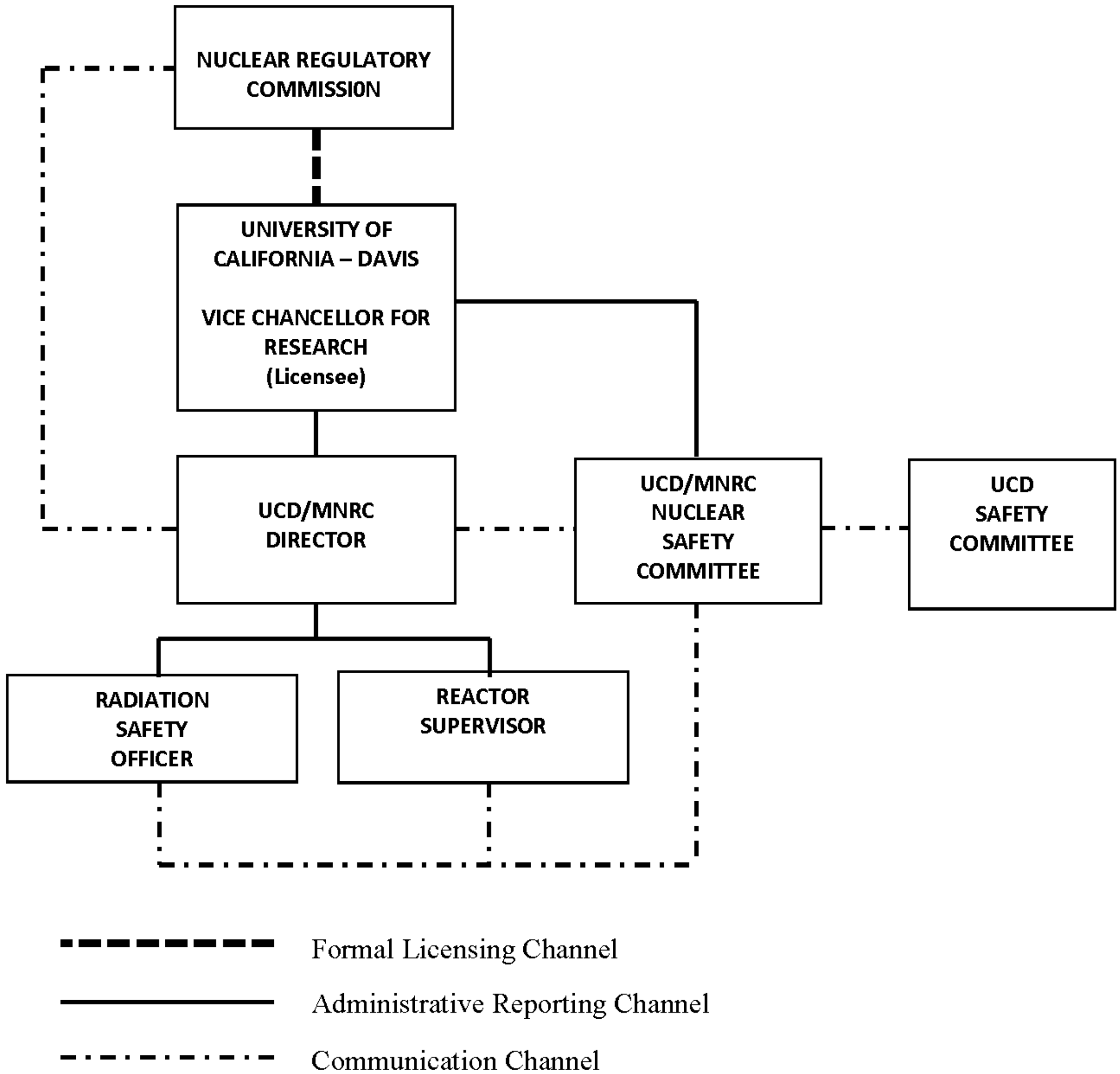

FIGURE 12.1 UCD/MNRC ORGANIZATION

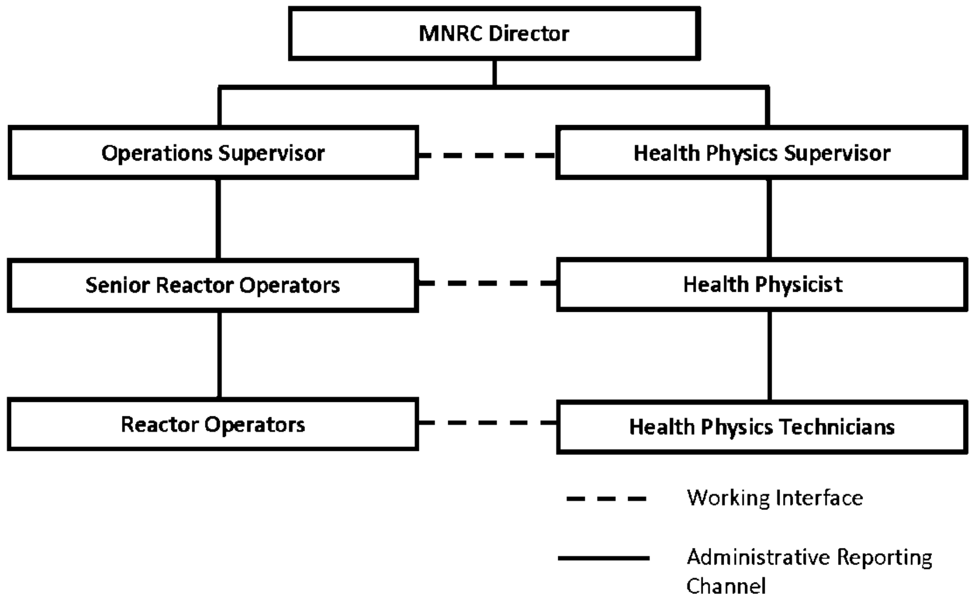

FIGURE 12.2 UCD/MNRC INTERNAL ORGANIZATION

12.1.2 Responsibility

a. UCD Vice Chancellor for Research – The UCD Vice Chancellor for Research is accountable for ensuring compliance with all licensing requirements in accordance with the USNRC codes and guides. The UCD Vice Chancellor for Research has delegated the implementation and enforcement authority for these requirements to the UCD/MNRC Director.

b. UCD/MNRC Director - The UCD/MNRC Director reports directly to the UCD Vice Chancellor for Research.

c. Operations Supervisor (Reactor Supervisor) – The Reactor Supervisor is responsible to the UCD/MNRC Director. The Reactor Supervisor reports directly to the UCD/MNRC Director on all matters concerning the reactor. The Reactor Supervisor is responsible for directing the activities of Senior Reactor Operators and Reactor Operators, and for the day-to-day operation and maintenance of the reactor. The Reactor Supervisor shall be licensed as a Senior Reactor Operator.

d. Health Physics Supervisor (Radiation Safety Officer) – The Radiation Safety Officer reports directly to the UCD/MNRC Director. The Radiation Safety Officer is responsible to the UCD/MNRC Director for directing the activities of Health Physics personnel, including development and implementation of the Radiation Safety Program.

e. Senior Reactor Operator – Senior Reactor Operators report to the Reactor Supervisor. Senior Reactor Operators are responsible for directing the activities of Reactor Operators on their assigned shift. Senior Reactor Operators shall be licensed at the Senior Reactor Operator level.

f. Health Physicist –The Health Physicist report to the Health Physics Supervisor. The Health Physicist is responsible for implementation of the Radiation Safety Program and for directing activities of the Health Physics Technicians.

g. Reactor Operator – Reactor Operators report to the Senior Reactor Operator on their assigned shift. Reactor Operators are primarily involved in the manipulation of reactor controls, monitoring of instrumentation and operation and maintenance of reactor related equipment. Reactor Operators shall be licensed at the Reactor Operator level.

h. Health Physics Technicians – Health Physics Technicians report to the Radiation Safety Officer. Health Physics Technicians are responsible for radiological monitoring, performing surveillance checks on radiological monitoring equipment throughout the UCD/MNRC facility, as well as taking environmental samples and providing radiological control oversight of operations involving radiation and/or contamination. Health Physics Technicians have the authority to interdict perceived unsafe practices.

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 Operator whenever needed. The call list shall include:

   (1) Management personnel

   (2) Health Physics personnel; and

   (3) Reactor Operations personnel

b. Reactor operator trainees shall be permitted to manipulate the controls of the reactor under the direct supervision of Licensed Reactor Operators

c. The minimum staffing when the reactor is not secured shall be:

   (1) A reactor operator in the control room

   (2) A second person in the facility area who can perform prescribed instructions

   (3) A Senior Reactor Operator readily available. The available senior reactor operator should be within thirty (30) minutes of the facility and reachable by telephone and;

   (4) A Senior Reactor Operator shall be present whenever a reactor startup is done, fuel is being moved or experiments are being placed in the reactor tank.

12.1.4 Selection and Training of Personnel

The UCD/MNRC Selection and Training Plan for Reactor Personnel (MNRC-009-DOC) contains the detailed information concerning the selection, training, licensing and re- qualification of reactor personnel. This plan addresses the qualifications, initial training, licensee responsibilities, and requalification of UCD/MNRC reactor operations personnel.

The UCD/MNRC training program complies with ANSI/ANS 15.4 - 1988 (Reference 12.2). The program's objective is to train, qualify, and re-qualify individuals for operation and maintenance of the reactor. The content of the training program covers the as-built and existing facility, significant facility modifications, current procedures, and administrative rules and regulations.

In addition to actual personnel training, Reactor Operators and Senior Reactor Operators are required to meet specific medical qualifications. The physical condition and the general health of UCD/MNRC reactor operations personnel shall be such that they are capable of properly operating under normal, abnormal and emergency conditions. The primary responsibility for assuring that medically qualified personnel are on duty rests with the UCD/MNRC Director. The health requirements set forth in the UCD/MNRC Selection and Training Plan for Reactor Personnel (MNRC-009-DOC) shall be used to determine the physical condition and general health of the individual. The designated medical examiner should be conversant with the medical requirements of this program.

In addition to the selection and training of reactor operations personnel, the UCD/MNRC provides formal annual training for all facility personnel in radiation protection topics, in items required by 10 CFR Part 19, in the As Low as Reasonably Achievable (ALARA) concept and in other related areas. The training is structured at different levels in order to meet the needs of different categories of facility staff and facility users. For more details in this aspect of UCD/MNRC personnel training, see Chapter 11, Section 11.1.3.4.

12.1.5 Radiation Safety

The purpose of the Radiation Safety Program is to allow the maximum beneficial use of radiation sources with minimum radiation exposure to personnel. Requirements and procedures set forth in this program are designed to meet the following fundamental principles of radiation protection:

- Justification - No practice shall be adopted unless its introduction produces a net positive benefit

- Optimization - All exposures shall be kept as low as reasonably achievable

- Limitation – the dose equivalent to individuals shall not exceed limits established by appropriate state and federal agencies. These limits shall include, but not be limited to those set forth in the Federal Regulations

All personnel using radiation sources shall become familiar with the requirements of the Radiation Safety Program and conduct their operations in accordance with them.

The Radiation Safety Program uses Reference 12.3 as a guide.

The details of the Radiation Safety Program can be found in Chapter 11

12.2 Review and Audit Activities

General Policy. It is the policy that nuclear facilities shall be designed, constructed, operated, and maintained in such a manner that facility personnel, the general public, and both university and non-university property are not exposed to undue risk. These activities shall be conducted in accordance with applicable government regulatory requirements.

The UCD Vice Chancellor for Research as the facility licensee has ultimate responsibility for assuring that the above policy is followed. The Nuclear Safety Committee (NSC) has been chartered to assist in meeting this responsibility by providing timely, objective, and independent reviews, audits, recommendations and approvals on matters affecting nuclear safety. The NSC is established in accordance with the guidance of Reference 12.1. The following describes the procedures, which govern the composition and conduct of the NSC.

12.2.1 Composition and Qualifications

The UCD/MNRC Vice Chancellor for Research shall appoint the chairman of the NSC. The NSC Chairman shall appoint a Nuclear Safety Committee of at least seven (7) members knowledgeable in fields which relate to nuclear safety.

12.2.2 Charter and Rules

The NSC shall conduct its review and audit/inspection functions in accordance with a written charter. This charter shall include provisions for:

a. Meeting frequency (the committee shall meet at least semiannually);

b. Voting rules

c. Quorums

d. A committee review function and an audit/inspection function;

e. Use of subcommittees; and;

f. Review, approval and dissemination for meeting minutes.

12.2.3 Review Function

The responsibilities of the NSC, or a designated subcommittee thereof, shall include but are not limited to the following:

a. Review approved experiments utilizing UCD/MNRC nuclear facilities;

b. Review and approve all proposed changes to the facility license, the Technical Specifications and the Safety Analysis Report, and any new or changed Facility Use Authorizations and proposed Class I modifications, prior to implementing (Class I) modifications, prior to taking action under the preceding documents or prior to forwarding any of these documents to the Nuclear Regulatory Commission for approval.

c. Review and determine whether a proposed change, test, or experiment would constitute an un-reviewed safety question or require a change to the license, to a Facility Use Authorization, or to the Technical Specifications. This determination may be in the form of verifying a decision already made by the UCD/MNRC Director;

d. Review reactor operations and operational maintenance, Class I modification records, and the health physics program and associated records for all UCD/MNRC nuclear facilities

e. Review the periodic updates of the Emergency Plan and Physical Security Plan for UCD/MNRC nuclear facilities

f. Review and update the NSC Charter every two (2) years;

g. Review abnormal performance of facility equipment and operating anomalies;

h. Review all reportable occurrences and all written reports of such occurrences prior to forwarding the final written report to the Nuclear Regulatory Commission; and

i. Review the NSC annual audit/inspection of the UCD/MNRC nuclear facilities and any other inspections of these facilities conducted by other agencies.

12.2.4 Audit/Inspection Function

The NSC, or a subcommittee thereof, shall audit/inspect reactor operations and health physics annually. The annual audit/inspection shall include, but not be limited to the following:

a. Inspection of the reactor operations and operational maintenance, Class I modification records, and the health physics program and associated records, including the ALARA program, for all UCD/MNRC nuclear facilities;

b. Inspection of the physical facilities at the UCD/MNRC;

c. Examination of reportable events at the UCD/MNRC;

d. Determination of the adequacy of UCD/MNRC standard operating procedures;

e. Assessment of the effectiveness of the training and retraining programs at the UCD/MNRC;

f. Determination of the conformance of operations at the UCD/MNRC with the facility's license and Technical Specifications, and applicable regulations;

g. Assessment of the results of actions taken to correct deficiencies that have occurred in nuclear safety related equipment, structures, systems, or methods of operation;

h. Inspection of the currently active Facility Use Authorizations and associated experiments;

i. Inspection of future plans for facility modifications or facility utilization;

j. Assessment of operating abnormalities; and

k. Determination of the status of previous NSC recommendations.

12.3 <u>Procedures</u>

Written procedures shall be prepared and approved prior to initiating any of the activities listed in this section. The procedures shall be approved by the UCD/MNRC Director. A periodic review of procedures will be performed and documented in a timely manner to assure they are current. Procedures shall be adequate to assure the safe operation of the reactor, but will not preclude the use of independent judgment and action should the situation require. The following sections list UCD/MNRC programs that will typically require reviewed written procedures.

12.3.1 Reactor Operations:

a. Startup, operation, and shutdown of the reactor;

b. Fuel loading, unloading, and movement within the reactor;

c. Control rod removal or replacement;

d. Routine maintenance of the control rod drives and reactor safety and interlock systems or other routine maintenance that could have an effect on reactor safety;

e. Testing and calibration of reactor instrumentation and controls, control rods and control rod drives;

f. Administrative controls for operations, maintenance, and conduct of irradiations and experiments that could affect reactor safety or core reactivity.

g. Implementation of required plans such as emergency or security plans; and

h. Actions to be taken to correct specific and foreseen potential malfunctions of systems, including responses to alarms and abnormal reactivity changes.

12.3.2 Health Physics

a. Testing and calibration of area radiation monitors, facility air monitors, laboratory radiation detection systems, and portable radiation monitoring instrumentation;

b. Working in laboratories and other areas where radioactive materials are used;

c. Facility radiation monitoring program including routine and special surveys, personnel monitoring, monitoring and handling of radioactive waste, and sampling and analysis of solid and liquid waste, and gaseous effluents released from the facility;

d. Monitoring radioactivity in the environment surrounding the facility;

e. Administrative guidelines for the facility radiation protection program to include personnel orientation and training;

f. Receipt of radioactive materials at the facility, and unrestricted release of materials and items from the facility, which may contain induced radioactivity or radioactive contamination;

g. Leak testing of sealed sources containing radioactive materials;

h. Special nuclear material accountability; and

i. Transportation of radioactive materials.

Changes to the written procedures of the above programs shall require approval of the UCD/MNRC Director. All such changes should be documented. Staff shall be trained on all changes made.

12.4 Required Action

12.4.1 Reportable Events

12.4.1.1 Safety Limit Violation

Actions to be taken in the case of a safety limit violation shall include cessation of reactor operations until resumption is authorized by the licensing authority, a prompt report of the violation to the licensing authorities and to the licensee, and a subsequent follow-up report which shall be reviewed by the NSC and then submitted to the licensing authority.

The follow-up report shall describe applicable circumstances leading to the violation, including causes and contributing factors that are known, effect of the violation upon reactor facility components, systems or structures, health and safety of personnel and the public, and corrective actions to prevent recurrence. Prompt reporting of the event shall be by telephone and confirmed by written correspondence within 24 hours. A written report is to be submitted within 14 days.

12.4.1.2 Release of Radioactivity

Actions to be taken in the event of a release of radioactivity from the operations boundary above allowable limits shall include returning the reactor to normal operating conditions or, if necessary to correct the occurrence, a reactor shutdown and no return to normal operation until authorized by the UCD/MNRC Director. There will also be a report to the licensee and licensing authority, and a review of the event and applicable reports by the NSC prior to submission of the required reports. Prompt reporting of the event shall be by telephone or similar conveyance within 24 hours to the NRC Operations Center. A written report is to be submitted to the NRC Document Control Desk, Washington, D.C. within 14 days.

12.4.1.3 Special Reports

Other events that will be considered reportable events are listed in this section. Appropriate reports shall be submitted to licensing authorities and such reports shall be reviewed by the NSC prior to submission. (Note: Where components or systems are provided in addition to those required by the Technical Specifications, the failure of these 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.)

Special reports are used to report unplanned events as well as planned major facility and administrative changes. The following classifications shall be used to determine the appropriate reporting schedule:

a. A report within 24 hours by telephone or similar conveyance to the NRC operations center of:

   (1) Any accidental release of radioactivity into unrestricted areas above applicable unrestricted area concentration limits, whether or not the release resulted in property damage, personal injury, or exposure;
   (2) Any violation of a safety limit;

(3) Operation with a limiting safety system setting less conservative than specified;

(4) Operation in violation of a Limiting Condition for Operation;

(5) Failure of a required reactor or experiment safety system component which could render the system incapable of performing its intended safety function unless the failure is discovered during maintenance tests or a period of reactor shutdown;

(6) Any unanticipated or uncontrolled change in reactivity greater than $1.00;

(7) An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of a condition which could have resulted in operation of the reactor outside the specified safety limits and;

(8) A measureable release of fission products from a fuel element.

b. A report within 14 days in writing to the NRC, Document Control Desk, Washington DC:

(1) Those events reported as required by Sections a. (1) through a. (8) above; and

(2) The written report (and, to the extent possible, the preliminary telephone report or report by similar conveyance) shall describe, analyze, and evaluate safety implications, and outline the corrective measures taken or planned to prevent recurrence of the event.

12.4.1.4 Other Reports

A written report shall be submitted within thirty (30) days to the NRC, Document Control Desk, Washington DC as a result of the following conditions:

a. Any significant variation of measured values from a corresponding predicted or previously measured value of safety-connected operating characteristics occurring during operation of the reactor;

b. Any significant change in the transient or accident analysis as described in the Safety Analysis Report (SAR);

c. A personnel change involving the positions of UCD/MNRC Director or UCD Vice Chancellor for Research; and

d. Any observed inadequacies in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused an existence or development of an unsafe condition with regard to reactor operations.

12.4.1.5 Annual Report

An annual report covering the activities of the reactor facility during the previous calendar year shall be submitted within six months following the end of each calendar year. Each annual report shall include the following information:

a. A brief summary of operating experiences including experiments performed, changes in facility design, performance characteristics and operating procedures related to reactor safety occurring during the reporting period, and results of surveillance tests and inspections;

b. A tabulation showing the energy generated by the reactor (in megawatt hours), hours the reactor was critical, and the cumulative total energy output since initial criticality;

c. The number of emergency shutdowns and inadvertent scrams, including reasons for the shutdowns or scrams;

d. Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safety of the operation of the reactor and the reasons for any corrective maintenance required;

e. A brief description, including a summary of the safety evaluations, of changes in the facility or in procedures, and of tests and experiments carried out pursuant to Section 50.59 of 10 CFR Part 50;

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

g. An annual summary of the radiation exposure received by facility operations personnel, by facility users, and by visitors in terms of the average radiation exposure per individual and the greatest exposure per individual in each group;

h. An annual summary of the radiation levels and levels of contamination observed during routine surveys performed at the facility in terms of average and highest levels; and

i. An annual summary of any environmental surveys performed outside the facility.

12.5 Records

Records of the following activities shall be maintained and retained for the periods specified below. The records may be in the form of logs, data sheets, or other suitable forms. The required information may be contained in single or multiple records, or a combination thereof.

12.5.1 Lifetime Records

Lifetime records are records 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.) The following are examples of lifetime records:

a. Offsite environmental monitoring surveys;

b. Fuel inventories and transfers;

c. Facility radiation and contamination surveys;

e. Radiations exposures for all personnel and;

f. Update, corrected and as-built drawings of the facility

12.5.2 Five Year Records

Records which are to be retained for a period of at least five years or for the life of the component involved whichever is shorter are as follows:

a. Normal reactor operation;

b. Principal maintenance activities

c. Those events reported as required by Section 12.4.1

d. Equipment and component surveillance activities required by the Technical Specifications;

e. Experiments performed with the reactor; and

f. Airborne and liquid radioactive effluents released to the environments and solid radioactive waste shipped off site.

12.6 Emergency Planning

The UCD/MNRC Emergency Plan (MNRC-001-DOC) contains detailed information concerning the UCD/MNRC response to emergency situations. The UCD/MNRC Emergency Plan is written to be in accordance with Reference 12.4. The information below will give a general overview of the emergency plan.

The UCD/MNRC Emergency Plan is designed to provide response capabilities to emergency situations involving the UCD/MNRC. The plan deals with the UCD/MNRC Facility, the spectrum of emergency situations and accident conditions that could arise within the facility, and the associated emergency responses that are required due to the unique nature of the reactor facility. Detailed emergency implementing procedures referenced in this plan. This approach provides the UCD/MNRC facility emergency staff the flexibility to cope with a wide range of emergency situations without requiring frequent revisions to the plan.

The responsibility for the plan rests with the UCD/MNRC Director who is also responsible for response to and recovery from emergencies. Implementation of the UCD/MNRC Emergency Plan on a day-to-day basis is the responsibility of the Senior Reactor Operator (SRO) on duty.

Provisions for reviewing, modifying, and approving emergency implementation procedures are defined in the UCD/MNRC Emergency Plan to assure that adequate measures to protect the staff and the general public are in effect at all times.

12.7 Security Planning

The UCD/MNRC Physical Security Plan (MNRC-003-DOC) contains detailed information concerning the UCD/MNRC security measures. The information below will give a general overview of this plan.

The UCD/MNRC Physical Security Plan provides the criteria and actions for protecting the facility from acts of intrusion, theft, civil disorder and bomb threats.

Overall responsibility for facility security rests with the UCD/MNRC Director, who is responsible for implementation of the plan. Implementation of the security plan on a day-to- day basis during hours of operation is the responsibility of the Senior Reactor Operator (SRO) on duty.

12.8 Quality Assurance

The UCD/MNRC Quality Assurance (QA) Program (MNRC-0045-DOC) contains detailed information concerning the UCD/MNRC QA Program elements and their implementation.

The UCD/MNRC QA Program provides criteria for design, construction, operation, and decommissioning of the UCD/MNRC reactor facility. The level of QA effort applied to UCD/MNRC reactor activities is consistent with the importance of these activities to safety. The activities included in the UCD/MNRC QA Program are those related to reactor safety and applicable radiation monitoring systems. The specific elements of the UCD/MNRC QA Program are the same as those listed in Reference 12.5.

12.9 Operator Training and Requalification Program

The UCD/MNRC Selection and Training Plan for Reactor Personnel (MNRC-009-DOC) has been established to train, qualify, and re-qualify individuals for operation and maintenance of the reactor. The content of the training shall cover the as-built and existing facility, significant facility modifications, current procedures, and administrative rules and regulations (Reference 12.2).

The program shall carry the trainee through documented stages of academic training and on-the-job training. The intended results shall be a candidate who anticipates conditions, who communicates well and who can accomplish required tasks during normal and abnormal operational situations. Licensing of a candidate is achieved after successful completion of the training and the following examinations. Written examinations covering the following categories shall be passed:

a. Nuclear Theory and Principles of Operation;
b. Facility Design and Operating Characteristics;
c. Facility Instrumentation and Control Systems;
d. Normal, Abnormal and Emergency Procedures;
e. Radiological Control and Safety;
g. Technical Specifications, to include bases for Senior Reactor Operator candidates;
h. Fuel Handling; and
i. Administrative Controls, Procedures and Regulations.

The minimum acceptance score in any category shall be established. Failures in no more than two categories can be made up by re-examination in only those categories. Failure in more than two categories requires repeating the entire examination. Regardless of the test results, if the individual's test record indicates a deficiency in a critical area that affects safety, security or operational functions, a remedial training program shall be administered to promptly correct the critical deficiency.

The objective of the re-qualification program is to refresh reactor operator's knowledge in areas of infrequent operation, to review facility and procedural changes, to address subject matter not reinforced by direct use, and to improve performance weaknesses. The program shall be designed to evaluate an operator's knowledge and proficiency for his duties. The program shall take into account the specialized nature and mode of operation of the UCD/MNRC reactor, and the background, skill, degree of responsibility, and participation of UCD/MNRC reactor operations personnel in activities related to reactor operations.

The requalification program shall consist of the following items:

- Schedule: The requalification program shall be conducted over a period not to exceed 24 months to be followed by successive two-year programs;

- Content: To formulate the basis for determining the contents of the re- qualification program, changes in jobs, tasks, and participation in related activities should be periodically reviewed. The following shall be adhered to:

1. Lectures: The re-qualification program shall include preplanned lectures in the categories listed in 12.9;

2. On the Job Training:To maintain active status, each licensed reactor operator shall manipulate the reactor controls and each licensed senior reactor operator shall either manipulate the reactor controls or direct the activities of individuals during reactor control manipulations for a minimum of four hours per calendar quarter;

3. Comprehensive Written Examination: A comprehensive written examination covering the categories listed in 12.9 shall be administered biennially to determine whether weaknesses exist and to identify categories for which retraining and retesting may be required;

4. Annual Operating Examination: An operational examination shall be given annually that requires the Senior Reactor Operator and the Reactor Operator to demonstrate an understanding of, and the ability to perform, the actions necessary to accomplish a comprehensive sample of the items listed below:

   a. Perform pre-startup procedures for the facility;

   b. Manipulate the console controls as required to operate the facility during normal, abnormal and emergency conditions;

   c. Identify annunciators and condition indicating signals and perform appropriate remedial actions;

   d. Identify the instrumentation systems and their significance;

   e. Describe the function of the facility's radiation monitoring system as it pertains to reactor operations;

   f. Demonstrate knowledge of significant radiation hazards and the steps taken to reduce personnel exposure;

   g. Demonstrate knowledge of the facility emergency plan including, as appropriate, the Senior Reactor Operator's or Reactor Operator's responsibility to decide whether the plan should be executed and the duties under the plan

   h. Demonstrate that the Senior Reactor Operator or Reactor Operator can function in the control room in such a way that the facility licensee's procedures are adhered to and that the limitations in its license and amendments are not violated;

5. Licensed operators shall be trained on changes to the facility and facility documentation, including Technical Specifications and procedures, before performing licensed duties that are affected by the changes; and

6. All licensed operators shall review the contents of all normal, abnormal, and emergency procedures on an annual basis

- Absence from Licensed Functions: An individual who has not actively performed licensed functions for four (4) hours per calendar quarter shall demonstrate to the Reactor Supervisor or the UCD/MNRC Director that his/her knowledge and understanding of the operation and administration of the UCD/MNRC facility are satisfactory before returning to licensed duties. This shall be accomplished through an interview and evaluation or a written or operational examination or a combination thereof. The individual shall be required to perform a minimum of six (6) hours of shift functions under the direction of a Senior Reactor Operator before returning to licensed duties.

Re-qualification examinations shall be administered by individuals knowledgeable of the UCD/MNRC operation.

12.10 Startup Plan

A start up plan is not required as this is not a new facility nor does this license renewal application request authorization of modifications that require verification of operability before normal operations are resumed.

12.11 Environmental Reports

An Environmental Assessment (EA) for the UCD/MNRC reactor operating license renewal application was prepared and was provided to the NRC staff separately.

CHAPTER 13

ACCIDENT ANALYSIS

Chapter 13 - Valid Pages
Rev. 11 06/10/20

all	Rev. 11	06/10/20

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13.0 ACCIDENT ANALYSIS

13.1 Introduction

In about 1980, the U.S. Nuclear Regulatory Commission requested an independent and fresh overview analysis of credible accidents for TRIGA® and TRIGA®-fueled reactors. Such an analysis was considered desirable since safety and licensing concepts had changed over the years. The study resulted in NUREG/CR-2387, Credible Accident Analysis for TRIGA® and TRIGA®-fueled Reactors (Reference 13.1). The information developed by the TRIGA® experience base and appropriate information from NUREG/CR-2387 serve as a basis for some of the information presented in this chapter of the UCD/MNRC Safety Analysis Report.

The reactor physics and thermal-hydraulic conditions in the UCD/MNRC TRIGA® reactor at a power level of 1 MW are established in Chapter 4. The core physics analysis demonstrates that the fundamental physical conditions in the UCD/MNRC reactor are preserved by an appropriate choice of the composition of mixed TRIGA® fueled cores.

The fuel temperature is a limit in both steady-state and pulse mode operation. This limit stems from the out-gassing of hydrogen from U-ZrH fuel and the subsequent stress produced in the fuel element cladding material. The strength of the cladding as a function of temperature sets the upper limit on the fuel temperature. Fuel temperature limits of 1100°C (with clad <500°C) and 930°C (with clad >500°C) for U-ZrH with a H/Zr ratio less than 1.70 have been set to preclude the loss of clad integrity (Section 4.5.4.1.3).

Nine credible accidents for research reactors were identified in NUREG-1537 (Reference 13.2) as follows:

- the maximum hypothetical accident (MHA);
- insertion of excess reactivity;
- loss of coolant accident (LOCA);
- loss of coolant flow;
- mishandling or malfunction of fuel;
- experiment malfunction;
- loss of normal electrical power;
- external events;
- mishandling or malfunction of equipment.

This chapter contains analyses of postulated accidents that have been categorized into one of the above nine groups. Some categories do not contain accidents which appeared applicable or credible for the UCD/MNRC TRIGA® reactor, but this was acknowledged in a brief discussion of the category. Some categories contain an analysis of more than one accident even though one is usually limiting in terms of impact. Any accident having significant radiological consequences was included.

For those events that do result in the release of radioactive materials from fuel, only a qualitative evaluation of the event is presented. Events leading to the release of radioactive material from a fuel element were analyzed to the point where it was possible to reach the conclusion that a particular event was, or was not, the limiting event in that accident category. The maximum hypothetical accident (MHA) for TRIGA® reactors is the cladding failure of a single irradiated element in air with no radioactive decay of contained fission products. Calculations supporting the analysis of this accident and several of the other accidents discussed in this chapter are contained in Appendix B.

13.2 Accident Initiating Events and Scenarios, Accident Analysis, and Determination of Consequences

13.2.1 Maximum Hypothetical Accident

13.2.1.1 Accident Initiating Events and Scenario

A single fuel element could fail at any time during normal reactor operation or while the reactor was shutdown, owing to a manufacturing defect, corrosion, or handling damage. This type of failure is infrequent, based on many years of operating experience with TRIGA® fuel, and such a failure would not normally incorporate all the necessary operating assumptions required to obtain a worst case fuel failure scenario.

For the UCD/MNRC TRIGA® reactor, the MHA has been defined as a cladding rupture of one highly irradiated fuel element with no decay followed by instantaneous release of fission products into the air. The failed fuel element was assumed to have been operated at the highest core power density for a continuous period of 1 year at 1 MW. This is the most severe accident for a TRIGA® and is analyzed to determine the limiting or bounding potential radiation doses to the reactor staff and to the general public in the unrestricted area.

A realistic scenario for the MHA is difficult to establish since fuel handling, the activity frequently associated with this accident, would be unlikely to occur immediately after reactor shutdown, and fuel elements would not be moved out of the reactor tank into air with no time to decay. Nevertheless, the accident has been analyzed for the UCD/MNRC TRIGA® in Appendix B and the results are summarized in this section.

13.2.1.2 Accident Analysis and Determination of Consequences

The fission product inventory used in the MHA is listed in Table B-1 of Appendix B. The data are for the volatile fission products present at shutdown in a fuel element run to saturation at the highest core power density.

For both accidents being analyzed in this chapter, a release fraction of 2.4×10^{-5} is assumed for the release of noble gases and halogens from the fuel to the cladding gap. This release fraction is developed in Chapter 4 and is based on the maximum measured fuel temperature (400 C) which corresponds to the average fuel temperature of the highest thermal output fuel element of the LCC core.

In addition, for the accident where the cladding failure occurs in air, it is very conservatively assumed that 25% of the halogens released to the cladding gap are eventually available for release from the reactor room to the outside environment. A release fraction of 2.4×10^{-5} was also assumed for a single fuel element failure 24 hours after shutdown in order to keep the radiological consequences conservative. In addition, it is assumed that 100% of the noble gases ultimately reach the unrestricted environment outside the reactor building. This value for the halogens is based on historical usage and recommendations from Appendix B References B.2, B.3, B.4, B.5, and B.6, where Reference B.2 recommends a 50% release of the halogens. References B.3 and B.4 apply a natural reduction factor of 50% due to plateout in the building. This latter 50% applied to the 50% of the inventory released from the fuel element cladding gap results in 25% of the available halogen inventory reaching the outside environment. It should be noted, however, that this value appears to be quite conservative based on the 1.7% gap release fraction for halogens quoted in References B.7 and B.8.

The Lawrence Livermore National Laboratory program "HotSpot" was used to determine the worst case radiation dose impact to the public for a variety of atmospheric stability class. Over the past several years the program has become the industry standard for relatively simple Gaussian plume modeling of radioactive material release.

Furthermore, it was assumed that all of the fission products were released to the unrestricted area instantaneously, which would maximize the dose rate to persons exposed to the plume during the accident and minimize the exposure time to receive the highest estimated dose from this accident. It is also assumed that the release height of the radioactive material to be the height of the floor of the reactor room (19 feet above ground level) not the 60 feet MNRC stack. This was done to keep the radiological results as conservative as possible. Additionally, a very calm wind speed of 1 m/s was selected as lower wind speeds in these types of releases produce greater radiological consequences. Dose conversion factors from FGR 13 (ICRP 60/70 series) were used to along with a breathing rate of 4.17e-4 m^3/s to calculated various doses to the receptors of interest (1.5 m above ground level). Calculations were done for Pasquill weather classifications A through F.

Worker dose was calculated by assume an instantaneous release of the radioactive source term into the reactor room and not being removed by the ventilation system. This was done to keep the radiological consequences conservative (overestimation). The reactor room volume was used to calculate the individual derived air concentration (DAC) values for each isotope of interest. The various DAC values and egress times in the reactor room were used to calculate the final worker dose consequences. Note the reactor room was assumed to be large enough to mimic a semi-infinite cloud for the external radiation sources (primarily radio-xenon). As the reactor room is much smaller than a semi-infinite cloud this method will over predict expected worker dose from external radiation.

1. Initial Fission Product Source;

   a. Reactor Power level - 1 MW;

   b. Operating time - 365 days;

2. Release Fractions;

   a. For the purposes of the maximum hypothetical accident and the accident where the pool water remains present, it was assumed that at the time of fuel cladding failure, a fraction of the radionuclide in the inventory given in Appendix B Table B-1 was instantaneously released into the air of the reactor room. In one scenario this instantaneous release occurs directly into the reactor room air, while in the other projected accident the pathway to the air requires migration through the pool water which will reduce the halogen release. Also, for the halogens, a further reduction in activity is expected to occur due to plateout in the building. Thus, the fraction (w_i) of the fission product inventory released from a single fuel element which reaches the reactor room air and then the atmosphere in the unrestricted environment outside the facility will be as follows:

$w_i = e_i \times f_i \times g_i \times h_i$; (1)

where:

e_i = the fraction released from the fuel to the fuel-cladding gap;

f_i = the fraction released from the fuel-cladding gap to the pool if water is present, or directly to the reactor room air if no water is present;

g_i = the fraction released from the pool water to the reactor room air; and

h_i = the fraction released from the reactor room air to the outside (unrestricted) environment.

b. Based on information provided previously in this appendix, e_i was set at 2.4×10^{-5} for both noble gases and halogens and at zero for other fission products. The values of f_i, g_i, and h_i for the two accident scenarios being considered are given in Table 13-1.

Table 13-1. Values of Release Fraction Components

Fission Product	f_i		g_i		h_i
	w/o pool water	w/ pool water	w/o pool water	w/ pool water	
Noble Gases	1.0	1.0	N/A	1.0	1.0
Halogens	0.5*	0.5*	N/A	0.05*	0.5

Offsite Dose: As can be seen below in table the offsite radiological consequences are low, which is consistent with the MHA of similar TRIGA reactors, primarily due to high retention of radionuclides in by the fuel itself during such an accident. The maximum expected CDE for the thyroid and TEDE occur under atmospheric D at a distance of 59 m from the MNRC reactor room. The thyroid CED is 48 mrem and the TEDE is 2.4 mrem, both well below the 10 CFR 20 annual limits for the public. The evaluation distances selected for table 13-2 were based on the closest point from the MNRC reactor room the site boundary (fence line), the closest inhabited building, and the closest residence.

* These values are conservative based on References B.7 and B.8, which quote a halogen release fraction of 0.017 from the cladding gap. Reference B.8 also quotes a halogen release from TMI-2 fuel to the reactor building of 0.006.

Table 13-2 Radiation Doses to Members of the General Public Under Different Atmospheric Conditions and at Different Distances from the MNRC Following a Fuel Element Cladding Failure in Air (MHA). All doses in mrem, CDE - Committed Dose Equivalent, TEDE - Total Effective Dose Equivalent

33 meters (MNRC Fence Line)

Weather Class	A	B	C	D	E	F
CD Thyroid	36	46	41	25	0.08	0
EDE	1.8	2.3	2.1	1.2	<0.01	0

93 meters (Closest Inhabited Building)

Weather Class	A	B	C	D	E	F
CD Thyroid	6.3	13	25	38	33	4.6
EDE	0.31	0.65	1.2	1.8	1.6	0.2

480 meters (Closest Residence)

Weather Class	A	B	C	D	E	F
CD Thyroid	0.25	0.55	1.3	2.9	6.7	16
EDE	0.01	0.03	0.06	0.13	0.30	0.7

Maximum Dose and Distance

Weather Class	A	B	C	D	E	F
CDE Thyroid	36 (@33m)	46 (@33m)	47 (@44m)	48 (@59m)	35 (@120m)	30(@220m)
TEDE	1.8 (@33m)	2.3 (@33m)	2.3 (@44m)	2.4 (@59m)	1.7 (@120m)	1.4(@220m)

Worker Dose:

As indicated by the results in Table 13-3, the occupational dose to workers who evacuate the reactor room within 5 minutes following the MHA should be approximately 300 millirem TEDE and 2,495 millirem Committed Dose Equivalent to the thyroid. If evacuation occurs within 2 minutes, as it no doubt will because the reactor room is small and easy to exit, the doses drop to 120 millirem TEDE and 998 millirem CDE. All of these doses are well within the NRC limits for occupational exposure as stated in 10 CFR 20.

Table 13-3 Worker dose for MHA

	CDE Thyroid (millirem)	TEDE (millirem)
2 minute room occupancy	998	120
5 minute room occupancy	2495	300

13.2.2 Insertion of Excess Reactivity

13.2.2.1 Accident Initiating Events and Scenarios

The most credible generic accident is the inadvertent rapid insertion of positive reactivity which could, if large enough, produce a transient resulting in fuel overheating and a possible breach of cladding integrity. Operator error or failure of the automatic power level control system could cause such an event to occur due to the uncontrolled withdrawal of a single control rod. Flooding or removal of beam tube inserts could also have a positive effect on reactivity but not as severe as removal of a control rod. In a separate scenario, a large reactivity insertion was postulated to create fuel cladding temperatures which might cause a metal-water reaction, but for many reasons this accident is not considered to be a safety risk in TRIGA® reactors.

13.2.2.2 Accident Analysis and Determination of Consequences

13.2.2.2.1 Maximum Reactivity Insertion

Raising the temperature of TRIGA® fuel has a strong, prompt negative reactivity effect, which can overcome a rapid reactivity insertion such as that produced by the firing of the transient rod or the accidental ejection of a high negative reactivity worth experiment. The quantity that captures this effect is the prompt negative temperature coefficient discussed in Section 4.5.4.2. There is a limit to the protection provided by this feedback, since the peak fuel temperature attained before the feedback terminates the transient increases with the magnitude of the inserted reactivity. The Nordheim-Fuchs model was used to compute the maximum reactivity pulse that can occur without exceeding the safety limit of 1100°C established in Section 4.5.4.1.3.

In the Nordheim-Fuchs model it is assumed the transient is so rapid that 1) the temperature rise is adiabatic and 2) delayed neutrons can be neglected. Thus, the model is given by the following set of coupled differential equations:

$$\frac{dn}{dt} = \frac{\rho-\beta}{l} \times n; \quad (2)$$

$$\rho(T) = \rho_o - \alpha(T) \times T; \quad (3)$$

$$\frac{dT}{t} = \frac{n}{C_\rho(T)} \quad (4)$$

Where n is the reactor power, ρ is the time-dependent reactivity, l is the neutron lifetime, β is the effective delayed neutron fraction, T is the core-average temperature, ρ_o is the reactivity insertion, α is the temperature feedback reactivity coefficient, and C_ρ is the whole-core heat capacity. Given values of β, l, and ρ_o, and expressions for α and C_ρ, this set of equations was solved numerically using simple finite difference techniques. The quantity of interest in the solution is ΔT, the difference between the maximum and initial values of the core-average fuel temperature. From the solution ΔT, the peak fuel temperature was found using the simple expression:

$$T_{peak} = T_o + PF \times \Delta T\ ;\ (5)$$

where T_o is the initial temperature and PF is the total peaking factor. In the equation just described, ρ_o is an input parameter and T_{peak} is the output, yet what is needed is the reverse; the object was to find the value of ρ_0 that yields T_{peak} = 1100°C. The object was attained by an iterative search. The search converged in no more than 3 iterations (estimates of ρ_o) because T_{peak} varies essentially linearly with ρ_o over a wide range.

The following input values were used for all the results displayed here:

β = 0.007;
l = 32 μs;
T_o = 20°C;
PF = 4.86.

Although some quantities, such as the peak reactor power, depend on the value of l, T_{peak} was found not to change with reasonable variations in l. The value of β is well known. The value used for T_o is the nominal zero-power temperature. The value of PF was selected to be 4.86 which is significantly larger than the 3.69 LCC is capable of producing. It is a conservative value for any other permissible core loading and rod bank position.

The reactivity insertion limit is shown for seven cases in Table 13-4. Each case has a different combination of fuel type and core-average burnup. The expressions for heat capacity, Cρ, as a function of temperature correspond to a minimum core size, 94 elements, and they were derived using the prescription in Reference 13.5. The insertion limit was found to be independent of Cρ. The insertion limit is sensitive to the prompt negative temperature coefficient, α. The curves in Figure 13.1 show that this coefficient varies with temperature, fuel type and fuel burnup. The different expressions for α are directly responsible for the differences in the reactivity insertion limit among the seven cases in Table 13-4.

TABLE 13-4 MAXIMUM REACTIVITY INSERTION AND RELATED QUANTITIES FOR VARIOUS FUELS AND BURNUPS

Fuel Type	Burnup (%)(MWD/rod)		Heat Capacity Cp (watt-sec/°C)	Prompt Negative Temperature Coefficient α (Δk/k°C)	Reactivity ρ_o ($)
20/20	0	0	7.12×10^4+143T	$4.91\times10^{-5}+1.93\times10^{-7}T-9.73\times10^{-11}T^2$	2.33
20/20	13	10	7.12×10^4+143T	$4.90\times10^{-5}+1.32\times10^{-7}T-7.82\times10^{-11}T^2$	2.16
20/20	33	27	7.12×10^4+143T	$5.24\times10^{-5}+7.45\times10^{-8}T-6.13\times10^{-11}T^2$	2.06
30/20	0	0	7.39×10^4+145T	$4.84\times10^{-5}+1.59\times10^{-7}T-7.3\times10^{-11}T^2$	2.23
30/20	15	20	7.39×10^4+145T	$4.71\times10^{-5}+9.13\times10^{-8}T-4.63\times10^{-11}T^2$	2.03
30/20	39	54	7.39×10^4+145T	$5.02\times10^{-5}+3.10\times10^{-8}T-2.24\times10^{-11}T^2$	1.92

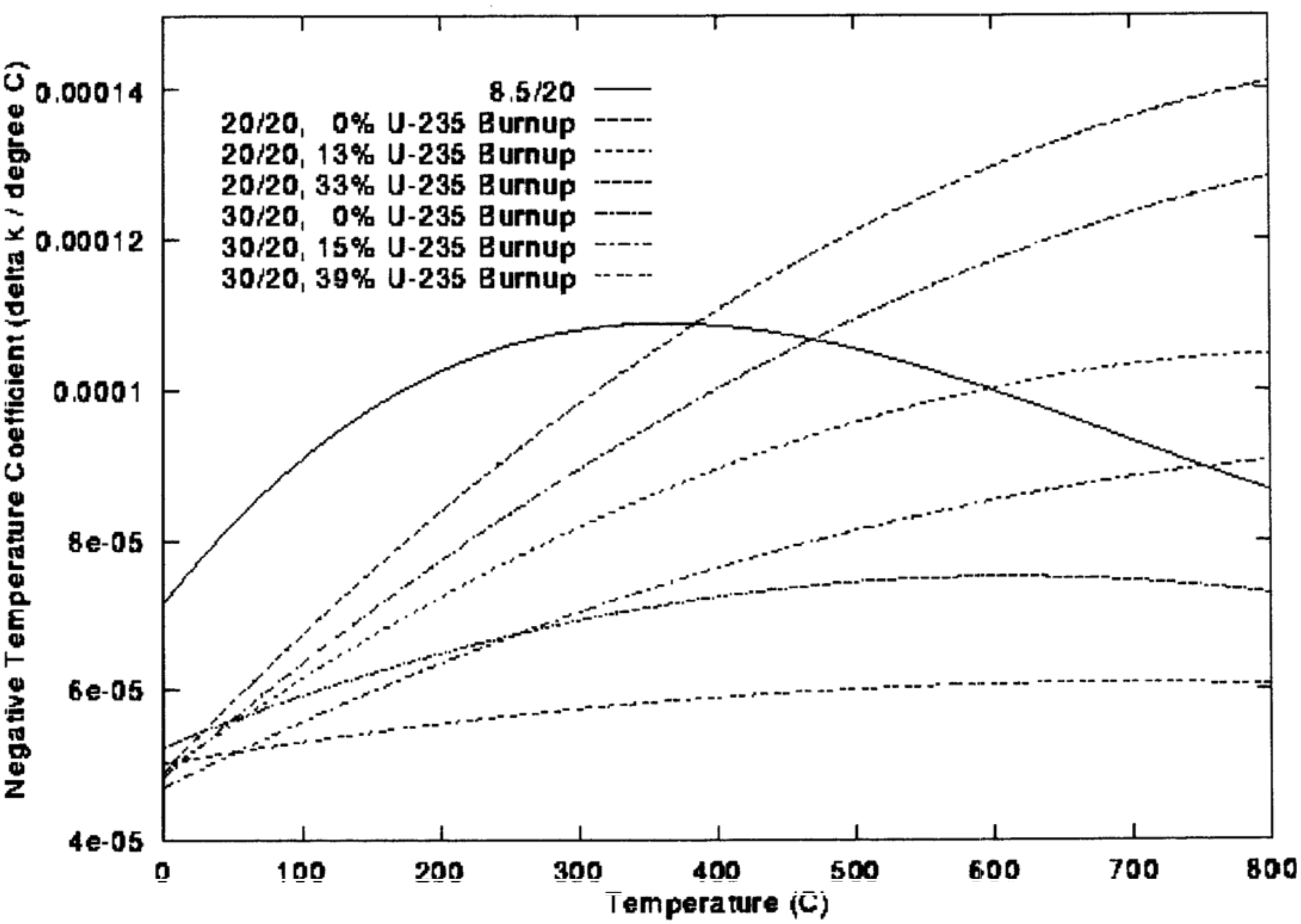

FIGURE 13.1 PROMPT NEGATIVE TEMPERATURE COEFFICIENT FOR TRIGA® FUELS

The worst-case result in Table 13-4, \$1.92, is considered as the maximum accidental reactivity insertion that could occur with no risk of fuel damage. There are at least two reasons why this is a conservative bound. One is that the core-average burnup, 39% ^{235}U, is greater than is likely to be achieved, which means that there will be more prompt feedback reactivity than was used for this case. The other reason is that the peaking factor is significantly larger than would be the actual case for a highly burned 30/20 loading.

13.2.2.2.2 Uncontrolled Withdrawal of a Control Rod

Operator error or failure of the automatic power level control system could cause one of the control rods to be driven out, starting at either high or low power levels. The maximum speed of a control rod is 1.78 cm/sec (42. in./min). The maximum single rod worth for the reference loadings of Section 4.5.5 is ~\$2.70, but a rod worth of \$3.00 for the 5 fuel followed control rod and \$2.50 for the transient rod was used here to allow for reasonable variations about the reference loadings. The reactivity worth was assumed to be linear along the length of the active 15 inches of their travel. Note the actual administratively limited rod speed is 24 in/min, the control rods are interlocked so the operator can only withdraw one at a time, and when in automatic power demand a maximum of 3 rods can be withdrawn at one time.

The initial reactor power levels of 100 W or 1.0 MW were analyzed using the single delayed neutron group model with the prompt jump approximation, a linear (ramp) reactivity increase results in the following equation for power as a function of time:

$$\frac{P(t)}{P_o} = \left(e^{-\lambda t}\right)[\beta/(\beta-\gamma t)]^{\left(1+\frac{\lambda\beta}{\gamma}\right)} \quad (6)$$

where: P(t) = power at time t
P_0 = initial power level
β= total delayed neutron fraction 0.0076
λ= one group decay constant = 0.405 (sec^{-1})
t = time (sec)
γ= linear insertion rate of reactivity (Δk/k-sec^{-1})

The SCRAM set point is 1.1 MW and a delay of 0.5 seconds is assumed between the set point being reached and the initiation of the controls dropping into the core.

Beginning at power level of 100 W a single rod being withdrawn until the SCRAM set point is reacted takes 6.82 seconds plus an additional 0.50 seconds for control drop to initiate. This corresponds to a reactivity insertion of \$1.025.

Beginning at power level of 1.0 MW a single rod being withdrawn until the SCRAM set point is reacted takes 0.60 seconds plus an additional 0.50 seconds for control drop to initiate. This corresponds to a reactivity insertion of \$0.154.

This reactivity insertion is much less than the limiting reactivity insertion derived in Section 13.2.2.2.1 for the pulse accident.

13.2.2.2.3 Uncontrolled Withdrawal of All Control Rods

Utilizing the same approach as in section 13.2.2.2.2 the consequences of the uncontrolled simultaneous withdrawal of all control rods is evaluated below. Since three control rods can be banked for reactor control, uncontrolled withdrawal of three control rods could be considered credible, but is bounded by the accidents analyzed.

Beginning at power level of 100 W a single rod being withdrawn until the SCRAM set point is reacted takes 1.23 seconds plus an additional 0.50 seconds for control drop to initiate. This corresponds to a reactivity insertion of $1.415.

Beginning at power level of 1.0 MW a single rod being withdrawn until the SCRAM set point is reacted takes 0.13 seconds plus an additional 0.50 seconds for control drop to initiate. This corresponds to a reactivity insertion of $0.515.

13.2.2.2.4 Beam Tube Flooding or Removal

In the event of flooding of one or more beam tubes, air or inert gas would be substituted with water. This will constitute a positive reactivity addition. It has been estimated that the worth of one flooded beam tube is about $0.25. This amount of excess reactivity is well below the limits discussed in Section 13.2.2.2.1; therefore, it does not represent a safety significant event.

During the removal of the in-tank section of a beam tube, air and graphite will be replaced by water because a portion of the graphite reflector is removed with this section of the beam tube. Again, replacement of the air/gas with water results in a positive increase in reactivity. On the other hand, replacement of graphite with water results in a negative effect on reactivity. The net result will be a smaller reactivity addition than for beam tube flooding so this action is of even less overall consequence.

13.2.2.2.5 Metal-water Reactions

Although metal-water reactions have occurred in some reactor accidents or destructive tests, the evidence from these events and laboratory experiments shows that a dispersed liquid metal is required for a violent chemical reaction to occur (References 13.1 and 13.7). The conditions for a solid metal-water reaction are not readily achievable in a reactor system such as the UCD/MNRC.

Water quench tests on TRIGA® fuel have been conducted to fuel temperatures as high as 1200°C without significant effect. Since the operating temperatures at 1 MW do not approach this temperature, this effect does not represent a safety risk. The only credible way in which temperatures high enough to allow metal-water reactions to be created in a TRIGA® reactor is through a large reactivity excursion. The limits set on excess reactivity preclude this.

13.2.3 Loss of Coolant Accident (LOCA)

13.2.3.1 Accident Initiating Events and Scenarios

Loss of coolant from the UCD/MNRC reactor could occur primarily through one of two scenarios, pumping water from the reactor tank or reactor tank failure. These scenarios are analyzed as part of this section.

13.2.3.2 Accident Analysis and Determination of Consequences.

13.2.3.2.1 Pumping of Water from the Reactor Tank

The intake for the primary-cooling-system pump is located about 3 ft below the normal tank water level. In addition, the line is perforated from about 8 in. below the normal tank water level to the intake line entrance. The intake for the purification-system pump is through a short flexible line attached to a skimmer that floats on the surface of the tank water. However, the length of the flexible line is such as to cause loss of pump suction if the tank water level is lowered about 4 ft. Thus, the reactor tank cannot be accidentally pumped dry by either the primary pump or the purification-system pump. Also, it is not possible for other cooling system or water cleanup system components to fail and syphon water from the tank since all of the primary-water-system and purification-system piping and components are located above the normal tank water level.

13.2.3.2.2 Reactor Tank Failure

A hole in or near the bottom of the reactor tank could cause the water level to drop below the top of the fuel elements. This event could occur either during reactor operation or while the reactor was shut down and unattended. There are no nozzles or other penetrations in the reactor tank below the normal water level, so the only mechanisms that could cause tank failure are corrosion of the tank or a mechanical failure. Leaks caused by corrosion would unquestionably be small leaks, which would be detected before the water level had lowered significantly. In such a case, makeup water could be supplied by the auxiliary make-up water system (AMUWS) until the leak is repaired.

Provisions to monitor for and collect tank leakage have been incorporated into the facility design. First, the tank is surrounded by corrugated metal. The corrugations provide a path to the bottom of the tank for any water leakage from the walls. Second, a drain, see Chapter 5, within the bulk shield surrounds the bottom of the tank. This drain will collect any water that may leak from the tank walls or bottom. Third, a duct leads from the drain to Radiography Bay 1 and the exit of this duct is periodically monitored for water leakage. If leakage is detected, the water could be easily collected at this point or diverted to the liquid holdup tank outside the building.

Consequences of a slow tank leak would be minimal and would require collection and containment of the water which leaked from the tank. This would be easily accomplished by using the existing liquid effluent control system described above. Small tank leaks due to corrosion are normally repairable using conventional techniques for patching aluminum, and thus it is expected that a leak could be located and fixed before there would be any significant loss of water from the tank.

An earthquake of much greater intensity than the Uniform Building Code Zone 3 earthquake appears to be the only credible mechanism for causing a large rupture in the tank, since the tank when supported by its associated biological shield structure was designed (with an importance factor of 1.5) to withstand this magnitude of earthquake. Even if such an event is assumed to cause very rapid loss of water while the reactor is operating at peak power; a reactor shutdown would be caused by voiding of water from the core, even if there was no scram.

A large rupture of the tank would obviously result in a more rapid loss of water than a leak due to corrosion or a minor mechanical failure in the tank wall. The UCD/MNRC reactor tank has no breaks in its structural integrity (i.e., there are no beam tube protrusions or other discontinuities in the reactor tank surface). In addition, the reactor core is below ground level. Thus the potential for most types of leaks is minimized.

Part of the historic 2 MW upgrade to the reactor included a new cavity (Bay 5) cut into the biological shield. This cut exposes the reactor tank wall below the reactor core level, and this introduces an increased possibility of draining water from the core area. While steps have been taken to minimize the probability of a tank rupture in this location, and it is believed that the likelihood of such a rupture is very low, an unplanned occurrence could nevertheless initiate such an event.

An analysis detailing the cooling capabilities required in the event of a complete LOCA. This analysis does not postulate the occurrence of a particular initiating sequence of events leading to all fuel elements in the core being uncovered. Instead, it simply assumes that the tank has ruptured and all the water is lost.

First there is the possibility of fuel clad rupture should the fuel temperature exceed design basis values. This event is covered in the analysis that follows, and focuses on the action of the EF-1 to prevent fuel temperatures from reaching safety limits. Second, there is a possibility of personnel exposure to radiation from the uncovered reactor core due to the direct beam from the core or from radiation scattered from the reactor room walls and ceiling. Finally, there is a chance that the lost water could cause ground water contamination. Both of these latter events are also analyzed as part of the LOCA evaluation.

13.2.3.2.2.1 Air Cooling

As with all TRIGA reactors operating under 1.5 MW (NUREG/CR-2387) no emergency core cooling is required in the event of instantaneous LOCA. The MNRC has not operated routinely above 1.0 MW for over a decade and was never operated 24/7. Therefore, the current fission product inventory in terms of decay heat after an instantaneous LOCA will be bound by the assumption the reactor operated at 1.0 MW the entire year before the LOCA event.

Unlike most other TRIGA reactors the MNRC has an unusually small reactor room due to its purpose built nature. Most other TRIGA reactors have large reactor halls. The consequence of this is that the volume of air in the reactor room acts as a substantial thermal buffer. In these cases, the initial increase in air temperature, from air cooling after a LOCA, is small and a typical industrial air conditioning system is likely to be able to remove more heat from the reactor room air than the core can add to the air after only one to two hours. During this period where decay out passes the cooling capacity of the air conditioner air temperature may only be raised a few tens of degrees. This would likely not be the case for MNRC.

The relatively small size (~7500 cu. ft.) of the reactor room will result in the air temperature increasing by over 100 degrees F (assuming no heat transfer to any other materials other than air in the reactor room) during the first hour of air cooling after the LOCA event. While the cooling capacity of the air conditioner will exceed the decay heat of the core after approximately one hour, it is unlike air conditioner will continue to operate with inlet air significantly hotter than what it was designed for. This leaves no ultimate heat sink (other than convection through the reactor room walls) for the decay heat of the core air temperature will continue to raise. Since this is the air that is available for cooling the core, this situation was analyzed in detail.

The air flow in the reactor room during normal operation is the following: an exhaust flow of 800 cfm passes through HEPA filters on the way to the stack, 500 cfm of which comes from the air conditioning system (1100 cfm outgoing, 1600 cfm returned) and 300 cfm comes from leaks into the reactor room from around doors or other leaks in the reactor room enclosure. Appendix D provides schematics of the reactor room and the exhaust and supply air ducts.

Although 1100 cfm is withdrawn from the room by the HVAC, and is refrigerated, and returned with an additional 500 cfm of air at ambient temperature, it will be assumed that during the LOCA event, this air flow continues but that the refrigeration fails due to an excessive heat load. (Note: If the HVAC fails, the reactor room exhaust fan will still be able to draw at least 500 cfm of ambient air in through the open HVAC damper.) Thus, 500 cfm (from the air conditioning) plus 300 cfm (from in-leakage into the reactor room) are continuously supplied to the reactor room at an ambient air temperature (~80°F) to match the 800 cfm exhaust that continues during the accident. This continuous flow of air provides the ultimate heat sink for the decay heat of the core.

13.2.3.2.2.2 Ground Water Contamination

As a result of activation of impurities in the primary cooling water, the water will contain small amounts of radionuclides depending on reactor power, reactor operating time and time since reactor shutdown. The equilibrium concentration of radioisotopes at 1 MW, are given in Table 13-5.

Next, a calculation was made to determine the length of time for the lost coolant to reach ground water. The relationship to determine the time (t) for water to move from a point under the reactor tank a distance, D, to ground water is:

$$t=D/(K \times I); \quad (7)$$

where:

- t = penetration time (sec.);
- D = depth of penetration with time (ft);
- I = hydraulic gradient = 1.0;
- K = hydraulic conductivity = 4.57×10^{-4} ft/sec (Reference 13.14).

If it is assumed that the ground water is 80 feet below the UCD/MNRC site, it would require more than 36 hours for it to be reached if the reactor tank containment were breached. The radionuclide concentrations present in the reactor tank water upon reaching the ground water were then calculated utilizing a 36 hour delay time. These values are presented in Table 13-5.

As shown, Aluminum-28, Magnesium-27, and Nitrogen-16 are gone by the time the tank water reaches the ground water, and most of the other radionuclides will have undergone some degree of decay during the first 36 hours. Decay will, of course, vary depending on the radionuclide, but Argon-41 activity would fall to less than 1.0×10^{-10} µCi/ml during the first 36 hours. Because of its low solubility in water, argon has no limiting water concentration under 10 CFR Part 20. However, this concentration level is well below the 10 CFR Part 20 air concentration limit for the unrestricted area. Since Argon-41 is only a concern from a dose standpoint when an individual is immersed in an Argon-41 cloud, and since the concentration in this situation is well below the air or cloud limit for the unrestricted area, Argon-41 is not a problem in the ground water.

The concentration of manganese-56, molybdenum-99, sodium-24 (after an additional 24 hours of decay), and cobalt-60 in the reactor primary water will be less than the 10 CFR part 20 effluence limits. This means that this accident would not result in any public health risks.

The estimated Hydrogen-3 (tritium) level is dependent upon how long the reactor has operated since initial startup and how much non-radioactive makeup water has been added prior to the LOCA. As shown in Table 11-4, the tritium concentration is measured to be at an equilibrium value of 2.0×10^{-2} µCi/ml for operating approximately 1,200 MW hours per year for nearly a decade. Likewise, the Co-60 concentration is measured to be at an equilibrium value of 1.0×10^{-6} µCi/ml for operating approximately 1,200 MW hours per year for nearly a decade. The tritium concentration in the water when it is released will be largely unchanged when and if the tank water reaches the ground water. Even so, the potential tritium dose to members of the general public who might consume the ground water will still be low because this accident will be a one-time event with a limited duration of release, and because only a limited amount of the 7,000 gallons of water potentially released from the reactor tank will likely escape from the radiography bays in the facility.

There will obviously also be a reduction in the tritium concentration when the reactor tank water mixes with the ground water. A dilution factor 20 is required to reduce the reduce the tritium concentrations to below the 10 CFR part 20 effluence limits. This level of dilution would easily be achieved as the closest well for drinking water is over one mile from the facility.

TABLE 13-5 PREDOMINANT RADIONUCLIDES IN PRIMARY COOLANT AT EQUILIBRIUM AND UPON REACHING GROUND WATER

Radionuclide	Half Life	Typical Concentration at 1MW (uCi/mL)	Concentration After 36 Hours of Decay (uCi/mL)	10 CFR Part 20 Effluent Limits (uCi/mL)
Aluminum-28	2.3 min	3.0×10^{-3}	0	n/a
Argon-41	1.8 hr	1.0×10^{-3}	$<1.0 \times 10^{-10}$	n/a
Hydrogen-3	12 yr	2.0×10^{-2}	2.0×10^{-2}	1.0×10^{-3}
Magnesium-27	9.46 min	2.0×10^{-4}	0	n/a
Molybdenum-99	2.75 d	1.0×10^{-5}	6.9×10^{-6}	2.0×10^{-5}
Cobalt-60	5.27 y	1.0×10^{-6}	1.0×10^{-6}	3.0×10^{-6}
Manganese-56	2.58 hr	5×10^{-5}	3.1×10^{-9}	7.0×10^{-5}
Sodium-24	14.96 hr	6.0×10^{-4}	1.1×10^{-4}	5×10^{-5}

13.2.3.2.2.3 Radiation Levels from the Uncovered Core

Even though there is a very remote possibility that the primary coolant and reactor shielding water will be totally lost, direct and scattered dose rates from an uncovered core following 1 MW operations have been calculated. Dose rates were calculated by constructing a simple, yet conservative, MNCP model of the MNRC reactor and facility. Details of this analysis are given in appendix B. The calculation of activity is given below. Dose rates were calculated for an individual directly over the core standing in the reactor room, inside the reactor room but not in direct line-of-sight of the core, just outside the reactor room, inside the control, at the MNRC fence line, in the closes building, and the closest in habited building.

Total Core Activity:

The total fission product activity as a function of time after shutdown was determined using the standard equation below (Reference B.16):

$$\text{Activity at time } \tau = 1.4\, P_o[(\tau - T_o)^{-0.2} - \tau^{-0.2}] \text{ Ci} \; ; (8)$$

where:

P_o = Thermal power (W);

τ = Time after reactor startup (d);

T_o = Time reactor operating at power P_o (d);

hence:

$\tau - T_o$ = Time after reactor shutdown (d).

The MNRC will operate a maximum 365 MWd per year; therefore, the operating profile used in the above equations was 8760 hours (365 days) at the full power of 1 MW. Increasing the operating time further makes little difference to the total fission product activity.

The resultant fission product activity and source term can be seen in Table 13-6.

Table 13-6 Total Fission Product Activity After Shutdown

Time After Shutdown	Total Activity (Ci)	Source Strength (γs^{-1})
10 seconds	8.15×10^6	3.02×10^{17}
1 hour	2.74×10^6	1.01×10^{17}
1 day	9.70×10^5	3.59×10^{16}
1 week	5.20×10^5	1.92×10^{16}
1 month	2.86×10^5	1.06×10^{16}

Dose Rate Directly Above the Core

The results are given in Table B-6 and are in general agreement with results for the 2 MW Torrey Pines TRIGA® Reactor (Reference B.18) scaled to 1.0 MW MNRC operations.

Table 13-7 Dose Rates Above the MNRC Reactor After a Loss of Pool Water Accident Following 1 MW Operations

Time After Shutdown	Effective Dose Equivalent Rate (rem/h)
10 seconds	1.53×10^4
1 hour	5.15×10^3
1 day	1.82×10^3
1 week	9.75×10^2
1 month	5.35×10^2

Dose Rate From Scattered Radiation in Reactor Room

The results below are given for a position located just inside the MNRC reactor room 2 meters from the edge of the reactor tank. This location was selected to be far enough away from the reactor tank so that there is no line-of-sight to the exposed core.

The results of the calculations for the scattered radiation dose rates can be seen in Table 13-8.

Table 13-8 Scattered Radiation Dose Rates in the MNRC Reactor Room After a Loss of Pool Water Accident Following 1 MW Operation

Time After Shutdown	Effective Dose Equivalent Rate (rem/h)
10 seconds	61.2
1 hour	20.6
1 day	7.3
1 week	3.9
1 month	2.1

The dose rates after a loss of pool water accident are provided in this section.

A position just outside (2 meters away) from the entrance to the reactor room is given to provide a worst-case estimate of personnel dose (Table 13-9) in the MNRC equipment room. Should this accident scenario occur personnel may need to enter these areas to replenish water to the reactor tank.

Table 13-9 Scattered Radiation Dose Rates in the MNRC Outside Reactor Room After a Loss of Pool Water Accident Following 1 MW Operation

Time After Shutdown	Effective Dose Equivalent Rate (rem/h)
10 seconds	1.66
1 hour	0.56
1 day	0.20
1 week	0.11
1 month	0.058

The MNRC control room is located at ground level approximately 14 m east of the core. The dose estimates below (Table 13-10) are conservative as no shielding credit is taken for the roof and walls of the reactor facility.

Table 13-10 Scattered Radiation Dose Rates in the MNRC Control Room After a Loss of Pool Water Accident Following 1 MW Operation

Time After Shutdown	Effective Dose Equivalent Rate (mrem/h)
10 seconds	370
1 hour	124
1 day	44
1 week	24
1 month	13

The shortest distance to the MNRC fence line is 28 meters. The dose estimate given below (Table 13-11) should be considered the highest possible doses possible for the public in this scenario.

Table 13-11 Scattered Radiation Dose Rates in the MNRC Fence Line After a Loss of Pool Water Accident Following 1 MW Operation

Time After Shutdown	Effective Dose Equivalent Rate (mrem/h)
10 seconds	320
1 hour	108
1 day	38
1 week	20
1 month	11

The closest building to the MNRC is an old industrial x-ray facility utilized by the Air Force. It is located 56 meters to the south of MNRC and has be uninhabited for nearly 20 years. No credit for building shielding is taken and the dose rates are presented below (Table 13-12).

Table 13-12 Scattered Radiation Dose Rates in Closest Public (not inhabited) Building After a Loss of Pool Water Accident Following 1 MW Operation

Time After Shutdown	Effective Dose Equivalent Rate (mrem/h)
10 seconds	152
1 hour	51
1 day	18

1 week	10
1 month	5

The closest inhabited building to the MNRC is a large conference center. It is located 94 meters to the north east of MNRC and is regularly used. No credit for building shielding is taken and the dose rates are presented below (Table 13-13).

Table 13-13 Scattered Radiation Dose Rates in Closest Public (inhabited) Building After a Loss of Pool Water Accident Following 1 MW Operation

Time After Shutdown	Effective Dose Equivalent Rate (mrem/h)
10 seconds	65
1 hour	22
1 day	8
1 week	4
1 month	2

13.2.4 Loss of Coolant Flow

13.2.4.1 Accident Initiating Events and Scenarios

Loss of coolant flow could occur due to failure of a key component in the reactor primary or secondary cooling system (e.g. a pump), loss of electrical power, or blockage of a coolant flow channel. Operator error could also cause loss of coolant flow.

Scenarios for loss of coolant flow events during operation are difficult to imagine since the bulk water temperature adiabatically increases at a rate of about 0.55°C/min at a power level of 1 MW. Under these conditions, the operator has ample time to reduce the power and place the heat-removal system into operation before any abnormal temperature is reached in the reactor water. A core inlet temperature alarm at 45°C and primary and secondary low flow alarms will alert the operator to an abnormal condition and should allow for corrective action prior to reaching the bulk water temperature limit.

13.2.4.2 Accident Analysis and Determination of Consequences

13.2.4.2.1 Loss of Coolant Flow Without Immediate Operator Action

If the reactor were operated without coolant flow for an extended period of time (and there was no heat removal by reactor coolant systems), voiding of the water in the core would occur and the water level in the tank would decrease because of evaporation. The sequence of events postulated for this very unlikely condition is as follows:

a) The reactor would continue to operate at a power of 1 MW (provided that the rods were adjusted to maintain power) and would heat the tank water at a rate of about 0.55°C/min until the water entering the core approached the saturation temperature (this would take 120 minutes, assuming an initial temperature near 35°C and adiabatic conditions). At this time, voids in the core would cause power oscillations and the negative void coefficient of reactivity would cause a reduction in power if control rods were not adjusted to maintain power;

b) If it is assumed that the operator or automatic control system maintained power at 1 MW, about 3180 kg/hr of water would be vaporized (assuming that the system is adiabatic except for the evaporation process), and the water level would decrease. It would take about 18 hours to heat and vaporize the entire tank at this rate. In fact, the reactor would shut down as the water level passed the top of the fuel.

It is considered inconceivable that such an operating condition would go undetected. Water level, water flow, and water temperature alarms would certainly alert the operator. Also, as the water level lowers, the reactor room radiation monitors will alarm. Because of all of these factors, water should be added to the tank to mitigate the problem.

13.2.4.3 Localized Loss of Coolant Flow

Blockage of the normal natural convection flow in an individual fuel coolant channel (from either above or below) could result in localized fuel overheating depending on the severity of the blockage. Foreign objects or debris (FOB) is tightly controlled in the reactor room. If FOB is dropped into the tank by someone in the reactor room the reactor operator's and the responsible individual in the reactor room immediate action is to SCRAM the reactor, in order to prevent the potential localized loss of coolant flow.

The other possible source of a significant localized loss of coolant flow if two elements began to simultaneously bow toward each other. It is possible for two fuel elements to touch each other and have both of them pass the 0.125 inch bowing criteria during a fuel inspection. Over the past 30 years MNRC has conducted approximately 1,000 fuel bowing examinations. No fuel elements have ever failed this criterion. Having two elements fail this criterion next to each other in opposing directions is not considered credible.

13.2.5 Mishandling or Malfunction of Fuel

13.2.5.1 Accident Initiating Events and Scenarios

Events which could cause accidents in this category at the UCD/MNRC reactor include 1) fuel handling accidents where an element is 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 fuel with subsequent cladding failure during steady state operations or pulsing; overheating might occur due to incorrect loading of fuel elements with different ^{235}U enrichments in a mixed core.

13.2.5.2 Accident Analysis and Determination of Consequences

13.2.5.2.1 Single Element Failure in Water

At some point in the lifetime of the UCD/MNRC reactor, used fuel within the core will be moved to new positions or removed from the core. Fuel elements are moved only during periods when the reactor is shut down. The most serious fuel-handling accident involves spent or used fuel that has been removed from the core and then dropped or otherwise damaged, causing a breach of the fuel element cladding and a release of fission products. As noted previously, the standard or accepted maximum hypothetical accident for TRIGA® reactors involves failure of the cladding of a single fuel element after extended reactor operations, followed by instantaneous release of the fission products directly into the air of the reactor room. A less severe, but more credible accident involving a single element cladding failure assumes the failure occurs underwater in the reactor tank 24 hours after reactor shutdown (i.e., 24 hours of decay has occurred). This accident has been analyzed in Appendix B and results in much lower doses to the public and the reactor staff than those estimated for the MHA. Overall the total amount of radioactive material released from this accident is 6-7 times less (from radioactive decay) than the MHA. It can be seen below that worker dose decrease roughly by the same magnitude relative to the MHA. This estimation of worker dose is further conservative as it assumes a release fraction base on the maximum element temperature while operating at 1 MW instead of being at ambient temperatures.

Table 13-14 Worker Dose for Single Element Failure in Water

	CDE Thyroid (millirem)	TEDE (millirem)
2 minute room occupancy	69	7.6
5 minute room occupancy	173	19.0

These doses are well below the 10 CFR 20 annual limits and should be used to inform worker decision if an accident such as this were to occur. For example, if fuel element leak were to be detected from the reactor control room during a fuel movement, the workers performing the fuel movement would have ample time (5-10 minutes) to place the element being moved in an approved location (inspection jig, back in the core, etc.) as opposed to simply dropping the element to the bottom of the tank in order to evacuate the reactor room as quickly as possible.

Offsite doses to the public are also expected to be 6-7 times less than the MHA. This would result in dose under the worst case condition of less than 10 mRem.

13.2.5.2.2 Fuel Loading Error

Under the current 1.0 MW core where only 20/20 and 30/20 elements are permitted there is no credible fuel loading error as the LCC in chapter 4 was established to intentionally produce the most peaking possible in a single element. In the LCC a fresh 30/20 element was place in the inner most fuel ring. The only fuel loading errors of some significance are considered not credible. A fuel element placed in the middle of the central irradiation facility may result in overheating of that element. This act would involve the removal of the aluminum slug normally located in that position and intentionally replacing it with a fuel element. When fuel movements occur at MNRC the aluminum slug is always in place making the accidental placement of a fuel element in the central irradiation facility impossible. The other non-credible fuel accident involves the placement of older low-burnup 8.5% fuel kept at MNRC inside the core. Once again the removal of an active fuel element and replacing it with an older 8.5 wt% element could not conceivably be done on accident. All 8.5 wt% elements look significantly different than the 20 and 30 wt% elements and are all kept in spent fuel storage.

13.2.6 Experiment Malfunction

13.2.6.1 Accident Initiating Events and Scenario

Improperly controlled experiments involving the UCD/MNRC reactor could potentially result in damage to the reactor, unnecessary radiation exposure to facility staff and members of the general public, and unnecessary releases of radioactivity into the unrestricted area. Mechanisms for these occurrences include the production of excess amounts of radionuclides with unexpected radiation levels, and creation of unplanned for pressures in irradiated materials which subsequently vent into reactor irradiation facilities or into the reactor building causing damage from the pressure release or an uncontrolled release of radioactivity. Other mechanisms for damage, such as corrosion and large reactivity changes, are also possible.

13.2.6.2 Accident Analysis and Determination of Consequences

Because of the potential for accidents which could damage the reactor if experiments are not properly controlled, there are strict procedural and regulatory requirements addressing experiment review and approval (Chapter 10). These requirements are focused on ensuring that experiments will not fail, but they also incorporate requirements to assure that there is no reactor damage and no radioactivity releases or radiation doses which exceed the limits of 10 CFR Part 20, should failure occur. For example, specific requirements in UCD/MNRC administrative procedures such as the Utilization of the University of California Davis/McClellan Nuclear Radiation Center Research Reactor Facility (MNRC- 0027) (Reference 11.7) establish detailed administrative procedures, technical requirements, and the need for safety reviews for all types of proposed reactor experiments.

Safety related reviews of proposed experiments usually require the performance of specific safety analyses of proposed activities to assess such things as generation of radionuclides and fission products (i.e., radioiodines), and to ensure evaluation of reactivity worth, chemical and physical characteristics of materials under irradiation, corrosive and explosive characteristics of materials, and the need for encapsulation. This process is an important step in ensuring the safety of reactor experiments and has been successfully used for many years at research reactors to help assure the safety of experiments placed in these reactors. Therefore, the process is expected to be an effective measure in assuring experiment safety at the UCD/MNRC reactor.

A specific limitation of less than $1.00 on the reactivity of individual moveable experiments placed in the reactor tank has been established and is safe because analysis has shown that pulse reactivity insertions of $1.75 in the UCD/MNRC reactor result in fuel temperatures which are well below the fuel temperature safety limit of 930°C (Section 13.2.2). In addition, limiting the worth of each moveable experiment to less than $1.00 will assure that the additional increase in transient power and temperature will be slow enough so that the fuel temperature scram will be effective. Likewise, an additional reactivity limitation of less than $1.75 for any single secured experiment and an absolute total reactivity worth of $1.92, including the potential reactivity which might result from malfunction, flooding or voiding, is safe because Section 13.2.2 shows that a maximum accidental reactivity of insertion of $1.92 will not result in fuel damage.

Limiting the generation of certain radionuclides in experiments and certain fission products in fueled experiments also helps to assure that occupational radiation doses (as well as doses to the general public) due to postulated experiment failure, with subsequent radionuclide or fission product release, will be within the limits prescribed by 10 CFR 20. A limit of 1.5 curies of I-131 through I-135 for a single fueled experiment is small compared to the approximately 8,500 curies of I-131 through I-135 which are present in the single fuel element failure analyzed in Section 13.2.1 (failure in air) and Section 13.2.5 (failure in water). In both cases, the occupational doses and the doses to the general public in the unrestricted area due to radioiodine are within 10 CFR 20 limits. Therefore, establishing conservative limits for radioiodine in experiments will result in projected doses well within 10 CFR 20 limits. Strontium-90 in a fueled experiment is limited to 0.005 curies which is far below the 34 curies present in the single fuel element failures mentioned above. Since no dose limits will be exceeded in the single element failure accidents, doses from experiments where the Strontium-90 is limited to 0.005 curies are expected to be safely within 10 CFR 20 limits.

Safety analyses have been performed which show that three pounds of TNT equivalent explosives may be safely irradiated in radiography Bays 1, 2, 3 and 4, provided the beam tube cover plates are at least 0.5 inch thick (Reference 13.16).

Southwest Research Institute (SRI) completed a safety analysis to determine the maximum amount of TNT equivalent explosive allowable in radiography Bay 3, (i.e., the amount that will not cause failure of the beam tube cover plate and will cause only repairable structural damage to the bay) (Reference 13.17). Bay 3 is the smallest in volume of all the radiography bays at the UCD/MNRC. The study concluded that Bay 3 can withstand a detonation of 6 pounds of TNT equivalent explosive with certain modifications. The study performed by SRI concluded that the Bay 3 door track must be strengthened. The recommended strengthening consists of welding three additional anchor bolt plates to the door track and bolting these plates into the wall with additional drilled anchor bolts. This strengthening assures that the door will respond in a ductile manner to an unexpected high blast load, absorbing the additional load with larger deflections, rather than responding in a brittle failure mode.

The UCD/MNRC completed a similar study to determine the maximum amount of TNT equivalent explosives allowable in all radiography bays (Reference 13.16). This study concluded that Bays 1, 2 and 4 can withstand a detonation of 6 pounds of TNT equivalent explosives without any damage provided the criteria in Table 13-15 are implemented in each bay. However, to meet category 1 protection requirements for 6 pounds of explosives, the west door of Bay 2 also requires modification by means of an additional wheel and post assembly. The analysis performed by the UCD/MNRC demonstrates that for 3 pounds of TNT equivalent explosives, no modifications are necessary to the radiography bay doors for Bays 1, 2 or 4. These doors will also respond in a ductile manner. As a result of the above studies, it is concluded that installation of beam tube cover plates with the thicknesses shown in Table 13-15 and implementing an explosives limitation of 3 pounds of TNT equivalent for each of the four radiography bays will satisfy the safety limitations established by the two previous safety analyses.

Table 13-15 Changes to Beam Tube Cover Plates

BAY	Cover Plate Thickness (in)	Explosive Location	Explosive distance* ref.@ 0°	Deflection (in)	Resistance (psi)	Ultimate Resistance (psi)
1	0.60	L/D = 100	13.00	0.294	43.6	108
2	0.60	L/D = 100	10.40	0.353	52.3	108
3	0.75	L/D = 100	8.80	0.433	125	168
4	0.60	L/D= 100	13.70	0.248	37	108

*Minimum distance from the beam tube cover plate to the explosive.

The Argon-41 Production Facility (see Chapter 10) can produce argon-41 in excess of the amounts analyzed in Appendix A. However, if the system releases argon-41, the gas will be contained in the reactor room and the existing reactor room ventilation system will be used in recirculation mode to prevent the release of argon-41 to the environment by recirculating the gas until it decays. The existing stack continuous air monitor will also be used to verify that none has been released outside the UCD/MNRC boundary.

If the system had a catastrophic failure and 4 Curies of argon-41 were released to the volume of the reactor room, the argon-41 concentration in the reactor room would be 2×10^{-2}µCi/ml and the gamma dose rate in the reactor room would be approximately 22 R/hr (based on a semi-infinite cloud, see following calculation). Though the release of 4 Ci of Ar-41 is indeed a significant release of radioactive material is several times lower than the amount of Ar-41 routinely made and released in the course of a year of single shift operations at MNRC. A 2-minute evacuation time would result in a whole body dose of 733 mrem. Actual dose would like be significantly less as the semi-infinite cloud model overestimates dose due to the small reactor room dimensions. Personnel would be evacuated from the reactor room and access would be restricted. The reactor room ventilation system (as described in Chapter 9) would be operated in the recirculation mode for approximately one day before the dose rate from argon-41 decays to less than 1 mR/hr. Therefore, the argon-41 discharge limit defined in the UCD/MNRC Technical Specifications will not be exceeded due to the recirculation mode of the reactor room ventilation system.

13.2.7 Loss of Normal Electrical Power

13.2.7.1 Accident Initiating Events and Scenarios

Loss of electrical power to the UCD/MNRC could occur due to many events and scenarios which routinely affect commercial power.

13.2.7.2 Accident Analysis and Determination of Consequences

Since the UCD/MNRC does not require emergency backup power systems (see Chapter 6) to safely maintain core cooling, there are no credible reactor accidents associated with the loss of electrical power. A backup power system is present at the UCD/MNRC which mainly provides conditioned power to the reactor console and control instrumentation. Therefore, the reactor will not automatically scram when there is a loss of normal electrical power. In fact, the backup power system is capable of providing electrical power for the reactor control and various operational measurements for a period of time after loss of normal electrical power and until its batter power supply is exhausted.

Loss of normal electrical power during operations is addressed in the reactor operating procedures, which require that upon loss of normal power an orderly shutdown is to be initiated by the operator on duty. The battery backup power will allow monitoring of the orderly shutdown of the reactor and verification of the reactors shutdown condition.

13.2.8 External Events

13.2.8.1 Accident Initiating Events and Scenarios

Hurricanes, tornadoes and floods are virtually nonexistent in the area around the UCD/MNRC reactor. Therefore, these events are not considered to be viable causes of accidents for the reactor facility. In addition, seismic activity in Sacramento is low relative to other areas of California (Chapter 2). Seismic activity has already been mentioned in connection with the postulated reactor tank damage 13.2.3.

The UCD/MNRC facility is surrounded by a security fence and a physical security plan is continuously in force for personnel and activities inside the fence. The reactor site is located in an Industrial Park on a former US Air Force Base where access and overall security is far stricter than the surrounding civilian business and residential areas. Therefore, accidents caused by human controlled events which would damage the reactor, such as explosions of other unusual actions, are considered to be of very low probability.

Since the UCD/MNRC reactor is located at the edge of the runway at the former McClellan AFB, airplane crashes involving the reactor may potentially cause reactor damage.

13.2.8.2 Accident Analysis and Determination of Consequences

A study of the probability of aircraft crashes which could cause reactor damage at the UCD/MNRC was conducted by GA Technologies as a part of the original Stationary Neutron Radiography System Proposal (Reference 13.19). The conclusions show that the calculated reactor damage probability due to aircraft accidents is 5×10^{-8} per reactor year. This value was obtained using conservative assumptions and the “best estimate” value is expected to be considerably lower than 5×10^{-8}. As can be seen in chapter 2, the total number of aircraft operations is down approximately an order of magnitude from when the original analysis was performed. Safety analysis of nuclear power reactors have generally concluded that a reactor damage probability due to an aircraft accident which is less than 1×10^{-7} per year does not represent a significant contribution to the overall reactor risk. Therefore, it is concluded that no specific aircraft accident and no radiological consequences need to be considered for the UCD/MNRC reactor.

13.2.9 Mishandling of Malfunction of Equipment

13.2.9.1 Accident Initiating Events and Scenarios

No credible accident initiating events were identified for this accident class. Situations involving an operator error at the reactor controls, a malfunction or loss of safety related instruments or controls and an electrical fault in the control rod system were anticipated at the reactor design stage. As a result, many safety features, such as control system interlocks and automatic reactor shutdown circuits, were designed into the overall TRIGA® Control System (Chapter 7).

TRIGA® fuel also incorporates a number of safety features (Chapter 4) which together with the features designed into the control system assured safe reactor response, including in some cases reactor shutdown.

Malfunction of confinement or containment systems would have the greatest impact during the maximum hypothetical accident (MHA), if they were used to lessen the impact of such an accident. However, as shown in section 13.2.1, no credit is taken for confinement or containment systems in the analysis of the MHA for the UCD/MNRC reactor. Furthermore, no safety consideration at the UCD/MNRC depend on confinement or containment systems, although simple confinement devices like a fume hood might be used as part of normal operations.

Rapid leaks of liquids have been previously addressed in Section 13.2.3. Although no damage to the reactor occurs as a result of these leaks, the details of the analysis provide a more comprehensive explanation.

13.3 Summary and Conclusions

Chapter 13 of the Safety Analysis Report contains a conservative analysis of many different types of hypothetical accidents as they relate to the UCD/MNRC reactor and the surrounding environment. Beginning with the maximum hypothetical accident and continuing on through an entire array of other accidents, it has been shown that the consequences of such accidents will not result in occupational radiation exposure of the UCD/MNRC staff or radiation exposure of the general public in excess of applicable NRC limits in 10 CFR Part 20. Furthermore, there is no projected significant damage to the reactor as an outcome of the accidents evaluated, except the damage or malfunction assumed as part of the different accident scenarios analyzed. Details of the assumptions used for each accident scenario and the specific consequences of each accident are presented in the text of this Chapter.

CHAPTER 14

Technical Specifications

14 Technical Specifications:

MNRC Technical Specifications are provided in a separate document.

CHAPTER 15

Financial Qualifications

15 Financial Qualifications:

MNRC financial qualifications are provided in a separate document.

CHAPTER 16

Other License Considerations

16 Other License Considerations:

16.1 Prior Use of Reactor Components

MNRC does not use any equipment previously utilized at other reactor facilities.

16.2 Medical Use of Non-Power Reactors

MNRC does not engage in nor is it licensed to engage in any activities for medical use of the facility.

APPENDIX A

RADIOLOGICAL IMPACT OF Ar-41, N-16, FISSION PRODUCTS AND ACTIVATED MATERIALS DURING NORMAL OPERATIONS

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APPENDIX A

RADIOLOGICAL IMPACT OF Ar-41, N-16, FISSION PRODUCTS, AND ACTIVATED MATERIALS DURING NORMAL OPERATIONS

A.1. INTRODUCTION

A.1.1 Purpose

The purpose of this appendix is to show the methods and calculations that were used to evaluate the production and concentrations and dose rates from Ar-41, N-16, and fission and activation products as a result of normal MNRC operation.

Argon-41 is produced by the reaction of thermal neutrons with the argon contained in air (≈1%) entrained in the reactor cooling water as it passes through the core and the air in the path of the radiography beams. This Ar-41 ends up in the reactor room and radiography bays and is subsequently released to the atmosphere through the facility exhaust stack.

Nitrogen-16 is produced by the reaction of fast neutrons with oxygen. The only N-16 source in the MNRC facility that needs consideration results from neutron interactions with the oxygen in the water molecule of the reactor cooling water as it passes through the core. The production of N-16 as a result of the reaction of neutrons with oxygen present in air, either in the beam path or air entrained in the reactor cooling water, is insignificant and has been neglected.

A portion of the N-16 produced in the core is eventually released from the top of the reactor tank into the reactor room. The half-life of N-16 is only 7.14 seconds so its radiological consequences outside the MNRC are insignificant.

Although not expected, the cladding on a fuel element could fail during normal operation as a result of corrosion or a manufacturing defect. Should a failure occur, a fraction of the fission products would be released to the reactor tank. Most of the halogens would remain in the cooling water while the noble gases, krypton and xenon, would be released to the reactor room and subsequently to the atmosphere through the exhaust stack. Although this operational occurrence is mentioned in this appendix, it is addressed in detail in Appendix B as part of the analysis of a single fuel element failure in water.

There will be a varying amount of neutron activation products generated due to neutron interactions with materials being intentionally irradiated in reactor irradiation facilities. Normally this will be fixed radioactivity and mainly a source of direct radiation to operations personnel.

A.1.2 Radiological Standards

In 1994, the U.S. Nuclear Regulatory Commission (NRC) implemented a major revision in 10 CFR 20 which incorporated many of the new dosimetry concepts published by the ICRP over the past several years. Since the new version of 10 CFR Part 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 will be based on the requirements of the new 10 CFR Part 20.

At the MNRC a cloud of finite size is more applicable, because room sizes are not large enough to meet the required dimensions for an infinite cloud for the noble gas routinely present, i.e., Ar-41. Because of this, the total effective dose equivalent at the concentration limit is correspondingly lower.

The current 10 CFR Part 20 concentration limits for Ar-41 are:

- For accessible regions inside the operations boundary 3 x 10^{-6} µCi/ml (1 DAC);
- For accessible regions outside the operations boundary 1 x 10^{-8} µCi/ml.

A.2 PRODUCTION RATE AND CONCENTRATION OF Ar-41 FROM BEAMS

Given the low neutron beam intensity in the neutron radiography bays the amount of Ar-41 produced is difficult to accurately measure. Therefore, the saturation Ar-41 activity is calculated by taking the average volume of each of the 4 MNRC neutron beamline and using the average beamline neutron radiography assumed to be at the MNRC radiography plane. The results of these analyses are given below.

Table A-1: MNRC Neutron Beamline Characteristics

	Equivalent Diameter (cm) @ Film Plane	Area cm^2	Length* cm	Beam Volume cm^3	Effective Target Mass of Argon (g)
Radiography Bay 1	58.4	2680	941	2.52 x 10^6	28.3
Radiography Bay 2	58.4	2680	762	2.04 x 10^6	23.0
Radiography Bay 3	25.4	510	581	2.96 x 10^5	3.3
Radiography Bay 4	58.4	2680	762	2.04 x 10^6	23.0

Table A-2: Resulting Ar-41 Production in MNRC Neutron Beamlines

	Flux (n/cm^2 s)	Ar-41 produced (uCi/s)	Bay Volume (ml)	Maximum Ar-41 concentration (uCi/ml)
Radiography Bay 1	3.0×10^6	2.39×10^{-3}	2.09×10^9	1.08×10^{-8}
Radiography Bay 2	3.0×10^6	1.94×10^{-3}	1.11×10^9	1.67×10^{-8}
Radiography Bay 3	4.3×10^6	4.00×10^{-4}	1.7×10^8	2.24×10^{-8}
Radiography Bay 4	3.0×10^5	1.94×10^{-4}	4.63×10^8	4.00×10^{-9}

The results above show that Ar-41 production and maximum concentrations show the radiological consequences of Ar-41 production in the MNRC beamline are very modest. The maximum concentration of Ar-41 on the radiography bays is over 100 times less than the DAC value for Ar-41. In reality the concentration in these radiography bays will be significantly less due to the air changes in the room and the fact the beamlines are not “on” all of the time the reactor is operating. These low Ar-41 levels are further confirmed by the fact that neutron radiographer annual whole body dose is essentially zero despite spending hundreds of hours per year in the radiography bays.

Table A-3: MNRC Neutron Beamline Air Change Rates

	cm^3/s	
Radiography Bay 1	7.88×10^5	1.35 air changes per hour
Radiography Bay 2	4.72×10^5	1.53 air changes per hour
Radiography Bay 3	2.36×10^5	5.00 air changes per hour
Radiography Bay 4	2.83×10^5	2.20 air changes per hour
Reactor Room	3.78×10^5	6.50 air changes per hour

A.3 PRODUCTION RATE OF Ar-41 FROM COOLANT WATER

Though the concentration of Ar-41 in the reactor primary water and in the air in the reactor room can be calculated, it is more prudent to provide measured Ar-41 concentrations.

Actual measurements of Ar-41 in the reactor room after the reactor had operated for about 9 hours at 1 MW (reactor room exhaust system on) showed Ar-41 concentrations averaging about 1.5×10^{-6} µCi/ml for areas which are occupied during normal work in the room. This provides strong indication that the vast majority of the Ar-41 effluences to the environment is from Ar-41 activation in the primary coolant and not the neutron beamlines. Using the semi-infinite cloud, the corresponding dose would be 1.25 mRem/hr. However, given the small size of the reactor room the semi-infinite cloud model will greatly over predict dose. In actuality the Ar-41 concentration in the reactor room contributes only slightly to the typical measured reactor room dose rate of 3-4 mrem/hr.

Actual measurements of Ar-41 in the primary coolant water (after 4 MW hours of operation) average 1.0×10^{-3} µCi/ml.

A.4. MAXIMUM IMPACT OF Ar-41 OUTSIDE THE OPERATIONS BOUNDARY

The Ar-41 will be discharged from the MNRC through the exhaust stack, which is 60 feet above ground level. Dilution with other building ventilation air and atmospheric dilution will reduce the Ar-41 concentration considerably before the exhaust plume returns to ground level locations which could be occupied by personnel or the general public.

It is important to note that only a modest amount of dilution is required to reduce the Ar-41 concentration to a level that is well below the 10 CFR Part 20 limit of 1×10^{-8} µCi/ml for unrestricted areas. Based on 2019 effluence data the MNRC operated for 1,430 hours at 1 MW and produced 27.6 Ci of Ar-41. This corresponds to an emission rate of 5.4×10^{-6} Ci/s. Based on a typical stack flow rate of 5678 CFM a concentration of 2.0×10^{-6} µCi/ml will be effluence during one MW operations. Though it is very unlikely MNRC will be operated again 24/7, the subsequent calculation will be based on a continuous effluence rate of 2.0×10^{-6} µCi/ml.

The radiography bay ventilation system provides both a significant dilution effect and increases the effective stack height by increasing the effluence exiting velocity. Both of which are taken credit for in this analysis. Therefore, the radiography bay ventilation system must be operated on a regular basis while the reactor is on. However, the radiography bay ventilation system does not need to be operated when the reactor is shutdown when no Ar-41 is being produced.

The Lawrence Livermore National Laboratory program “HotSpot” was used to determine the worst case radiation dose impact to the public for a variety of atmospheric stability classes. Over the past several years the program has become the industry standard for relatively simple Gaussian plume modeling. The program was also used to provide the down field centerline maximum concentration position. Based on this distance the maximum concentration at ground level was calculated using the established Gaussian plume model equation found below. Dispersion coefficients were calculated based on the equations provided in the HotSpot user manual. Ground level maximum concentrations were selected as the most appropriate to evaluate compliance with 10 CFR 20 appendix B effluence limit concentration of Ar-41 as the prevailing wind direction will take MNRC effluence the majority of the time (chapter 2) into an area of the McClellan air field that is flat, controlled access, and largely free of any elevated buildings.

For the modeling in HotSpot the following inputs and assumptions were made:

-Average wind speed is assumed to be 3.4 m/s (chapter 2).
-MNRC Stack height is 18.2 m.
-Emission rate of Ar-41 of 5.4 x 10^{-6} Ci/s.
-Effluence exit velocity of 16 m/s based on 5678 cfm effluence rate and stack exit diameter of 0.5 meters which produces an effective stack height of 25 m in all scenarios.
-The "sample time" was kept at the default value of 10 minutes.
-Standard/rural terrain was used (conservative).

Hotspot unfortunately does not provide maximum centerline effluence concentration which is a value of concern. Hotspot was used to determine the ground level downwind distance where the maximum concentration is expected to occur. Then the following established Gaussian Plume model equation was used to determine the maximum Ar-41 concentration in order to compare the results to regulatory limit of 1 x 10^{-8} uCi/ml. Sigma y and sigma z values were determined based on the formulas provided in the HopSpot user manual.

$$\chi_{(max,0)} = \frac{Q}{\pi u \sigma_z \sigma_y} exp\left(-\frac{1}{2}\frac{H^2}{\sigma_z^2}\right) = \frac{Ci}{m^3} = \frac{\mu Ci}{cm^3} \;;$$

where:

Q = Emission Rate
H = 25m effective stack height (18.2 m physical stack height);
u = Mean Wind Speed (m/s);
σ_y = diffusion coefficient in the y-axis
σ_z = diffusion coefficient in the z-axis

Table A-4: MNRC Offsite Ar-41 Radiological Consequence for Continuous 1 MW Operations with Plume Rise

Atmospheric Stability Classification	*Sigma y* (m)	Sigma z (m)	χ_{max} (μCi/cm^3)	Distance to Max Dose (m)	Maximum Dose per Year (mrem)
Very Unstable (A)	19.3	17.6	5.4×10^{-10}	88	3.7
Moderately Unstable (B)	23.8	18	4.5×10^{-10}	150	3.1
Slightly Unstable (C)	23.9	17.3	4.3×10^{-10}	220	2.9
Neutral (D)	27.1	16.6	3.6×10^{-10}	340	2.4
Slightly Stable (E)	40.2	17.1	2.5×10^{-10}	690	1.2
Moderately Stable (F)	52.3	15.8	1.7×10^{-10}	1400	0.8

The results given above show that even under 24/7 operations at the maximum licensed steady state power Ar-41 effluence will not exceed 10 mRem per year for the maximum exposed individual located at the highest concentration year round. The most likely average annual atmospheric condition is class "C" slightly unstable which results in a maximum dose of 2.9 mrem per year which corresponds to a delusion factor of 4640 from the stack effluence to the maximum ground level concentration. The worst case atmospheric condition is class "A" very unstable which results in a maximum dose of 3.7 mrem per year which corresponds to a dilution factor of 3703 from the stack effluence to the maximum ground level concentration.

The results in table A-4 are given to demonstrate that the physical height of the stack alone (no plume rise) is sufficient to result in enough dilution at ground level to remain under the 10 mrem effluence limit. Furthermore, under the most unfavorable condition (i.e. class A stability and no plume rise) does the Ar-41 concentration result in concentration above 1.0×10^{-9} uCi/ml which is 1/10th of the 10 CFR 20 appendix B limit.

A.5. Good Engineering Practice Considerations

The MNRC was constructed in the late 1980s by the US Air Force. Though no specific documentation exists, it appears that the *Guideline for Determining of Good Engineering Practice Stack Height* (an EPA guidance document) was followed. The two major concerns are the MNRC's close proximity to a large building to the south and the low height of the MNRC stack relative to the reactor room. The concern is that either of these condition may result in routinely occurring excessive building wakes that will result in a significant underestimation of the effects of MNRC Ar-41 effluence.

The building to the south of MNRC stands approximately 36 feet in height and is slightly wider than it is tall. The facility was originally used to perform neutron radiography on the F-117 using a large Cf-252 source. The building is essentially upwind of the prevailing wind encountered at MNRC (figure A-1).

Figure A-1 Overhead View of MNRC and Adjacent Facility of Concern

The good engineering practice calls for a stack to be located at a distance of at least 5 time the height of the nearby building of concern. Based on a google map measurement the close edge of the neighboring building is located 197 feet from the MNRC stack. This distance is just slightly larger than 5 times the height (36 feet) of the building of concern of 180 feet. Though some degree of a wake effect may still take place because this criterion only just met, it is likely to be relatively small.

The other criterion of concern is that the MNRC effective stack height may not be 2.5 times the height of the nearest building. In the case of the MNRC the nearest building is the reactor room itself, to which the stack is physically mounted to. Given the fact that the MNRC is stack is only 18.2 m in height it is considered “de minimis” and the actual stack height should be used in calculating emission limitations. Therefore, the dose to the maximum exposed individual was calculated again taking no credit for plume rise. In this case 24/7 1 MW operations do not result in offsite doses in excess of 10 mRem/hr.

Table A-5: MNRC Offsite Ar-41 Radiological Consequence for Continuous 1 MW Operations without Plume Rise

Atmospheric Stability Classification	$Sigma\ y$ (m)	Sigma z (m)	χ_{max} (µCi/cm^3)	Distance to Max Dose (m)	Maximum Dose per Year (mrem)
Very Unstable (A)	13.8	12.6	1.0×10^{-9}	63	7.2
Moderately Unstable (B)	15.9	12	8.4×10^{-10}	100	5.9
Slightly Unstable (C)	17.5	12.6	8.1×10^{-10}	160	5.6
Neutral (D)	19.0	12.3	7.2×10^{-10}	240	4.9
Slightly Stable (E)	27.0	12.1	5.0×10^{-10}	460	3.0
Moderately Stable (F)	36.3	11.8	3.6×10^{-10}	950	1.9

APPENDIX B

RADIOLOGICAL IMPACT OF ACCIDENTS

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APPENDIX B

RADIOLOGICAL IMPACT OF ACCIDENTS

B.1 Maximum Hypothetical Accident (MHA)

Numerous safety committees that have reviewed TRIGA® reactor operations have considered potential accidents including rapid insertion of reactivity, loss of heat removal, loss of coolant, metal-water reactions, rearrangement of fuel, fuel aging, and accidents during handling of irradiated fuel. Chapter 13 of this document discusses such accidents. This appendix addresses the consequences of the accepted maximum hypothetical accident (previously called the design basis accident) for a TRIGA® reactor: a cladding rupture of one fuel element with no decay and subsequent instantaneous release of fission products into the air. This is commonly referred to as a single element failure in air. This is also the most severe of all accidents for TRIGA® reactors and is analyzed to examine potential radiation doses to the reactor staff and the general public. A less severe, but more credible accident involving a single element cladding failure in water during 1 MW operation and 24 hours after the reactor has shut down after extended 1.0 MW operation.

At some point in the lifetime of the MNRC reactor, used fuel within the core may be moved to new positions or removed from the core. Fuel elements are moved only during periods when the reactor is shut down. The most serious fuel-handling accidents involve spent or used fuel that has been removed from the core and then dropped or otherwise damaged, causing a breach of the fuel element cladding and a release of fission products. As noted previously, the standard or accepted maximum hypothetical accident for TRIGA® reactors involves failure of the cladding of a single fuel element after extended reactor operations, no time to decay, and subsequent release of the fission products directly into the air of the reactor room. While a credible scenario for this accident is hard to establish, it will be assumed that such an event can take place and does so immediately after reactor operation with a fuel element that has been run at full power (1 MW) for a period of 1 year. This operating history is very conservative in nature as it is unlikely the facility is ever operated more than 2,000 MWhrs in a year for the remainder of the facility’s life. The intent of the fuel element failure in water during 1 MW (maximum steady-state power) accident to provide a better understanding of the radiological consequences of a more realistic accident than the MHA. The intent of the fuel element failure in water 24 hours after shutdown is to provide a better understanding of the radiological consequences of damaging a fuel element while it is being handles remotely in the reactor tank.

The fission product inventory at shutdown is listed in Table B-1. The data are for the volatile fission products contained in a fuel element run to saturation at the highest core power density.

For both accidents being analyzed in this appendix, a release fraction of 2.4×10^{-5} is assumed for the release of noble gases and halogens from the fuel to the cladding gap. This release fraction is developed in Chapter 4 and is based on the maximum measured fuel temperature (400 C) which highest average fuel temperature expected operating at 1 MW. This temperature is likely an overestimate given the measured (via instrument fuel element) of the hottest element is closer to 330 C. This thermal output is the considered to be the maximum that can be achieved by the LCC.

In addition, for the accident where the cladding failure occurs in air, it is very conservatively assumed that 25% of the halogens released to the cladding gap are eventually available for release from the reactor room to the outside environment. A release fraction of 2.4×10^{-5} was also assumed for a single fuel element failure 24 hours after shutdown, even though the element would be at ambient temperatures, in order to keep the radiological consequences conservative.

Table B-1 Source Terms for One-Element Accident for MHA

Group I		Group II	
Nuclide	Activity (Ci)	Nuclide	Activity (Ci)
Br-83	113.4	Kr-83M	113.4
Br-84	208.4	Kr-85M	272.9
Br-84M	35.3	Kr-85	3.7
Br-85	272.9	Kr-87	541.6
I-131	611.4	Kr-88	751.0
I-132	911.8	Kr-89	954.1
I-133	1417.4	Xe-131M	8.6
I-134	1656.5	Xe-133M	40.0
I-134m	77.0	Xe-133	1417.4
I-135	1328.6	Xe-135M	232.7
		Xe-135	1383.6
		Xe-137	1296.8
		Xe-138	1332.8
TOTAL	**6632.7**		**8348.6**

Table B-2 Source Terms for One-Element Accident for 24-hour decay

Group I		Group II	
Nuclide	Activity (Ci)	Nuclide	Activity (Ci)
Br-83	5.1e-3	Kr-83M	2.2e-4
Br-84	0	Kr-85M	1.29
Br-84M	0	Kr-85	3.7
Br-85	0	Kr-87	0
I-131	540	Kr-88	0.16
I-132	0.024	Kr-89	0
I-133	447	Xe-131M	7.9
I-134	0	Xe-133M	25.3
I-134m	0	Xe-133	1171
I-135	34.4	Xe-135M	0
		Xe-135	99.0
		Xe-137	0
		Xe-138	0
TOTAL	**1021.4**		**1308.4**

This value is based on historical usage and recommendations from References B.2, B.3, B.4, B.5, and B.6, where Reference B.2 recommends a 50% release of the halogens. References B.3 and B.4 apply a natural reduction factor of 50% due to plate out in the building. This latter 50% applied to the 50% of the inventory released from the fuel element cladding gap results in 25% of the available halogen inventory reaching the outside environment. However, this value appears to be quite conservative based on the 1.7% gap release fraction for halogens quoted in References B.7 and B.8.

For the single element accident in water, it is conservatively assumed that most of the halogens released from the cladding gap remain in the water and are removed by the demineralizer system. However, a small fraction, approximately 2.5% of the total halogens released to the cladding gap are, in this case, assumed to escape from the reactor tank water into the reactor room air, which is more conservative than assuming total (100%) solubility of the halogens as is sometimes done for TRIGA® reactors (Reference B.18). However, even assuming 2.5% halogen release from the pool water will almost certainly result in an overestimate of the actual radioiodine activity released into the room because of the use of a 50% halogen gap release fraction rather than the 1.7% documented in References B.7 and B.8. In addition, about 50% of the halogens released from the water are expected to plate out in the reactor building before reaching the outside environment (References B.3 and B.4).

The experience at TMI-2, along with some experiments, indicate that the 50% halogen release fraction is much too large. Smaller releases, possibly as little as 0.6% of the iodine reaching the cladding gap may be released into the reactor room air due in part to a large amount of the elemental iodine reacting with cesium to form CsI, a compound much less volatile and more water soluble than elemental iodine (Reference B.8).

The Lawrence Livermore National Laboratory program “HotSpot” was used to determine the worst case radiation dose impact to the public for a variety of atmospheric stability classes. Over the past several years the program has become the industry standard for relatively simple Gaussian plume modeling of radioactive material release.

Furthermore, it was assumed that all of the fission products were released to the unrestricted area instantaneously, which would maximize the dose rate to persons exposed to the plume during the accident. It is also assumed that the release height of the radioactive material to be the height of the floor of the reactor room (19 feet above ground level) not the 60 feet MNRC stack. This was done to keep the radiological results as conservative as possible. Additionally, a very calm wind speed of 1 m/s was selected as lower wind speeds in these types of releases produce greater radiological consequences. Dose conversion factors from FGR 13 (ICRP 60/70 series) were used to along with a breathing rate of 4.17e-4 m^3/s to calculated various doses to the receptors of interest (1.5 m above ground level). Calculations were done for Pasquill weather classifications A through F.

Worker dose was calculated by assuming an instantaneous release of the radioactive source term into the reactor room and not being removed by the ventilation system. This was done to keep the radiological consequences conservative. The reactor room volume was used to calculate the individual derived air concentration (DAC) values for each isotope of interest. The various DAC values and egress times in the reactor room were used to calculate the final worker dose consequences. Note the reactor room was assumed to large enough to mimic a semi-infinite cloud for the external radiation sources (primarily radio-xenon). As the reactor room is much smaller than a semi-infinite cloud this method will over predict expected worker dose from external radiation.

1. Initial Fission Product Source;

    a. Reactor Power level - 1 MW;

    b. Operating time - 365 days;

2. Release Fractions;

    a. For the purposes of the maximum hypothetical accident and the accident where the pool water remains present, it was assumed that at the time of fuel cladding failure, a fraction of the radionuclide in the inventory given in Table B-1 was instantaneously released into the air of the reactor room. In one scenario this instantaneous release occurs directly into the reactor room air, while in the other projected accident the pathway to the air requires migration through the pool water which will reduce the halogen release. Also, for the halogens, a further reduction in activity is expected to occur due to plateout in the building. Thus, the fraction (w_i) of the fission product inventory released from a single fuel element which reaches the reactor room air and then the atmosphere in the unrestricted environment outside the facility will be as follows:

$$w_i = e_i \times f_i \times g_i \times h_i ;$$

where:

e_i = the fraction released from the fuel to the fuel-cladding gap;

f_i = the fraction released from the fuel-cladding gap to the pool if water is present, or directly to the reactor room air if no water is present;

g_i = the fraction released from the pool water to the reactor room air; and

h_i = the fraction released from the reactor room air to the outside (unrestricted) environment.

b. Based on information provided previously in this appendix, e_i was set at 2.4 x 10^{-5} for both noble gases and halogens and at zero for other fission products. The values of f_i, g_i, and h_i for the two accident scenarios being considered are given in Table B-3.

Table B-3. Values of Release Fraction Components

Fission Product	f_i		g_i		h_i
	w/o pool water	w/ pool water	w/o pool water	w/ pool water	
Noble Gases	1.0	1.0	N/A	1.0	1.0
Halogens	0.5*	0.5*	N/A	0.05	0.5

Offsite Dose: As can be seen below in table B-4 the offsite radiological consequences are low, which is consistent with the MHA of similar TRIGA reactors, primarily due to high retention of radionuclides in by the fuel itself during such an accident. The maximum expected CDE for the thyroid and TEDE occur under atmospheric D at a distance of 59 m from the MNRC reactor room. The thyroid CED is 48 mrem and the TEDE is 2.4 mrem, both well below the 10 CFR 20 annual limits for the public. The evaluation distances selected for table B-4 were based on the closest point from the MNRC reactor room the site boundary (fence line), the closest inhabited building, and the closest residence.

Table B-4 Radiation Doses to Members of the General Public Under Different Atmospheric Conditions and at Different Distances from the MNRC Following a Fuel Element Cladding Failure in Air (MHA). All doses in mrem, CDE - Committed Dose Equivalent, TEDE - Total Effective Dose Equivalent

33 meters (MNRC Fence Line)

Weather Class	A	B	C	D	E	F
CDE Thyroid	36	46	41	25	0.08	0
TEDE	1.8	2.3	2.1	1.2	<0.01	0

93 meters (Closest Inhabited Building)

Weather Class	A	B	C	D	E	F
CDE Thyroid	6.3	13	25	38	33	4.6
TEDE	0.31	0.65	1.2	1.8	1.6	0 22

480 meters (Closest Residence)

Weather Class	A	B	C	D	E	F

CDE Thyroid	0.25	0.55	1.3	2.9	6.7	16
TEDE	0.01	0.03	0.06	0.13	0.30	0.71

Maximum Dose and Distance

Weather Class	A	B	C	D	E	F
CDE Thyroid	36 (@33m)	46 (@33m)	47 (@44m)	48 (@59m)	35 (@120m)	30(@220m)
TEDE	1.8 (@33m)	2.3 (@33m)	2.3 (@44m)	2.4 (@59m)	1.7 (@120m)	1.4(@220m)

Worker Dose:

As indicated by the results in Table B-5, the occupational dose to workers who evacuate the reactor room within 5 minutes following the MHA should be approximately 300 millirem TEDE and 2,495 millirem Committed Dose Equivalent to the thyroid. If evacuation occurs within 2 minutes, as it no doubt will because the reactor room is small and easy to exit, the doses drop to 120 millirem TEDE and 998 millirem CDE. All of these doses are well within the NRC limits for occupational exposure as stated in 10 CFR 20.

Table B-5 Worker dose for MHA

	CDE Thyroid (millirem)	TEDE (millirem)
2 minute room occupancy	998	120
5 minute room occupancy	2495	300

Furthermore, if the pool water remains in place and there is 24 hours of decay, the 5 minute and 2 minute occupational doses (TEDEs) drop to 7.6 and 19.0 millirem, respectively.

Offsite radiological consequences will be much less significant than the MHA given the significantly smaller source term and that for this release the source term would like be effluenced through the stack not directly from the reactor room.

Table B-6 Radiation Dose Rate from the Core Following a LOCA

	CDE Thyroid (millirem)	TEDE (millirem)
2 minute room occupancy	69	7.6
5 minute room occupancy	173	19.0

B.2 Single Element Cladding Failure in Water

While the above analysis of the MHA clearly shows that the MNRC can be subjected to current MHA criteria and remain within dose limits established by the NRC for occupational radiation exposure and exposure of the general public, results of more realistic accident scenarios are summarized in Tables B-6. In this accident, it is assumed that the pool water remains in the reactor tank (thus lowering the halogen dose significantly) and that the cladding failure occurs 24 hours after reactor shutdown. This time is selected as the minimum time to between a reactor shutdown and a fuel handling maneuver. The source term itself is overestimated as the same release fraction is used as in the MHA where the leaking fuel element was assumed to be 400 C instead of at ambient temperatures.

Since most of the halogens will be retained in the primary coolant water, the majority of the activity will end up in the demineralizer resin beds. If it is assumed that all of the halogens in the primary coolant are removed in one resin bottle and that they can be represented as a point source, the dose rate can be estimated by:

$$D = 6\ CEN;$$

where:

D = Dose, R/hr at 1 ft;
C = Number of curies retained in resins;
= (Total Ci x release fraction) x 0.975;
= 1021 Ci x 2.4 x 10^{-5} x 0.975*;
= 0.025 Ci x 0.975;
= 0.024 Ci;
E = Energy of source in MeV = 1;
N = Number of photons/dis = 1.

From this equation and the above values, the dose rate 1 ft from a single resin bottle will be 144 mR/hr. If necessary, temporary shielding could be placed in the demineralizer resin area to protect personnel during resin replacement. However, in reality, this is a worst case estimate because the halogens will be distributed over more than one resin bottle and thus will not create a point source geometry, which maximizes the dose rate. In addition, decay could be used to reduce the halogen dose rate before personnel would need to be in the area. But, should this operation be required, personnel will be closely monitored and exposures kept within limits specified by 10 CFR Part 20.

As discussed in Chapter 11, an area radiation monitor is located inside the demineralizer resin cubicle to alert personnel if the radiation level in the area reaches the alarm point.

B.3 Radiation Dose Rate from the Core Following a Loss of Coolant Accident

B.3.1 Introduction

Even though there is a very remote possibility that the primary coolant and reactor shielding water will be totally lost, direct and scattered dose rates from an uncovered core following 1 MW operations have been calculated. Dose rates were calculated by constructing a simple, yet conservative, MNCP model of the MNRC reactor and facility. The calculation of activity is given below. Each radioactive decay is assumed to produce a single 1 MeV gamma-ray. The MCNP model utilizes a homogenized reactor (26 cm in height by 38 cm in diameter) to simplify the geometry greatly while approximating for some degree of self-shielding that will take place within uranium of the reactor. The 9 foot concrete shielding that constitutes the base of the reactor tank and the minimum 9 foot concrete cylinder surround the reactor tank were modeled. The 6 inch thick concrete walls of the reactor room were modeled however the roof of the facility was not modeled because it is essentially a thin steel corrugated structure that would provide little shielding or scatter. A 1-inch-thick iron disk 1 meter in diameter was placed 3 meters above the top of the reactor tank to simulate the MNRC “bridge” where the control drives are located. This disk was used to simulate the scatter that would take place during this event that would contribute to the dose rate of individuals not directly located in direct line-of-sight with the reactor. No other structural materials were included in the model, such as walls and ceilings. This was done to minimize the attenuation of scattered gamma-ray to yield conservative dose estimates. Attenuation and scatter in air were allowed for in this model.

Dose rates were calculated for an individual directly over the core standing in the reactor room, inside the reactor room but not in direct line-of-sight of the core, just outside the reactor room, inside the control, at the MNRC fence line, in the closes building, and the closest inhabited building.

Total Core Activity

The total fission product activity as a function of time after shutdown was determined using the standard equation below (Reference B.16):

$$\text{Activity at time } \tau = 1.4\, P_o[(\tau-T_o)^{-0.2} - \tau^{-0.2}] \text{ Ci ;}$$

where:

P_o = Thermal power (W);

τ = Time after reactor startup (d);

T_o = Time reactor operating at power P_o (d);

hence:

$\tau - T_o$ = Time after reactor shutdown (d).

The MNRC will operate a maximum 365 MWd per year; therefore, the operating profile used in the above equations was 8760 hours (365 days) at the full power of 1 MW. Increasing the operating time further makes little difference to the total fission product activity. The resultant fission product activity and source term can be seen in Table B-7.

Table B-7 Total Fission Product Activity After Shutdown		
Time After Shutdown	Total Activity (Ci)	Source Strength (γs^{-1})
10 seconds	8.15×10^{6}	3.02×10^{17}
1 hour	2.74×10^{6}	1.01×10^{17}
1 day	9.70×10^{5}	3.59×10^{16}
1 week	5.20×10^{5}	1.92×10^{16}
1 month	2.86×10^{5}	1.06×10^{16}

B.3.2 Dose Rate Directly Above the Core

The results are given in Table B-8 and are in general agreement with results for the 2 MW Torrey Pines TRIGA® Reactor (Reference B.18) scaled to 1.0 MW MNRC operations.

Table B-8 Dose Rates on the MNRC Reactor Top After a Loss of Pool Water Accident Following 1 MW Operations	
Time After Shutdown	Effective Dose Equivalent Rate (rem/h)
10 seconds	1.53×10^{4}
1 hour	5.15×10^{3}
1 day	1.82×10^{3}
1 week	9.75×10^{2}
1 month	5.35×10^{2}

B.3.3 Dose Rate From Scattered Radiation in Reactor Room

The results below are given for a position located just inside the MNRC reactor room 2 meters from the edge of the reactor tank. This location was selected to be far enough away from the reactor tank so that there is no line-of-sight to the exposed core.

The results of the calculations for the scattered radiation dose rates can be seen in Table B-9.

Table B-9 Scattered Radiation Dose Rates in the MNRC Reactor Room After a Loss of Pool Water Accident Following 1 MW Operation

Time After Shutdown	Effective Dose Equivalent Rate (rem/h)
10 seconds	61.2
1 hour	20.6
1 day	7.3
1 week	3.9
1 month	2.1

B.3.4 Dose Rate at Other Locations After a Loss of Pool Water Accident

The dose rates after a loss of pool water accident are provided in this section.

A position just outside (2 meters away) from the entrance to the reactor room is given to provide a worst-case estimate of personnel dose (Table B-10) in the MNRC equipment room or on the first level of the MNRC room. Should this accident scenario occur personnel could need to enter these areas to establish water to the reactor tank.

Table B-10 Scattered Radiation Dose Rates in the MNRC Outside Reactor Room After a Loss of Pool Water Accident Following 1 MW Operation

Time After Shutdown	Effective Dose Equivalent Rate (rem/h)
10 seconds	1.66
1 hour	0.56
1 day	0.20
1 week	0.11
1 month	0.058

The MNRC control room is located at ground level approximately 14 m east of the core. The dose estimates below (Table B-11) are conservative as no shielding credit is taken for the roof and walls of the reactor facility.

Table B-11 Scattered Radiation Dose Rates in the MNRC Control Room After a Loss of Pool Water Accident Following 1 MW Operation

Time After Shutdown	Effective Dose Equivalent Rate (mrem/h)
10 seconds	370
1 hour	124
1 day	44
1 week	24
1 month	13

The shortest distance to the MNRC fence line is 28 meters. The dose estimate given below (Table B-12) should be considered the highest possible doses possible for the public in this scenario.

Table B-12 Scattered Radiation Dose Rates in the MNRC Fence Line After a Loss of Pool Water Accident Following 1 MW Operation

Time After Shutdown	Effective Dose Equivalent Rate (mrem/h)
10 seconds	320
1 hour	108
1 day	38
1 week	20
1 month	11

The closest building to the MNRC is an old industrial x-ray facility utilized by the Air Force. It is located 56 meters to the south of MNRC and has be uninhabited for nearly 20 years. No credit for building shielding is taken and the dose rates are presented below (Table B-13).

Table B-13 Scattered Radiation Dose Rates in Closest Public (not inhabited) Building After a Loss of Pool Water Accident Following 1 MW Operation	
Time After Shutdown	Effective Dose Equivalent Rate (mrem/h)
10 seconds	152
1 hour	51
1 day	18
1 week	10
1 month	5

The closest inhabited building to the MNRC is a large conference center. It is located 94 meters to the north east of MNRC and is regularly used. No credit for building shielding is taken and the dose rates are presented below (Table B-13).

Table B-14 Scattered Radiation Dose Rates in Closest Public (inhabited) Building After a Loss of Pool Water Accident Following 1 MW Operation	
Time After Shutdown	Effective Dose Equivalent Rate (mrem/h)
10 seconds	65
1 hour	22
1 day	8
1 week	4
1 month	2

B.4 Historical Fuel Accident at MNRC

The MNRC has experienced (in the late 1990s) a single fuel element leakage in its operational history. The incident occurred approximately 30 minutes after starting up the reactor for 1 MW operations. There was no drop reactor tank level associated with this incident making the severity of this accident somewhere between the MHA and a single fuel element 24 hours after shutting down. No one was inside the reactor room during the incident so no worker uptakes took place. The event was quickly detected by the reactor room and stack CAMs which acted to switch the ventilation path to “recirculation” mode before becoming saturated. The total offsite release of radioactivity was thought to be small but could not be determined with any degree of accuracy.

Of interest was the dose rate measured outside of the reactor room door just after the fuel leak was measure to be 40 mRem/hr and produced no significant (<1 mrem/hr) dose readings at the MNRC fence line.

APPENDIX C

PROBABILISTIC ASSESSMENT OF THE AIRPLANE CRASH RISK FOR MCCLELLAN AIR FORCE BASE TRIGA® REACTOR

Appendix C - Valid Pages

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NOTE: This appendix was extracted from the Stationary Neutron Radiography System (SNRS) proposal to Sacramento Air Logistics Center, McClellan AFB, California, Volume I - Design and Management, Submitted March 31, 1986 by GA Technologies. The original authors were W. H. Strohmayer and M. G. Stamatelatos.

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SUMMARY

An assessment of the risk of aircraft accidents at a TRIGA® radiography facility (stationary neutron radiography system) located at the McClellan Air Force Base in Sacramento, California was performed to evaluate its contribution to the overall reactor risk. The facility is partially below grade, with the top of the reactor core located 2 ft below ground level. The core itself is surrounded by a cylinder of reinforced concrete [redacted] thick, and it has a 7-ft concrete mat below.

The most severe credible accident sequence has been estimated to be the direct impact on the reactor building by a heavy aircraft leading to structure penetration and major damage to the reactor; i.e., the release of gaseous fission product nuclides from the gap between the fuel and the cladding as a result of fuel element breach. The doses related to this release are below 10 CFR 20 limits and much below the 10 CFR 100 limits which would actually apply under reactor accident conditions.

The present analysis consisted of the evaluation of three probabilities the product of which yields the sought-after probability. These are the probability of aircraft impact onto the reactor building and its vicinity, the conditional probability of building penetration when subject to impact, and the probability of major reactor damage given reactor structure penetration.

Each of the three probabilities was calculated by using conservative methods and data. Therefore, the resulting product, i.e., the probability of major reactor damage due to aircraft accidents, which was evaluated to be 5 x 10^{-8} per reactor year, is an upper limit. The "best estimate" value is expected to be considerably lower.

Probabilistic safety analyses of nuclear power reactors have generally concluded that a reactor damage probability due to an aircraft accident less than 10^{-7} per year does not represent a significant contribution to the overall reactor risk.

In a "partially below grade" configuration, the design safety requirement for the SNRS (specification 4.2.2 in solicitation No. F04606-86-R-0266) is that <u>either</u> (1) the probability of a significant radiological accident is less than 10^{-7} per year, <u>or</u> (2) the maximum credible radiological accident will have inconsequential effects, considering both air and water releases as per 10 CFR 20 and ANSI 15.7 limits. Our analysis has shown that the GA TRIGA® reactor can actually meet <u>both</u> requirements. Therefore, the aircraft radiological accident is an "incredible" event

1. INTRODUCTION

The objective of this assessment is to estimate the risk of aircraft accidents at a stationary neutron radiography system (SNRS) facility using a 1-MW TRIGA® nuclear reactor located at the McClellan Air Force Base in Sacramento, California.

The main concern motivating this assessment is the potential release to the public of radioactive material as a result of an aircraft striking the reactor building or its vicinity.

Over the past several decades, probabilistic methods have gained increasing use for evaluating the risks of nuclear power reactors. Probabilistic risk assessment (PRA) is aimed at evaluating the probability of adverse reactor conditions leading to prompt or delayed severe health hazards to the public.

As a result of numerous reactor studies, it has been concluded that airplane crashes become a significant contributor to public risks when the probability of a significant aircraft radiological accident exceeds approximately 1 in 10,000,000 reactor years; i.e., 10^{-7} per reactor year. If this probability is smaller than 10^{-7}, the aircraft accident risk becomes insignificant or, equivalently, the aircraft radiological accident becomes a non-credible accident scenario.

In a "partially below grade" configuration, the design safety requirement for the SNRS (specification 4.2.2 in solicitation No. F04606-86-R-0266) is that <u>either</u> (1) the probability of a significant radiological accident is less than 10^{-7} per year, <u>or</u> (2) the maximum credible radiological accident will have inconsequential effects, considering both air and water releases per 10 CFR 20 and ANSI 15.7 limits. Our analysis has shown that the GA TRIGA® reactor can actually meet <u>both</u> requirements. Therefore, the aircraft radiological accident is an "incredible" event.

The analysis performed here has been realistic (best estimate) where possible and very conservative where there were data limitations. The derived conclusions about aircraft accident-related risk are therefore very conservative from the safety standpoint.

2. ASSESSMENT METHODOLOGY

The probability P per year that an airplane crash will lead to critical reactor damage is essentially the product of three probabilities:

1. P_c, the probability of an airplane crash at the reactor site,

2. $P_{p/c}$, the conditional probability that the reactor building will be penetrated as a result of an airplane crash.

3. $P_{d/p}$, the conditional probability of critical damage to the reactor as a result of airplane-crash-induced reactor building penetration,

$$P = P_c \cdot P_{p/c} \cdot P_{d/p}. \quad \text{(C-1)}$$

The crash probability, P_c, is the product of three factors: the number of aircraft movements, the accident probability per aircraft movement per unit area, and the effective area of the target of interest.

$$P_c = \sum_i \sum_j \sum_k \left(N_{ijk} C_{ijk} A_{ijk}\right) , \quad \text{(C-2)}$$

where N_{ijk} = number of annual movements of type j for aircraft type i in flight pattern k,

C_{ijk} = crash probability per movement of type j for aircraft type i in flight pattern k,

A_{ijk} = effective target area associated with the structure of interest for aircraft type i in movement type j and flight pattern k.

The types of aircraft for an Air Force Base (Reference C.1) include fighters, trainers, heavy aircraft, and other miscellaneous aircraft. The aircraft movements include takeoff, landing, and closed pattern maneuvers. Flight patterns refer to flights to and from different runways.

In the present analysis the aircraft were grouped into two categories: those exceeding 12,500 lb in gross weight and the lighter ones which represent an insignificant fraction of the total operations. Three flight maneuvers were considered: namely, takeoff, landing, and closed patterns. There is only one flight path of interest since there is only one runway in the TRIGA® vicinity.

The conditional probability of penetration as a result of an airplane crash is generally evaluated for both direct and indirect hits. The latter refer to the impact of missiles generated as a result of the airplane crash in the immediate vicinity of the target of interest, as well as the possibility of lateral aircraft skid into the structure.

For the present analysis it was conservatively estimated that indirect hits are inconsequential because of the target configuration and building structure (Figure C.1). The reactor building (with wall thickness of 3.5 ft or more) is surrounded on all sides by rooms and, therefore, it was estimated that no missile generated by an aircraft crash in the immediate vicinity of the TRIGA®

building could penetrate to and through the reactor walls. Thus, only direct hits were considered in the analysis.

For direct hits the conditional probability of structure penetration by an aircraft was calculated based on the method of Reference C.2. This method very conservatively evaluates the conditional probability of penetration of a reinforced concrete wall when impacted by a heavy aircraft at full flight speed. The probability is given as a function of wall thickness as shown in Figure C.2.

The conditional probability of radionuclide release from the reactor by fuel element breach due to reactor structure penetration by an impacting aircraft ($P_{d/p}$) is difficult to evaluate analytically because it strongly depends on the collision history and on the likelihood that all or most of the reactor water will be lost and that a radioactive release from the damaged fuel will enter the atmosphere or groundwater resulting in radiological environmental hazard.

The McClellan AFB TRIGA® core is surrounded on all lateral sides and on the bottom by a continuous concrete structure [redacted] thick. Therefore, total loss of coolant is highly unlikely since this structure is designed to comply with seismic requirements. Also, Reference C.2 conservatively estimates that 7 ft of reinforced concrete are impenetrably by an aircraft. If the reactor water is not lost, then it will be an effective radionuclide filter. Water scrubbing under accident conditions was estimated to be a very efficient radionuclide remover when estimating nuclear reactor accident source terms (Reference C.3).

Nevertheless, the present analysis took the very conservative approach of assigning $P_{d/p}$ the value of unity. This approach could be fine-tuned at a later time when a “best estimate” value (less than one) could be used instead. This is, however, not necessary to meet the SNRS aircraft accident design specification.

[redacted]

FIGURE C.1: TRIGA ® REACTOR BUILDING

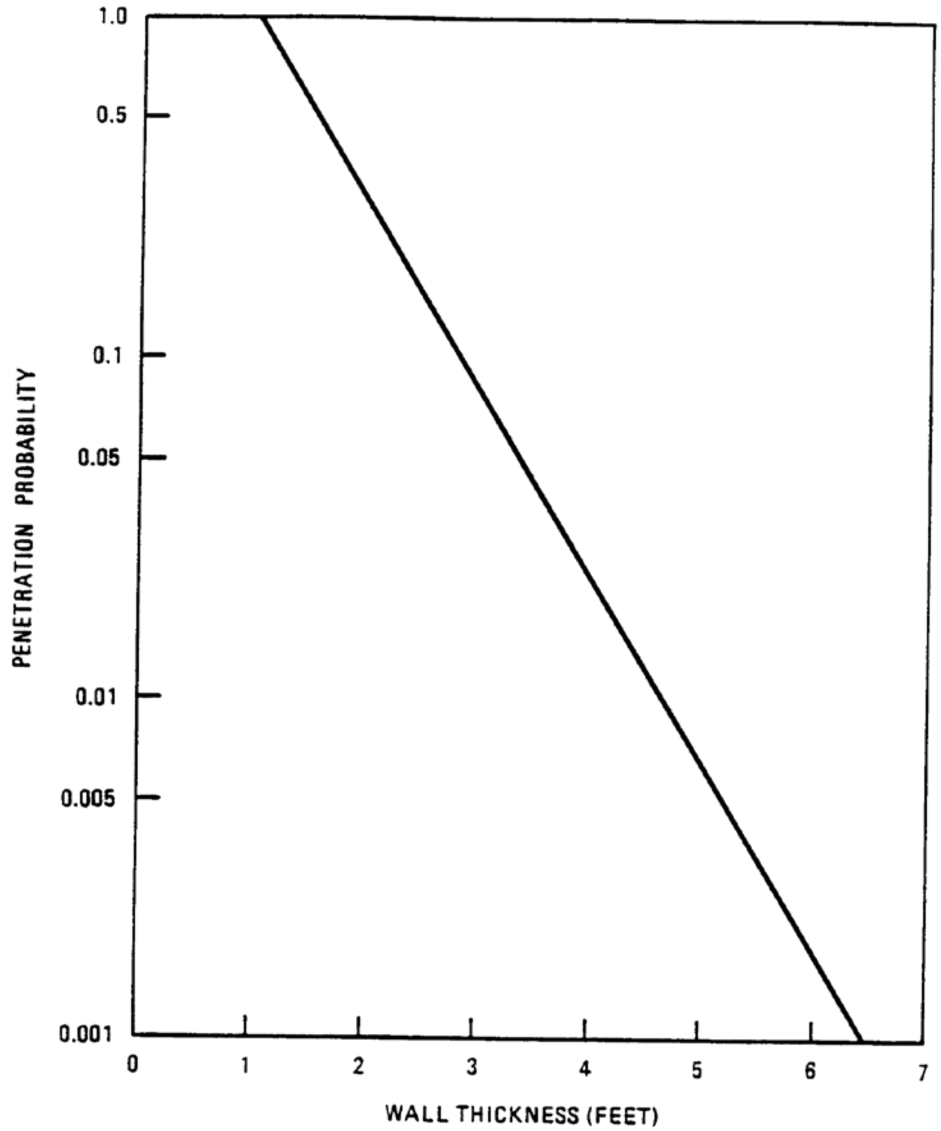

FIGURE C.2: CONDITIONAL WALL PENETRATION FOR HEAVY AIRCRAFT IMPACT

3. ANALYSIS

The proposed location of the McClellan SNRS TRIGA® is shown in Figure C.3. The reactor building is located approximately 2000 ft east of the main north-south runway and 4000 ft north of the runway's southern end. Figure C.4 shows the air traffic patterns at McClellan.

The aircraft crash probability was evaluated using Equation C-2. We considered essentially only one class of aircraft, those with a gross weight in excess of 12,500 lb. Lighter aircraft operations at McClellan form an insignificant part of total operations. We also considered three maneuvers (takeoff, landing, and closed patterns) and one flight path (one runway).

Table C-1 shows data accumulated for USAF aircraft crash probabilities as a function of distance from the runway (References C.4 and C.5). These probabilities have been combined to generate an overall probability of crash per square mile per aircraft movement within an area with a radius of five miles from the runway. This probability, evaluated to be 10^{-8} per square mile per movement, was used as the probability of crash for closed patterns. Based on information from Reference C.6, the probability of crash during a landing maneuver is 1.5×10^{-7} per square mile per movement. We assumed that takeoffs have a negligible contribution to the crash probability at the reactor site because the reactor is always at an orientation of 180 deg away from the takeoff flight path.

Table C-2 shows operations data for McClellan AFB. These data, in raw form, were collected for the seven-month period from February 1 through August 31, 1981. Assuming that the air activity in Reference C.1 is fairly constant in time, the annual activity was obtained by multiplying the seven months data by 12/7. This step was omitted in the analysis of Reference C.7, which was also performed for McClellan AFB. We subdivided the aircraft activity into categories: closed pattern and landings. The total annual activities for all aircraft are also given in Table C-2.

TABLE C- 1 AIRCRAFT (USAF) CRASH PROBABILITIES (References C.4 and C.5)	
Distance from Runway (miles)	Crash Probability (10^{-8}/square mile movement)
0 to 1	5.7
1 to 2	2.3
2 to 3	1.1
3 to 4	0.42
4 to 5	0.4

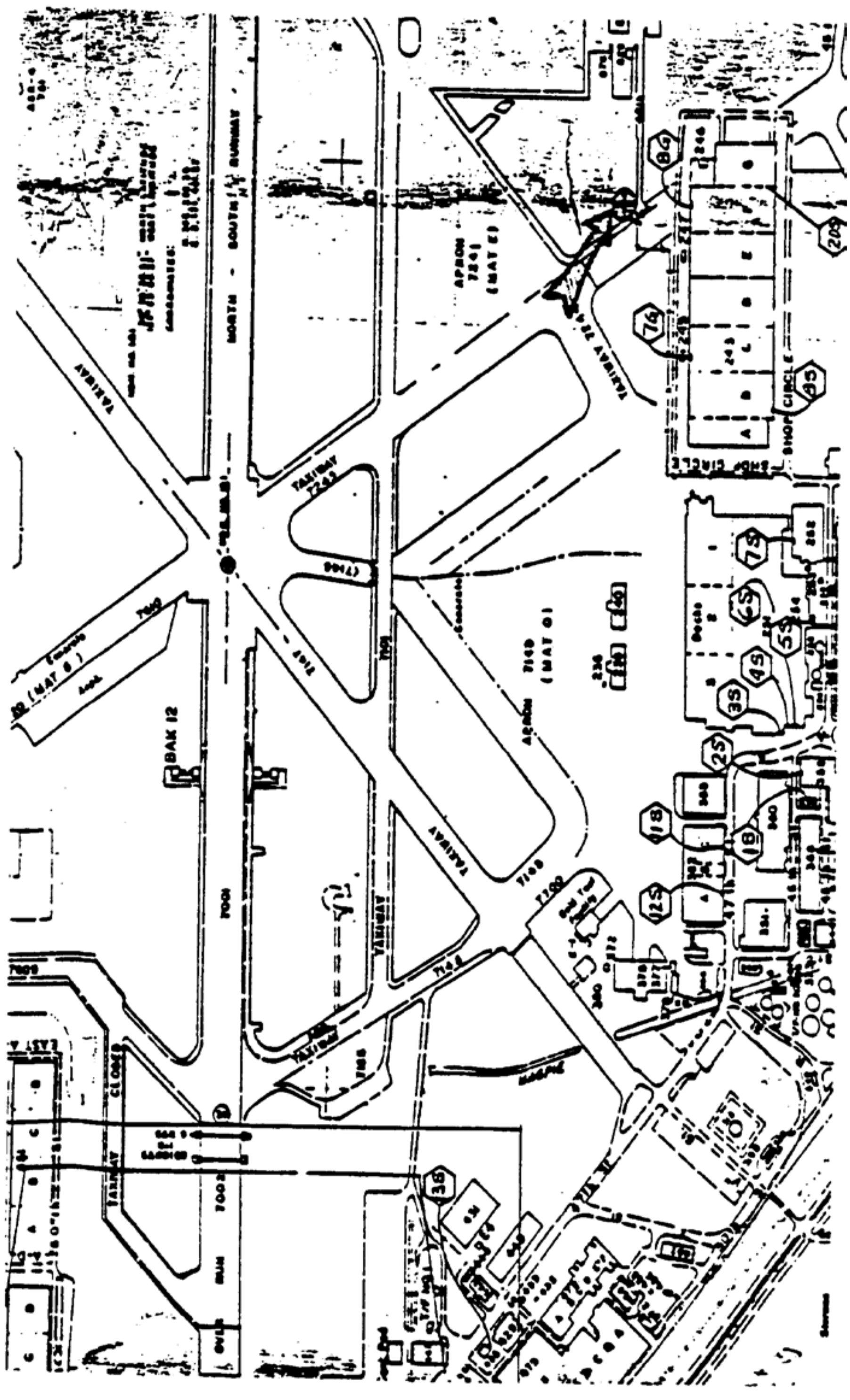

FIGURE C.3 TRIGA ® LOCATION

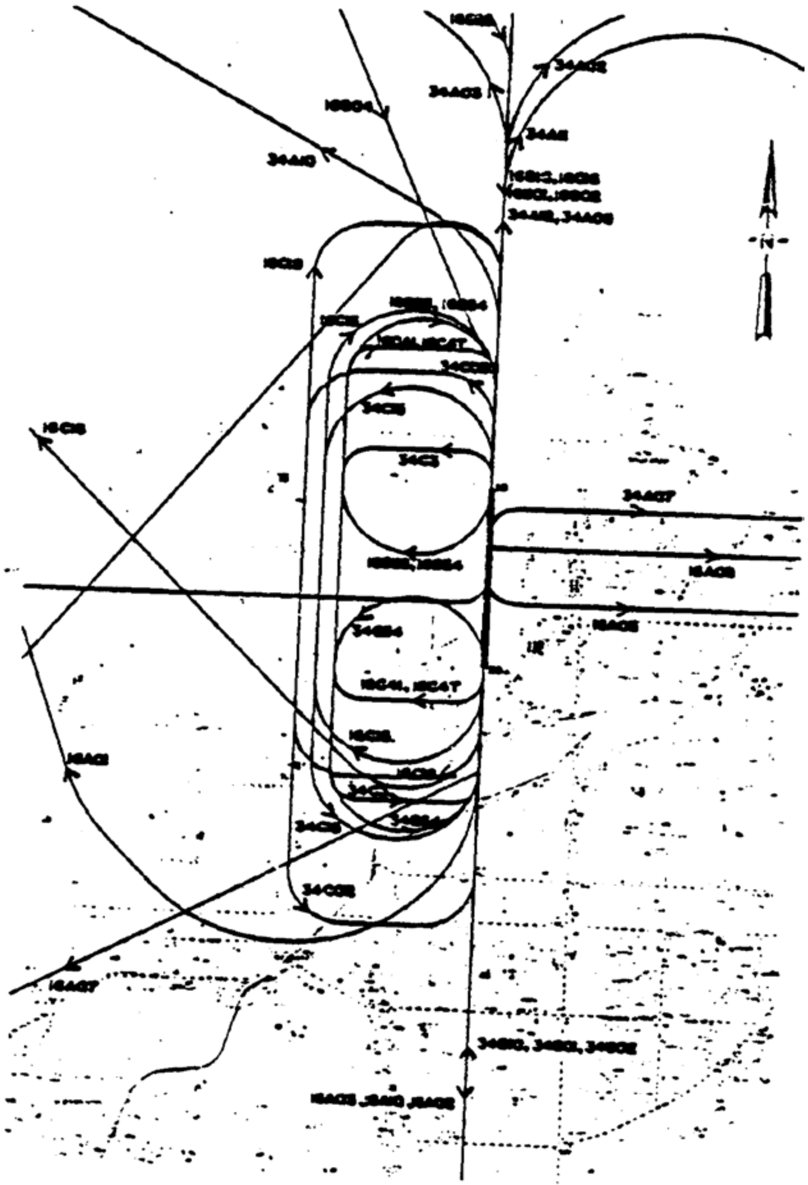

FIGURE C. 4 MCCLELLAN AFB AIR TRAFFIC PATTERNS

TABLE C-2 AIRCRAFT MOVEMENTS AT MCCLELLAN AFB FOR THE SEVEN-MONTH PERIOD FEBRUARY 1 THROUGH AUGUST 31, 1981 (Reference C.2)								
Maneuver	Fighter		Trainer		Heavy		Miscellaneous	
	7-Month	Annual	7-Month	Annual	7-Month	Annual	7-Month	Annual
Takeoff	3,226	5,530	2,218	3,802	3,768	6,459	6,870	11,777
Landing	3,226	5,530	2,218	3,802	3,768	6,459	6,870	11,777
Closed patterns	12,154	20,835	7,447	12,766	18,560	31,817	9,387	16,092
Total	18,606	31,895	11,883	20,370	26,096	44,735	23,127	39,636

The target area for the crash probability calculation was evaluated from exact drawings of the proposed SNRS TRIGA® facility and the "shadow" area was calculated using a glide angle of 20 deg (Reference C.6). The base dimensions of the reactor building are 72 ft 3 in. X 85 ft 6 in. with a height of 33 ft. Thus the effective target area is:

$$A_{eff} = \frac{(72.25)(85.5)ft^2}{(5280\ \ ft/mi)^2} + \frac{(33)(85.5)ft^2}{\tan(20^o)(5820\ \ ft/mi)^2} = 5\ \text{x}\ 10^{-4}\ \ mi^2.$$

This effective area does not include the room above the reactor which houses the control rod drives and the refueling crane (Figure C.1). This room was deemed to be more vulnerable to airplane crash structural damage than the rest of the reactor building (thinner walls), and it was treated separately from the rest of the reactor building.

The calculation of the probability of an aircraft crash on the reactor building yields:

$$\begin{aligned} P_c = &\ (81{,}510\ \ closed\ \ patterns/year)\left(1\ \text{x}\ 10^{-8}\frac{crash}{closed\ \ pattern - mi^2}\right)(5\ \text{x}\ 10^{-4} mi^2) \\ &+ (27{,}568\ \ landings/year)\left(1.5\ \text{x}\ 10^{-7}\frac{crash}{landing - mi^2}\right)(5\ \text{x}\,10^{-4}\ \ mi^2) \\ &= 2.5\ \text{x}\ 10^{-6}\ \ crashes/year \end{aligned}$$

Estimates of the conditional probability of penetration given a crash are given in Figure C.2. For the case where the minimum thickness of reinforced concrete is 3.5 ft and the aircraft weight is greater than 12,500 lb the conditional probability of penetration is approximately 0.045/crash. Thus the uncorrected probability of breach of the reactor building structure is:

$$\Delta\acute{P}_1 = (0.045/crash)(2.5\ \ \text{x}\,10^{-6}\ \ crash/year),$$

$$\Delta\acute{P}_1 = 1.1\ \ \text{x}\,10^{-7}/year.$$

This number requires correction because only approximately one-third of the target area has the minimum wall thickness of 3.5 ft, while the remainder has a minimum wall thickness of 8 ft which, according to Reference C.2, is impenetrable to an airplane crash. Applying the one-third correction factor to the previous calculation, we obtain the probability of core damage due to the breach of the reactor building structure:

$$\Delta P_1 = \ (1/3)(1.1 \ \text{x}\,10^{-7})/year,$$

$$\Delta P_1 = 3.7 \ \ \text{x} \ \ 10^{-8}/year.$$

The refueling room, located above the reactor, has thin walls which are estimated to yield no significant missiles upon impact. However, a three-ton crane is in position over the core for at most two days every six months. The rest of the time, this crane is stowed away. Only when the crane is in position can it contribute to the likelihood of core damage by an impact-generated missile. The probability of this event was estimated from the annual probability of crashing into the refueling room using the fraction of time when the crane is in place. The effective target area of the refueling room is 4.4 x 10^{-5} mi^2, whence the crash probability for this room is 2.18 x 10^{-7}/year. Since the room is a thin-walled structure, we can assume that the conditional probability of penetration is unity. The fraction of time the crane is in place is 0.01 (i.e., four days per year). Thus an additional contribution to probability of core damage due to a missile generated from a collision with the refueling crane is:

$$\Delta P_2 = (0.01)(2.18 \ \ \text{x} \ \ 10^{-7})/year \ \ = \ \ 2.2 \ \ \text{x}\,10^{-9}/year$$

The probability of an aircraft crash directly onto the opening at the top of the reactor vessel will now be evaluated. Since it is assumed that the refueling room structure has unit conditional probability of penetration, the effective target area is the opening area at the top of the reactor vessel, which has a diameter of 7.5 ft. This area is 2.3 x 10^{-6} mi^2. Therefore, the third contribution to the probability of damage to the reactor core, a crash directly onto the pool opening, is:

$$\Delta P_3 = \left(\frac{2.5 \ \ \text{x} \ \ 10^{-6}\frac{crashes}{year}}{5 \ \ \text{x} \ \ 10^{-4} mi^2}\right)(2.3 \ \ \text{x} \ \ 10^{-6} \ \ mi^2)(1 \ \ penetration/crash)(1 \ \ release/penetration).$$

or

$$\Delta P_3 = 1.2 \ \ \text{x} \ \ 10^{-8}/year.$$

The total probability of air crash-related core damage is then the sum of these separate probabilities:

$$P = \ \ \Delta P_1 + \Delta P_2 + \Delta P_3,$$

which yields:

$$P = 5.1 \ \ \text{x} \ \ 10^{-8}/year.$$

This value is considerably less than 10^{-7}, the threshold value for a significant risk contributor.

It should be borne in mind that the value of P calculated here is very conservative in view of the fact that the probability of critical damage to the reactor given structural penetration ($P_{d/p}$) was assumed to be unity. For a best estimate, inclusion of a more realistic value of $P_{d/p}$ could lead to a reduction in the value of P by one order of magnitude or more.

4. CONCLUSIONS

The reactor damage probability due to aircraft accidents was conservatively calculated to be 5×10^{-8}, a factor of two below the probability of a credible aircraft radiological accident. This, by itself, satisfies the SNRS aircraft safety requirement.

Since the closure of the Air Force base in the year 2000 the number of overall flight operations has reduced by approximately an order of magnitude (chapter 2). In general, the type of aircraft has migrated from larger high speed military airplanes to smaller lower speed civilian aircraft. The originally calculated 5×10^{-8} probability of a creditable aircraft radiological accident is now more likely 5×10^{-9} which is considered strongly non-credible.

To further reinforce the statement that the aircraft radiological accident is an "incredible" event, several comments can be made regarding the potential radionuclide release during the air crash-induced most severe TRIGA® accident.

The most severe credible accident for a TRIGA® is expected to be the simultaneous breach of integrity of the majority of the fuel elements and the complete loss of water from the reactor tank.

Insofar as melting of the reactor core following complete loss of coolant is not feasible for the TRIGA®, the only radiological hazard associated with the above most severe credible accident is due to gaseous radioisotopes that could be released from the gap between the nuclear fuel and the cladding in the event of fuel element breach. Aerosolization of the nuclear fuel is highly unlikely because the TRIGA® fuel elements consist of a $UZrH_x$ metallic matrix with outstanding retention capability not only structurally but also for the entrapped fission and activation products.

Release of radionuclides into the ground by coolant loss is also unlikely. Since the reactor is surrounded from the sides and the bottom by a thick reinforced concrete structure that is not penetrable by a direct aircraft crash, the only mode of water loss can be due to air crash-generated missiles which are not capable of penetrating or cracking the reinforced concrete reactor cradle. If water is released from the reactor by severing one of the beam conduits connecting the reactor with the surrounding rooms without cracking the floor of the reactor structure, contaminated water cannot be released into the environment because the floor and part of the walls of the room adjacent to the reactor core are waterproof.

Therefore, the most credible air crash-induced TRIGA® accident may involve partial release of gaseous radionuclides directly into the reactor atmosphere by partial exposure of damaged fuel elements and, indirectly, through the water remaining in the core which has been shown to be a very effective radionuclide "scrubber."

APPENDIX D

LAYOUT OF REACTOR AND REACTOR HALL

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D.2 Elevation View Showing Reactor Tank, Reactor Hall, Inlet And Outlet Ventilation Ducts D-3

D.1 Layout of Reactor and Reactor Hall

Personnel at MNRC have supplied sketches of the relationship between the reactor tank, reactor hall, and the inlet and exhaust ventilation ducts. These are exhibited in Figures D.1 and D.2. Figure D.1 is important for two reasons. First, it shows why it is reasonable to assume that a portion of the hot air plume is diverted from the reactor hall into the 500 cfm exhaust duct at the top of the reactor tank. Second, it shows that the exhaust from the 1600 cfm return duct should be diverted from across the core to an angular direction (perhaps 30° to 45°). Such a redirected air path minimizes the disturbance of the plume as it rises into the reactor hall and does not interfere with the action of the 500 cfm exhaust duct to withdraw a portion of the plume before it enters the reactor hall. In the redirected path, the 1600 cfm continues to promote rapid mixing of the reactor hall air.

Figure D.2 provides additional information concerning the physical relationship of the hot air plume from the core and the various inlet and exhaust ducts. It should be noted that an additional 300 cfm of air is supplied to the reactor hall by leaks through the walls, ceiling, and around the two doors in the east wall.

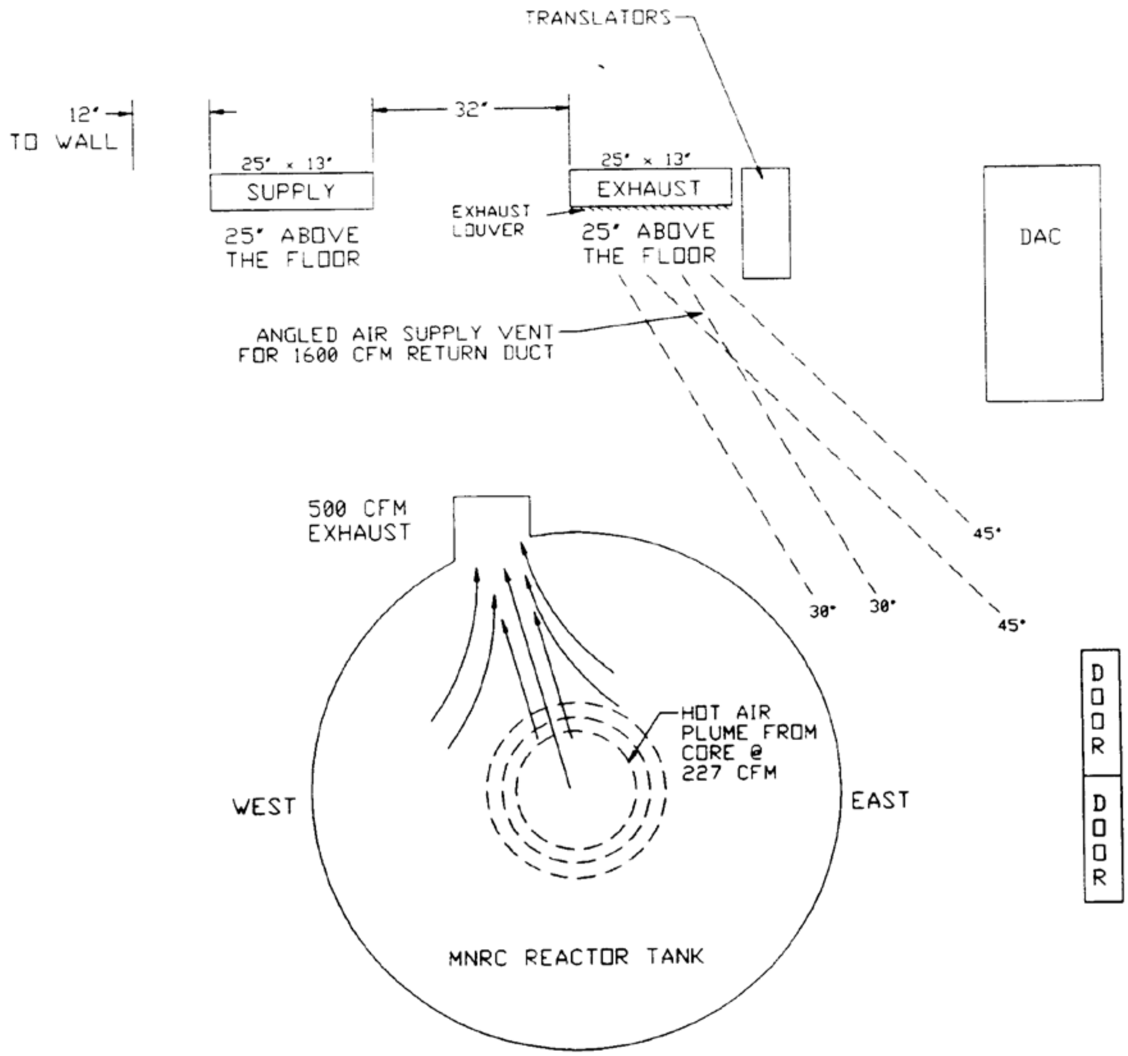

Figure D. 1: FLOOR PLAN OF REACTOR HALL SHOWING RELATION OF WALL MOUNTED AIR SUPPLY DUCT AND 500 CFM EXHAUST DUCT

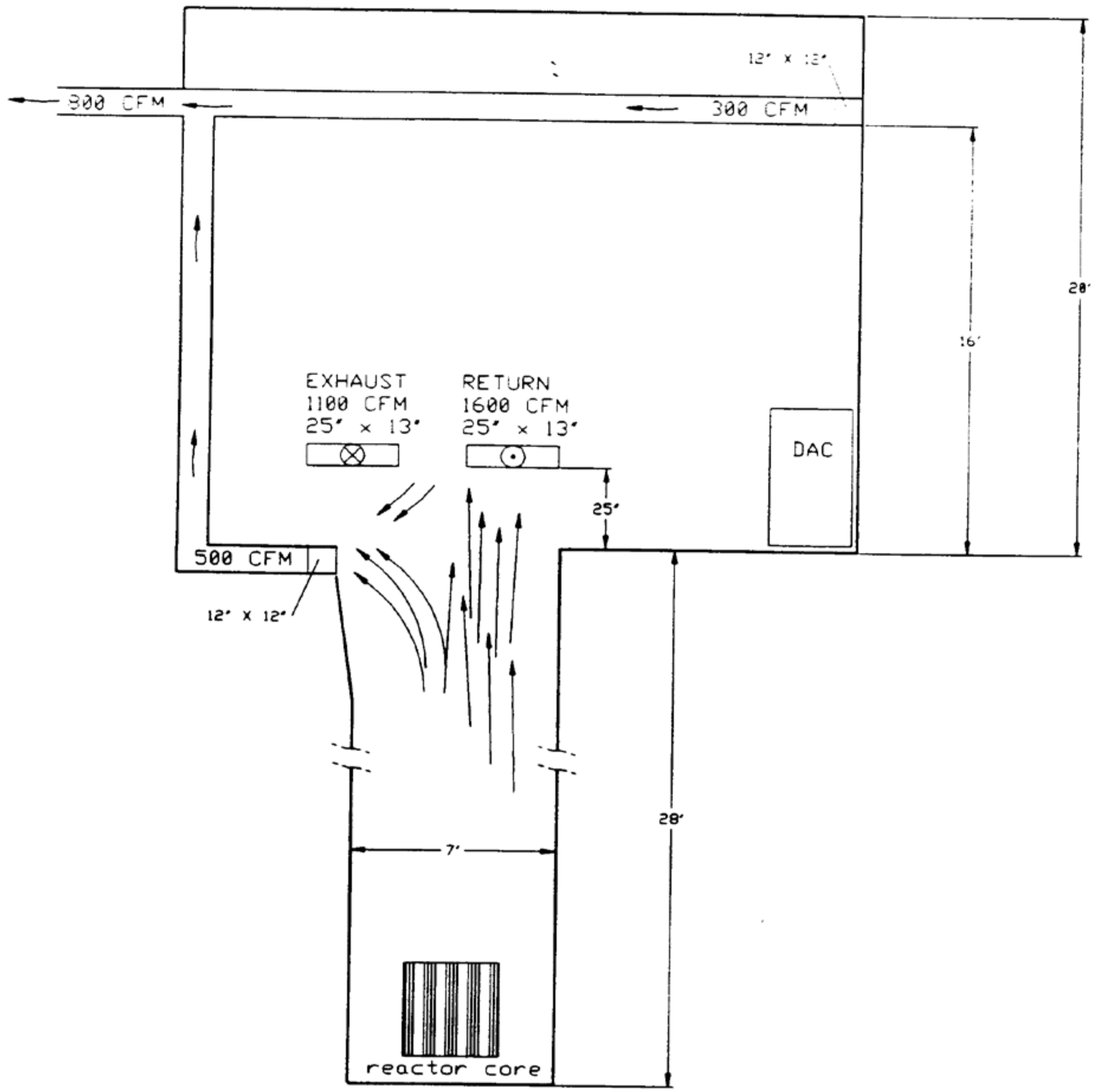

Figure D. 2: ELEVATION VIEW SHOWING REACTOR TANK, REACTOR HALL, INLET AND OUTLET VENTILATION DUCTS