REED COLLEGE
REED RESEARCH REACTOR
LICENSE NO. R-112
DOCKET NO. 50-288

LICENSE RENEWAL APPLICATION

SAFETY ANALYSIS REPORT,
TECHNICAL SPECIFICATIONS,
ENVIRONMENTAL CONSIDERATIONS, AND
OPERATOR REQUALIFICATION PROGRAM

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUR OR DENOTED BY BRACKETS

REED COLLEGE Portland, Oregon 97202

REACTOR FACILITY

August 29, 2007

US Nuclear Regulatory Commission
Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

Docket: 50-288
Subject: License Renewal

This is Reed College's application for a renewal of the operating license for the Reed Research Reactor, License R-112, Docket 50-288. Reed College is a private non-profit educational institution incorporated in the State of Oregon. The applicant information is:

Reed Institute (dba Reed College)
3203 SE Woodstock Blvd.
Portland, OR 97202

Reed is applying to be licensed as a Class 104 facility for an additional 20 years. As part of the relicensing, Reed is upgrading the licensed power level to 500 kW. The license will be used for primarily educational activities in Portland, Oregon and the surrounding region.

Enclosed are the supporting documents, including;

1. Updated Safety Analysis Report (SAR)
2. Technical Specifications
3. Financial Qualifications and Decommissioning per 10CFR50.33(f)(2) and 10CFR50.33(k)
4. Environmental Report
5. Radiation Protection Plan
6. Operator Requalification Plan
7. Fire Plan
8. Administrative Procedures

There are no changes to Reed College's Emergency Plan or Physical Security Plan being implemented as part of the relicensing, so they are not included in this submittal. The most recent versions of each plan remains in effect. Reed College is not owned, controlled, or dominated by an alien, foreign corporation, or foreign government.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on August 29, 2007.

Stephen G. Frantz

Stephen G. Frantz
Reactor Director

A020

3203 SE Woodstock Blvd., Portland, OR 97202-8199 503-777-7222 Fax: 503-777-7274 reactor@reed.edu

NRR

Safety Analysis Report

August 2007

Reed Research Reactor
3203 SE Woodstock Boulevard
Portland, Oregon 97202
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License R-112
Docket 50-288

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Chapter 1

The Facility

Reed Research Reactor
Safety Analysis Report

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Chapter 1

The Facility

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1 THE FACILITY

1.1 Introduction

This safety analysis report supports an application to the U.S. Nuclear Regulatory Commission (NRC) by Reed College for the utilization of a TRIGA®-fueled research reactor. The reactor is owned and operated by Reed College for the purpose of performing neutron irradiation services for a wide variety of scientific applications. The reactor is known as the Reed Research Reactor (RRR). The license number is R-112, Docket 50-288.

The Reed Research Reactor (RRR) is owned and operated by Reed College, a private undergraduate educational institution located in Portland, Oregon. The reactor was obtained in 1968 through a grant from the United States Atomic Energy Commission and is currently operated under Nuclear Regulatory Commission License R-112 and the regulations of Chapter 1, Title 10, Code of Federal Regulations. The facility supports education and training, research, and public service activities. The reactor is in a building constructed for that purpose and which is situated adjacent to the Psychology Building near the southeast corner of the Reed College campus in southeast Portland. The a campus has approximately 1,300 students while the city of Portland has approximately 560,000 people, as described in Chapter 2, Site Characteristics.

This report is based on the *Safety Analysis Report, Reed Reactor Facility* for the initial operation of the reactor at 250 kW thermal power, and subsequent analyses supporting steady-state operations to a maximum of 500 kW. The RRR is a non-pulsing reactor.

This report addresses safety issues associated with operation of the reactor at steady-state power levels up to 500 kW. This report reflects the as-built condition of the facility, and includes experience with the operation and performance of the reactor, radiation surveys, and personnel exposure histories related to operations to a maximum of 250 kW steady-state power. The consequence of routine generation of radioactive effluent and other waste products from steady-state operation to a maximum of 500 kW is addressed in Chapter 11, Radiation Protection and Waste Management. Radiation worker and public doses from radiation associated with routine operations are well within the limits of Title 10, Code of Federal Regulations, even under extremely conservative scenarios. The consequence of accident scenarios from operation at 500 kW steady-state power is presented in Chapter 13, Accident Analysis. The consequences of accidents postulated to occur under extremely conservative conditions are well within limits. Therefore, analysis demonstrates that there is still a "reasonable assurance that the reactor can be operated at the designated location without undue risk to the health and safety of the public."

The description of the reactor core and thermal hydraulic analysis presented in Chapter 4, Reactor Description, the Secondary Cooling System in Chapter 5, Reactor Coolant

Systems, and the Reactor Control System in Chapter 7, Instrumentation and Control Systems are based on 500 kW operations.

Throughout the document most measurements have been metric equivalents, i.e., listing the dimensions in centimeters in addition to inches. Since the facility was constructed using traditional unit, these are generally the correct one. The metric equivalents are included as an aid to understanding.

1.2 Summary & Conclusions on Principal Safety Considerations

Design basis parameters of the RRR are (1) power level and (2) fuel loading required to achieve desired power. Limits on the amount of fuel loaded in the core and on the maximum power level ensure the RRR is an inherently safe reactor.

1.2.1 Safety Considerations

As of July 2007, there were over seventy TRIGA® reactors in use or under construction at universities, government and industrial laboratories, and medical centers in 24 countries. Historically, analysis and testing of TRIGA® fuel has demonstrated that fuel cladding integrity is not challenged as long as stress on the cladding remains within yield strength for the cladding temperature. Elevated TRIGA® fuel temperatures evolve hydrogen from the zirconium matrix, with concomitant pressure buildup in the cladding. Therefore, the strength of the clad as a function of temperature establishes the upper limit on power. Power less than limiting values will ensure clad integrity [1] and, therefore, contain the radioactive materials that are produced by fission in the reactor core.

As a natural-convection cooled system, heat removal capacity is well defined as long as the primary coolant is sub-cooled, restricting potential for film boiling. Limiting the potential for film boiling assures that fuel and clad temperatures are not capable of challenging cladding-integrity. The maximum heat generated within a fuel element and the bulk water temperature determine the propensity for film boiling. The design basis analysis in Chapter 4, Reactor Description, indicates that steady-state operation at power levels greater than 500 kW in natural convective flow will not lead to film boiling.

Negative fuel temperature feedback inherently limits the operation of the reactor. Increases in fuel temperature associated with operation-at-power regulate the maximum possible steady-state power, as described in Chapter 4, Reactor Description. This chapter also shows that the negative temperature coefficient is a function of the fuel composition and core geometry. Within established core systems, the negative temperature coefficient is rather constant with temperature. Excess fuel (above the amount required to establish a critical condition) is required to overcome the negative temperature feedback as operation at power causes the fuel to heat up. Consequently, maximum possible power using

TRIGA® fuel is controlled by limiting the amount of fuel loading. Limits on total fuel loading and excess reactivity ensure that the maximum power level will not lead to conditions under which design basis temperatures are possible.

1.2.2 Consequences of Normal Operations

As indicated in Chapter 11, Radiation Protection and Waste Management, radiation sources are discharged from the reactor facility in gaseous (airborne), liquid or solid form. These forms are treated individually in subsections of Chapter 11. Airborne radiation sources consist mainly of argon-41 and nitrogen-16, with argon-41 the major contributor to off-site dose. Limits on argon-41 are tabulated in Appendix B of 10 CFR Part 20.

A general limit on off-site doses from gaseous effluents is also contained in 10CFR20.1101. Radiation protection programs, effectively establishing a limit of 10 mrem per year to the public from radon-222 and its progeny.

Argon-41 is the major contributor to radiation exposure incident to the operation of the RRR. Argon-41 is attributed to neutron activation of natural argon (in air) in the reactor bay atmosphere, rotary specimen rack adjacent to the core, and dissolved in primary coolant. Argon-41 has a 1.8-hour half-life. Calculations in Chapter 11 based on 500 kW steady-state continuous operations show that doses in the reactor bay remain below inhalation DAC. A full-year exposure to equilibrium argon concentration for 500 kW operations under normal atmospheric conditions would lead to a dose less than the applicable limits.

Nitrogen-16 is the major contributor to radiation fields directly over the reactor pool during operation. Nitrogen-16 is produced by a fast neutron reaction with oxygen (as a natural component of water in the core). Nitrogen-16 has a 7.1-second half-life, and consequently does not remain at concentrations capable of contributing significantly to off-site dose. Chapter 11 shows that radiation dose rates directly above the reactor pool during expected operations at levels up to 500 kW are within required levels for a radiation area as defined in 10 CFR Part 20. Installed monitoring systems provide information necessary to identify appropriate access controls.

No liquid radioactive material is routinely produced by the normal operation of the RRR except for miscellaneous neutron activation product impurities in the primary coolant. Non-routine liquid radioactive contamination may be produced during decontamination or maintenance activities (such as resin changes.) There are no drains in the reactor bay, and any liquid radioactive waste is absorbed into a solid medium (e.g., a paper towel or other absorbent) and shipped off-site for burial.

Most of the impurities found in the primary cooling system are deposited in the mechanical filter and demineralizer resins. Therefore, these materials are dealt with as solid waste. The only radionuclides observed are trace quantities of cesium-137, cobalt-

60, etc. Even unfiltered, untreated primary coolant would meet the liquid effluent limit without further dilution.

1.2.3 Consequences of Potential Accidents

Chapter 13, Accident Analysis, recognizes three classes of accidents for which analysis is required. The maximum hypothetical accident (MHA) is a fuel element failure with maximum release of fission product inventory, from which the radioactive materials can migrate into the environment. Complete loss of coolant from the reactor pool is the second accident analyzed. The final accident is an insertion of the maximum available positive reactivity. Analysis demonstrates the consequences of these reactor accidents are acceptable, and doses to the public are well below limits established by 10 CFR Part 20.

1.3 General Description of the Facility

1.3.1 Geographical Location

The reactor is located on the campus of Reed College, in the City of Portland, in Multnomah County, Oregon. The licensee controls access to Reed College facilities and infrastructure. City and college maps are supplied in Chapter 2, Site Characteristics. The reactor is located in [redacted] the Psychology building. Latitude and longitude, building plans, universal Transverse Mercator coordinates, population details, etc. are provided in Chapter 2.

The operations boundary of the reactor facility encompasses the reactor room and control room. The site boundary encompasses the entire building and 250 feet (76 m) from the center of the reactor pool, including the Psychology and Chemistry buildings.

1.3.2 Principal Characteristics of the Site

The Portland terraces, which compose the largest physiographic subunit in the East Portland area, were formed by the ancestral Columbia and Willamette Rivers during a time when the rivers were flowing at higher levels than at present. Most of the East Portland area is underlain by bedrock of the Troutdale formation. The depth of bedrock at the reactor site is unknown but it may be hundreds of feet. A boring, 47 feet deep, was made at the reactor site and it was found that the subsurface materials at this site are sand and silt, representing lake or fluvial deposition. The water Table in the test boring was observed to stand at a depth of 46 feet. Temperatures and weather patterns are mild at the reactor site; hurricanes, tornadoes, and sieches are not considered credible.

The local seismological conditions in the neighborhood of the site are generally favorable. No fault is known to exist near the site. However, following the practice of

Portland architects the construction of the building and reactor pit were designed to resist lateral forces of Zone II as specified in the Uniform Building Code.

1.3.3 Principal Design Criteria, Operating Characteristics, & Safety Systems

The RRR TRIGA® reactor is a water-moderated, water-cooled thermal reactor operated in an open below-ground construction pool. The reactor is fueled with heterogeneous elements clad with aluminum or stainless steel, consisting of nominally 20% enriched uranium in a zirconium hydride matrix. In 1968, the RRR TRIGA® was licensed to operate at a steady-state thermal power of 250 kW. Application is made concurrently with this license renewal to operate up to a maximum steady-state thermal power level of 500 kW. Reactor cooling is by natural convection. The 250 kW-core consists typically of 64 fuel elements, each containing as much as [redacted] grams of uranium-235. The reactor core is in the form of a right circular cylinder of abou[redacted] radius and [redacted] depth, positioned with axis vertical on one focus of a 10 foot (3 m) by 15 foot (4.6 m) tank with a 5 foot (1.5 m) radius on each long end. Criticality is controlled and shutdown margin assured by three control rods in the form of aluminum or stainless-steel clad boron carbide or borated graphite. A sectional view of a typical TRIGA® reactor is shown in Figure 1.1.

1.3.4 Engineered Safety Features

The design of the RRR TRIGA®, licensed in 1968, imposed no requirements for engineered safety features. As discussed in Chapter 13, Accident Analysis, and from previous analysis, neither forced-cooling flow nor shutdown emergency core cooling is required for operation at steady-state thermal power as high as 1,900 kW, a large margin over the 500 kW steady-state operations.

1.3.5 Instrumentation and Control (I&C) and Electrical Systems

Instruments and controls are described in Chapter 7, Instrumentation and Control Systems, with the electrical power system described in Chapter 8, Electrical Power Systems. The reactor instrument and control systems include the reactor control system, process instruments, reactor protection system, and radiation safety monitoring systems. As previously noted, there are no engineered safety features at the RRR and, therefore, no associated instrumentation. The bulk of the reactor instrumentation and control systems are hard-wired analog systems (primarily manufactured by General Atomics) and widely used at various NRC-licensed facilities.

1.3.5.1 Reactor Control System

The reactor control system includes the mechanical and electrical systems for control rod drives and instruments that monitor control rod position. Each control rod can be independently manipulated by pushbutton console controls. One control rod can be operated in an automatic mode to regulate reactor power according to the linear channel and period feedback. The power meters provide interlocks for rod control and scram capability. A scram can also be actuated manually via a button on the console.

Three neutron detection instruments measure reactor power separately: a wide-range logarithmic channel, a multi-range linear channel, and a percent power channel. These provide at least two indications of reactor power from source range to power range. The nuclear instruments of the reactor protection system are integrated into the reactor control system through the automatic power level control system and through rod control interlocks. Since the core is cooled by natural convection, no engineered safety features are necessary for safe reactor shutdown.

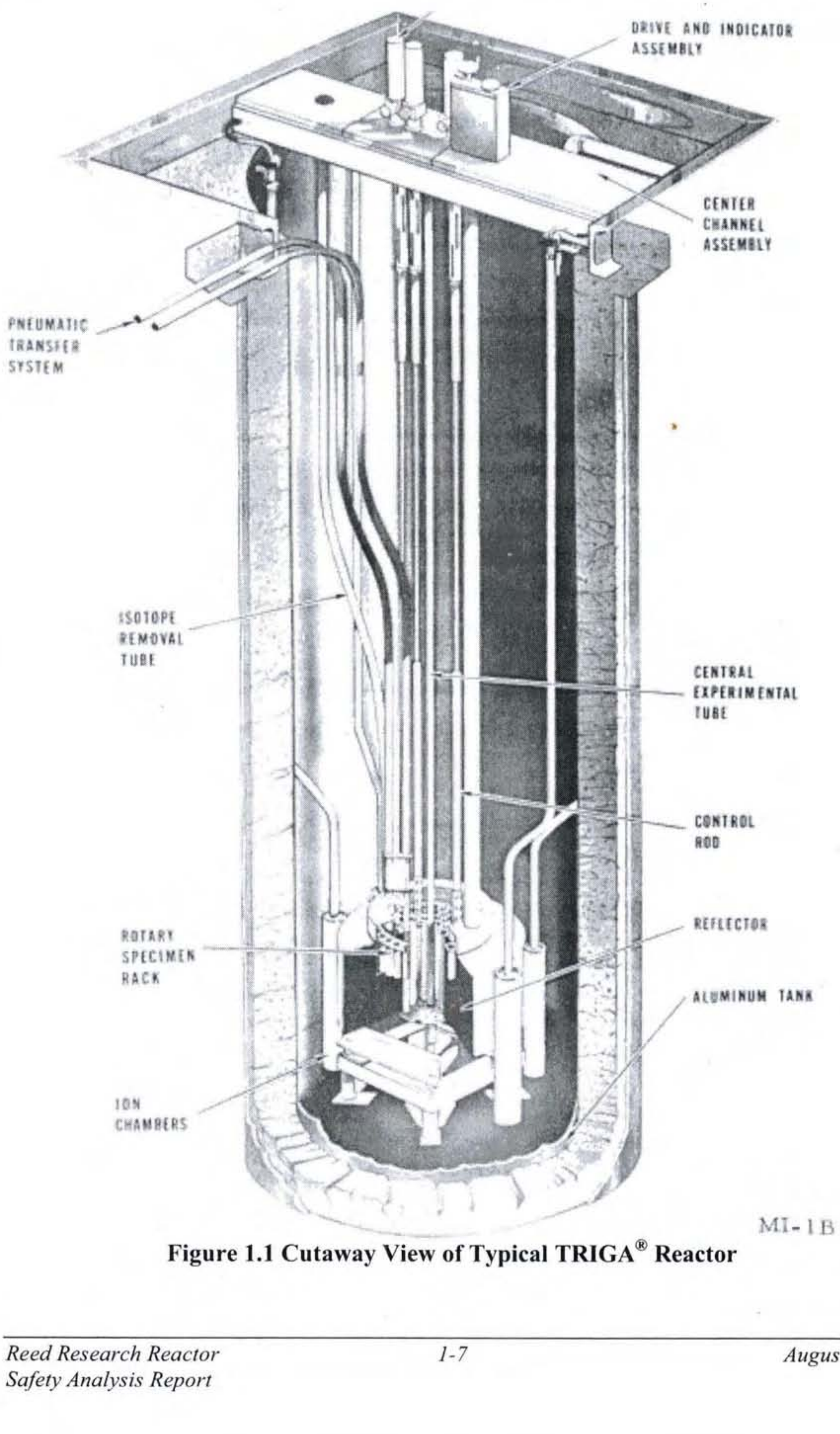

Figure 1.1 Cutaway View of Typical TRIGA® Reactor

1.3.5.2 Radiation Safety Monitoring Systems

Radiation monitors are installed to monitor radiological conditions at the facility. One monitor is stationed at the southwest side of the pool with a 2 mR/hr alarm. A continuous air monitor measures radioactive particulates in the bay and will trip ventilation isolation when alarmed. Ventilation is described in Chapter 9, Auxiliary Systems.

1.3.5.3 Electrical Power

Primary electrical power is provided through the Reed College power grid, supplied by commercial generators. Loss of electrical power will de-energize the control rod drives, causing the rods to fall by gravity into the core and placing the reactor in a subcritical configuration. Since the core is cooled by natural convection, no emergency power is required for reactor cooling systems. Loss of electrical power does not represent a potential hazard to the reactor. Backup battery systems are provided for required emergency lighting [redacted]

1.3.5.4 Reactor Protection System

The reactor protection system is designed to ensure reactor and personnel safety by initiating a scram if the reactor exceeds operating parameters. Two power meters can initiate a scram if measured power exceeds 110% of licensed power. A bar above the control rod drive switches allows the scram system to be actuated manually by the reactor operator at the controls.

1.3.6 Reactor Coolant and Other Auxiliary Systems

The reactor coolant and auxiliary systems are very simple in design and operation. Detailed descriptions of the coolant and auxiliary systems equipment and operation are provided in Chapters 4, 5, and 13 of this report.

1.3.6.1 Reactor Coolant System

During full power operation, the nuclear fuel elements in the reactor core are cooled by natural convection of the primary tank water. To remove the bulk heat to the environment, the primary water is circulated through a heat exchanger where the heat is transferred to a secondary cooling loop. A cleanup loop maintains primary water purity with a filter and demineralizer to minimize corrosion and production of long-lived radionuclides that could otherwise occur. The primary coolant provides shielding directly above the reactor core.

1.3.6.2 Secondary Cooling System

The secondary cooling system provides the interface for heat rejection from the primary coolant system to the environment. The secondary system is an open system, with the secondary pump discharging through a primary-to-secondary heat exchanger, then through a forced-draft cooling tower.

1.3.6.3 Makeup

Makeup water is provided from the municipal water supply and run through a purification filter before being added to the pool. Secondary makeup water comes directly from the municipal water supply.

1.3.7 Radioactive Waste Management and Radiation Protection

Operation of the RRR TRIGA® produces (low concentration) routine discharges of radioactive gases and small quantities of solid waste. Details of the waste management and radiation protection procedures at the reactor are provided in Chapter 11, Radiation Protection and Waste Management, of this report.

1.3.7.1 Gaseous Waste

Maintaining negative pressure in the reactor bay controls concentrations of radioactive gases during operations. An exhaust fan maintains negative pressure in the reactor bay to ensure that discharges are controlled under analyze conditions.

1.3.7.2 Liquid Waste

The RRR facility does not regularly create or release liquid waste.

1.3.7.3 Solid Waste

Solid waste is very limited in volume and specific activity. Solid wastes include ion-exchange resin used in reactor-water cleanup, contaminated tools, lab-ware, samples and sample handling material for completed experiments, and anti-contamination clothing associated with reactor experiments and surveillance or maintenance operations. Shipments of solid waste to commercial disposal facilities are made infrequently. Solid waste shipments are coordinated with the Environmental Health and Safety Office.

1.3.8 Experimental Facilities and Capabilities

Standard experimental facilities at the RRR TRIGA®, as supplied by the vendor, General Atomics, include the central thimble, rotary specimen rack, and pneumatic specimen tube. Samples can also be lowered into the pool near the core for individually designed in-pool irradiations. Experimental facilities are described in Chapter 10, Experimental Facilities and Utilization.

1.3.8.1 Central Thimble

The reactor is equipped with a central thimble for access to the point of maximum flux in the core. The central thimble consists of an aluminum tube that fits through the center holes of the top and bottom grid plates terminating with a plug below the lower grid plate. The tube is anodized to retard corrosion and wear. The thimble is approximately 20 feet (6.1 m) in length, made in two sections, with a watertight tube fitting. Although the shield water may be removed to allow extraction of a vertical thermal-neutron and gamma-ray beam, four 0.25 inch (6.3 mm) holes are located in the tube at the top of the core to prevent expulsion of water from the section of the tube within the reactor core.

1.3.8.2 Rotary Specimen Rack

A 40-position rotary specimen rack (RSR) is located in a well in the top of the graphite radial reflector. A rotation mechanism and housing at the top of the reactor allows the specimens to be loaded into indexed positions and also allows rotation of samples for more uniform exposure across a set of co-irradiated samples. The RSR allows large-scale production of radioisotopes and for activation and irradiation of multiple material samples with neutron and gamma ray flux densities of comparable intensity.

1.3.8.3 Pneumatic Specimen Tube

A pneumatic transfer system, permitting applications with short-lived radioisotopes, rapidly conveys a specimen from the reactor core to a remote receiver. The in-core terminus is located at location F-5 in the outer ring of fuel-element positions.

1.4 Shared Facilities and Equipment

Electrical systems are serviced by the Reed College power grid, as described in Chapter 8, Electrical Power Systems. Building heating and ventilation systems use centralized campus supplies for steam heating. A description of environmental controls is provided in Chapter 9, Auxiliary Systems. Potable water is provided to a deep sink in the reactor bay, which discharges to sewerage. Water and sewerage is addressed in Chapter 3, Design of Structures, System, and Components, with controls on discharge to sewerage addressed in Chapter 11, Radiation Protection and Waste Management.

1.5 Comparison with Similar Facilities

The design of the fuel for the RRR TRIGA® is similar to that for fuels used in 70 reactors in 24 nations [2]. Of the total number of reactors, 45 are currently in operation or under construction with 22 rated for steady-state thermal powers of 500 kW or greater.

In the United States 26 TRIGA® reactors have been built, with 19 currently in operation.

Major design parameters for the RRR TRIGA® are given in Table 1.1. Fuel for the RRR is standard TRIGA® fuel having 8.5% of uranium, by weight, enriched up to 20% in the uranium-235 isotope. 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 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 and terminated quickly and safely.

Table 1.1 Major Design Parameters

Parameter	Value
Max steady-state thermal power	500 kW
Maximum excess reactivity	2.8 % Δk/k
Number of control rods	3
Regulating rods	1
Shim rods	2
Minimum shutdown margin	0.7 % Δk/k
Integral fuel-moderator material	$U\text{-}ZrH_{1.6\text{-}1.7}$
Reactor cooling	Natural convection
Number fuel elements	83
Uranium enrichment	Up to 20% uranium-235
Uranium content	8.5%
Shape	Cylindrical
Length	[redacted]
Diameter	[redacted]
Cladding	0.51 cm (0.020 in) 304 SS or Aluminum
External moderator	Light water

1.6 Summary of Operations

The RRR facility is a unique and valuable tool for a wide variety of research and educational applications for the Reed community and the greater Portland area. The reactor is normally operated for up to a few hours each workday for educational purposes with longer runs as appropriate for experiments. The usual power level is 240 kW. The average energy output per year is on average less than 50 MW-hours. According to the analysis in this report, there are no limitations on operating schedule. The operating history for a typical year is shown in Table 1.2.

Table 1.2 Operating History 2005-2006

Parameter	Value
Times Critical	340
Days Operated	120
MW-hrs	42
Irradiation Requests	45

1.7 Compliance with Nuclear Waste Policy Act of 1982

Compliance with Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982 for disposal of high-level radioactive waste and spent nuclear fuel is effected through Fuel Assistance Contract DE-AC06-76ER02063 between Reed College and the U.S. Department of Energy. [2] The DOE retains title to the fuel and is obligated to take spent fuel and/or high-level waste for storage and reprocessing.

1.8 Facility Modifications and History

Criticality was first achieved in 1968.

All neutron instrumentation has been replaced, over the course of 1998-2000, with new meters from Sorrento nuclear instruments. Various other upgrades have been performed, the major ones being summarized in tabular form (Table 1.3) to illustrate the timeline.

Table 1.3 Major Facility Modifications

Year	Activity
1968	Construction completed, fuel loaded, initial criticality
1994	Replaced heat exchanger with plate-type system, and installed new secondary pump and cooling tower to replace lake-based cooling
1995	Added supplemental HV power to linear and log-n channels
1998	Replaced linear channel display with new Sorrento NMP-1000 meter
2000	Replaced percent power and log-n meters with Sorrento NP-1000 and NLW-1000
2001	Upgraded facility security system
2003	Installed a Honeywell Multitrend for data logging to replace chart recorders

1.9 References

1) NUREG-1282, "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA® Reactors," U.S. Nuclear Regulatory Commission, 1987.
2) US Department of Energy Fuel Assistance Contract DE-AC06-76ER02063

STANDARD FORM 30, JULY 1966
GENERAL SERVICES ADMINISTRATION
FED. PROC. REG. (41 CFR) 1-16.101

AMENDMENT OF SOLICITATION/MODIFICATION OF CONTRACT

1. AMENDMENT/MODIFICATION NO. M011
2. EFFECTIVE DATE
3. REQUISITION/PURCHASE REQUEST NO.
4. PROJECT NO. (If applicable)

5. ISSUED BY CODE
U. S. Department of Energy
Richland Operations Office
P.O. Box 550
Richland, WA 99352

6. ADMINISTERED BY (If other than block 5) CODE

7. CONTRACTOR NAME AND ADDRESS CODE FACILITY CODE
(Street, city, county, state, and ZIP Code)
Reed College
3203 S.E. Woodstock
Portland, OR 97202

8.
☐ AMENDMENT OF SOLICITATION NO. ______
DATED ______ (See block 9)
☒ MODIFICATION OF CONTRACT/ORDER NO. DE-AC06-76ER02063
DATED 6/01/68 (See block 11)

9. THIS BLOCK APPLIES ONLY TO AMENDMENTS OF SOLICITATIONS
☐ The above numbered solicitation is amended as set forth in block 12. The hour and date specified for receipt of Offers ☐ is extended, ☐ is not extended.
Offerors must acknowledge receipt of this amendment prior to the hour and date specified in the solicitation, or as amended, by one of the following methods:
(a) By signing and returning ____ copies of this amendment; (b) By acknowledging receipt of this amendment on each copy of the offer submitted; or (c) By separate letter or telegram which includes a reference to the solicitation and amendment numbers. FAILURE OF YOUR ACKNOWLEDGEMENT TO BE RECEIVED AT THE ISSUING OFFICE PRIOR TO THE HOUR AND DATE SPECIFIED MAY RESULT IN REJECTION OF YOUR OFFER. If, by virtue of this amendment you desire to change an offer already submitted, such change may be made by telegram or letter, provided such telegram or letter makes reference to the solicitation and this amendment, and is received prior to the opening hour and date specified.

10. ACCOUNTING AND APPROPRIATION DATA (If required)

11. THIS BLOCK APPLIES ONLY TO MODIFICATIONS OF CONTRACTS/ORDERS
(a) ☐ This Change Order is issued pursuant to ______
The Changes set forth in block 12 are made to the above numbered contract/order.
(b) ☐ The above numbered contract/order is modified to reflect the administrative changes (such as changes in paying office, appropriation data, etc.) set forth in block 12.
(c) ☒ This Supplemental Agreement is entered into pursuant to authority of The Atomic Energy Act of 1954, as amended, and Public Law of 95-91
It modifies the above numbered contract as set forth in block 12.

12. DESCRIPTION OF AMENDMENT/MODIFICATION

1. The period of performance's expiration date specified in Article III is corrected to read June 30, 1988, rather than June 3, 1988, as typed in M010.

2. Item 1. of Article IV - <u>CONSIDERATION</u> is revised to read as follows in order to include the nuclear material in two new fuel elements recently provided to Reed College by DOE:

   DOE will loan to the Contractor without charge, pursuant to 3. below, for use, burnup and normal loss approximately 2.294 kg of uranium - 235 at an enrichment of less than 20% U^{235} contained in reactor fuel and approximately [redacted] grams of uranium-235 at an enrichment of approximately 93% contained in the fission chambers.

Continued on Page 2

Except as provided herein, all terms and conditions of the document referenced in block 8, as heretofore changed, remain unchanged and in full force and effect.

13. ☐ CONTRACTOR/OFFEROR IS NOT REQUIRED TO SIGN THIS DOCUMENT ☒ CONTRACTOR/OFFEROR IS REQUIRED TO SIGN THIS DOCUMENT AND RETURN 4 COPIES TO ISSUING OFFICE.

14. NAME OF CONTRACTOR/OFFEROR
Marshall W. Cronyn
(Signature of person authorized to sign)

17. UNITED STATES OF AMERICA
BY
(Signature of Contracting Officer)

15. NAME AND TITLE OF SIGNER (Type or print)
Marshall W. Cronyn
Vice President - Provost

16. DATE SIGNED 7/14/84

18. NAME OF CONTRACTING OFFICER (Type or print)

19. DATE SIGNED 3/15/84

★U.S.GPO:1980-0-311-153/5520

3. The document entitled "Special Terms Relating to Special Nuclear Material Furnished to Educational Institutions," made a part of the contract by Article IV, is deleted and the attached updated "Special Terms Relating to Special Nuclear Materials Furnished to Educational Institutions" is made a part of this contract.

4. Attachment 1 relating to the dissemination of scientific and technical information is deleted and the attached new Attachment 1 is made a part of the contract.

CONTRACT NO. AT(45-1)-2063

TITLE: RESEARCH REACTOR ASSISTANCE

THIS CONTRACT, entered into as of the 19th day of June, 1968, effective as of the First day of June, 1968, by and between the UNITED STATES OF AMERICA (hereinafter referred to as the "Government"), as represented by the UNITED STATES ATOMIC ENERGY COMMISSION (hereinafter referred to as the "Commission"), and THE REED INSTITUTE OPERATING REED COLLEGE (hereinafter referred to as the "Contractor").

WITNESSETH THAT:

WHEREAS, the Commission desires to provide the Contractor assistance in the reactor program, as hereinafter provided; and

WHEREAS, this contract is authorized by law, including the Atomic Energy Act of 1954;

NOW, THEREFORE, the parties agree as follows:

ARTICLE I - DEFINITIONS

1. The term "Contracting Officer" means the person executing this contract on behalf of the Government and includes his successor or any duly authorized representative of such person.

2. The term "Commission" means the United States Atomic Energy Commission or any duly authorized representative thereof, including the Contracting Officer, except for the purpose of deciding an appeal under the article entitled "Disputes".

3. The term "reactor" means the pool-type nuclear reactor owned by the Contractor and located in Multnomah County near Portland, Oregon.

ARTICLE II - SCOPE OF WORK

The Contractor shall, in accordance with its proposals dated April 15, 1967, May 2, 1967 and April 11, 1968, conduct programs of research and education in nuclear science and engineering utilizing the reactor which it shall have built and installed on its campus.

ARTICLE III - THE PERIOD OF PERFORMANCE

The term of this contract shall commence on June 1, 1968 and shall end on May 31, 1969; provided, however, that the term of this contract may be extended for additional periods by the mutual written agreement of the parties. EXTENDED

ARTICLE IV - CONSIDERATION

In consideration of the performance by the Contractor of the research and development activities described in Article II and other obligations assumed by the Contractor under this contract:

1. The Commission will loan to the Contractor, without charge, pursuant to 3. below, for use, burnup and normal loss, approximately [redacted] kg of uranium-235 at an enrichment of approximately 20% contained in reactor fuel and a small amount of uranium-235 at an enrichment of approximately 93% contained in fission chambers.

2. The Commission will reimburse the Contractor in a sum equal to the invoice cost but not to exceed $52,000 for the fabrication of fuel elements.

3. The Commission will waive its charges for use, burnup and normal (non-negligent) loss of the uranium-235 contained in the fuel elements and fission chambers. The waiver of charges is to commence at the time of transfer of the fabricated materials to the Contractor. The Commission will also waive its charges for the quantities of uranium-235 not recovered in reprocessing performed subsequent to the return of the materials if such reprocessing is performed in Commission facilities.

4. The Commission will, during the term of this contract, accept at such places as the Commission may determine, return of the fuel elements without charge to the Contractor, for reprocessing or will arrange to provide funds for reprocessing in commercial facilities.

5. The Commission will, during the term of this contract, reimburse the Contractor for costs incurred in returning spent fuel elements for reprocessing, including rental or fabrication of shipping containers, as mutually agreed to by the parties.

6. The financial assistance to be provided by the Commission is contingent upon the Contractor obtaining the necessary licenses to construct and operate the reactor.

ARTICLE V - SPECIAL NUCLEAR MATERIAL

The document entitled "Special Terms Relating to Special Nuclear Material Furnished to Educational Institutions," attached hereto, is hereby incorporated into this contract by reference.

ARTICLE VI - REPORTS

The Contractor will furnish the Commission with a current list of all published reports embodying the results of activities involving the facility, and upon Commission request will furnish the Commission with copies of the reports themselves. Further, the Contractor will furnish the Commission, at its request and without cost to it, reports of results of the Contractor's other activities or investigations involving the facility. Information contained in such reports may be used by the Commission and may be distributed by it for use by others.

Annual Report
David Jones
~~to~~ Richland Office
10/3/68

ARTICLE VII - PATENTS

1. Whenever after delivery of the material to the Contractor under this contract, and during the term of this contract, any invention or discovery is made or conceived by the Contractor or its employees:

a. Relating to improvements in the design, construction or operation of the facility in which the material is used; or

b. Relating to a method or process useful in the production or utilization of special nuclear material, resulting in whole or in part from work involving the use of the material but not inventions resulting from the utilization of radiation from this material in the performance of work for others than the Commission.

The Contractor shall furnish the Commission with complete information thereon; and the Commission shall have the sole power to determine whether or not and where a patent application shall be filed, and to determine the disposition of the title to and rights under any application or patent that may result. The judgment of the Commission on these matters shall be accepted as final; and the Contractor, for itself and for its employees, agrees that the inventor or inventors will execute all documents and do all things necessary or proper to carry out the judgment of the Commission.

2. No claim for pecuniary award or compensation under the provisions of the Atomic Energy Acts of 1946 and 1954 shall be asserted by the Contractor or its employees with respect to any invention or discovery covered by the foregoing paragraph.

ARTICLE VIII - PATENT INDEMNITY

The Contractor agrees to include in all subcontracts or purchase orders relating to the fuel elements a provision indemnifying the Government against liability for the use of any invention or discovery and for the infringement of any Letters Patent arising by reason of the purchase, use or disposal of items manufactured or supplied, or services performed under the purchase order or subcontract.

ARTICLE IX - DISCLOSURE OF INFORMATION

1. It is mutually expected that the activities under this contract will not involve Restricted Data or other classified information or material. It is understood, however, that if in the opinion of either party this expectation changes prior to the expiration or termination of all activities under this contract, said party shall notify the other party accordingly in writing without delay. In any event, the Contractor shall classify, safeguard, and otherwise act with respect to all Restricted Data and other classified information and material in accordance with applicable law and the requirements of the Commission, and shall promptly inform the Commission in writing if and when Restricted Data or other classified information or material becomes involved. If and when Restricted Data or other classified information or material becomes involved, or in the mutual judgment of the parties it appears likely that Restricted Data or other classified information or material may become involved, the Contractor shall have the right to terminate performance of the work under this contract and in such event the provisions of this contract respecting termination for the convenience of the Government shall apply.

2. The Contractor shall not permit any individual to have access to Restricted Data or other classified information, except in accordance with the Atomic Energy Act of 1954, as amended, and the Commission's regulations or requirements.

3. The term "Restricted Data" as used in this Article means all data concerning the design, manufacture, or utilization of atomic weapons, the production of special nuclear material, or the use of special nuclear material in the production of energy, but shall not include data declassified or removed from the Restricted Data category pursuant to section 142 of the Atomic Energy Act of 1954, as amended.

ARTICLE X - DISPUTES

1. Except as otherwise provided in this contract, any dispute concerning a question of fact arising under this contract which is not disposed of by agreement shall be decided by the Contracting Officer, who shall reduce his decision to writing and mail or otherwise furnish a copy thereof to the Contractor. The decision of the Contracting Officer shall be final and conclusive unless within thirty days from the date of receipt of such copy, the Contractor mails or otherwise furnishes to the Contracting Officer a written appeal addressed to the Commission. The decision of the Commission or its duly authorized representative for the determination of such appeals shall be final and conclusive unless determined by a court of competent jurisdiction to have been fraudulent, or capricious or arbitrary, or so grossly erroneous as necessarily to imply bad faith, or not supported by substantial evidence. In connection with any appeal proceeding under this clause, the Contractor shall be afforded an opportunity to be heard and to offer evidence in support of its appeal. Pending final decision of a dispute hereunder, the Contractor shall proceed diligently with the performance of the contract and in accordance with the Contracting Officer's decision.

2. This "Disputes" clause does not preclude consideration of law questions in connection with decisions provided for in paragraph 1 above: Provided, That nothing in this contract shall be construed as making final the decision of any administrative official, representative, or board on a question of law.

ARTICLE XI - OFFICIALS NOT TO BENEFIT

No member of or delegate to Congress or resident commissioner shall be admitted to any share or part of this contract or to any benefit that may arise therefrom, but this provision shall not be construed to extend to this contract if made with a corporation for its general benefit.

ARTICLE XII - EQUAL OPPORTUNITY

During the performance of this contract, the Contractor agrees as follows:

1. The Contractor will not discriminate against any employee or applicant for employment because of race, creed, color or national origin. The Contractor will take affirmative action to ensure that applicants are employed, and that employees are treated during employment, without regard to their race, creed, color or national origin. Such action shall include, but not be limited to, the following: Employment, upgrading, demotion or transfer; recruitment or recruitment advertising; layoff or termination; rates of pay or other forms of compensation; and selection for training, including apprenticeship. The Contractor agrees to post in conspicuous

places, available to employees and applicants for employment, notices to be provided by the Contracting Officer setting forth the provisions of this nondiscrimination clause.

2. The Contractor will, in all solicitations or advertisements for employees placed by or on behalf of the Contractor, state that all qualified applicants will receive consideration for employment without regard to race, creed, color, or national origin.

3. The Contractor will send to each labor union or representative of workers with which he has a collective bargaining agreement or other contract or understanding, a notice, to be provided by the agency contracting officer, advising the labor union or workers' representative of the Contractor's commitments under Section 202 of Executive Order No. 11,246 of September 24, 1965, and shall post copies of the notice in conspicuous places available to employees and applicants for employment.

4. The Contractor will comply with all provisions of Executive Order 11,246 of September 24, 1965, and of the rules, regulations, and relevant orders of the Secretary of Labor.

5. The Contractor will furnish all information and reports required by Executive Order No. 11,246 of September 24, 1965, and by the rules, regulations, and orders of the Secretary of Labor, or pursuant thereto, and will permit access to his books, records, and accounts by the contracting agency and the Secretary of Labor for purposes of investigation to ascertain compliance with such rules, regulations, and orders.

6. In the event of the Contractor's noncompliance with the nondiscrimination clauses of this contract or with any of such rules, regulations, or orders, this contract may be cancelled, terminated, or suspended in whole or in part and the Contractor may be declared ineligible for further Government contracts in accordance with procedures authorized in Executive Order No. 11,246 of September 24, 1965, and such other sanctions may be imposed and remedies involved as provided in Executive Order No. 11,246 of September 24, 1965, or by rule, regulation, or order of the Secretary of Labor, or as otherwise provided by law.

7. The Contractor will include the provisions of Paragraphs (1) through (7) in every subcontract or purchase order unless exempted by rules, regulations, or orders of the Secretary of Labor issued pursuant to Section 204 of Executive Order No. 11,246 of September 24, 1965, so that such provisions will be binding upon each subcontractor or vendor. The Contractor will take such action with respect to any subcontract or purchase order as the contracting agency may direct as a means of enforcing such provisions including sanctions for noncompliance: Provided, however, that in the event the Contractor becomes involved in, or is threatened with, litigation with a subcontractor or vendor as a result of such direction by the contracting agency, the Contractor may request the United States to enter into such litigation to protect the interests of the United States.

ARTICLE XIII - CONVICT LABOR

In connection with the performance of work under this contract, the Contractor agrees not to employ any person undergoing sentence of imprisonment at hard labor

ARTICLE XIV - CONTRACT WORK HOURS STANDARDS ACT - OVERTIME COMPENSATION

This contract, to the extent that it is of a character specified in the Contract Work Hours Standards Act (40 U.S.C. 327-330), is subject to the following provisions and to all other applicable provisions and exceptions of such Act and the regulations of the Secretary of Labor thereunder.

1. Overtime requirements. No contractor or subcontractor contracting for any part of the contract work which may require or involve the employment of laborers or mechanics shall require or permit any laborer or mechanic in any workweek in which he is employed on such work to work in excess of eight hours in any calendar day or in excess of forty hours in such workweek on work subject to the provisions of the Contract Work Hours Standards Act unless such laborer or mechanic receives compensation at a rate not less than one and one-half times his basic rate of pay for all such hours worked in excess of eight hours in any calendar day or in excess of forty hours in such workweek, whichever is the greater number of overtime hours.

2. Violation; liability for unpaid wages; liquidated damages. In the event of any violation of the provisions of paragraph 1, the Contractor and any subcontractor responsible therefor shall be liable to any affected employee for his unpaid wages. In addition, such Contractor and subcontractor shall be liable to the United States for liquidated damages. Such liquidated damages shall be computed with respect to each individual laborer or mechanic employed in violation of the provisions of paragraph 1 in the sum of $10 for each calendar day on which such employee was required or permitted to be employed on such work in excess of eight hours or in excess of the standard workweek of forty hours without payment of the overtime wages required by paragraph 1.

3. Withholding for unpaid wages and liquidated damages. The Contracting Officer may withhold from the Government Prime Contractor, from any moneys payable on account of work performed by the Contractor or subcontractor, such sums as may administratively be determined to be necessary to satisfy any liabilities of such Contractor or subcontractor for unpaid wages and liquidated damages as provided in the provisions of paragraph 2.

4. Subcontracts. The Contractor shall insert paragraphs 1 through 4 of this clause in all subcontracts, and shall require their inclusion in all subcontracts of any tier.

5. Record. The Contractor shall maintain payroll records containing the information specified in 29 CFR 516.2 (a). Such records shall be preserved for three years from the completion of the contract.

ARTICLE XV - COVENANT AGAINST CONTINGENT FEES

The Contractor warrants that no person or selling agency has been employed or retained to solicit or secure this contract upon an agreement or understanding for a commission, percentage, brokerage, or contingent fee, excepting bona fide employees or bona fide established commercial or selling agencies maintained by the Contractor for the purpose of securing business. For breach or violation of this warranty the Government shall have the right to annul this contract without liability or in its discretion to deduct from the contract price or consideration, or otherwise recover, the full amount of such commission, percentage, brokerage or contingent fee.

ARTICLE XVI - PAYMENTS

The Commission will pay the amounts prescribed in Article IV hereof upon submission by the Contractor of such invoices or vouchers as are satisfactory to the Commission. Such payments shall not prejudice or otherwise affect adversely any of the Government's rights under the contract.

ARTICLE XVII - EXAMINATION OF RECORDS

1. The Contractor agrees that the Comptroller General of the United States or any of his duly authorized representatives shall, until the expiration of three years after final payment under this contract, have access to and the right to examine any directly pertinent books, documents, papers and records of the Contractor involving transactions related to this contract.

2. The Contractor further agrees to include in all his subcontracts hereunder a provision to the effect that the subcontractor agrees that the Comptroller General of the United States or any of his duly authorized representatives shall, until the expiration of three years after final payment under the subcontract, have access to and the right to examine any directly pertinent books, documents, papers, and records of such subcontractor, involving transactions related to the subcontract. The term "subcontract" as used in this clause excludes (a) purchase orders not exceeding $2,500 and (b) subcontracts or purchase orders for public utility services at rates established for uniform applicability to the general public.

ARTICLE XVIII - CIVIL RIGHTS ACT OF 1964

The Contractor agrees to comply with the Atomic Energy Commission's Regulation (Part 4 of Title 10, Chapter I, Code of Federal Regulations) as amended, effectuating the provisions of Title VI of the Civil Rights Act of 1964.

ARTICLE XIX - SOVIET BLOC CONTROLS

In connection with the contract activities, the Contractor agrees to comply with the requirements set forth in Attachment 1 of this contract relating to the countries listed herein. From time to time, by written notice to the Contractor, the Commission shall have the right to change the listing of countries in Attachment 1 upon a determination by the Commission that such change is in conformance with national policy. The Contractor shall have the right to terminate its performance under this contract upon at least sixty (60) days prior written notice to the Commission if the Contractor determines that it is unable without substantially interfering with its policies as an educational institution or without adversely affecting its performance, to continue performance of the work under this contract as a result of a change in Attachment 1 made by the Commission pursuant to the preceding sentence. If the Contractor elects to terminate performance, the provisions of this contract respecting termination for the convenience of the Government shall apply.

IN WITNESS WHEREOF, the parties have executed this contract.

UNITED STATES OF AMERICA

UNITED STATES ATOMIC ENERGY COMMISSION

By: C. R. Gualheim

Title: ACTING Director
Laboratory & University Division

THE REED INSTITUTE OPERATING
REED COLLEGE

By: Ross B. Thompson

Title: Acting President

Chapter 2

Site Characteristics

Reed Research Reactor Safety Analysis Report

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Chapter 2

Site Characteristics

Table of Contents

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2 SITE CHARACTERISTICS

This chapter describes the site characteristics of the Reed Research Reactor (RRR) on the Reed College campus and their relation to the safety and operation of the reactor.

2.1 Geography and Demography

2.1.1 Site Location and Description

2.1.1.1 Specification and Location

The reactor is located on the campus of Reed College, in the City of Portland, in Multnomah County, Oregon. Portland is a major city on the junction of the Willamette and Columbia rivers, just across the border from the state of Washington and 50 miles north-northwest of Salem, the state capital. The Reed College campus is approximately 100 acres, and is located in the southeastern section of Portland in the Eastmoreland neighborhood.

The RRR is located on the east side of the Reed campus. The reactor is south of the Reed Lake, the nearest body of water. The location of the reactor site relative to major highways and bodies of water can be seen in Figure 2.2.

The latitude and longitude of the RRR is [redacted] In Universal Transverse Mercator coordinates the site is at [redacted]

2.1.1.2 Boundary and Zone Area Maps

Figures 2.1 and 2.2 illustrate the location of the RRR with respect to the State of Oregon and the city of Portland. Figure 2.3 illustrates the location of the RRR within the Reed College campus.

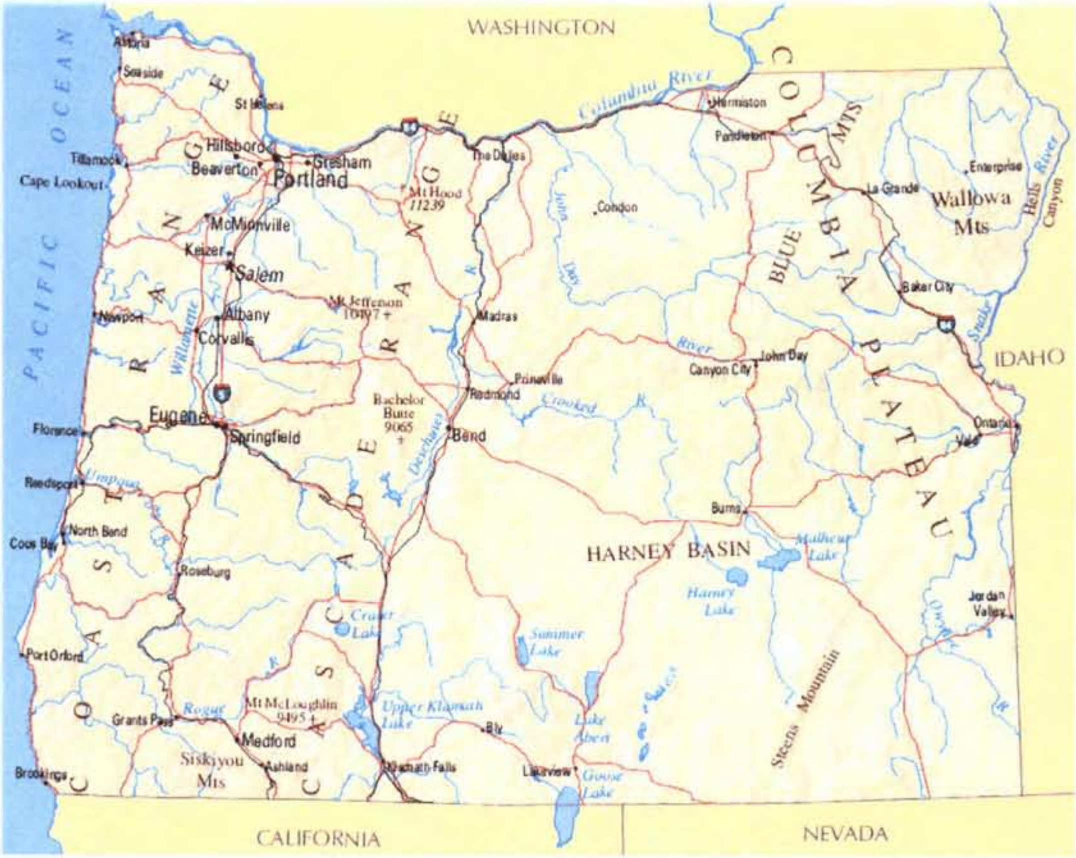


Figure 2.1 State Map of Oregon

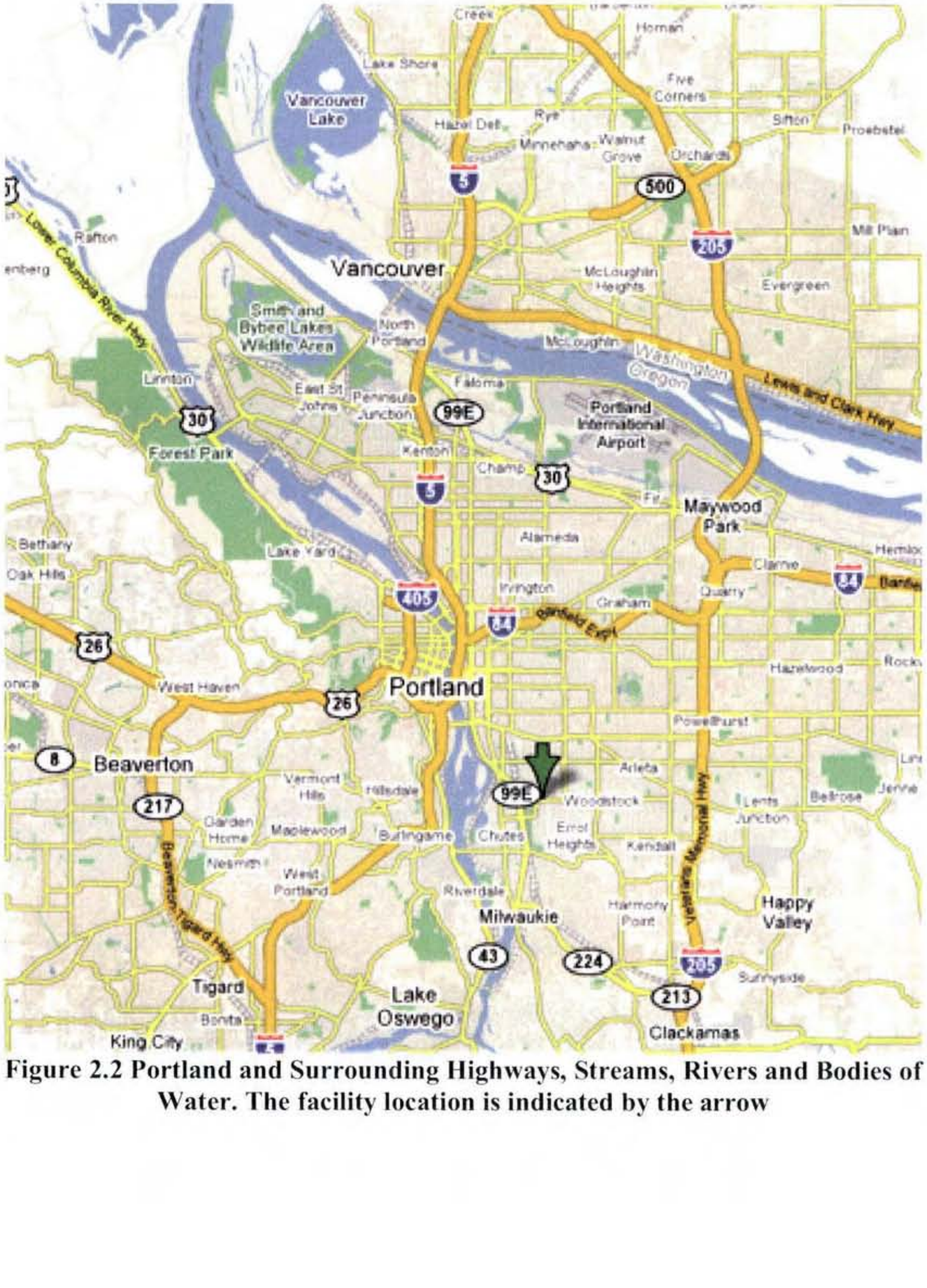
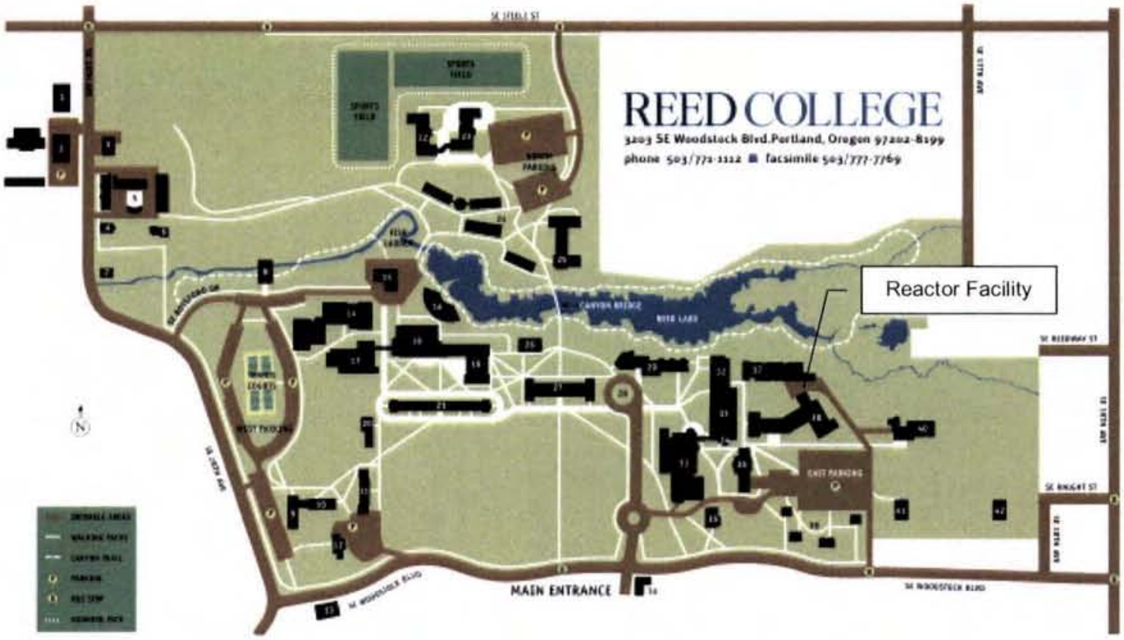


Figure 2.2 Portland and Surrounding Highways, Streams, Rivers and Bodies of Water. The facility location is indicated by the arrow



1 Birchwood apartments
2 Theatre annex, Reed warehouse
3 28 West: community safety, residence life, SEEDS, switchboard
4 Garden House (residence hall)
5 Reed College apartments
6 Chinese House (residence hall)
7 Farm House (residence hall)
8 Theatre
9 Scholz (residence hall)
10 Foster (residence hall)
11 MacNaughton (residence hall)
12 Prexy (music building)
13 Parker House
14 Watzek Sports Center: dance studio, fitness room, gymnasiums, mat room, pool, racquetball courts, squash courts
15 Physical plant: facilities services
16 Cerf Amphitheatre
GRAY CAMPUS CENTER (17–19)
17 Kaul Auditorium, Gray Center Lounge, conference and events planning office
18 Bookstore, commons dining room, conference and private dining rooms, convenience store, food service office, mail services, student activities office, student organizations offices
19 Student union: judicial board, Paradox Café, Sound Kollektiv, student body offices
20 Anna Mann (residence hall)
21 Old Dorm Block (residence halls, W to E): Ladd, Abington, Kerr, Westport, Eastport, Doyle, Quincy, Winch (Capehart)
22 Naito Hall (residence hall)
23 Sullivan Hall (residence hall)
24 Griffin, McKinley, Woodbridge, Chittick (residence halls, W to E)
25 Bragdon Hall (residence hall)
26 Health and counseling center
27 Eliot Hall: chapel; classrooms; faculty offices; offices of admission, business, college relations, controller, dean of the faculty, dean of student services, development, financial aid, human resources, institutional research, international student programs, president, printing services, public affairs, registrar, special programs, treasurer
28 Eliot Circle
29 Vollum College Center: classrooms, faculty offices, lecture hall, lounge
30 Willard House
31 Hauser Memorial Library: classrooms, Douglas F. Cooley Memorial Art Gallery, faculty offices, instructional media center, thesis tower
32 Knowlton Laboratory of Physics: classrooms, faculty offices, labs
33 Griffin Memorial Biology Laboratory: classrooms, faculty offices, labs
34 Paradox Lost Café
35 Greywood: alumni & parent relations, campus information, career services, CRIS
36 Educational Technology Center (ETC): classrooms, computing and information services, faculty multimedia lab, faculty offices, information resource centers (IRCs), telecommunications
37 Scott Laboratory of Chemistry: classrooms, faculty offices, labs
38 Psychology: classrooms, faculty offices, labs
39 Woodstock language houses (residence halls, W to E): Russian, German, French, Spanish
40 Studio art: ceramics studio, gallery
41 Johansen House: faculty offices, quantitative skills center, science center, writing center
42 Center for Advanced Computation

Figure 2.3 Reactor Facility Location within the Campus

2.1.1.3 Population Distribution

Portland is a major metropolitan center of Oregon state, with a population of approximately 540,000 in 2003, with a growth of 1.8% from 2000 to 2003. In the 2000 census, Portland had a population of 529,121. The city is situated in Multnomah County, across the Columbia river from Vancouver, WA, which had an approximate population of 150,000 in 2003 and a growth of 5.6% from 2000 to 2003. Table 2.1 summarizes population data from the 2000 census [1] for Portland and the surrounding areas. A population density map [2] for the Portland metropolitan region is given in Figure 2.4.

Table 2.1 Population and Location of Surrounding Communities

City	Direction from Reed Campus	Distance (air miles)	Population
Beaverton	W	8.4	76,129
Camas, WA	NE	13.4	12,534
Clackamas	SE	5.8	6,177
Gladstone	SSE	7.1	11,438
Gresham	E	9.8	90,205
Hillsboro	WNW	17.7	70,186
Lake Oswego	S	4.6	35,278
Milwaukie	SSE	2.4	20,490
Oregon City	SSE	8.6	25,754
Tigard	SW	7.7	41,223
Tualatin	SSW	9.3	22,791
Vancouver, WA	N	11.0	143,560
West Linn	S	8.0	22,261

The nearest permanent residences to the reactor are about 700 feet (215 m) from the reactor, located in both the northeast and south directions. A grouping of Reed College dormitories, housing around 30 students from August to May, are located approximately 500 feet (150 m) south the reactor. Locations of campus buildings are shown in Figure 2.3.

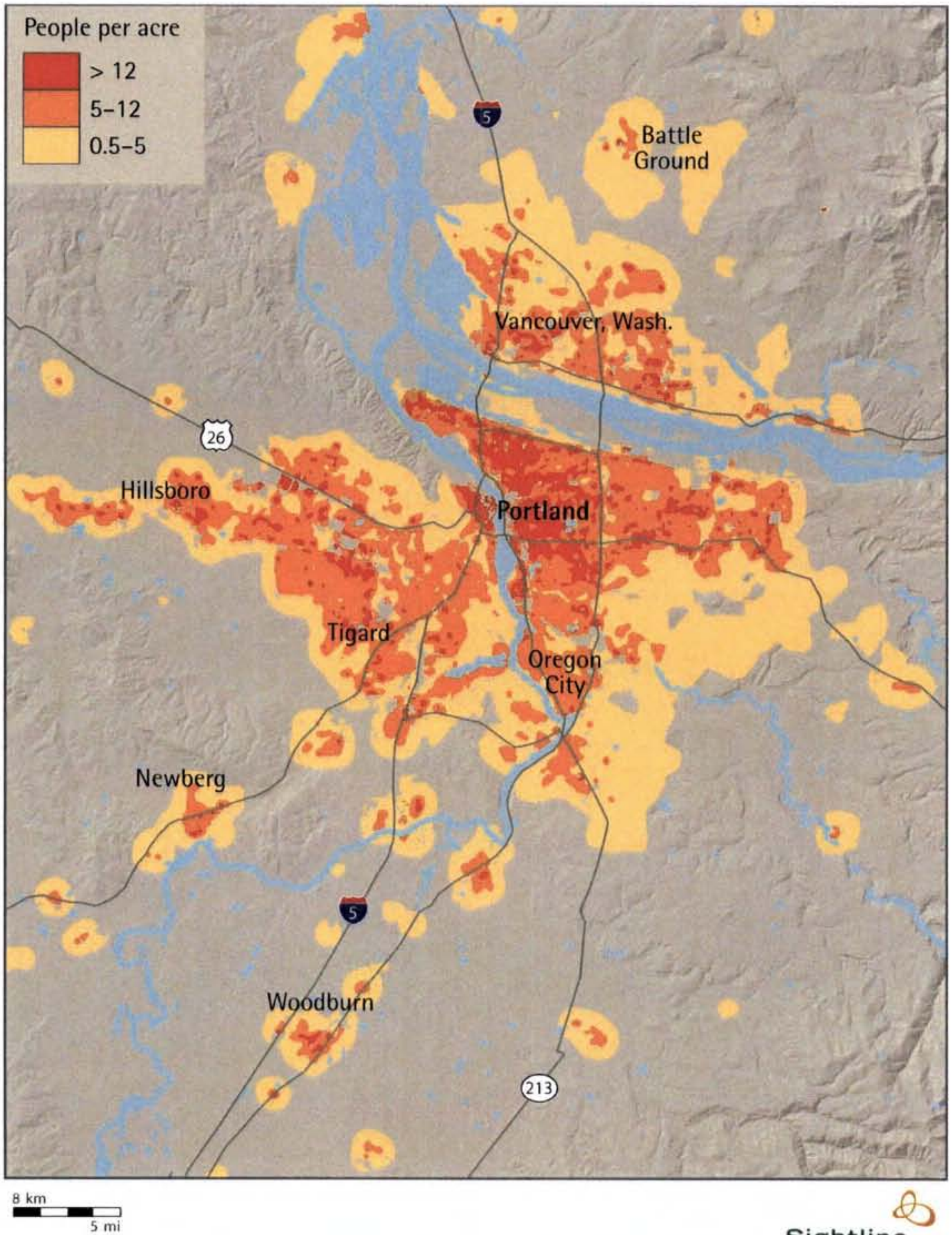


Figure 2.4 Population Density in the Portland Metropolitan Region

2.2 Nearby Industrial, Transportation, & Military Facilities

2.2.1 Locations and Routes

Figure 2.5 shows nearby industrial and transportation facilities. The nearest railway to the reactor site is approximately 0.5 miles (0.8 km) west of the reactor. The nearest rail yard is approximately one mile (1.6 km) northwest of the reactor. There are no refineries, mining facilities, or fuel storage facilities near the Reed campus. Water transportation occurs on the Willamette River, located approximately one mile west of the campus. Approximately 0.5 miles (0.8 km) northwest of campus is an industrial area where multiple small manufacturing plants are located. None of the nearby manufacturing plants produce materials that pose a reactor safety concern.

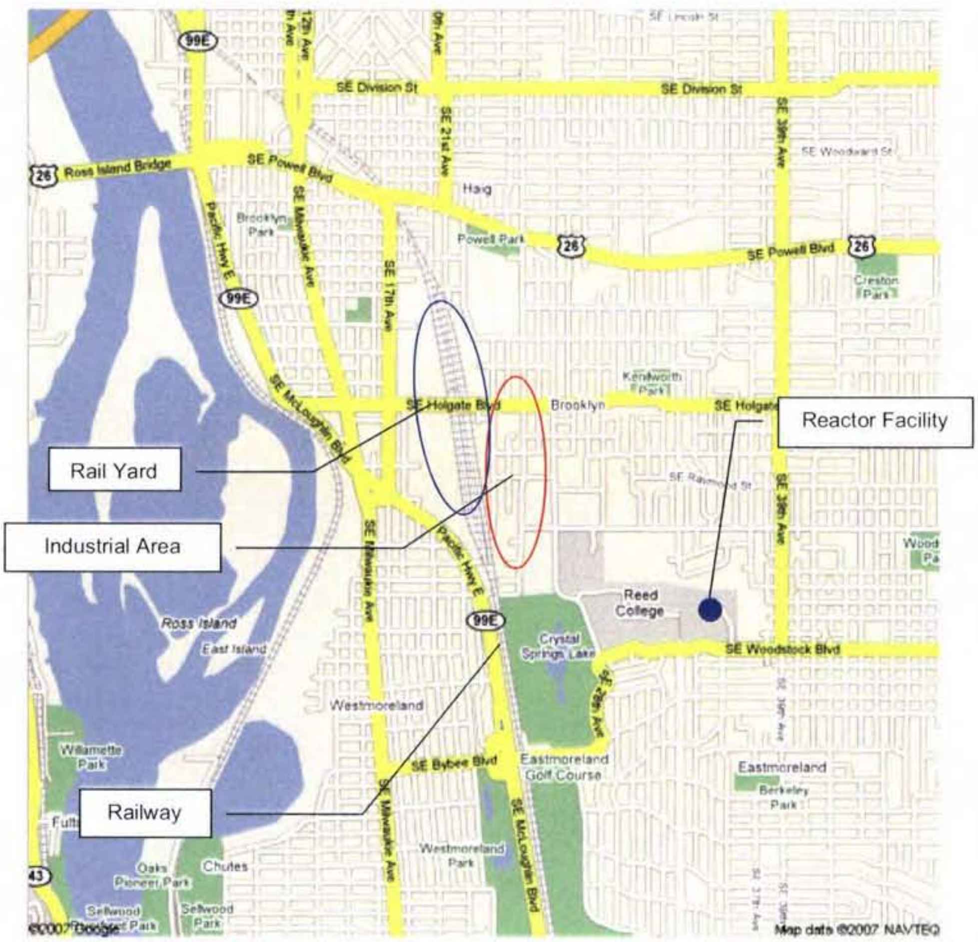


Figure 2.5 Railways Near the Reed Campus

2.2.2 Air Traffic

The Portland International Airport (PDX) is located approximately 10 miles (16 km) north-northwest of campus. PDX aircraft movements were projected to be 369,00 per year in 2006. The reactor is not located within the trajectory of any of the airport's runways.

2.2.3 Analysis of Potential Accidents

There are no nearby industrial, transportation, or material facilities that could experience accidents affecting the safety of the nuclear reactor.

2.3 Meteorology

Portland is located in the Willamette Valley, which in turn is located between the Cascade Mountain range and the Coastal Mountain range. The climate is characterized by cool, wet winters, and warm, dry summers. The region does not experience a significant amount of snowfall or severe weather.

2.3.1 General and Local Climate

2.3.1.1 Monthly Temperatures

Monthly temperature values for the Portland area are shown in Table 2.2. [3] Values are taken from 1971 to 2000. Monthly averages and daily extremes are given for each month. The normal minimum daily temperature extreme is 34.2°F in January and the normal maximum daily temperature extreme is 79.7°F, occurring in August. Extreme temperatures have ranged from 8°F to 107°F in the sample period.

Table 2.2 Monthly Temperatures

	Monthly Normals, °F			Daily Extremes, °F	
Month	Maximum	Minimum	Mean	Maximum	Minimum
Jan	45.6	34.2	39.9	63	12
Feb	50.3	35.9	43.1	71	9
Mar	55.7	38.6	47.2	77	19
Apr	60.5	41.9	51.2	90	30
May	66.7	47.5	57.1	100	35
Jun	72.7	52.6	62.7	100	41
Jul	79.3	56.9	68.1	104	45
Aug	79.7	57.3	68.5	107	44

Sep	74.6	52.5	63.6	105	37
Oct	63.3	45.2	54.3	92	26
Nov	51.8	39.8	45.8	73	13
Dec	45.4	35.0	40.2	65	8
Annual	62.1	44.8	53.5	107	8

2.3.1.2 Precipitation

Precipitation values, also taken from 1971 to 2000, are shown in Table 2.3. [4] The normal annual precipitation, calculated over the years 1971 to 2000, for the Portland area is 37.07 inches. The range of total annual precipitation is, however, considerable. A low of 26.11 inches was recorded in Portland in 1929 and a high of 67.24 inches was recorded in 1882. More than three-fourths of the annual precipitation falls during the six-month period October through March. In the Portland area, July and August are the driest months, with averages less than 1 inch per month. November, December, and January constitute the wettest period with around 5 inches per month. Table 2.4 [5] summarizes solid precipitation data for 1961-1990.

Table 2.3 Monthly Rainfall in Inches

	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Annual
Mean	5.07	4.18	3.71	2.64	2.38	1.59	0.72	0.93	1.65	2.88	5.61	5.71	37.07
Extreme 24 hr	2.33	2.16	1.54	1.25	1.45	1.46	1.06	1.47	2.03	2.44	2.69	2.08	2.69

Table 2.4 Monthly Solid Precipitation (Snow, Ice Pellets, Hail) in Inches. T denotes a trace amount was measured

	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Annual
Max. Monthly	41.4	13.2	12.9	T	0.6	T	0.0	T	T	0.2	8.2	15.7	41.4
Extreme 24 hr	10.6	6.4	7.7	T	0.5	T	0.0	T	T	0.2	7.4	8.0	10.6

2.3.1.3 Wind Stability

Wind rose data are available for the Portland International Airport weather station (station ID 24229, operated by the National Weather Service). Annual average data from this station is presented in the wind rose in Figure 2.6. [3] Data are taken from 1961 to 1990.

2.3.1.4 Humidity

Values for average relative humidity are given in Table 2.5. Data are from the National Climatic Data Center. [4] Humidity data are given for morning (M) values, measured at 4 A.M. local time, and for afternoon (A) values, measured at 4 P.M. local time.

Table 2.5 Relative Humidity

	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Annual
A	85	84	85	86	85	83	81	82	85	89	88	86	85
M	75	66	59	55	53	49	44	44	48	62	74	78	59

2.3.1.5 Severe Weather Phenomena

Tornadoes are infrequent in the Willamette Valley. Most of the Willamette Valley tornadoes are classified as F0 on the Fujita scale, with wind speeds reaching 72 mph. Two significant tornadoes in the Portland area are described here. An F2 tornado (wind speeds of 113 to 157 mph) occurred on December 8, 1993 near Newberg, Oregon in Washington County. Damage was not extensive. On April 5, 1972, an F3 tornado (wind speeds of 158 to 206 mph) struck Portland and Vancouver, Washington. The tornado covered about nine miles, injuring 300 people and killing six. This tornado was the most devastating in Oregon's recorded weather history, which dates to 1871. [5]

Snowstorms are infrequent in the Portland area. Freezing rain and ice storms are more common, due to effects of air flow through the Columbia River Gorge. While ice storms are more common in the Gorge to the east of Portland, they can affect the eastside and even downtown Portland. Ice storms in the Portland area typically cause power outages and road closures, neither of which are threatening to reactor safety. The dates of the most recent significant ice storms are summarized: December 28 to January 6 2004; January 16-18 1996; February 2-4 1996; December 26-30 1996; January 6-7 1991; January 5 1986; January 9-10 1979; January 9-10 1978; January 11-12 1973; and February 4-6 1972. [3] None have caused damage to the Reactor building.

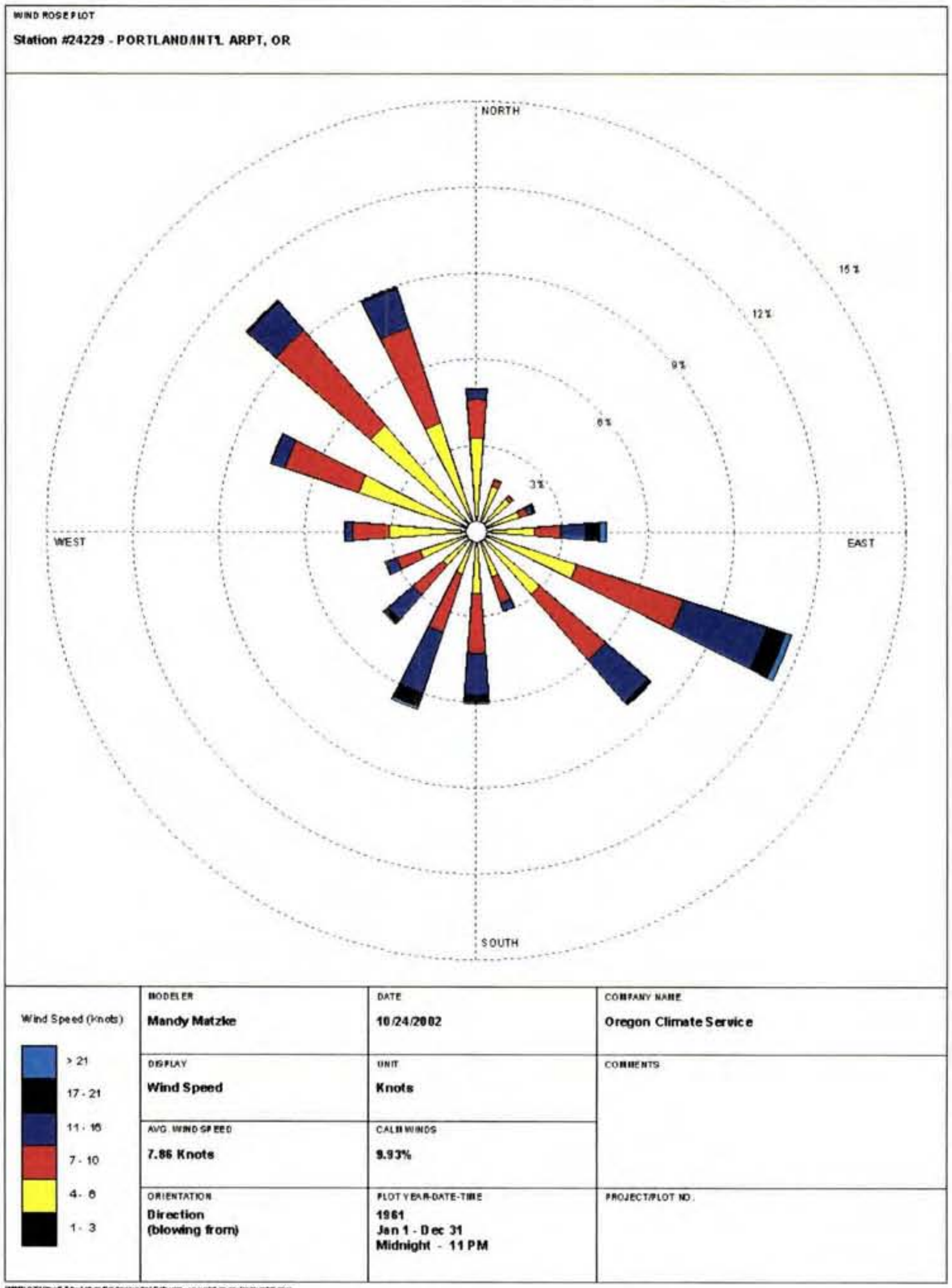


Figure 2.6 Portland Wind Rose (Annual Average)

2.3.2 Site Meteorology

Currently, monthly wind rose data are readily available from the National Water and Climate Center division of the Natural Resources Conservation Service of the United States Department of Agriculture (USDA). Meteorological information is not recorded on-site; however, the National Weather Service has multiple meteorological measuring stations in the Portland area, data from which is accessible to the public.

2.4 Hydrology

The nearest waterway to the reactor site is Crystal Springs Creek, located at the bottom of the ravine to the north of the reactor site. It flows westward through the campus, then south through the municipal golf course and part of Sellwood; and in the southern part of Sellwood it joins Johnson Creek which flows southward to join the Willamette River at Milwaukie. Crystal Springs Creek is fed by springs that issue from near the base of a terrace scarp where the regional water table is intercepted by the land surface. Most of the water is credited to ground water discharge. The water level of the lake stays nearly constant year-round, as the springs are self-regulating. It has been estimated that the total annual runoff of Crystal Spring Creek is in the neighborhood of 4,000 acre feet per year. The reactor site is on the bank of the ravine above Crystal Springs creek, and not at a high risk of being flooded by the spring-fed creek. [6]

Tsunamis are not a significant hazard to Portland, as the city is over 50 miles (80 km) from the Pacific coast. The reactor site is not located near any significant dams. The 100-year flood plain for waterways near the reactor is shown in Figure 2.6. [7]

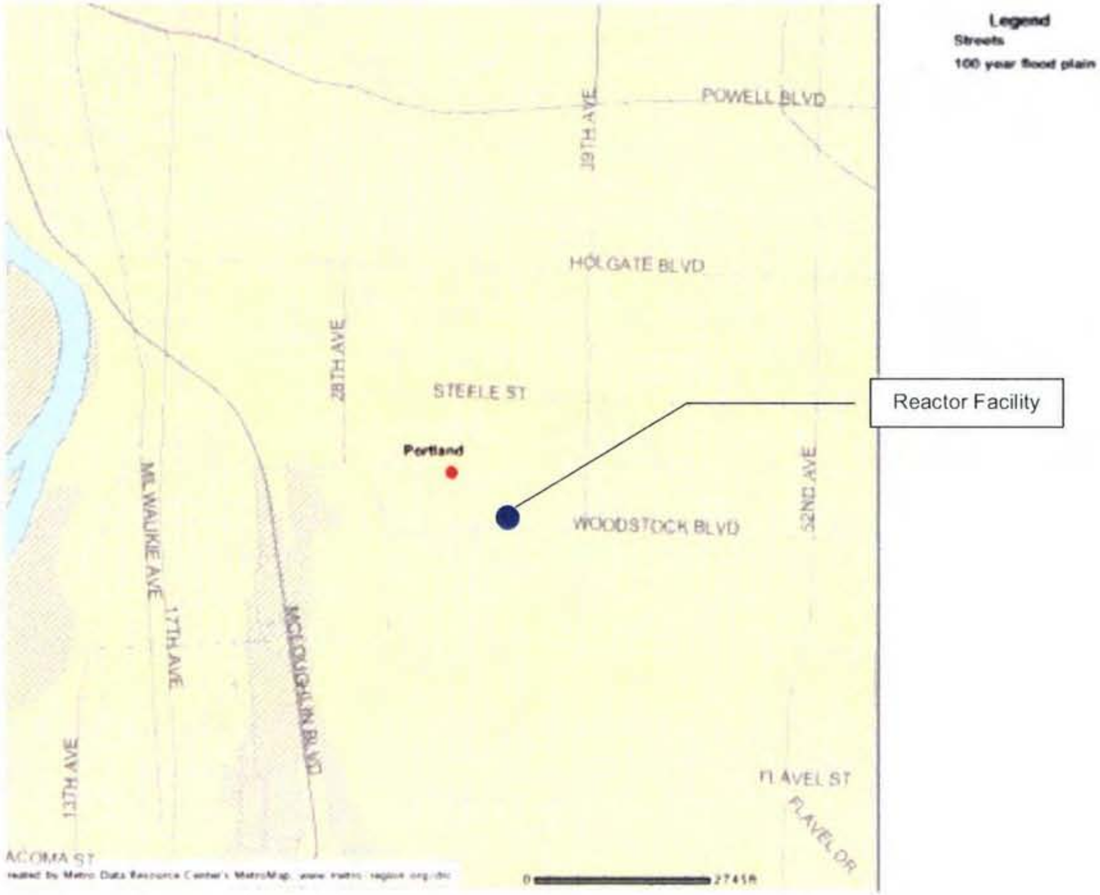


Figure 2.6 Reed Campus 100-year Flood Plain

2.5 Geology, Seismology, and Geotechnical Engineering

2.5.1 Regional Geology

Two tectonic plates are active in Western Oregon. Oregon is located on the North American Plate. The Juan de Fuca oceanic plate is located off the Oregon coast, and is being subducted beneath the North American plate at about 36 mm (1.4 inches) per year This subduction produces shallow, deep, and great thrust earthquakes. [8]

There are six geologic units in the Portland area which may serve as foundation materials. These are, in the order of decreasing geologic age: The Miocene Columbia River basalt; Pliocene Troutdale formation (conglomerate, sandstone and siltstone); Pliocene-Pleistocene Boring lavas of basaltic composition; Pleistocene loess (or windblown silt) and lake deposits of gravel, sand, and clay; and recent alluvium. The foregoing geologic units are summarized in the stratigraphic column below (Table 2.6), after Allen (1932):

Table 2.6 Stratigraphic Column, Portland Area

Age		Thickness of Unit (Feet)
Recent	Alluvium, sand and silt	0-50
Pleistocene		
	Alluvium, sand and silt	0-100
	Lacustrine deposits, gravel, sand	0-150
	Loess, windblown silt	0-60
Pliocene or Pleistocene	Boring lavas, basalt	0-800
Pliocene	Troutdale conglomerate, sandstone	0-1000
Miocene	Columbia River basalt	0-2000

Most of the East Portland area is underlain by bedrock of the Troutdale formation. This material also makes up much of Mt. Tabor, Kelly Butte, and other hills in the East Portland area. Boring lavas are the most important unit at Rocky Butte, but they compose only small portions of Mt. Tabor and Kelly Butte.

One of the most important geologic features in East Portland is a series of terraces which were cut by the Willamette and Columbia Rivers on Pleistocene lake silts and gravels. These terraces occur at elevations of approximately 100 feet (30 m), 200 feet (60 m), and 275 feet (83 m). The unconsolidated deposits of gravel, sand, and silt comprise sediments hundreds of feet thick. Recent sand and silt in the Willamette and Columbia River valleys rarely rise higher than the 50 foot (15 m) elevation.

From the structural point of view the Portland area is relatively simple. Broad folds with small to moderate dips trend generally north-westward. West of the Willamette River is the Portland West Hills' anticline. The Willamette syncline lies to the east of this.

No major faults are known to exist in the Portland area. Minor faults with small displacements may cut Columbia River basalt in West Portland. Deformations that produced the anticlines and synclines in the lavas and overlaying Troutdale beds must have ceased before the extrusion of the Boring lavas, because the latter are not deformed. The top of the Columbia River basalt in East Portland is at a depth of 1000 feet (305 m) below sea level. [6]

2.5.2 Site Geology

The Reed College campus is on a stream-carved terrace. The reactor site slopes gently to the north from an elevation of approximately 145 feet (44 m). The campus is situated on the unconsolidated deposits of the Willamette basin. The depth to bedrock is unknown from well log data or geophysical surveys; it may well be hundreds of feet.

In July of 1966, Shannon and Wilson, Foundation Engineers, made a boring to a depth of 47 feet (14.3 m) at the reactor site, north of the existing psychology building. The log of this boring indicates that the subsurface materials at the reactor site are sand and silt, the one admixed with the other in most horizons. This sand and silt section represents lake or fluvial deposition. [6]

The Portland Hills fault is located west of the reactor site, along the eastern margin of the Portland Hills west of the city. Investigations in 1993 did not reveal any evidence of fault activity in the Holocene or late Pleistocene activity. The fault is approximately 40 km (25 miles) long. The slip rates of the Portland Hills fault are considered to be 0.05 mm (0.002 inches) per year to 0.2 mm (0.008 inches) per year, which are comparable to other potentially active faults in the region. [9]

The Lackamas Creek fault is located east of the reactor site, along the eastern margin of the Portland Basin, near Lackamas Creek. No evidence of activity of this fault in the Holocene era has been observed. The fault is approximately 10 km (6 miles) long. Slip rates assigned to the fault by the Geomatrix analysis are 0.05 mm (0.002 inches) per year to 0.2 mm (0.008 inches) per year. [9]

2.5.3 Seismicity

In the past 150 years, there have been three significant earthquakes with epicenter near Portland: (1) the magnitude 5.4 earthquake in 1877, (2) the magnitude 5.5 earthquake in 1962, and (3) the magnitude 5.5 earthquake in 1993. Historically, earthquakes occurring in the Puget Sound region have also caused minor damage in Portland. Table 2.7 summarizes historic earthquakes that have occurred in the Portland area. [10]

Figure 2.7 shows the epicenters of earthquakes recorded from 1841 to 2002. Many of these are small magnitude (< 3.0) earthquakes, and not historic events. [11]

Table 2.7 Historic Earthquakes Near Portland, OR

Date	Approx. Location	Magnitude
Oct 12, 1877	Portland, OR	5.4
Feb 4, 1892	Portland, OR	5.6
Jul 19, 1930	Salem, OR	5.0
Dec 29, 1941	Portland, OR	5.0
Apr 13, 1949	Olympia, WA	7.1
Dec 16, 1953	Portland, OR	5.0
Nov 17, 1957	Tillamook, OR	5.0
Sep 16-17, 1961	Mt. St. Helens, WA	4.8, 4.0, 5.1
Nov 5, 1962	Portland, OR	5.5
Mar 7, 1963	Salem, OR	4.6
Oct 1, 1964	Portland, OR	4.1
May 28, 1981	Goat Rocks, WA	4.6, 5.0

Mar 1, 1982	Elk Lake, WA	4.4
Mar 25, 1993	Scott's Mill, OR	5.6
Jan 30, 2000	Condon, OR	4.1
Feb 28, 2001	Olympia, WA	6.8
Jun 29, 2002	Mt. Hood, OR	4.5
Jul 12, 2004	Newport, OR	4.9

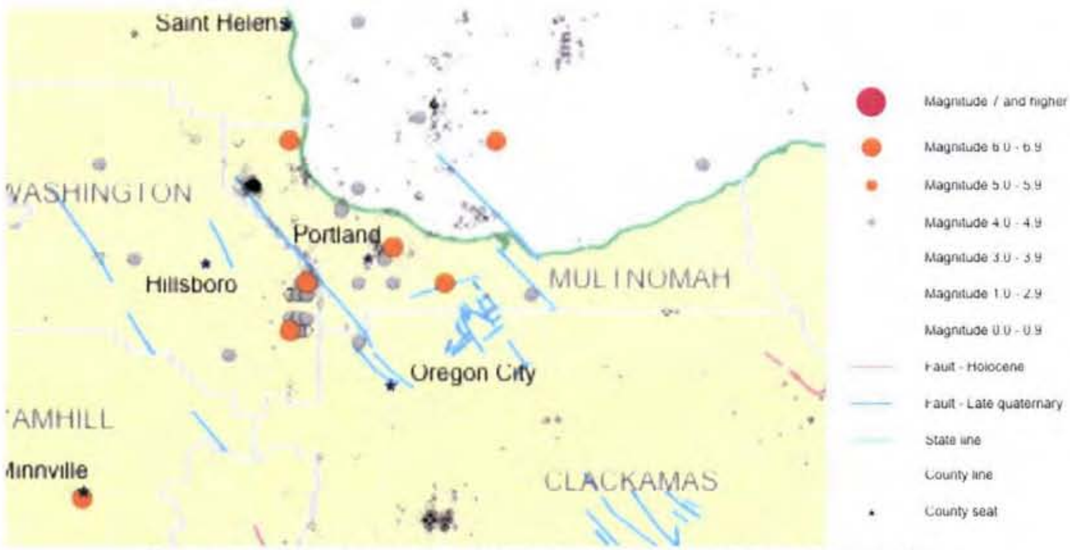


Figure 2.7 Earthquakes near Portland Oregon, 1841-2002

2.5.4 Maximum Earthquake Potential

The primary geological structure in the Portland area is the Cascade Range, which runs north-south through Oregon, from Northern California to Washington state. The Cascade Range is home to a host of volcanoes; the nearest to Portland is Mount Hood. Other Oregon volcanoes are Mount Jefferson, Three Sisters, Newberry and Crater Lake. Mount Saint Helens, in Washington, is the only of the Cascade Range volcanoes that exhibits an above-normal level of background seismicity. The major volcanoes of the Cascade Range are monitored by the United States Geological Survey (USGS). The last eruption of Mount Hood was in the 1790s; Mount Hood is considered to be the most active of the Oregon volcanoes. [8]

Recent geophysical studies indicate that the crustal faults beneath the Portland metropolitan region could generate crustal earthquakes of Richter Local Magnitude (M_L) 6.5 or larger. The recurrence period of a M_L 6.5 crustal earthquake in the Portland area is estimated to be about 1,000 years; this is based on the historical record. The historical record also shows that Cascadia subduction zone earthquakes, of up to a moment magnitude 9, have occurred and could occur in the future. The recurrence period of such an earthquake has not been established. [12]

2.5.5 Vibratory Ground Motion

Figures 2.8, 2.9, and 2.10 depict ground shaking at the ground surface in the Portland metropolitan area. They incorporate the site-response effects of soils, unconsolidated sediments, and shallow rock. The probabilistic maps, Figures 2.8 and 2.9, are for a moment magnitude (MW) 9.0 earthquake along the megathrust of the Cascadia subduction zone. They are for the two return periods of building code relevance, 500 and 2,500 years. Figure 2.10 depicts a hypothetical MW 6.8 event on the Portland Hills fault. The maps were prepared by the State of Oregon, Department of Geology and Mineral Industries. A Cornell-McGuire hazard analysis was used to calculate the probabilistic ground motions. [12]

Figure 2.8 is a map of the probabilistic peak horizontal acceleration at ground surface in the Portland, Oregon area in the 500 year return period. The Reed campus is located in a region of peak acceleration estimated at 0.25 to 0.30 times the acceleration of gravity. The acceleration of gravity is a constant 980 cm/sec/sec; hence 0.25 g corresponds to an acceleration of 245 cm/sec/sec. Figure 2.9 is a map of the probabilistic peak horizontal acceleration at ground surface in the Portland, Oregon area in the 2,500 year return period. Here the Reed campus is located in an area where the peak acceleration at ground surface is estimated to be 0.5 to 0.6 g, or 490 to 588 cm/sec/sec. [12]

Figure 2.10 estimates the peak horizontal acceleration at the ground surface for a magnitude 6.8 earthquake emanating from the Portland Hills fault. The Reed campus is located in a region estimated at 0.6 to 0.7 g. [12]

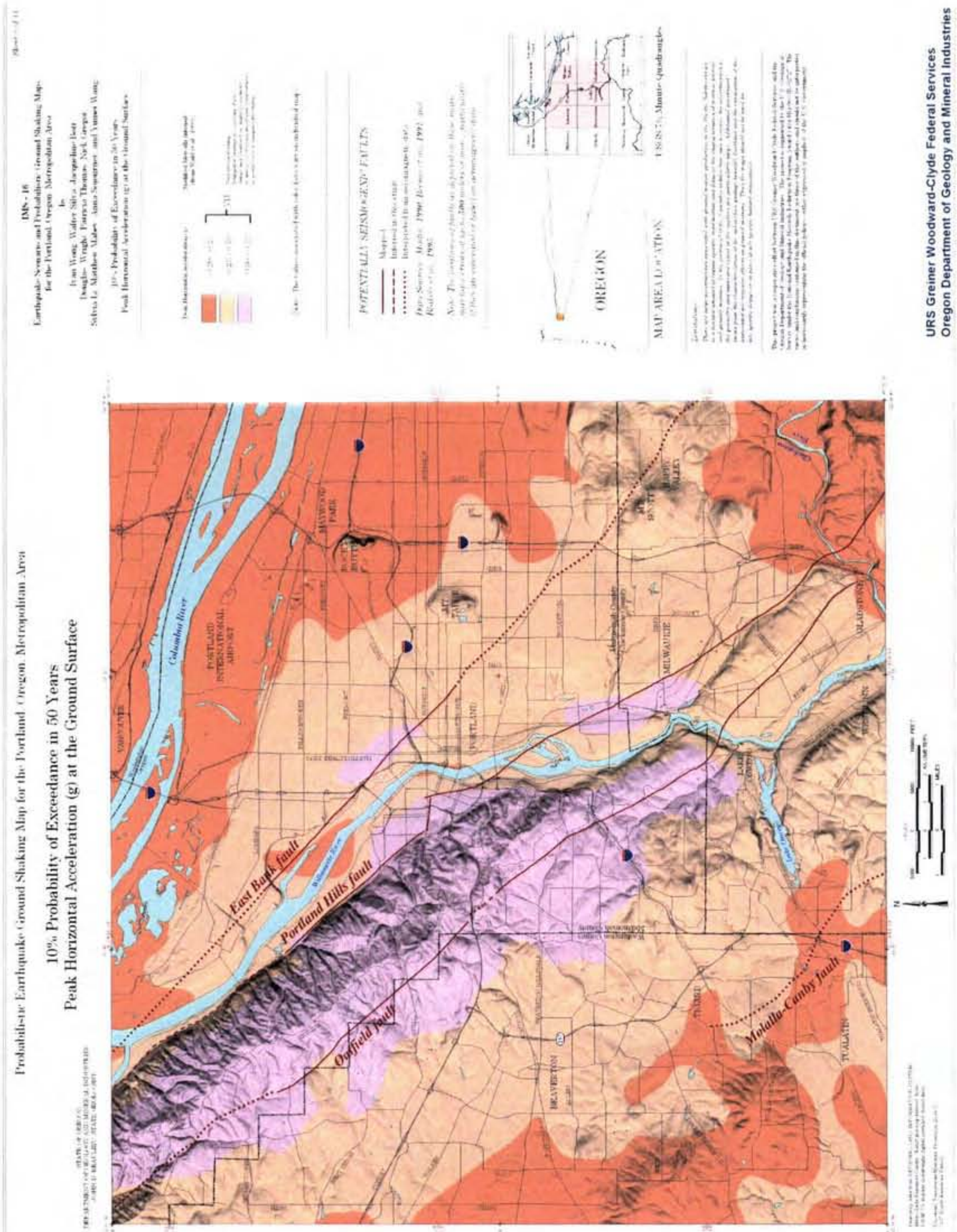


Figure 2.8: 500 year return Probabilistic Earthquake Ground Shaking Map for the Portland Area

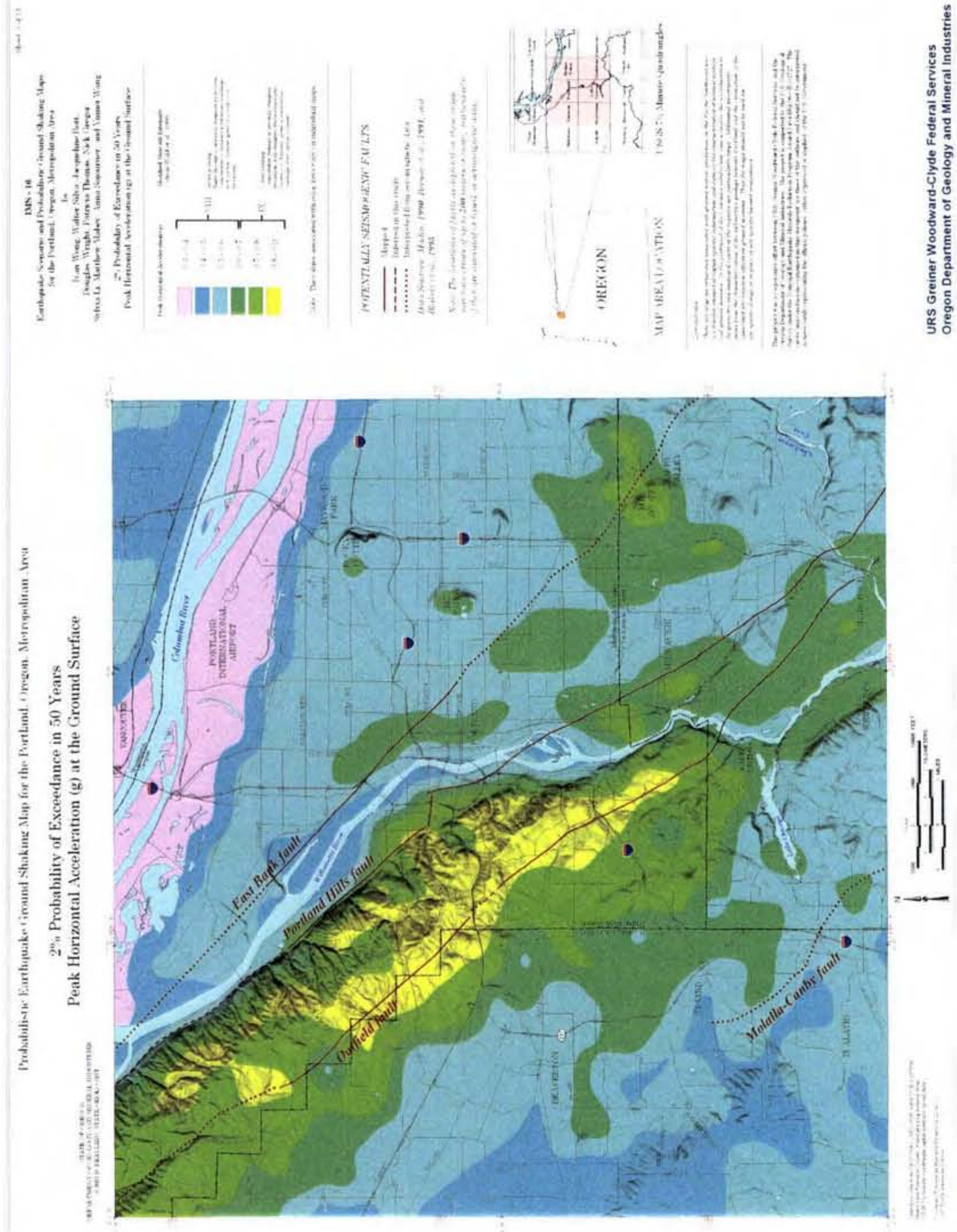


Figure 2.9: 2500 year Return Earthquake Scenario Ground Shaking Map for the Portland Area

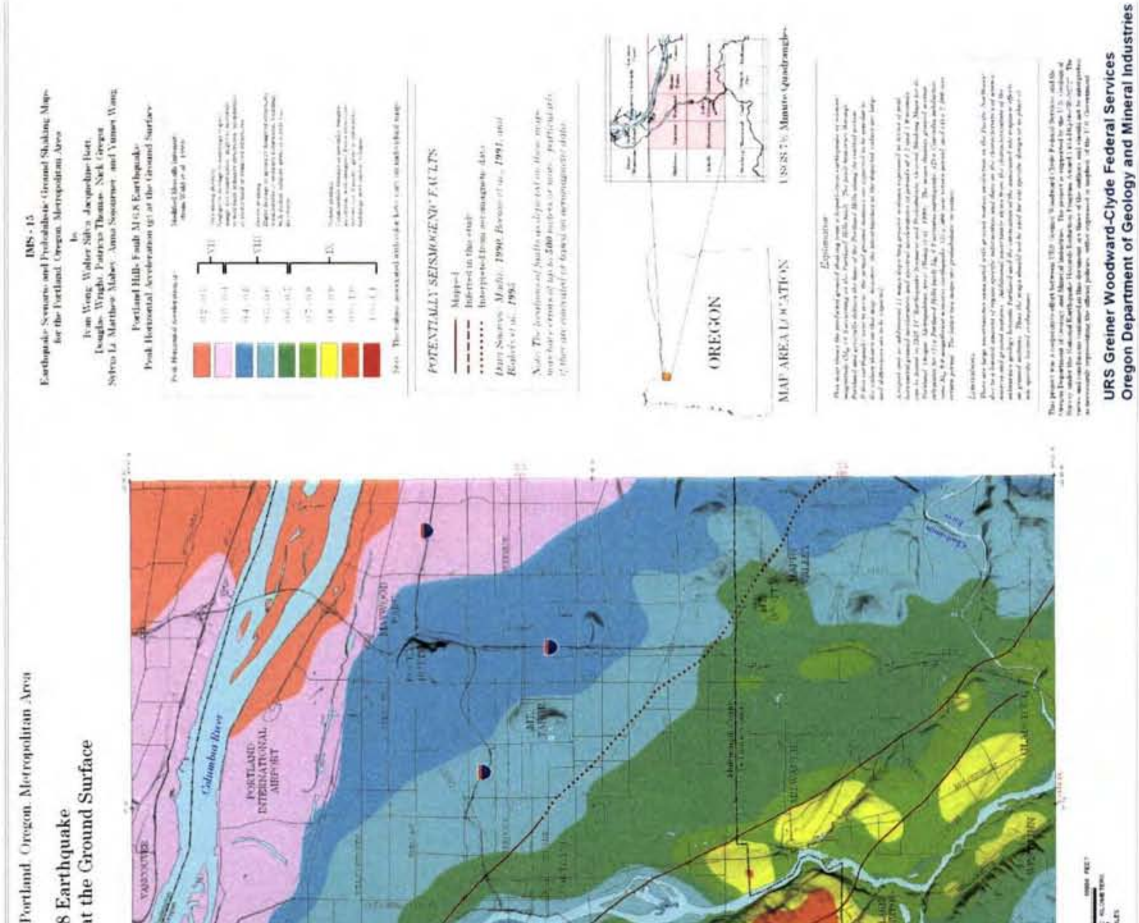


Figure 2.10: Earthquake Scenario Ground Shaking Map for the Portland Area

2.5.6 Surface Faulting

The Reed campus lies to the east of the Portland Hills fault. Earthquakes occurring near the fault are mapped in Figure 2.7. Notable earthquakes occurring near the fault are summarized in Table 2.6.

2.5.7 Liquefaction Potential

As discussed in 2.4.2, the subsurface materials at the reactor site are sand and silt, the one admixed with the other in most horizons. This sand and silt section represents lake or fluvial deposition. The water table in the test boring was observed to stand at a depth of 46 feet (14 m). [6]

The Natural Resources Conservation Service classifies the soil at the reactor site as a Latourell complex, which is characterized as being a well drained loam, with its parent material being medium textured alluvium. A typical profile of the soil reveals it to be loam for 0 to 56 inches (0 to 1.4 m) from the surface, and from 56 to 66 inches (1.4 to 1.7 m) a very gravelly sandy loam. [13]

Liquefaction occurs when soils are saturated with water. Since the soil at the reactor site is well drained loam, liquefaction has a low potential for occurrence.

2.6 References

1) U.S. Census Bureau, United States Census 2000.

2) Sightline Institute, *Portland-Area Population Density Map, 2005,* www.sightline.org/maps, August 2007.

3) Oregon Climate Services, www.ocs.orst.edu, August 2007.

4) NCDC, "*Comparative Climatic Data for the United States through 2006,*" National Climatic Data Center, National Oceanic and Atmospheric Administration, Asheville, North Carolina, 2006.

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6) Safety Analysis Report, Reed Reactor Facility, 1967.

7) Metro, *Natural Hazards Resource Program,* http://geode.metro-region.org/hazardmap/metromap.cfm, August 2007.

8) USGS, *Pacific Northwest geologic mapping and urban hazards*, http://geomaps.wr.usgs.gov/pacnw/rescasp1.html, August 2007.

9) Geomatrix Consultants, 1995, *Final Report: Seismic Design Mapping, State of Oregon:* Prepared for the Oregon Department of Transportation, Salem, Oregon, Personal Services Contract 11688, January, 1995.

10) PNSN, 2007, Pacific Northwest Earthquake Catalog Search, 1970 to Present: www.pnsn.org, University of Washington, Pacific Norhtwest Seismograph Network.

11) Niewendorp, Clark A. and Mark E. Neuhaus, *Map of Selected Earthquakes for Oregon, 1841 through 2002*, Oregon Department of Geology and Mineral Industries.

12) State of Oregon, Department of Geology and Mineral Industries, *Interpretive Map Series IMS-16,* 2000.

13) USDA, *Natural Resources Conservation Service,* http://websoilsurvey.nrcs.usda.gov, August 2007.

Chapter 3

Design of Structures, Systems, and Components

Reed Research Reactor Safety Analysis Report

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Chapter 3

Design of Structures, Systems, and Components

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3 DESIGN OF STRUCTURES, SYSTEMS, & COMPONENTS

This chapter describes the principal architectural and engineering design criteria for the structures, systems, and components that are required to ensure reactor facility safety and protection of the public.

3.1 Design Criteria

The Reed Research Reactor (RRR) is sited on the campus of Reed College in Portland, Oregon. It is located in a building constructed for that purpose and adjacent to the Psychology Building. The original reactor installation in 1968 used fuel and components manufactured by General Atomics (GA), and the specifications to which structures were built were those stated by GA. Specific design criteria were not stated. All building modifications and equipment additions were in conformance with the building codes in existence at the time.

The basic design goal of a TRIGA® reactor is integrity of the fuel by cladding that will act as a physical containment system for fission products. Fuel design prevents the release of radioactive fission products during routine reactor operation and potential accident conditions. The prompt negative temperature coefficient of reactivity of TRIGA® fuel is the basic parameter that allows safe usage of the fuel, as it results in a temperature-dependent decrease in the number of absorptions of neutrons by uranium-235, producing a feedback that places a physical limitation on fuel temperature below danger levels. Limits on the amount of fuel loaded in the core (i.e., reactivity) establish a maximum steady state power level, which limits the maximum fuel temperature, the major constraint on safe operation of TRIGA® fuel. Fuel design is detailed in Chapter 4, Reactor Description.

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

The reactor control system maintains safe shutdown conditions. Since operational limits prevent achieving conditions that could lead to fuel element failure, control system response speed is not significant to protection of fuel integrity. System design is discussed in Chapter 4, Reactor Description and Chapter 7, Instrumentation and Control Systems.

Building and structure design for meteorological, hydrological, and seismic effects are discussed in the following sections.

3.2 Meteorological Damage

The RRR is protected from damage by high winds or tornadoes by virtue of the [redacted] concrete structure surrounding the reactor tank and the [redacted] construction of the pool itself. The facility has endured approximately forty years of local weather conditions with no meteorological damage. Hurricanes, tsunamis, and seiches do not occur in the Portland area.

Only a small number of tornadoes, one every few years, have been reported in Oregon. Based on the small probability of occurrences, postulated low intensity, intermittent reactor operation and low fission-product inventory, no criteria for tornadoes have been established for the RRR.

3.3 Water Damage

As discussed in Chapter 2, the flood plain of the local rivers does not come near the reactor site. However, even if flooding occurred, reactor safety would not be an issue since the core is located in a water pool.

3.4 Seismic Damage

No faults are known to exist near the RRR site. However, following the practice of Portland architects the construction of the building and reactor pit were designed to resist lateral forces of Zone II as specified in the Uniform Building Code when the reactor was installed. This ensures that the reactor can be returned to operation without structural repairs following an earthquake likely to occur during the lifetime of the plant. Failure of the reactor tank and loss of the coolant in the event of a very large earthquake has been considered in Chapter 13, Accident Analysis, and the consequences found acceptable for the standpoint of public safety.

3.5 Systems and Components

The reactor facility design uses a defense-in-depth concept to reduce and control the potential for exposure to radioactive material generated during reactor operation. Fuel cladding is the principal barrier to the release of radioactive fission products. Shielding is provided (including reactor pool water) to control potential personnel exposures to radiation associated with reactor during operation or activated material. The control rods assure that safe shutdown conditions are maintained when reactor operation is not required. If radioactive material releases associated with reactor operations occur, a controlled ventilation system minimizes exposure to reactor personnel and the public.

Cladding integrity is ensured by the fuel system (fuel rod and core design). Fuel cladding surrounding individual fuel elements is the primary barrier to the release of radioactive fission products. The fuel system maintains cladding integrity through established limits on reactivity and power such that cladding integrity will not be challenged.

Shutdown reactor conditions are initiated and maintained by the control rod scram system. Since inherent shutdown mechanisms of the TRIGA® prevent unsafe excursions, the TRIGA® system does not rely on speed of control as paramount to the safety of the reactor. The control system ensures maintenance of reactor shutdown conditions, as well as control of power level during operation.

Although there are no required engineered safety features for this reactor due to low operating power and good fission product retention in the fuel, a controlled ventilation system maintains a negative air pressure in the bay to reduce the consequences of airborne radiological release. The ventilation system is described in Chapter 9, Auxiliary Systems.

3.6 Control Rod Scram System

The RRR, operating at up to 500 kW thermal power, is designed to be operated with three standard control rods. The control rods are nominally 1.25 inches (3.18 cm) outside diameter, 20 inches (50.8 cm) long, and are clad with 30 mil (0.076 cm) stainless steel or aluminum. The control rod material is either boron carbide or borated graphite. During operation, the rods are held in place by electromagnets, and are withdrawn or inserted by motor-driven gear mechanisms. Upon a scram signal, power to the electromagnets is interrupted and the control rods descend by gravity into the core. The rods have a maximum drop time of 1 second from fully withdrawn to fully inserted positions. Details of the control rod scram system are addressed in Chapter 4, Reactor Description, and Chapter 7, Instrumentation and Control.

Chapter 4

Reactor Description

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Reactor Description

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4 REACTOR DESCRIPTION

4.1 Reactor Tank

The reactor core is located at the bottom of an aluminum tank which is 10 feet (3 m) wide and 15 feet (4.6 m) long with a 5 foot (1.5 m) radius at each end. The tank is 25 feet (7.6 m) deep and is bolted at the bottom to a 24 inch (61 cm) thick poured concrete slab. The tank has a minimum wall thickness of 0.25 inches (0.64 cm) and is surrounded by approximately [redacted] of concrete.

The tank is water proofed by continuous welded joints; the integrity of the joints was verified by X-ray testing, pressure testing, dye-penetraent checking, and soap-bubble leak testing. For corrosion protection, the outside of the tank is coated with a double layer of tar and felt. A 2 inch (5 cm) by 2 inch (5 cm) aluminum channel used for mounting the neutron detectors and underwater lights is welded around the top of the tank.

The top of the tank is surrounded by a steel frame 11 feet (3.4 m) wide and 16 feet (4.9 m) long, which is fabricated of 10 inch (25.4 cm) structural-steel channel and is recessed in the top of the shield structure. The tank is filled with demineralized water to a depth of 24.5 feet (7.5 m), providing approximately 20 feet (6 m) of shielding water above the top of the core.

4.1.1 Center-Channel Assembly

Support for the various irradiation facilities, the control rod drive mechanisms, and the tank covers is provided by the center-channel assembly at the top of the reactor tank (Figure 4.1). This assembly consists of two 8 inch (20 cm) structural-steel channels with six 16 inch (41 cm) wide by 0.625 inch (1.59 cm) thick steel cover plates bolted end-to-end to the flanges of the channels with socket-head screws. The assembly has the shape of an inverted U, is 11 feet (3.3 m) long, and is positioned directly over the center of the reactor. The assembly is attached by two steel angle brackets at each end to the 10 inch (25.4 cm) steel channels that form the recessed frame around the top of the tank. Each angle bracket is attached to each channel with four 1/2-13 by 1.5 inch (3.8 cm) stainless steel machine bolts. The brackets are made of 6 inch (15 cm) by 6 inch (15 cm) steel angle, 0.5 inches (1.3 cm) thick and 6 inches (15 cm) long. The channel assembly is designed to support a shielded isotope cask, weighing 4.5 tons (4100 kg), placed over the specimen-removal tube.

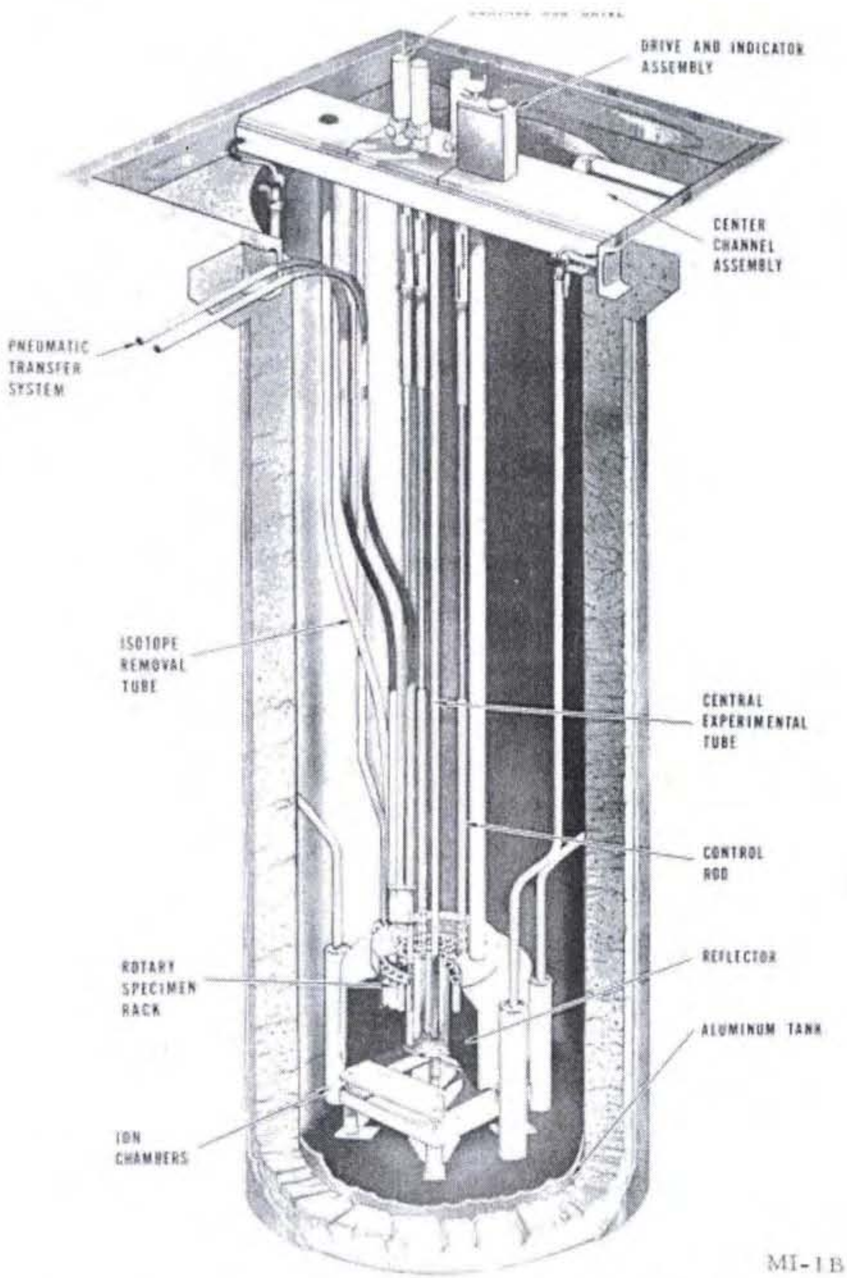

Figure 4.1 Cutaway view of a typical TRIGA® Mark I reactor

4.1.2 Reactor-Tank Covers

The top of the reactor tank is closed at one end by four hinged covers that are flush with the floor. The covers are made of aluminum grating formed from 0.1875 inch (0.77 cm) by 1.5 inch (3.8 cm) aluminum bars. A sheet of 0.25 inch (0.635 cm) thick plastic is inserted in the bottom of each grating section to limit the entry of foreign matter into the tank while still permitting visual observation of the reactor. The plastic sheets slide into channels on the underside of the gratings and are easily removed for cleaning. A gap around the perimeter of the plastic permits adequate venting of the small quantities of gas that may be released during reactor operation. Each cover is fastened by a stainless steel hinge to the recessed 10 inch (25.4 cm) channel around the top of the tank. The hinge is attached to the channel by 1/4-20 by 0.5 inch (1.27 cm) screws. The center channel assembly provides support for the covers when they are closed. Each cover is provided with two flush lifting handles.

4.2 Basic Reactor Components

The core assembly is a right circular cylinder consisting of a compact array of cylindrical fuel-moderator elements, a central thimble, a neutron source, and control rods, all positioned vertically between two grid plates which are fastened to the reflector assembly. The outer region of the core may contain some graphite dummy elements. The reflector surrounds the core and is composed of graphite with a radial thickness of about 12 inches (30 cm) encased in an aluminum can.

The control rods are guided by guide tubes that are inserted through the top grid plate and attached to the bottom grid plate by means of a special locking device. The core is cooled by natural convection of the water that occupies about one-third of the core volume. Shielding above the core is provided by approximately 20 feet (6 m) of water.

4.2.1 Reflector Platform

The reflector platform is a square, all-welded aluminum-frame structure. It rests on four legs that are held down by aluminum anchor bolts welded to the bottom of the aluminum tank. Oversized bolt holes permit some horizontal adjustment during initial installation.

4.2.2 Reflector

The reflector surrounding the core (Figure 4.2) consists primarily of a ring-shaped block of graphite having an inside diameter of approximately [redacted] Water is kept from contact with the graphite by a welded aluminum container which encases the entire reflector. Provision for the isotope-production facility (rotary specimen rack) is made in the form of a ring-shaped well in the top of the reflector. The rotary specimen rack mechanism does not penetrate the sealed reflector assembly at any point.

The reflector assembly rests on the reflector platform. Support is provided by two aluminum channels welded to the bottom of the reflector container. Four holes in the lower flanges of the channels are used to attach the reflector to the reflector platform with 0.5 inch (1.27 cm) stainless steel bolts and nuts. The reflector platform is bolted to the tank bottom. The reflector housing has an inside diameter of [redacted] Aluminum shims under the reflector platform provide vertical adjustment of approximately 2.25 inches (5.7 cm). When adjustment and leveling have been completed, the position is secured by tightening the 5/8-11 anchor bolts in the tank bottom.

Three lugs with 2 inch (5 cm) diameter holes are provided for lifting the reflector assembly, which weighs approximately 1700 lb (770 kg).

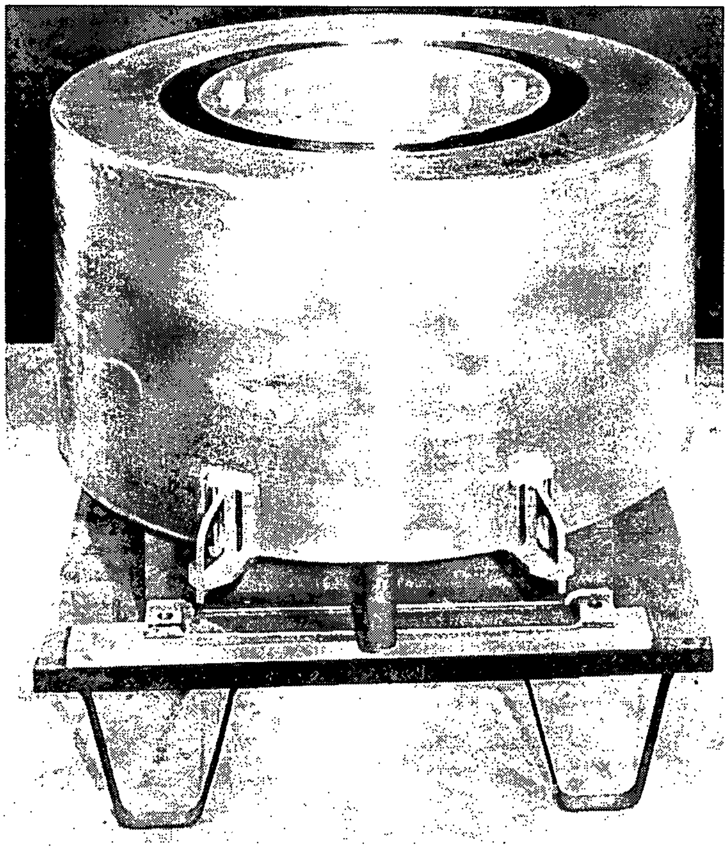

Figure 4.2 Reflector

4.2.3 Grid Plates

The top grid plate (Figure 4.3) is made of aluminum and is 19.44 inches (49.35 cm) in diameter and 0.75 inches (1.9 cm) thick. This plate provides accurate lateral positioning of the core components. It rests on six pads welded to the top of the reflector container. Two stainless steel dowel pins, which fit tightly in the pads and loosely in the grid plate, orient the grid plate. Four hexagonal-head captive screws of anodized aluminum measuring 0.5625 inches (1.43 cm) across the flats secure the top grid plate to the reflector. They also serve as hold down screws for the clamps that secure the rotary specimen rack in its location. In addition to the central thimble, ninety fuel element locations, distributed in five circular rings, are provided. Each element location is a 1.505

inch (3.823 cm) diameter hole through the plate, which is anodized to retard wear and corrosion.

Cooling water passes through the differential area between the triangular spacer block on the top of each fuel element and the round holes in the grid plate. The nominal diametral clearance between the tips of the spacer blocks and the grid plate is 0.010 to 0.020 inches (0.025 to 0.051 cm). The center hole in the top grid plate, which is 1.515 in. (3.848 cm) in diameter, serves as a guide for the central thimble.

The bottom grid plate (Figure 4.3), in addition to providing accurate spacing between the fuel-moderator elements, carries the entire weight of the core. This plate is of aluminum and is 16 inches (40.6 cm) in diameter and 0.75 inches (1.9 cm) thick. It is supported by six L-shaped lugs welded to the underside of the reflector container. Two stainless steel dowel pins, which fit tightly in the lugs and loosely in the plate, orient the grid. Four hexagonal-head captive screws of anodized aluminum and measuring 0.5625 inches (1.43 cm) across the flats secure the grid to the support lugs. The size of the bottom grid plate allows it to be inserted and removed through the core void when the top grid plate and other core components are removed. Ninety holes, 0.281 inches (0.714 cm) in diameter countersunk 90° to a diameter of 0.625 in. (0.159 cm), are machined in alignment with the holes in the top grid plate. The countersink supports the lower end fixture of the fuel-moderator element. The holes in both grid plates also orient and support the three control-rod guide tubes. The central hole, which is 1.562 inches (3.969 cm) in diameter, serves as a clearance hole for the central thimble.

Thirty-one holes, 0.5625 inches (1.43 cm) in diameter and oriented in three circular bands concentric with the central-thimble hole, provide a water-passage area through the lower grid plate. However, most of the water used to cool the core flows by natural convection into the lower core plenum through the annular space provided between the top of the bottom grid and the bottom of the reflector. Also, the lower grid plate is anodized after machining to retard wear and corrosion.

Foil-insertion holes, 0.314 inches (0.798 cm) in diameter, are drilled at various positions through both grid plates. These holes make possible the insertion of foils into the core to obtain flux measurements.

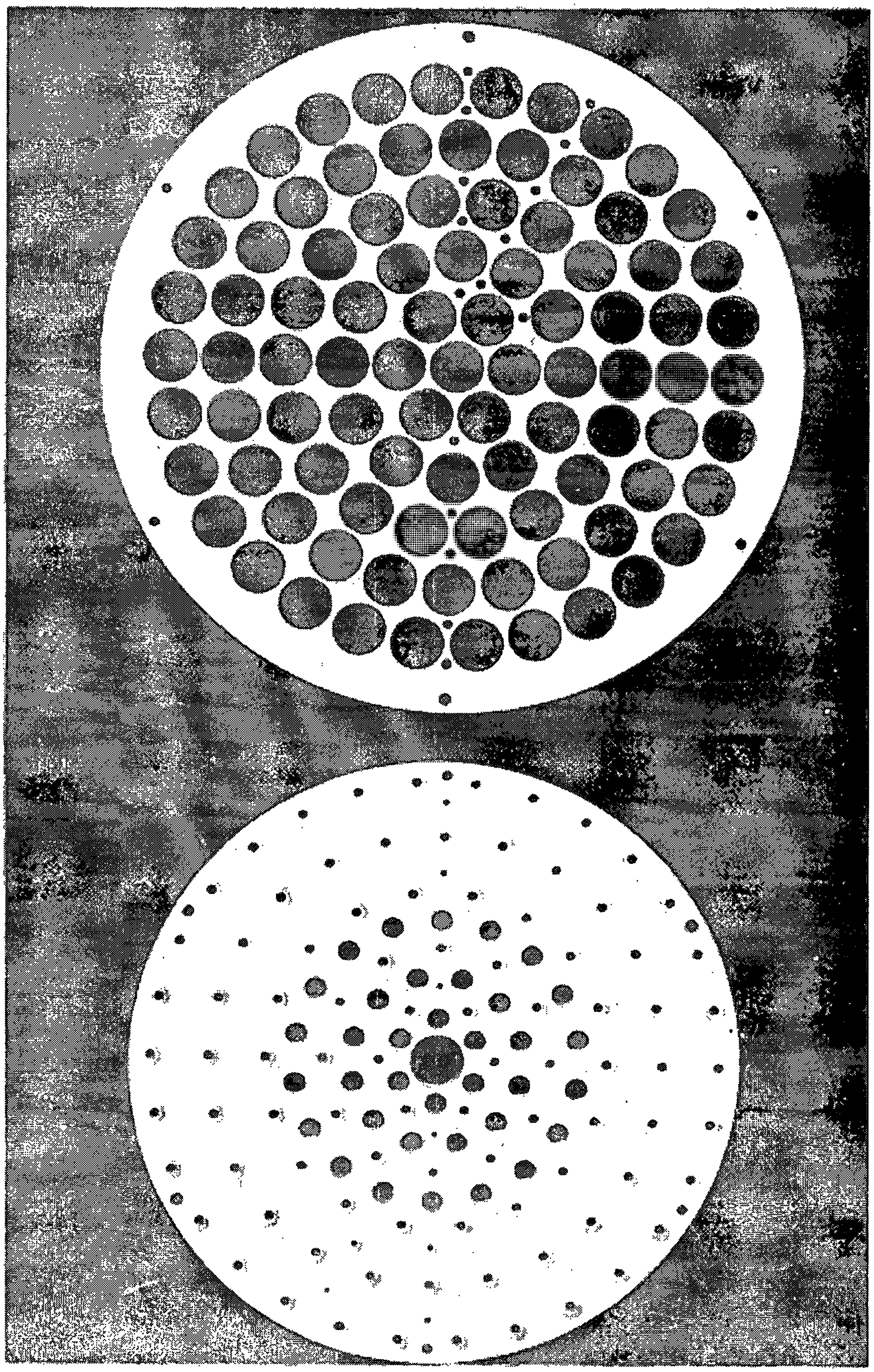

Figure 4.3 Upper and Lower Grid Plates

4.2.4 Aluminum Fuel-Moderator Elements

The active part of each fuel-moderator element (Figure 4.5) is [redacted]. The fuel is a solid, homogeneous mixture of hydrided uranium-zirconium alloy containing 8.5 wt-% uranium enriched to 20 wt-% in uranium-235. The hydrogen-to-zirconium atom ratio is approximately 1:1.

Each element is clad with a 0.030 inch (0.076 cm) thick aluminum or stainless steel can, and all closures are made by heliarc welding. Two 4 inch (10 cm) 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. Aluminum or stainless steel end fixtures are attached to both ends of the can, making the overall length of the fuel-moderator element approximately [redacted] inche[redacted] Fuel elements for use only in steady-state operation are anodized clear.

The lower end fixture of the fuel element 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 which permits cooling water to flow through the upper grid plate. This spacer block has a clear anodized aluminum finish for a standard, fully loaded, non-pulsing fuel moderator element, i.e., one containing approximately [redacted] of uranium-235. For an element that has three-quarters of the normal uranium content, the spacer has a green-anodized finish. For an element containing half the normal fuel loading, the spacer is red; and for a one-quarter-loaded element, it is orange. The fuel loading of an element may also be determined by means of grooves machined into the top end fixtures of the elements. The total weight of a fully loaded fuel element i[redacted]). Figure 4.4 shows the top markings for fuel element loading.

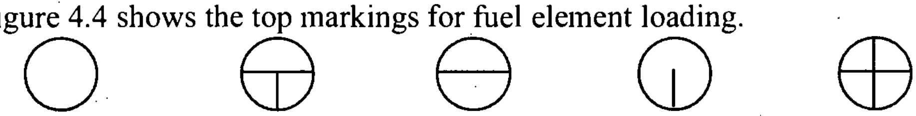

Figure 4.4 Fuel Loading

[redacted]

4.2.5 Stainless Fuel-Moderator Elements

The stainless fuel-moderator elements (Figure 4.6) are similar to the aluminum ones described. The significant difference is that they are clad with stainless steel rather than aluminum.

[redacted]

4.2.6 Graphite Dummy Elements

Graphite dummy elements occupy the grid positions not filled by fuel-moderator elements and other core components. The graphite dummy elements are canned in aluminum and have aluminum end fixtures and spacer blocks. These elements are of the same dimensions as the fuel moderator elements, but are filled entirely with graphite. Each graphite dummy element weighs 2.8 pounds (1.27 kg) and is anodized after assembly. The spacer blocks have a blue-anodized finish to make the dummy elements easily distinguishable from fuel-moderator elements. When a dummy element is properly installed in the core, the top of the triangular spacer block is about level with the top of the top grid plate.

4.2.7 Control Rod Guide Tube

The three control rod guide tubes (Figure 4.7) that are provided in a standard core can fit into any of the 90 fuel positions. The tubes are supported from the lower grid plate and are laterally positioned by both grid plates; they extend approximately 10.25 inches (26 cm) above the top grid plate. The outside diameter of the guide tubes is 1.495 inches (3.797 cm).

Water passage through a control rod guide tube is provided by a large number of holes evenly distributed over its entire length. The guide-tube assembly is made of anodized aluminum to increase resistance to wear and corrosion. The lower end of the assembly is a cylindrical rod whose axis is concentric with the axis of the tube. The rod fits into the holes in the lower grid plate for lateral positioning. A locking device is built into the lower end of the assembly.

After insertion in the core, a tube is locked in place with a special wrench (Figure 4.7) made for this purpose. The wrench slides easily into the guide tube, fitting by means of a hexagonal rod into the stainless steel lock screw in the lower end of the tube. Turning the lock screw clockwise until it is tight will extend a locking wire under the bottom grid plate. With the wire protruding, the tube assembly cannot be accidentally removed from the grid. The tube can be removed from the grid only after the locking screw has been turned counterclockwise five or six turns.

Figure 4.7 Control-rod guide tube and removal tool

4.2.8 Control Rods

The shim, safety, and regulating control rods (Figure 4.8) are sealed aluminum tubes containing powdered boron carbide as the neutron absorber, or poison. The rods have an outside diameter of 1.25 inches (3.175 cm). The upper end of each of the three control rods is a male, threaded 1/2-13 connection, which screws into the extension rod that is connected to the control rod drive assembly at the top of the reactor tank. A 0.093 inch (0.2362 cm) diameter pin, which is inserted through the lower extension rod and into the threaded upper end of the control rod, prevents the parts from working loose. In a similar fashion, 0.250 inch (0.635 cm) diameter pins are used in the connection of the upper and lower extension rods. All control rods are approximately 20 inches (50 cm) long and have a vertical travel of approximately 15 inches (38 cm).

Figure 4.8 Control Rod

4.2.9 Control Rod Drives

The drive assemblies for the control rods are fastened to a mounting plate located on the center channel. The standard control rods have electrically driven rack-and-pinion drives. Electrical connectors are provided on the bridge to permit easy disconnection and removal of a rod drive.

Rack-and-pinion drives (Figure 4.9) are used to position the control rods. Each drive consists of a single-phase, reversible motor, a magnet rod-coupler, a rack-and-pinion-gear system, and a ten-turn potentiometer which provides an indication of rod position. The pinion gear engages a rack attached to a draw tube supporting an electromagnet. The magnet engages an iron armature attached above the water level to the end of a long connecting rod that terminates at its lower end in the control rod. The magnet, its draw tube, the armature, and the upper portion of the connecting rod are housed in a tubular barrel. This barrel extends below the reactor water level, with the lower end of the barrel serving as a mechanical stop to limit the downward travel of the control rod assembly. Partway down the upper portion of the connecting rod, i.e., just below the armature, is a piston that travels within the barrel assembly. Since the upper portion of the barrel (Figure 4.10) is well-ventilated by large slotted openings, the piston moves freely in this range; but when the piston is within 2 inches (5 cm) of the bottom of its travel, its movement is restrained by the dashpot action (Figure 4.11) in the lower end of the barrel. This dashpot action reduces bottoming impact when rods are dropped by removal of magnet current during a scram. The piston is connected to the neutron absorber by aluminum rod sections bolted together (Figure 4.12).

Clockwise rotation of the motor shaft (as viewed from the pinion) raises the draw-tube assembly. When the electromagnet attached to the draw tube is energized, the armature and connecting rod rise with the draw tube, and the control rod is withdrawn from the reactor core. When the reactor is scrammed, the electromagnet is deenergized and the armature is released. The armature, connecting rod, and control rod drop by gravitational force to reinsert the neutron poison into the reactor core.

The safety rod drive motor is non synchronous, single phase, and electrically reversible and can insert or withdraw a rod at a rate of about 19 inches (48.3 cm) per minute. The shim rod drive motor is non synchronous, single phase, and electrically reversible and can insert or withdraw a rod at a rate of about 11 inches (27.9 cm) per minute. The regulating-rod drive, which has a stepper type motor, is driven a maximum of 24 inches (61 cm) per minute. Electrical dynamic braking and static braking on the motors are used to provide fast stops and to limit coasting or over travel.

Limit switches mounted on each drive assembly stop the rod-drive motor at the top and bottom of travel and provide switching for console indicator lights, which show:

1. When the magnet is in the UP position;

2. When the magnet (and thus the control rod) is in the DOWN position; and
3. When the magnet is in contact with the control-rod armature as sensed by the down limit switches.

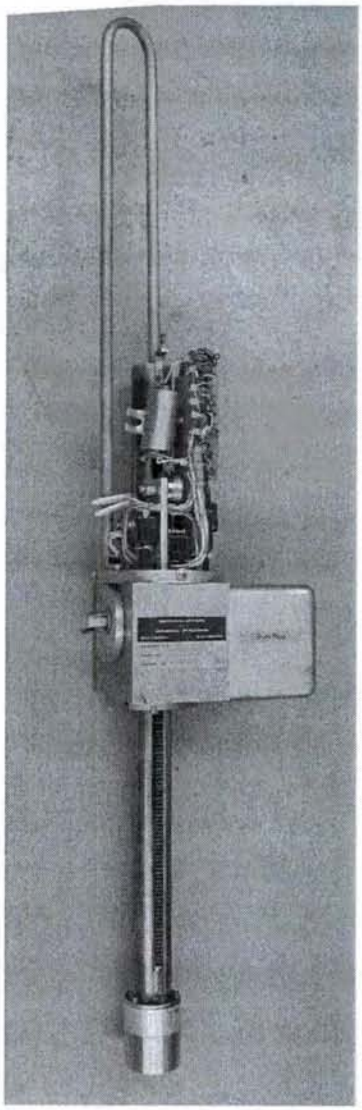

Figure 4.9 Control Rod Motor

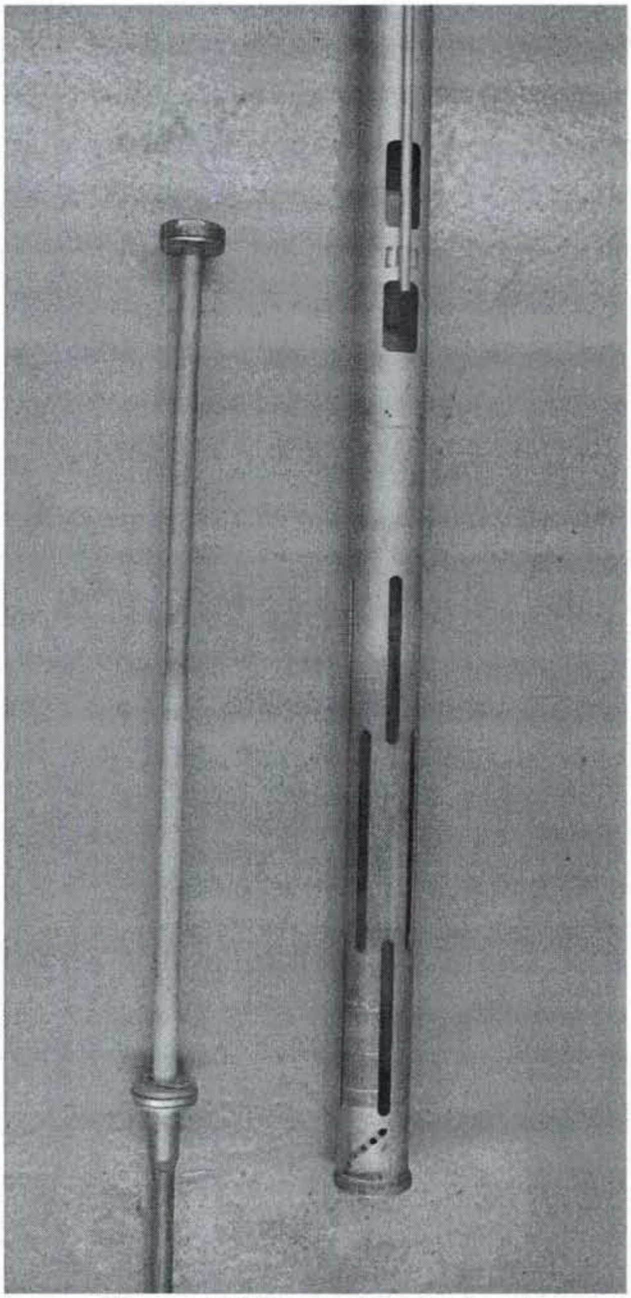

Figure 4.10 Control Rod Barrel

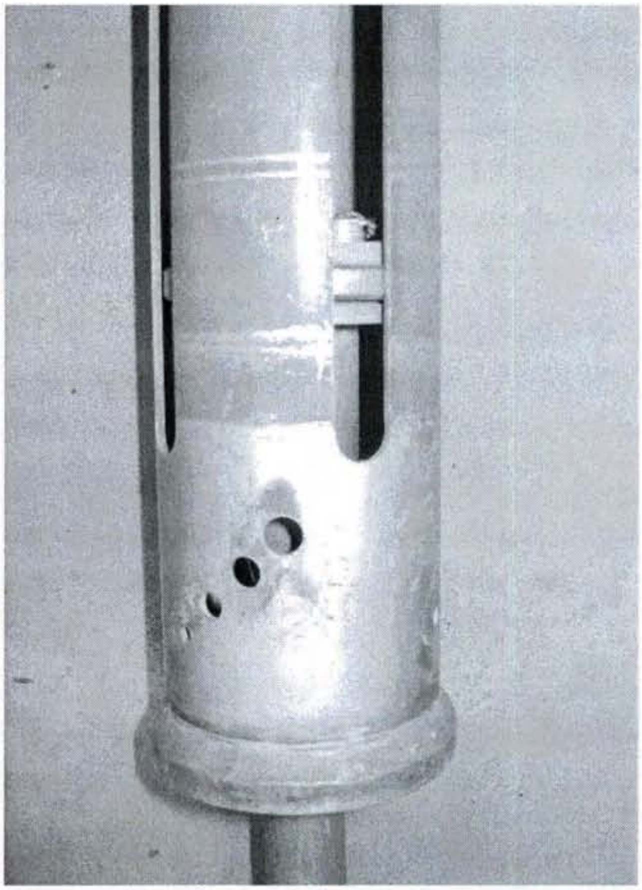

Figure 4.11 Dash Pot

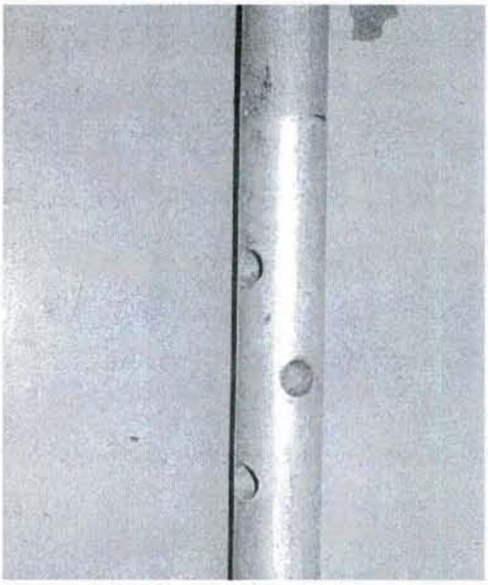

Figure 4.12 Control Rod Attachments

4.2.10 Source Holder

The source holder (Figure 4.13) is an anodized-aluminum rod assembly, with a cavity to contain the neutron source. The dimensions of this assembly permit it to be installed in any of the fuel locations in the core, [redacted]

[redacted]

Figure 4.13 Source Holder

The source holder is cylindrical, with a small shoulder at the upper end. This shoulder supports the assembly on the upper grid plate, the rod itself extending down into the core region. The rod clears the lower grid plate by about 0.5 inches (1.3 cm). The neutron source is contained in a cavity in the lower portion of the rod assembly and is located approximately at the vertical center of the core. This cavity is cylindrical, 1.17 inches (3 cm) in diameter, and 5.93 inches (15 cm) deep.

The upper and lower portions of the rod are screwed together, and each is provided with 1.325 inch (3.37 cm) flats for wrench application. A soft aluminum ring (Type 1100-0) seals the cavity against water leakage. This sealing ring is to be used only once; it should be replaced each time the source rod is disassembled and assembled.

4.3 Neutron Detectors And Mounting

The three neutron detectors, one compensated ion chamber, one uncompensated ion chamber, and the fission neutron detector (Figure 4.14) are each enclosed in a seal-welded, pressure-tested aluminum container. The electrical connections for each chamber are contained in a 0.75 inch (1.9 cm), offset aluminum pipe, which terminates above the water level at the side of the tank. A flanged, gasketed joint is provided below the offset. The overall length of the assembly is approximately 23 feet (7 m).

The chambers are located adjacent to the core reflector. To facilitate removal and reinstallation, each chamber is positioned inside an aluminum guide tube. These tubes are attached to the outer edge of the reflector assembly.

Each ion-chamber tube assembly is attached to the aluminum channel around the upper perimeter of the tank with a U-bolt mounted on a simple aluminum bracket.

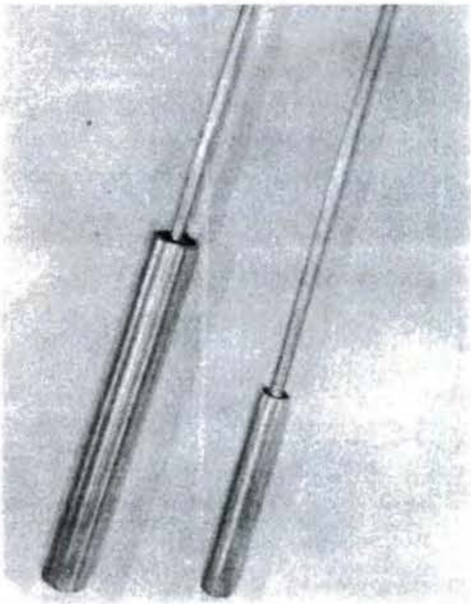

Figure 4.14 Neuron Detectors

4.4 Underwater Lights

The reactor core is illuminated by four waterproof lights in the reactor tank (Figures 4.15 and 4.16). Each light assembly consists of a 300 W, 110 V sealed-beam light enclosed in a waterproof aluminum housing equipped with a SHALDA No. 1306D lens. A 0.75 inch (1.9 cm) aluminum pipe from the housing to the top of the tank forms a support for the housing and serves as a waterproof conduit for the electrical wiring. The lights are supported from the aluminum channel at the top of the tank by an aluminum clamp assembly and are connected to the electrical receptacles at the top of the tank.

Figure 4.15 Underwater Light

4.5 Fuel-Storage Racks

Fuel-storage racks (Figure 4.16), each capable of holding [redacted] fuel elements, are located underwater along the walls of the reactor tank to provide temporary storage for fuel-moderator or graphite dummy elements. Each rack is 20 inches (50.8 cm) high and 22.5 inches (57 cm) wide, with 1.625 inch (4.13 cm) diameter cutouts, and is made of 16-gauge aluminum. [redacted] Stainless steel nuts (3/4-10) and spring-lock washers are used in the fastening.

[redacted]

4.6 Limiting Design Bases

4.6.1 Steady State Operation

General Atomics utilized a mixed core of stainless steel and aluminum-clad fuel from 1960 when they were first authorized to use a limited number of stainless steel clad together with aluminum-clad elements until cessation of operations. The mixture was authorized as long as fuel temperature in the mixed aluminum and stainless steel core did not exceed 550 °C (1022 °F). This was authorized by Amendment 9 to License No. R-38 in Oct., 1960. Change #1 to License No. R-38, dated Sept. 1965, authorized General Atomic to use stainless steel, aluminum, Hasteloy X or Incoloy 800 up to a full core loading. In addition Amendment No. 31 to Section 4.0 of GA TRIGA Mark I (R-38) Technical Specifications (dated March 1994) authorize various cladding materials and thicknesses, including a mixture of aluminum and stainless steel clad fuel. Consequently, since a mixed core of aluminum and stainless steel was used in the Mark I reactor for more than 35 years at a thermal power greater than the RRR reactor, it is concluded that the health and safety of the public will not be endangered by operating with mixed stainless steel and aluminum fuel.

The elements are spaced so that about 33% of the core volume is occupied by water. This fuel-to-water ratio in the core was selected because calculations show that it gives very nearly the minimum critical mass. At the present time, the RRR contains 64 active fuel elements for steady-state operation at 250 kW. The fuel inventory consists of 55 aluminum-clad elements and 9 stainless steel clad elements. There are currently eighty-four fuel-element positions available in the lattice; the unused positions are normally occupied by graphite dummy elements, i.e., elements in which the uranium-zirconium-hydride fuel is replaced by graphite, but are available for addition of fuel in the future for an upgrade to steady-state operation at 500 kW.

The limit for TRIGA® fuel is dictated by temperature. This limit is dependent on the type of TRIGA® used. The RRR has both aluminum-clad low hydride (H/Zr ratio less than

1.5) fuel and stainless steel high hydride (H/Zr ratio greater than 1.5) fuel. The majority of which is aluminum-clad. The TRIGA® fuel with low hydride ratio has a lower temperature limit than the high hydride fuel. Figure 4.17 indicates that the higher hydride compositions are single phase and are not subject to large volume changes associated with the phase transformations at 530 °C (986 °F) in the lower hydrides. The high hydride limit stems from the out-gassing of hydrogen from U-ZrH fuel and the subsequent stress produced in the fuel element clad material. It should be noted, however, that the higher hydrides lack any significant thermal diffusion of hydrogen [1].

The results of General Atomic's experimental and theoretical determinations [2,3] show that fuel element integrity is not compromised for cladding temperatures at or less than 500 °C (932 °F). Reviews of these experiments and determinations can be found in NUREG-1282 [4] and NUREG 0988 [5].

4.6.2 Dynamic Behavior of Reactor

This section will consider the behavior of the reactor as a result of the sudden insertion of a large amount of excess reactivity into the core. General Atomic performed testing and evaluation of TRIGA® by undertaking a high-power transient test program under controlled experimental conditions on the prototype reactor. A special license was obtained from the AEC for this series of tests. Some of the salient features of the tests are summarized here [6]. The test was performed using the Torrey Pines TRIGA® Mark I reactor identical in construction to the RRR. The only difference is that the Torrey Pines reactor had two safety rods worth $2.50 and $2.00, a pneumatically driven regulating rod worth $2.50 and a shim rod worth $4.50.

A $2.00 step reactivity insertion has been demonstrated, without deleterious effects either to the reactor or to operating personnel in the immediate vicinity of the reactor. This $2.00 insertion yielded a reactor period of 10 ms and a peak power of approximately 250 MW. This excess reactivity was rapidly compensated by the large prompt negative temperature coefficient, which is an inherent characteristic of this reactor core. Within 30 seconds after initiation of the transient, the reactor power level had returned to an equilibrium of 200 kW. The total energy release in the prompt burst was approximately 10 MW-s. The maximum transient fuel temperature was about 360 °C (680 °F).

Curves of the transient power level and of the fuel temperature during this transient are shown in Figure 4.18. No boiling was observed in the reactor tank and no disturbance of the shielding-water surface was noted during the $2.00 transient. The integrated radiation dose that an individual would have received had he stood immediately over the reactor tank during this power transient would have been 21 mrem.

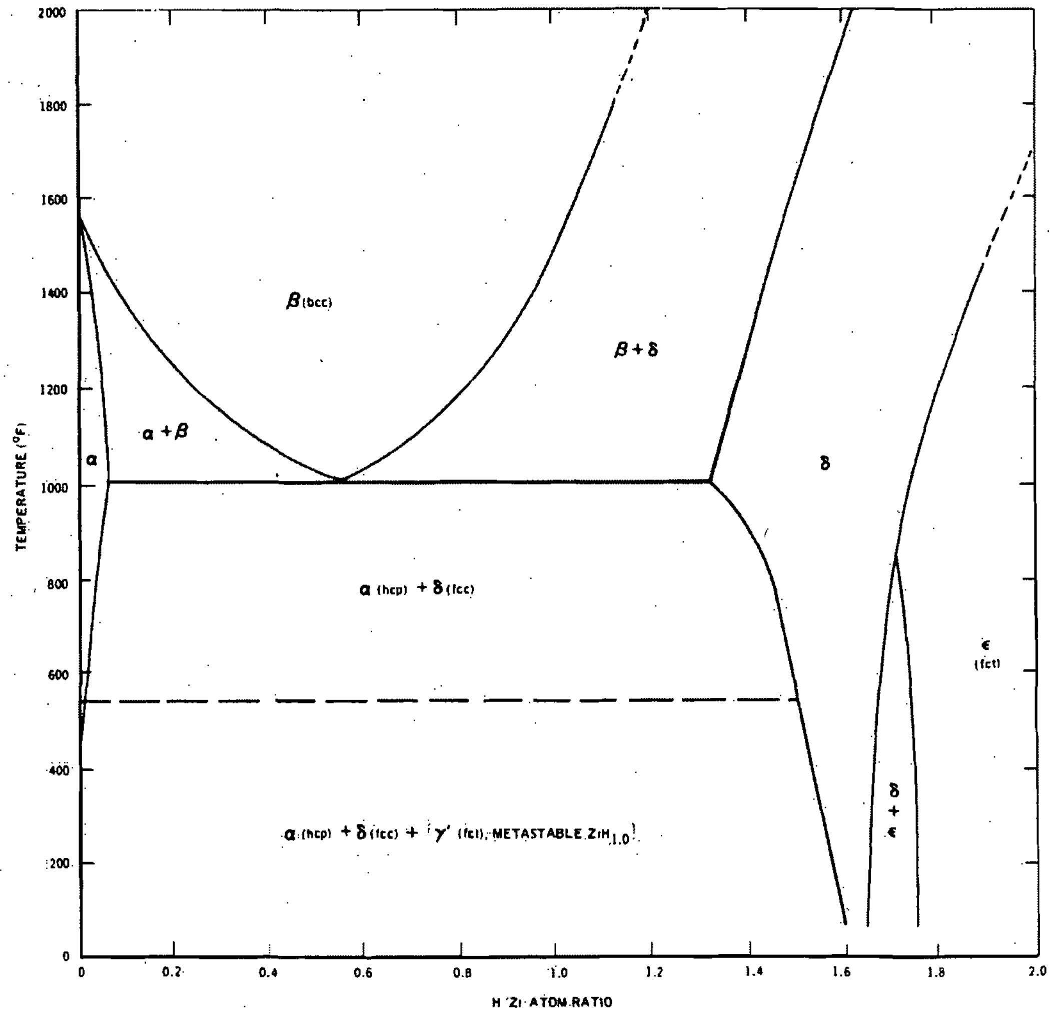

Figure 4.17 Zirconium Hydride Phase Diagram, Showing Boundary Determination [7]

During the quasi-equilibrium experiments on the prototype TRIGA®, the reactor was operated at a power of 330 kW for a period of approximately one hour with no indication of instability or bulk boiling in the reactor core. The data obtained in these experiments provide an experimental value of 80 ± 5 μsec for the effective neutron lifetime for this reactor. The temperature coefficient measured in the quasi-equilibrium experiments can be fitted to good approximation by a constant over the experimental temperature range. This temperature coefficient has been measured to be $0.016 reactivity loss per degree centigrade rise in fuel temperature.

The core consists mostly of a aluminum-clad fuel with a H:Zr ratio of 1.0. Zirconium occurs in two crystalline forms; alpha (stable below 860 °C (1580 °F) and Beta (stable above 860 °C (1580 °F)) (Figure 4.18). The alpha phase is close-packed hexagonal and does not absorb any large amount of hydrogen. The small amount of hydrogen it does take up forms a solid solution with it. Absorption of more hydrogen at elevated temperatures (>530 °C) cause a transition of part solid to the beta phase, which is body-centered cubic and in which hydrogen is added can go into solid solution up to an H:Zr ratio of 1.0. If more hydrogen is added than is required to saturate the beta phase, the precipitation of the gamma hydride, which has a ratio of H:Zr ≤~1.5 begins.

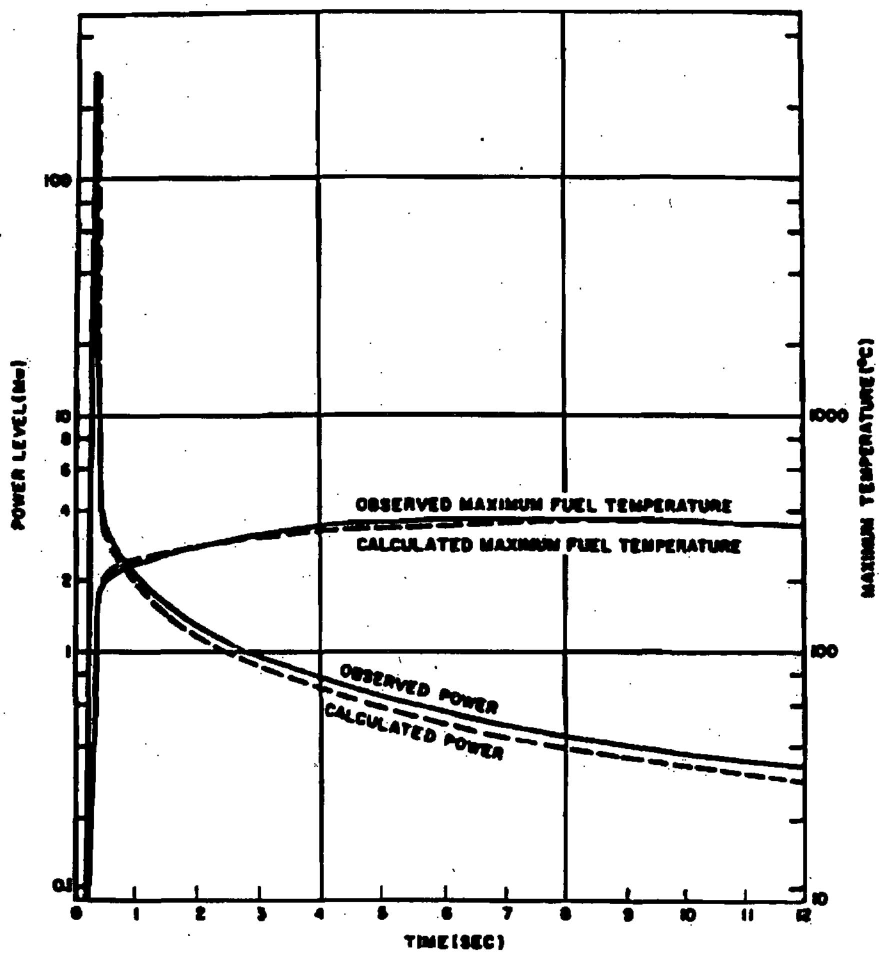

Figure 4.18 $2.00 Reactivity Transient

A loss in the integrity of the fuel element cladding could arise from a buildup of excessive pressure between the fuel and the cladding if the fuel temperature exceeds the safety limit. The heating of air, fission product gases, and hydrogen causes the pressure from the dissociation of the fuel-moderator. The magnitude of this pressure is determined by the temperature of the fuel element and by the hydrogen content. Experience when operation of TRIGA®-fueled reactors at power levels up to 1500 kW shows no damage to the fuel due in thermally induced pressures.

Thermal cycling tests have been performed to verify fuel matrix stability with respect to swelling or elongation. Simnad [3] has described these tests with temperatures in the range 500 °C (932 °F) to 725 °C (1337 °F). He has explained why there are no important changes in length or diameter of the test samples even though a small phase transition did occur at 653 °C (1207 °F) (orhorthombic to tetragonal). For a TRIGA® fuel with fuel temperatures ≤ 200 °C (392 °F), there is no phase charge or other transition to produce elongation or swelling in the fuel matrix.

Under long term, high burnup conditions of irradiation, the possibility would exist for hydrogen migration and accumulation of fission products in the fuel. Simnad [3] has treated these features at length and demonstrated that none of these effects is important for fuel temperatures below 500 °C (932 °F), especially if the reactor is not pulsed as is the case for the RRR. A temperature of 500 °C (932 °F) is well above the fuel temperatures characteristic of a TRIGA® operating at 250 kW.

On the basis of the evidence presented above, it is concluded that there is no hazard associated with a rapid insertion of as much as $2.00 excess reactivity in the RRR. From the above experiment the following reactivity limits can be justified:

4.6.2.1 Excess Reactivity

The objective of limiting excess reactivity is to prevent the fuel element temperature safety limit from being reached by limiting the potential reactivity available to the reactor for any condition of operation. The maximum power excursion that could occur would be one resulting from inadvertent rapid insertion of the total available excess reactivity. Limiting the fuel loading of the RRR reactor to $3.00 excess reactivity under clean-cold critical conditions will assure that the fuel temperature will not reach the maximum fuel temperature of 530 °C (986 °F) where a phase change resulting in great enough internal pressure to cause cladding failure occurs [2,3].

4.6.2.2 Shutdown Margin

Requiring a minimum shutdown margin of $0.50 with the highest worth control rod fully withdrawn, the highest worth non secured experiment in its most reactive state, and the reactor in the cold critical condition without xenon, assures that the reactor can be shut down from any operating condition.

4.6.2.3 Reactivity limits on experiments

Limiting the worth of a single experiment to $1.00 assures that sudden removal of the experiment will not cause the fuel temperature to rise above the critical temperature level of 530 °C (986 °F). Limiting the worth of all experiments in the reactor and in the associated experimental facilities at one time to $1.00 will also assure that removal of the total worth of all experiments will not exceed the fuel element temperature safety limit of 500 °C (932 °F).

4.6.3 Stainless Steel Clad Fuel

Transition to stainless steel clad fuel elements will not change neutronic evaluations presented for the RRR, however calculations performed by General Atomics and confirmed by experiments indicate that no cladding damage occurs at peak fuel temperatures as high as 1175 °C (2150 °F) for high-hydride-type (U-$ZrH_{1.65}$) stainless steel clad fuel [2,3]. Therefore, for a future core with only stainless steel clad fuel for operation at up to 500 kW, fuel temperature limits of 1,100 °C (with clad < 500 °C (932 °F)) and 930 °C (with clad > 500 °C (932 °F)) for U-ZrH with a H/Zr ratio less than 1.70 have been set to preclude the loss of clad integrity.

4.7 Bibliography

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2) *Simnad, M.T., Foushee, F.C. and West, G.B., "Fuel Elements for Pulsed TRIGA Research Reactors," Nuclear Technology, 28:31, 1976.*

3) *Simnad, M.T., "The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel," GA-4314, E-117-833, GA Technologies, Inc., San Diego, CA, 1980.*

4) *"Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors," Docket No. 50-163, GA Technologies, Inc., U.S. Nuclear Regulatory Commission, August 1987.*

5) *Safety Evaluation Report related to the renewal of the operating license for the research reactor at the Omaha Veterans Administration Medical Center, Docket No. 50-131, U.S. Nuclear Regulatory Commission, July, 1983.*

6) *R.S. Stone and H.P. Sleeper, "TRIGA Transient Experiments," Interim Report General Atomic GA 531, September 19, 1958.*

7) *K.E. Moore and M.M. Nakata, "Phase Relationships in Alpha-plus-Delta Region of the Zr-H System," USAEC Report AI-AEC-12703, Atomics International, September 30, 1968.*

Chapter 5

Reactor Coolant Systems

Reed Research Reactor Safety Analysis Report

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Chapter 5

Reactor Coolant Systems

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5 REACTOR COOLANT SYSTEMS

The Reed Research Reactor (RRR) is located at the bottom of an [redacted] 25 foot (7.6 m) deep open-top aluminum pool, which holds 25,000 gallons (95,000 L) of shielding and cooling water. Due to the small size and low power of the RRR, the primary coolant system is not a necessary safety system of the facility, but is used for maintaining efficient operations. The water in the tank is used to moderate the reactor, to cool the fuel rods during operation, and to shield the reactor room from radiation. In the unlikely event that the pool was emptied, design analysis of TRIGA® fuel shows that it may be cooled by natural convection in air without risk of fuel failure.

5.1 Summary Description

The primary cooling system serves the following five major functions:

1. Provides a means of dissipating heat generated in the reactor;
2. Reduces radioactivity in the water by removing nearly all particulate and soluble impurities;
3. Maintains low conductivity in the water in order to minimize corrosion of reactor components, especially the fuel elements;
4. Maintains the optical clarity of the primary water; and
5. Shields reactor bay from radiation generated in the core.

Figure 5.1 shows the primary cooling system and Figure 5.2 shows the secondary cooling system.

The primary system contains purified water and is open to the atmosphere. The reactor core is cooled by natural convection alone. To assist in temperature control during extended operation, bulk heat is transferred by forced convection across a heat exchanger to the secondary cooling system. The secondary cooling system then transfers the heat to the environment via a cooling tower, using service water treated with caustic and algaecide to prevent corrosion and biological growth. This cooling system combination provides enough heat removal for normal operation. Makeup for both water systems comes from the municipal water system, though water entering the primary system passes through a preliminary filter before entering the pool.

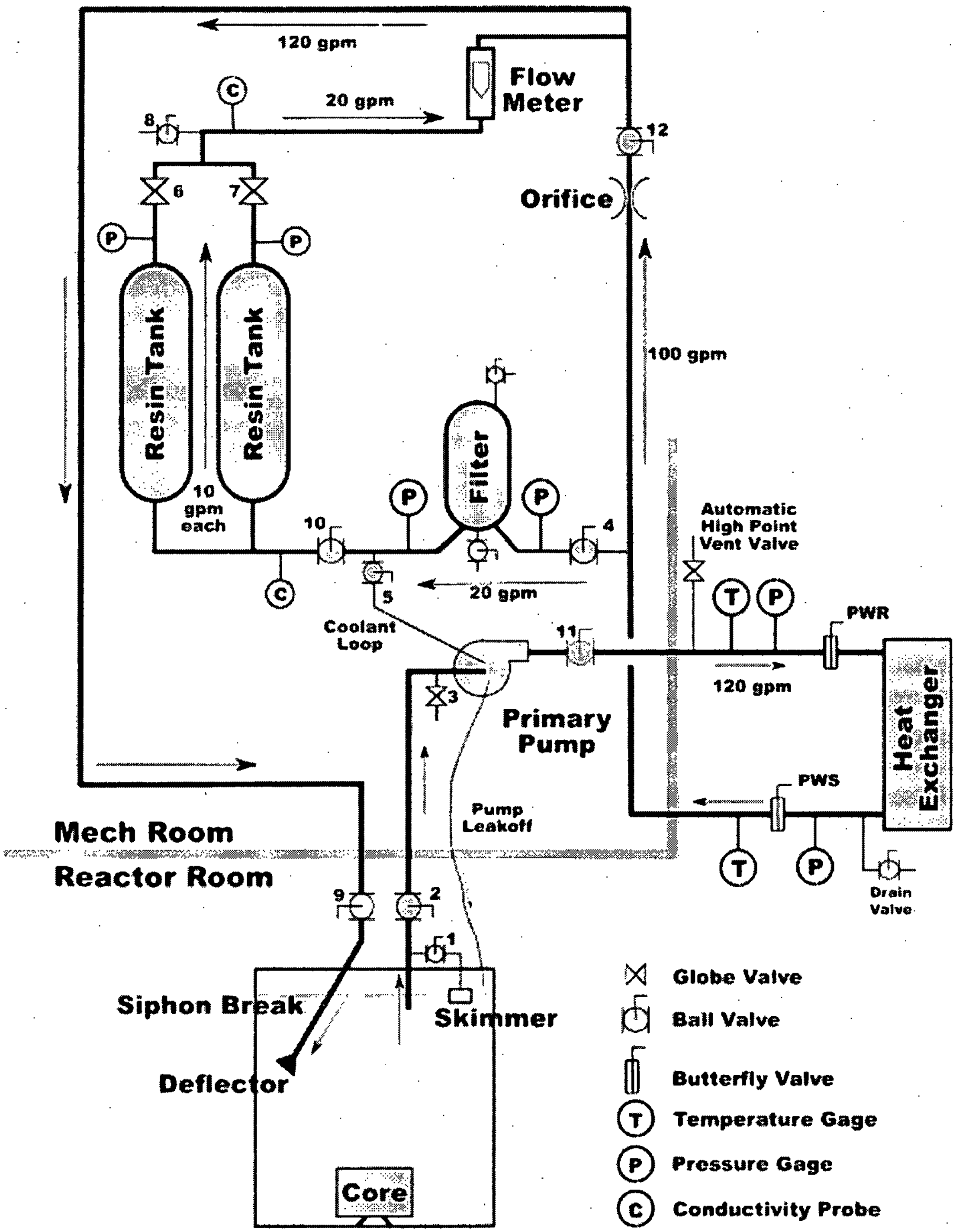

Figure 5.1 Primary Cooling System

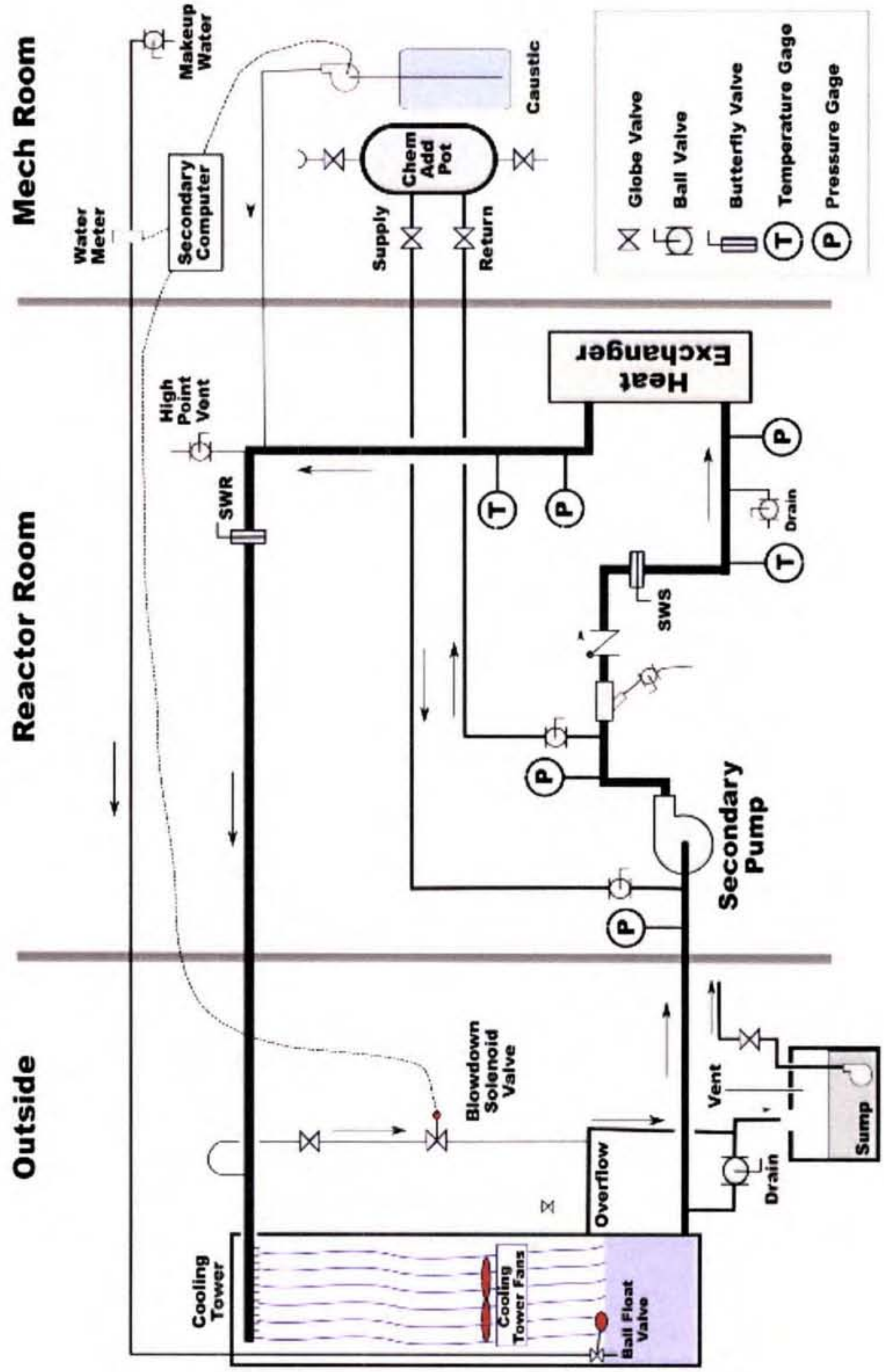

Figure 5.2 Secondary Cooling System

5.2 Primary Coolant System

Principal functional requirements of the primary coolant system are to transfer heat from the reactor core out of the facility by way of the secondary cooling system and to provide radiation shielding directly above the reactor core. Primary bulk water is kept below 55°C in order to prevent damage to the demineralizer. At temperatures above this level, the resin may break down and be dispersed in the reactor pool, threatening corrosion of reactor systems. In order to monitor temperature, a thermocouple in the pool reads out in the control room and alarms at a temperature below 40°C. The radiation shielding requirement is fulfilled by keeping at least 16 feet (4.9 m) of water directly above the reactor core. This is a pool level of approximately 20 feet (6 m).

The system consists principally of a pump, heat exchanger, fiber cartridge filter, mixed-bed type demineralizer, and flow meter connected by suitable aluminum piping and valving, as shown in Figure 5.1. The primary system has two suction inlets in the reactor pool; one large intake pipe located 28 inches (71 cm) below the pool surface, and a skimmer that collects foreign particles floating on the pool surface.

5.2.1 Skimmer

A surface skimmer (Figure 5.1) collects foreign particles that float on the surface of the reactor-tank water. The skimmer is connected to the main water suction line by 1 inch (2.54 cm) diameter piping and a valve.

The skimmer is an all-plastic, 8 inch (20.3 cm) diameter cylinder that contains a basket in its upper end. This cylinder fits over a disk rigidly supported by the 1 inch (2.54 cm) diameter piping. Water at the surface of the tank flows over the top of the floating cylinder, so that large floating foreign particles are deposited in the basket. Particles small enough to pass through the basket screen are collected in the filter cartridges located downstream in the purification loop.

5.2.2 Pump

The water system pump is a centrifugal-type with a stainless steel body and impeller. The pump is driven by a directly-coupled induction motor. The suction of the pump is a 2 inch (5 cm) flanged-pipe connection, and the discharge is a 1.5 inch (3.8 cm) flanged-pipe connection.

5.2.3 Heat Exchanger

From the primary pump, the water flow enters a plate-type heat exchanger that acts to transfer heat from the primary to the secondary coolant. The two coolants are separated by thin metal plates to maximize surface area for heat transfer. A temperature gauge and a pressure gauge are located at each inlet and outlet of the heat exchanger (primary inlet, primary outlet, secondary inlet, secondary outlet) and butterfly valves allow isolation of inlet and outlet of each coolant system in case of heat exchanger damage to prevent mixing of coolant water.

The heat exchanger was installed in 1994. It is an Alfa Laval Thermal, Inc. plate-type heat exchanger rated for 250°F (120°C) and 150 psi (1 MPa). There are 69 plates for a heat transfer area of 111 ft^2 (10.3 m^2).

5.2.4 Cleanup Loop

After the primary coolant passes through the heat exchanger, approximately 20 gallons per minute (88 liters per minute) are piped through the cleanup loop before returning to the main coolant flow loop.

The filter removes insoluble particulate matter from the reactor water system. It uses replaceable fiber cartridges (i.e., the cartridges are removed from the filter vessel and replaced when they become clogged, rather than being back-flushed and reused). Three filter cartridges of 25-micron ratings are available for the filter vessel. In addition to improving the optical clarity of the water in the reactor tank, the removal of solid particles from the water by the filter extends the operating life of the demineralizer resin. The filter will become slightly radioactive in use and will be disposed of in accordance with Chapter 11, Radiation Protection and Waste Management.

Two pressure gauges are provided in the filter line, one before the filter and one after the filter. These gauges can be used to measure the pressure drop across the filter as an aid in determining the extent of filter clogging.

The prime function of a demineralizer is to maintain the conductivity of the water at a sufficiently low level to prevent corrosion of the reactor components exposed to the water, particularly the fuel elements. A demineralizer performs this function by removing soluble impurities from the water.

The demineralizer is a mixed-bed type that removes both positive and negative ions from the circulating water. The positive ions are replaced by hydroxyl (OH) ions and the negative ions by hydrogen (H) ions. The OH and H ions combine to form water. Consequently, any contaminants in the water are concentrated on the resin and replaced by pure water. Any radioactive ions in the water are therefore absorbed and concentrated in the resin bed. In normal use, a demineralizer will become slightly radioactive.

Each demineralizer unit contains 3 ft^3 (85 L) of an intimate mixture of anion resin and cation resin.

There are two conductivity probes in the water system. One, located upstream from the demineralizer, measures the conductivity of the water leaving the reactor tank. The other, located downstream from the demineralizer, measures the conductivity of the water as it leaves the demineralizer and thus indicates whether the demineralizer is operating properly or whether the resin has become depleted. Each conductivity probe consists of a titanium-palladium electrode conductivity cell mounted in the water system through a threaded pipe fitting. Each cell is connected to a microprocessor based conductivity meter that provides a local display of both inlet and outlet conductivity. The probes measure water conductivity of 0 to 10 microSiemens/cm from 0° to 100°C.

Connections to the demineralizer are made with Victaulic snap-type couplings. These couplings effect a seal by means of specially-grooved pipe nipples, a neoprene-rubber gasket, and a toggle-type coupling. The couplings may be easily disconnected by opening the toggle joint.

The flowmeter is mounted downstream from the demineralizer. It has a range of 0 to 28 gpm (0-123 lpm). The meter is operated by the flow of water, which forces the flow rotor upward in the tube. Changes in the rate of flow produce a corresponding change in the height of the rotor. The flow rate can then be determined by comparing the height of the rotor with the meter scale. Removable pipe plugs are located at the top and bottom of the meter to allow cleaning of the internal parts of the meter without removal of the meter from the line. Inlet and outlet connections have 1 inch (2.54 cm) standard female pipe threads.

5.2.5 Orifice

To establish proper flow through the water-purification loop, a stainless steel orifice assembly is installed in the piping from the heat exchanger.

5.2.6 Piping And Valves

All piping and fittings in the water system are of aluminum alloy. In the main water circuit, the piping is primarily of nominal 2.5 inch (6.35 cm) Schedule 40 pipe. The water-purification loop is primarily of nominal 1 inch (2.54 cm) and 1.5 inch (3.8 cm) Schedule 40 pipe. The piping unions have 0.0625 inch (0.159 cm) thick polyethylene gaskets for sealing.

The ball valves in the system are of aluminum construction, with synthetic rubber seals. The gate and globe valves are also of aluminum construction and have Teflon packings.

All of the water is returned to the pool through a deflector nozzle, which creates a swirling current in the pool that increases the time it takes water to travel from the bottom of the pool to the surface, and thus provides ample time for the decay of radioactive nitrogen-16.

Water level is normally kept approximately 8.7 inches (22 cm) below the top of the reactor tank. To prevent a malfunction in the primary system from draining the pool, the primary inlet is approximately 28 inches (71 cm) below water level, and there is a siphon break in the return pipe 40 inches (102 cm) below water level. A pool level meter activates a red light on the console, in the control room, in the entrance hallway, and outside the facility at water levels more than 2 inches (5 cm) above or below normal.

5.3 Secondary Cooling System

The secondary cooling system circulates water from the heat exchanger through the cooling tower. The water utilized in the secondary system is normal municipal water that has been treated with caustic and algaecide to minimize corrosion and biological growth, respectively. The system consists of a centrifugal pump, the heat exchanger, and the cooling tower, as well as an automatic caustic addition system and an algaecide feed loop. The algaecide loop consists of a feed pot to which algaecide is added on a regular basis. The cooling tower is located on the outside of the facility. The heat exchanger and secondary pump are located inside the reactor bay room, and the algaecide loop and the caustic reservoir are located in the mechanical room. Water is added to the cooling tower automatically by a float-operated valve, and overflow is drained into the sanitary sewer. To prevent freezing in winter the secondary can be manually drained to the sewer.

5.4 Makeup Water System

Makeup water for the primary system is provided by the municipal water supply. It is passed through a particulate filter and a carbon filter unit before entering the pool in order to extend the cleanup loop replaceable part lifetimes.

5.5 Nitrogen-16 Control System

The primary cooling system returns water to the pool through a diffuser nozzle. This diffusion pushes the pool water into a spiraling pattern, gently swirling the water and slowing its ascent to the top of the pool. This current provides the radioactive isotope nitrogen-16, with its half-life of 7.1 seconds, more than enough time to decay before reaching the surface. As a result, radiation levels in the reactor bay room remain low, even during periods of extended operation.

Chapter 6

Engineered Safety Features

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6 ENGINEERED SAFETY FEATURES

The Reed Research Reactor does not require or have any Engineered Safety Features.

Chapter 7

Instrumentation and Control Systems

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Chapter 7

Instrumentation and Control Systems

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7 INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Summary Description

The reactor is operated from a console located in the control room, at which the operator has a clear view into the reactor bay through large windows, and has all instrumentation necessary to monitor reactor operation and radiation safety close at hand. Instrumentation is either mounted on the console, near at hand, or in the reactor bay with readout clearly visible through the window.

The console allows operation of the reactor with interlocks preventing rapid reactivity insertion. The console allows for of automatic rod control, which modulates the movement of the least reactive control rod to keep power within a certain percentage of the currently selected linear range. The reactor instrumentation is all solid-state circuitry.

7.2 Design of Instrumentation and Control System

7.2.1 Design Criteria

The instrumentation and control system is designed to provide:

1. Complete information on the status of the reactor and reactor-related systems;
2. A means for manually withdrawing and inserting control rods;
3. Automatic scrams in response to excessive power levels;
4. Manual scram capability in case of emergency; and
5. Monitoring of radiation and airborne radioactivity levels.

Additional parameters not necessary for the reactor protection system are also monitored and displayed.

7.2.2 Design-Basis Requirements

The primary design basis for the Reed Research Reactor (RRR) is the safety limit on reactor power, designed to keep reactor fuel below a safe operating temperature. To prevent exceeding the safety limit, automatic scrams are provided for high power conditions, although none are required for reactor safety and none are taken credit for in this SAR. Interlocks limit the magnitude of reactivity insertions.

7.2.3 System Description

Reactor power is measured by three neutron detectors: a fission chamber, a compensated ion chamber, and an uncompensated ion chamber. The signal from the fission chamber is used by the wide range logarithmic channel. The compensated ion chamber is used by the multi-range linear channel. The uncompensated ion chamber runs a full-scale percent-power channel. A schematic is presented in Figure 7.1.

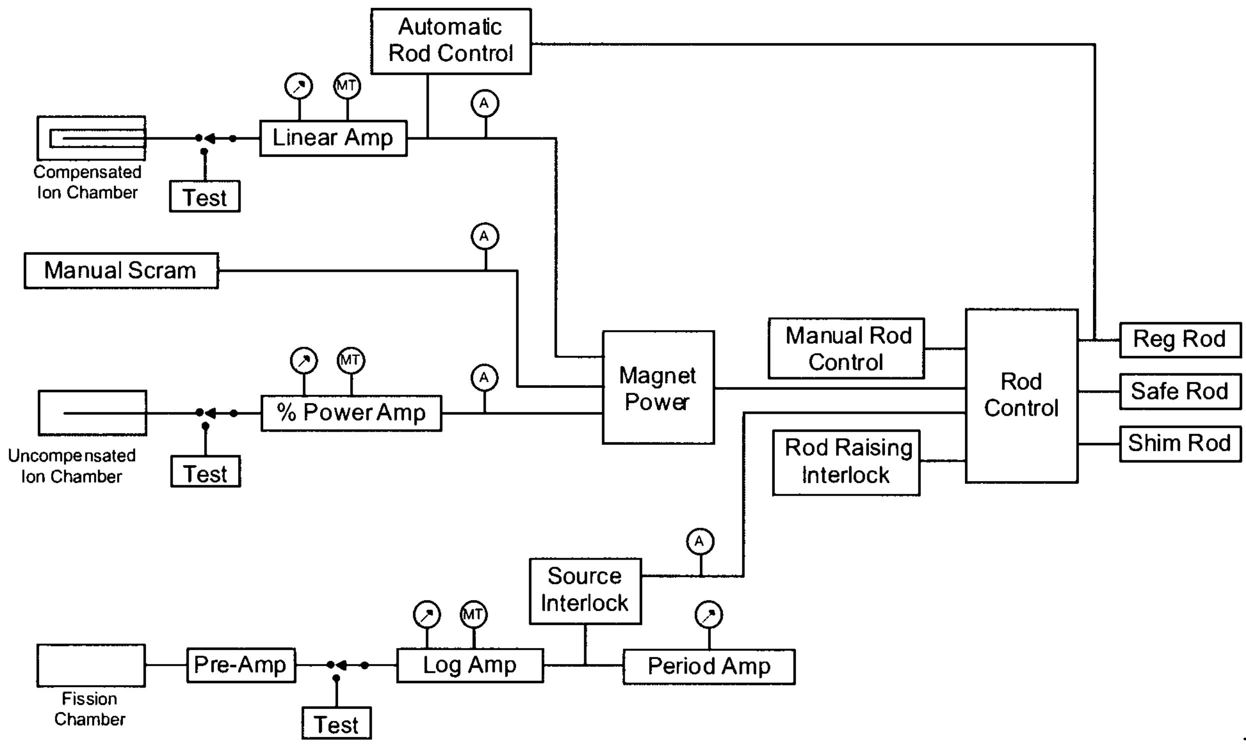

Figure 7.1 Instrumentation and Control

7.2.3.1 Fission Chamber

The fission chamber (called the Logarithmic Channel or the Log Channel) provides a continuous indication from 10E-8 to 100% power. It is lined with highly enriched uranium-235 and operates in the proportional region of the gas filled detector curve. Neutrons from the core interact with the uranium-235 lining to produce fission fragments, which ionize the fill gas. Gammas from the core (and background) ionize the gas in both chambers. At powers below 0.1% of full power the circuitry distinguishes the neutron induced fission fragments from gammas by means of a pulse height discriminator. At low powers most of the gammas come from the decay of fission products in the core, which is not indicative of reactor power. The signal is displayed as a percentage of full power. There is no count per second display. At powers above 0.1% the circuitry changes to display current signal like an ion chamber. At high powers there is no discrimination for gammas since the gamma signal is much smaller than the neutron signal, and the gammas are mostly coming from fission, which is proportional to power anyway. The Log Channel also displays the reactor period.

The only safety related feature of the Log Channel is the Source interlock, which ensures that rods cannot be withdrawn if there is no neutron-induced signal. Once the reactor is above 5 watts the Log Channel is no longer needed.

7.2.3.2 Linear Channel

The compensated ion chamber (called the Linear Channel) provides an indication from 0 to 120% of 10 ranges, up to full power. It has an outer chamber for the primary signal, and an inner chamber for the compensating signal. The outer chamber is lined with boron-10; the inner chamber is not. Fission neutrons from the core interact with boron in the outer chamber, releasing alpha particles that ionize the fill gas. Gammas from the core (and background) ionize the gas in both chambers. Electronics are used to subtract the inner chamber signal from the outer chamber signal, resulting in a signal that is proportional to the neutron signal, and thus the reactor power.

The Linear Channel has multiple linear ranges that slightly overlap. The channel automatically ranges up to the next highest (less sensitive) range when the signal is above 90% of the current range. It automatically ranges to the down (to a more sensitive) range when the signal is below 10% of the current range. It is also possible to manually select an individual range.

The Linear Channel is required to be operable with a high power scram whenever the reactor is not in the shutdown mode.

7.2.3.3 Percent Channel

The uncompensated ion chamber (called the Percent Channel) provides an indication from 0 to 120% of full power. It has only one chamber, and it is lined with boron-10. Fission neutrons from the core interact with boron, releasing alpha particles that ionize the fill gas. Gammas from the core (and background) also ionize the gas.

At low powers most of the gammas come from the decay of fission products in the core, which is not indicative of reactor power. This makes the Percent Channel inaccurate below approximately 1% of full power. At high powers the gamma signal is much smaller than the neutron signal, and the gammas are mostly coming from fission, which is proportional to power anyway. Thus there is no need for compensation for gammas.

The Percent Channel is required to be operable with a high power scram whenever the reactor is not in the shutdown mode.

The relative overlap of the three neutron detectors is show in Figure 7.2.

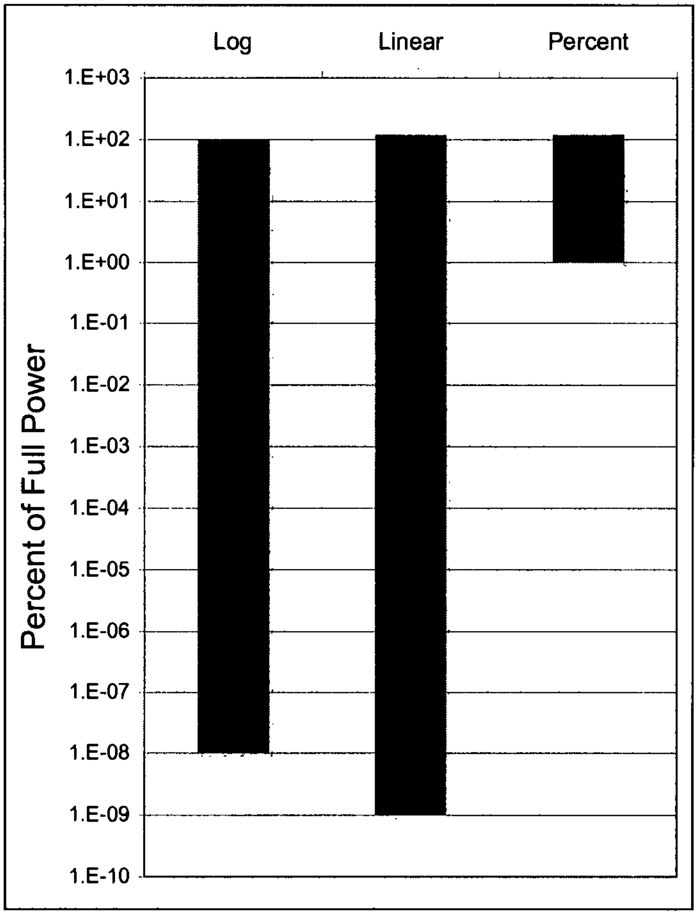

Figure 7.2 Relative Ranges of the Power Indications

7.2.4 System Performance Analysis

The system performance of the current instrumentation and control systems is excellent. Reliability has been high, with few unanticipated reactor shutdowns. Since daily checkouts are performed, any discrepancies would be observed and corrected in a prompt manner. The isolated outputs of the neutron channels allow the data to be utilized by other devices without concern over those devices affecting the channels.

7.2.5 Conclusion

The current instrumentation and control systems outperform the original equipment supplied with the reactor, while meeting all of the necessary design bases for the facility.

The human design factors used in control room development allow the reactor to be operated by a single individual. Checkout and testing procedures ensure that all equipment is maintained in operational status.

7.3 Reactor Control System

Three control rods are required for reactor operations to meet reactivity control requirements: a shim rod, a regulating rod, and a safety rod. These are positioned by control rod drives mounted on the reactor top center channel. The three rods share identical control circuitry and provide coarse and fine power control. All rods can be individually scrammed if necessary, or all three can be manually scrammed by the operator.

Each rod is controlled by a rack-and-pinion drive (Figure 7.3), with the rack mounted on a drawtube extending approximately 12 inches (30 cm) below the center channel. At the bottom of the draw tube is an electromagnet which, when actuated, connects the draw tube to the control rod armature and allows rod withdrawal and insertion. The draw tube and top of the armature are housed in a tubular barrel that extends below the water surface. Just below the connection to the magnet on the control rod armature is a piston that travels within the barrel assembly. Vents in the top portion of the barrel enable the water to escape, allowing the piston to move freely, but the bottom two inches (0.8 cm) restrain the motion by dashpot action, providing cushioning for the control rod mechanism in the event of a scram.

Rod position is indicated by a ten-turn potentiometer that sends motor position indication to the console. Position is indicated in percentage of total travel. Rod position is the same as motor position if the armature is connected. Connection is indicated by an annunciator on the console, which is controlled by the system of limit switches on each motor. The three limit switches, motor-full-out, motor-full-in, and rod-full-in, light the rod control pushbuttons at the motor-out and motor-in positions and turn off the contact light if the rod-in and motor-in switches do not agree. If the connection is broken, the motor immediately drives in after the rod to reconnect to the armature. The limit switches are shown in Figure 7.4.

The shim and safety rod drive motors are non-synchronous, single-phase, and electrically-reversible. The regulating rod drive motor is a stepper motor. Normal rod motion speed is about 11 inches (28 cm) per minute for the shim rod, 19 inches (48 cm) per minute for the safety rod, and 24 inches (61 cm) per minute for the regulating rod.

The regulating rod can be put in automatic rod control, which disables manual rod control and engages a rod control servo which moves the rod to keep the percentage on the current decade of the multi-range linear channel within 2% of the percent demand set by the operator. The servo stops the motor from moving if either the motor-up or motor-down lights are engaged, and will not move it again unless the rod is moved manually to clear the limit switch.

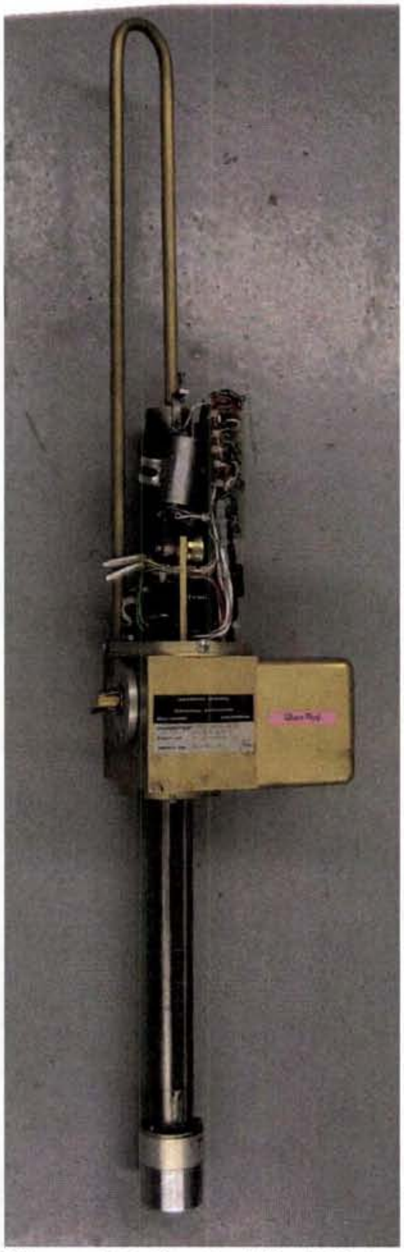

Figure 7.3 Control Rod Drive

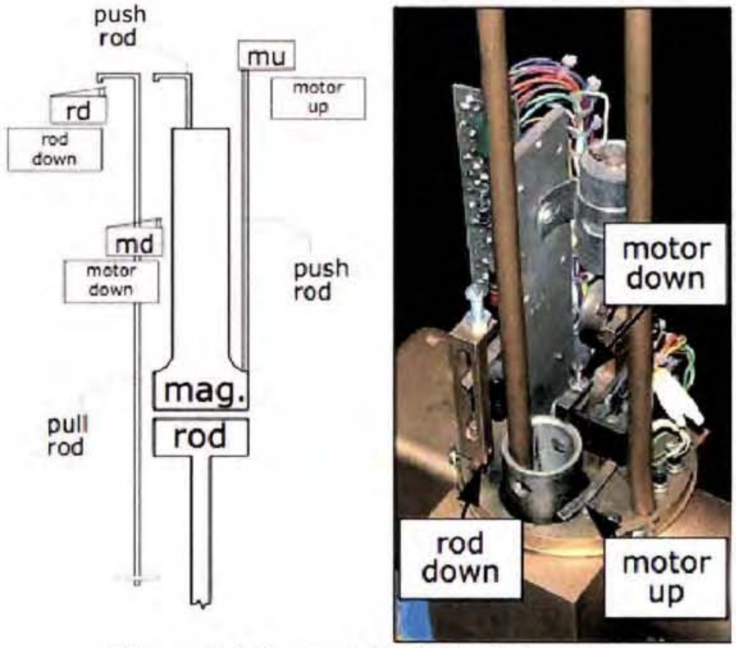

Figure 7.4 Control Rod Limit Switches

Two interlocks are built into the control system of the reactor to prevent improper operation. These interlocks are hard-wired into the control rod drive circuitry. They are stated below:

1. No control rod withdrawal is possible unless a neutron-induced signal is present on an instrumentation channel. This interlock prevents the possibility of raising the rods with no neutrons in the core, which could cause an uncontrolled power increase when neutrons are introduced to the core; and

2. Simultaneous manual withdrawal of two or more control rods is not possible. This interlock prevents violation of the maximum reactivity insertion rate of the reactor.

7.4 Reactor Protection System

The reactor protection system will initiate a reactor scram if any of several measured parameters are outside their safety system settings. The reactor scram effectively shuts down the reactor by de-energizing the rod drive electromagnets, causing the control rods to drop into the reactor core by gravity. The reactor operator may manually scram the reactor by means of a scram bar on the console. The scrams required for operation of the reactor are a high power scram on the percent power channel, a high power scram on the linear channel, and a manual scram. All of these scrams are tested daily before operation.

7.5 Engineered Safety Features Actuation Systems

There are no engineered safety features actuation systems. Control rod insertion is provided by gravity and core cooling is provided by natural convection in water or air. Therefore, Engineered Safety Features systems are not required in this design.

7.6 Control Console and Display Instruments

Data from the linear channel is displayed on a Sorrento NMP-1000, data from the logarithmic channel on a NLW-1000, and the percent power channels on a NP-1000. The information is also displayed on a Honeywell Multitrend Analyzer.

Control rod indication is displayed on three labeled displays mounted in the console. Position is displayed as 0 to 100% of withdrawal with 0.1% resolution.

When a reactor scram occurs, the corresponding annunciator lights up. The annunciators only reset when the console key is moved to the reset position.

7.7 Radiation Monitoring Systems

A radiation area monitor (RAM) is mounted in the reactor room and is easily visible through the windows by the operator. It has a local visible and audible alarm that can be seen and heard in the control room. The RAM is an energy-compensated Geiger-Mueller.

A continuous air monitor (CAM) is mounted in the reactor bay and samples the air for radioactive particulates. Air from the reactor bay is passed through a paper particulate filter in close proximity to a detector. The readout from the unit is mounted within reach of the operator at the console. The CAM has a 30-minute average (slow) alarm, a 1-minute average (fast) alarm, and a beta net count rate alarm. Similar units sample air from the exhaust stack, through which all air from the facility passes, and may be used as a backup if the CAM fails.

Chapter 8

Electrical Power Systems

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Chapter 8

Electrical Power Systems

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8 ELECTRICAL POWER SYSTEMS

Primary electrical power is provided through the Reed College power system which is supplied through the municipal grid by commercial generators. Main power lines traverse underground tunnels, thus inhibiting tampering. Loss of electrical power automatically places the reactor in a subcritical, secured configuration. Loss of electrical power will de-energize the control rod drives, causing the rods to fall by gravity into the core, and therefore does not represent a potential hazard to the reactor. Since the core is cooled by natural convection, no emergency power is required for reactor cooling systems.

8.1 Normal Electrical Power Systems

The design basis for the normal electrical power systems is to provide sufficient current for normal operations. The reactor has no exclusive electrical supply and distribution, but derives from the building transformers. Supplied power is standard 60 Hz AC, 240 V, three-phase current. A schematic representation of the electrical power system is provided in Figure 8.1. Loads are distributed between two panels, which can be shut off directly in the mechanical equipment room or remotely by key in the entrance hallway. The breakers are divided into vital and non-vital loads. Shutting off the vital load circuit breaker also cuts power to the non-vital load breaker. Although they are labeled "vital" and "non-vital", no electrical systems are required for reactor safety or are taken credit for in this SAR.

8.2 Emergency Electrical Power Systems

The Reed Research Reactor does not require electrical power to maintain a safe, shutdown condition, because the control rods are designed to rest in a fully-shutdown position when electrical power is cut from the electromagnets. Therefore, there is no emergency electrical power system at the facility. The building is equipped with emergency exit lighting as required by Portland fire codes. There are uninterruptible power supplies for most of the instrumentation, but none are required for reactor safety or are taken credit for in this SAR.

Psychology Building Room 135

Main Panel
Breaker 2MB1 Nuclear Reactor Panel

Mechanical Room

Panel B (Non-Vital)			
Cooling Tower Pump	1	2	Cooling Tower Sump Pump
	3	4	Heat Trace
	5	6	Chem Pump, Control Panel
Cooling Tower	7	8	Cooling Tower Damper
	9	10	Lights – Reactor Room North Wall
	11	12	Lights – Reactor Room North of Tank
Cooling Tower Heat	13	14	Outlets – N/S sides of Tank, Pool Lights
	15	16	Outlets – E/W sides of Tank, RAM
	17	18	Outlets – Reactor Room Hall, Loft, Mech Room
Primary Pump	19	20	Lights – Reactor Room South
	21	22	Lights – Reactor Room Center/East
	23	24	Reactor Panel B Shunt Trip
Spare	25	26	Rabbit Blower
Spare	27	28	Rabbit Solenoid Valves
Spare	29		
Spare	31	32	Lights – Reactor Room Center/West
		34	Spare

Panel A (Vital)			
1	Spare	Spare	2
3	Spare	Spare	4
5	Spare	Spare	6
7	Spare	Spare	8
9	Lights - Outside, Outlets - control, reactor, & mech room, Pool level alarm	APM / GSM,utlets - control & reactor rooms, & loft	10
11	Lights - mech room & control room, Fire panel, Evacuation alarm	Spare	12
13	Lights - Hall & loft	Spare	14
15	Heating, Ventilation Isolation, Outlet - control room over console	Spare	16
17	Spare	Console	18
19	Reactor Panel A Shunt Trip	Spare	20
21	Spare	Reactor Room Exhaust Fan	22
25	Spare	Reactor Room Supply Fan	24

Figure 8.1 Electrical Panels

Chapter 9

Auxiliary Systems

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Chapter 9

Auxiliary Systems

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9 AUXILIARY SYSTEMS

The systems covered in this chapter are not directly required for reactor operation, but are used in support of the reactor for normal and emergency operations.

9.1 Heating, Ventilation, and Air Conditioning Systems

Heating of the reactor bay and control room is provided by steam from the Reed College Physical Plant. The thermostat is automatically controlled at the Physical Plant. The reactor building does not have an air conditioning system.

The reactor bay was specifically designed for handling radioactive materials. A ventilation system moves air through the reactor room, the control room, and the mechanical equipment room. In normal operation (Figure 9.1), a fan draws air from the loft over the facility entry hallway and moves it into the reactor room through two vents for a total airflow of approximately 1,330 cubic feet per minute (630 L/s). This air is drawn from the reactor room by an exhaust fan, and either recirculates or goes up the exhaust stack, which by technical specifications releases at least 12 feet (3.7 m) above ground level to allow for decay of radiological emissions. If the system is switched over to isolation mode (Figure 9.2), the input fan shuts down and the exhaust fan draws reactor room air through a HEPA filter at an airflow of approximately 150 cfm (70 L/s). Routine maintenance and service of these systems is the responsibility of the Reed College Facilities Services office.

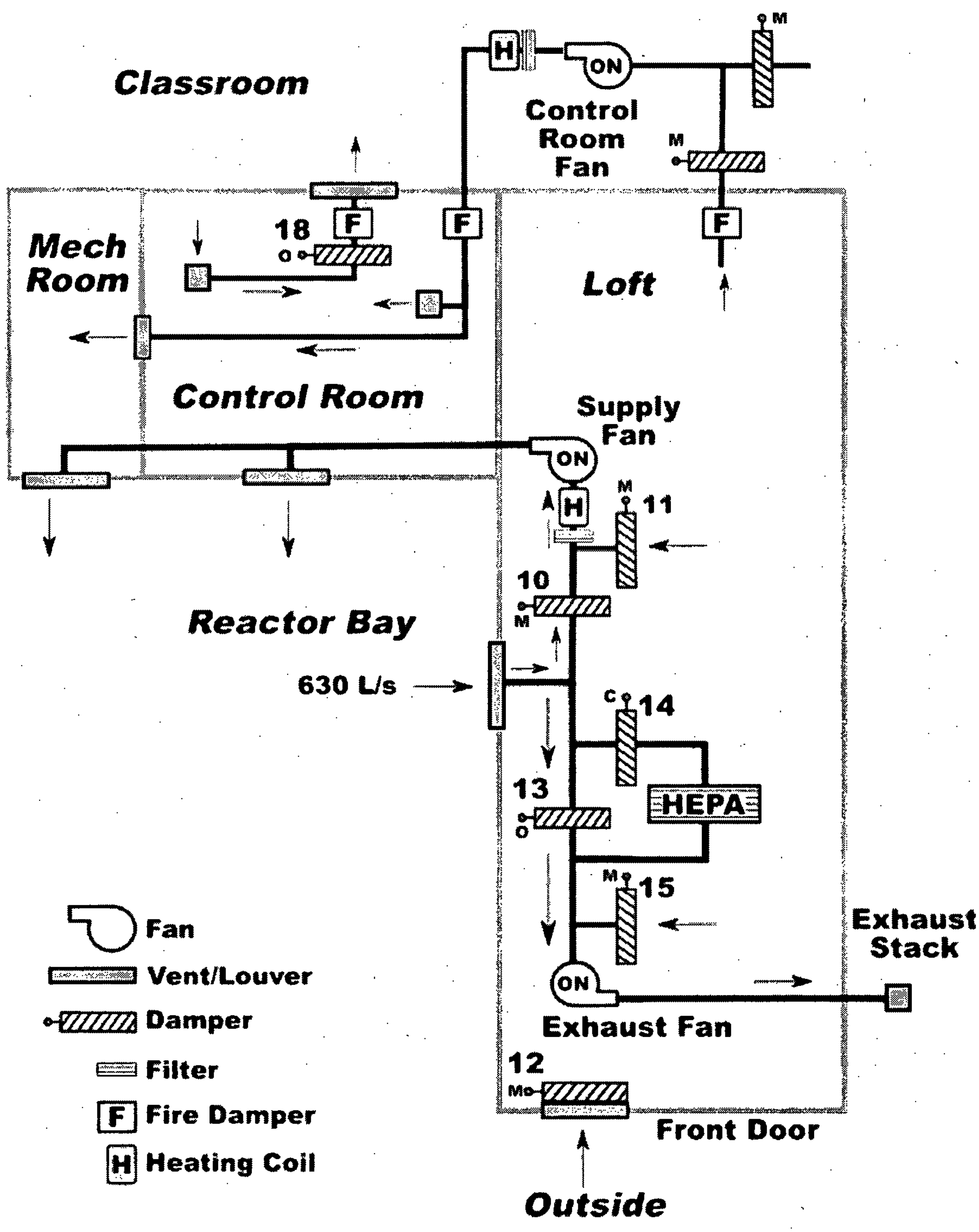

Figure 9.1 Ventilation System in Normal Operation

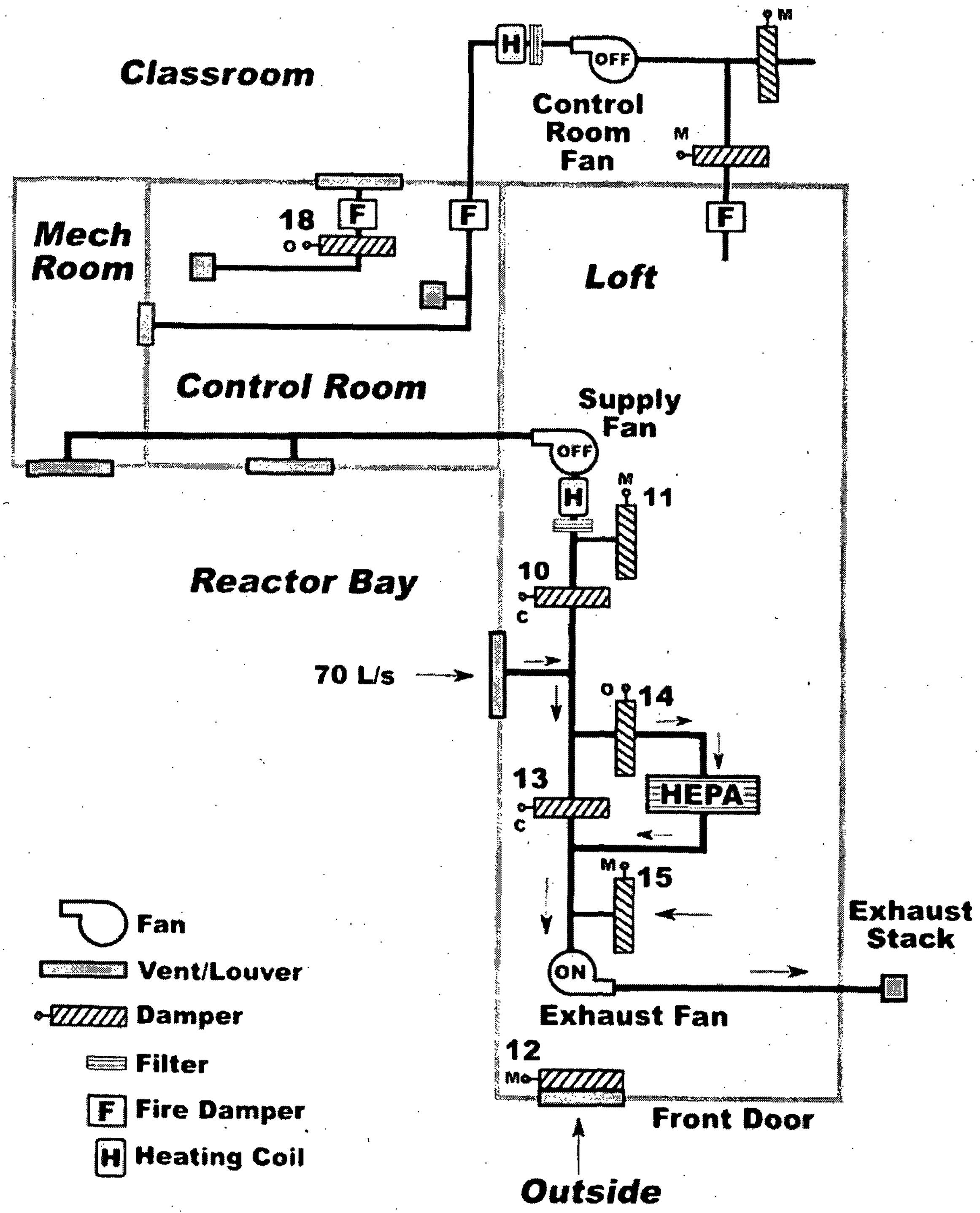

Figure 9.2 Ventilation System in Isolation

9.2 Handling and Storage of Reactor Fuel

[redacted] The racks are fabricated of aluminum and allow only for single row spacing of up to ten elements. Spacing in the rack is sufficiently far apart to prevent accidental criticalities.

[redacted] A licensed operator must be at the controls of the reactor while fuel movement is underway.

For fuel inspection, there is a periscope that can be mounted in the reactor tank for visual inspection of the elements. It is a rigid pole with a mounting point for a small video camera and an attached guide for the element.

Fuel replacement may be accomplished by moving spent or lightly burned fuel rods into a shielded container under water using the fuel-handling tool. A suitable container can be lowered into the pool by use of the 4-ton (3.6 metric ton) crane.

9.3 Fire Protection Systems and Programs

Fire protection systems are maintained and serviced by services contracted by Reed College Community Safety which maintain fire protection services for the entire campus. The building fire alarm system is part of a campus-wide network. [redacted] The reactor has three pull-stations, one on the north wall of the reactor bay, one in the exit hall, and one in the adjacent radiochemistry laboratory. There is a smoke detector in the control room, two in the reactor room, and one in the mechanical equipment room.

There are four fire extinguishers readily available to reactor personnel. Halon extinguishers are located in the control room and on the south wall of the reactor room for electrical fires and ABC dry chemical fire extinguishers are located on the north wall of the classroom area and by the emergency exit stairwell. Reed College Environmental Health & Safety contracts with an off-campus agency for fire extinguisher testing and maintenance on a regular basis.

9.4 Communication Systems

Telephones at the facility share a common line, and are serviced by the Reed College switchboard. Phones are located in the Director's office, the control room, and the reactor room; additional lines can be connected elsewhere in the facility as needed. Connection to the campus Ethernet for computers in the facility is maintained by Reed College Computer User Services. In addition, there is a public address system allowing communication from the control room into the reactor bay, and an intercom system.

9.5 Possession and Use of Byproduct, Source, and Special Nuclear Material

Reportable quantities of radioactive materials are possessed under the College's State Radioactive Materials license and the Reactor License. The reactor fuel is the property of the Department of Energy. Several radioactive sources are owned by Reed College. Radioactive materials, including special nuclear material (SNM), are inspected for contamination and inventoried on a semiannual basis. Several areas are designated for storage of these materials.

Byproduct material produced in the reactor for research purposes is transferred to the College's state license and recorded on the irradiation documentation. The state license is maintained by the Reed College Office of Environmental Health and Safety (EHS), and administered by the Radioactive Materials (RAM) Committee. Only individuals listed under the license are permitted to receive materials. Normally, a member of the reactor staff is also approved by the RAM Committee to receive byproduct and special nuclear material under the state license. Possession limits are set by the state, and the RAM Committee determines use limits. Transfers off-campus to other licensees must first go through EHS. The facility has several sources for research and instrumentation calibration purposes that are possessed under this license. Low-level wastes generated under the State of Oregon license are disposed of under the state license. Disposal of low-level wastes generated under the reactor license is coordinated with EHS. Short-lived isotopes (half-life less than 90 days) are decayed in storage. Longer-lived isotopes are disposed of at US Ecology's facility in Richland, Washington.

SNM inventory is reported to the Nuclear Assurance Corporation under Reporting Identification Symbol (RIS) ZSW. The reactor fuel comprises the bulk of SNM at the facility. This fuel is owned by the Department of Energy and possessed under the Reactor Facility license R-112. The possession limit for 250 kW is set by the license R-112 at 2,500 g uranium-235 in enrichments less than 20%. The license also allows possession of a [redacted]4-Ci americium-beryllium neutron source, and no more than 11 g of uranium-235 in the fission chambers.

[redacted]

9.6 Cover Gas Control in Closed Primary Coolant Systems

The Reed reactor has an open primary coolant system and hence has no cover gas control. Nitrogen-16 is controlled as described in Chapters 5 and 11 by forcing convection cooling flow from the reactor core into a helical pattern (to enhance time delays for more decay).

9.7 Other Auxiliary Systems

9.7.1 Reactor Bay Crane

A manual chain-fall crane in the reactor bay is used to manipulate loads of up to 4 tons (3.6 metric tons). Its use is administratively controlled.

9.7.2 Associated Laboratories

In the rear of the reactor building is the RRR radiochemistry laboratory, featuring sample preparation facilities and several fume hoods for wet chemistry work. In an adjacent room is the gamma spectroscopy laboratory, which features several high-purity germanium detectors and associated electronics for neutron activation analysis. There is also a dedicated scintillation detector for counting health physics wipes.

Chapter 10

Experimental Facilities

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Chapter 10

Experimental Facilities

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10 EXPERIMENTAL FACILITIES AND UTILIZATION

10.1 Summary Description

The Reed Research Reactor (RRR) provides educational and training services to support the scientific curriculum at Reed College and the education of the community about nuclear science and radiology. The laboratory science programs at Reed College are among the top five in the nation, and the RRR plays a part in allowing in-depth training for all students in the field of nuclear physics and engineering. The main experimental technique utilized at the RRR is neutron activation analysis.

Sectional views of the reactor are shown in Chapter 1, The Facility. Principal experimental features of the RRR facility include:

- Central thimble
- Rotary specimen rack
- Pneumatic transfer system
- Single-element replacement
- Gamma irradiation facility

New experiments are reviewed and approved by the RRC prior to operations. The Reactor Director or Supervisor may schedule for performance an approved experiment or an experiment of any type previously reviewed by the committee.

10.2 Experimental Facilities

The RRR is a flexible, multi-use facility with irradiation facilities inside the core boundary, in the reflector, outside the reflector, and outside the biological shielding. One of the in-core facilities is a pneumatic sample delivery system capable of providing samples directly to the neutron activation analysis laboratory.

10.2.1 Central Thimble

The reactor is equipped with a central thimble for access to the point of maximum flux in the core. A removable screen at the top end of the thimble allows gas relief and prevents objects from falling through the reactor tank covers.

The central thimble is an aluminum tube that fits through the center holes of the top and bottom grid plates terminating with a plug at a point approximately 7.5 inches (19 cm) below the lower grid plate. The tube is anodized to retard corrosion and wear. Although the shield water may be removed to allow extraction of a vertical thermal-neutron and gamma-ray beam, four 0.25 inch (0.64 cm) holes are located in the tube at the top of the core to prevent expulsion of water from the section of the tube within the reactor core. Dimensions of the tube are 1.5 inch OD (3.81 cm), with 0.083 inch (0.21 cm) wall thickness and an inside diameter of 1.33 inches (3.38 cm). The thimble is approximately 24.5 feet (7.5 m) in length, made in three sections joined with watertight tube fittings.

10.2.2 Rotary Specimen Rack

A forty-position rotary specimen rack (Lazy Susan, or LS) is located in a well in the top of the graphite radial reflector. The LS allows large-scale production of radioisotopes and activation and irradiation of multiple material samples with neutron and gamma ray flux densities of comparable intensity. Specimen positions are 1.25 inches (3.18 cm) diameter by 10.8 inches (27.41 cm) depth. Samples are manually loaded from the top of the reactor through a water-tight tube into the LS. The rack may be rotated (repositioned) manually from the top of the reactor, and a motor allows continuous rotation at about 1.17 rpm during irradiation. Figure 10.1 is an image of the LS during construction, and Figure 10.2 is an image of the rotation control mechanism and motor housing.

The rotary specimen rack, which surrounds the core, consists of an aluminum rack for holding specimens during irradiation. This rack is located inside a ring-shaped, seal-welded aluminum housing. The rack is rotated on a stainless steel ball-bearing assembly consisting of Stellite balls, Type 304 stainless steel races, and spring-type spacers of Type 302 stainless steel. It supports 40 evenly spaced, tubular aluminum containers, open at the top and closed at the bottom, which serve as receptacles for specimen containers. The maximum internal space in each of the 40 tubes is 1.25 inches (3.18 cm) in diameter by 10.8 inches (27.41 cm) in length. Each location can hold one TRIGA irradiation tube, or two tubes if they are properly screwed together as shown in Figure 10.3.

Four of the tubes, spaced 90° apart, have perforations in their walls. One of these four perforated tubes has a 0.625 inch (1.6 cm) diameter hole in the bottom. This hole permits periodic testing of the bottom of the rotary specimen-rack housing to determine the extent of any accumulation of condensation or leaking water. Each of the four perforated tubes can be loaded with a suitable porous container filled with a water-absorbing agent to dry any condensation that may occur as a result of high humidity in the reactor area and low operating temperature.

Each tube on the rack is oriented with respect to the specimen removal tube by a single locking rod. The ring is rotated from a drive at the top of the reactor, rotation being transmitted through a drive shaft inside a tubular housing to a sprocket-and-chain drive in the rotary specimen rack housing.

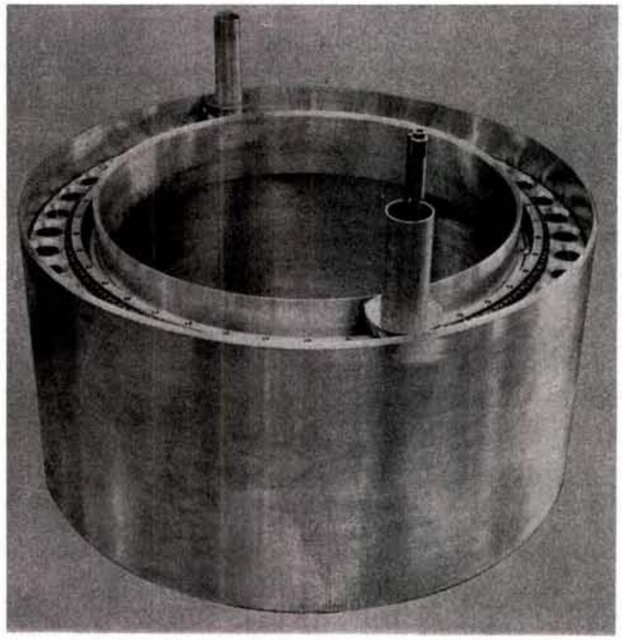

Figure 10.1 Rotary Specimen Rack (Lazy Susan)

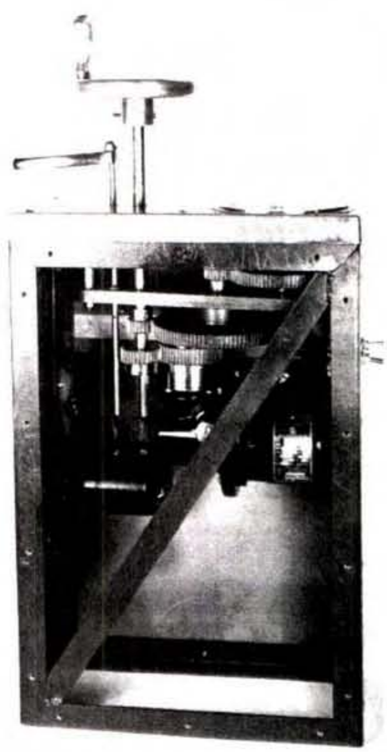

Figure 10.2 Rotary Specimen Rack Motor

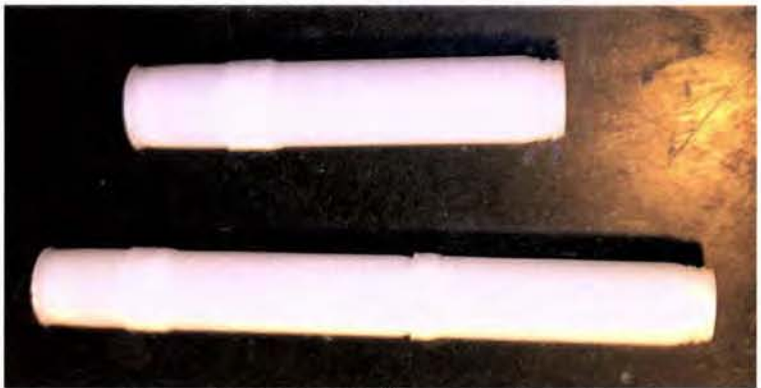
Figure 10.3 TRIGA® Irradiation Tubes

10.2.3 Pneumatic Transfer System

A pneumatic transfer system, permitting experiments involving short-lived radioisotopes, rapidly conveys a specimen from the reactor core to a remote receiver. The in-core terminus is normally located in the outer ring of fuel-element positions. The sample capsule (rabbit) is made of polyethylene, with an internal diameter of approximately 0.7 inches (1.8 cm) and a length of 4.5 inches (11.4 cm). It is conveyed to a receiver/sender station via aluminum tubing nominally 1.25 inch OD (3.18 cm) and at least 1.08 inches (2.74 cm) ID, with radii of curvature no less than 2 feet (61 cm). The in-tank and in-core portion of the pneumatic transfer system is illustrated in Figure 10.5. This system, shown schematically in Figure 10.4, consists of the following major components:

- A specimen capsule ("rabbit")
- A blower-and-filter assembly
- A valve assembly
- A terminus assembly
- A receiver assembly
- A control assembly
- Tubing fittings

The system is controlled from the receiving area and may be operated either manually or automatically (i.e., with an electric timing device incorporated into the system so that the specimen capsule is ejected automatically from the core after a predetermined length of time. Four solenoid-operated valves control the air flow. The system operates on a pressure differential, drawing the specimen capsule into and out of the core by vacuum. Thus, the system is always under a negative pressure so that any leakage is always into the tubing system. All the air from the pneumatic system is passed through a HEPA filter before it is discharged to the exhaust stack.

When the pneumatic transfer system is in the manual mode, a relay in the relay box located near the blower-and-valve assembly is energized, applying power to the blower motor. With the IN/OUT switch positioned at IN, the valves are energized, which sets the air flow pattern through the specimen tube into the core. The specimen capsule should then be inserted in the receiver assembly and the receiver door latched. The air flow will transfer the capsule into the terminus assembly in the core. For removal of the capsule from the core, the IN/OUT switch must be positioned on OUT. This reverses the air flow by deenergizing all four valves, with the blower still energized, so that the capsule is driven out of the core into the receiver assembly.

To minimize the possibility of inadvertently leaving a specimen capsule in the core, the system is designed so as to keep the blower energized during irradiation; this serves as an audible reminder that the system is in use. If the blower is turned off, the valves will change to positions that will cause the capsule to be removed from the core, thus preventing excessive irradiation and consequent embrittlement of the capsule.

In the automatic mode, as in the manual, the relay is energized, applying power to the blower motor, while the valves remain deenergized. Timing is controlled by a clock, with intervals up to 5 minutes. For a specimen capsule to be inserted into the core, the capsule must be placed in the receiver assembly, the receiver door latched, the clock set at a predetermined time interval, and the red timer button pressed. This operation automatically energizes all valves, reverses the air flow, and drives the capsule into the core. After the preset time interval has elapsed, the clock automatically deenergizes all valves, returning the air flow to the OUT condition and driving the capsule out of the core.

10.2.3.1 Valve Assembly

Adjacent to the blower assembly, four solenoid-operated valves (see Figure 10.4) for 2.25 inch (5.7 cm) tubing are mounted on a common bracket. In the deenergized condition, valves 2 and 4 are open and valves 1 and 3 are closed. Valves 2 and 3 open to the Mechanical Room, and valves 1 and 4 are connected by flexible hoses through the plenum chambers and filter to the blower suction. No special maintenance of the valves is required except for periodic inspection of the electrical equipment and oiling of the moving parts.

10.2.3.2 Terminus Assembly

The terminus assembly (Figure 10.5) is located in the reactor tank. The bottom part, a double tube, extends into the reactor core. The terminus support is shaped like the tip of a fuel moderator element and can therefore fit into any fuel location in the core lattice. The prescribed location for the terminus assembly is in the outer ring of the lattice. Approximately 6 inches (15 cm) above the top grid plate, the double tube branches into two separate tubes, both of which extend to the top of the reactor. The tubes are made of anodized aluminum and have an outside diameter of 1.25 inches (3.2 cm). The distance between the center lines of the two tubes is 4.5 inches (11.4 cm). The overall length of the terminus assembly is approximately 12.8 feet (3.9 m). Two 90° bends, 1.25 inch (3.18 cm) diameter tubes connect the assembly with the tubing at the reactor; one tube connects the terminus assembly to the receiver, and the other connects it to the blower assembly.

To counteract its buoyancy and keep it firmly in place in the core, the terminus assembly is weighted. The bottom of the internal tube is equipped with an aluminum spring shock absorber to absorb the impact of the specimen container when it is driven into position.

10.2.3.3 Receiver-Sender Assembly

The specimen capsule is inserted in and removed from the pneumatic system through an aluminum door in the receiver-sender assembly (Figure 10.6). This door is hinged on the upper side and has a latch on the lower side. When latched, the door will stop the ejected capsule in the receiver assembly.

10.2.3.4 Blower-and-Filter Assembly

The blower-and-filter assembly (Figure 10.8) is installed on a wall-mounted steel angle support in the Mechanical Room. The assembly consists of a blower, a manifold, plenum chambers, and a filter. The blower exhausts the system air into a vent pipe that discharges outside the building.

The blower is driven by a 1.5 hp motor and is equipped with sealed ball bearings that have permanent lubrication and thus require no regular maintenance. Brushes on the motor are inspected for wear semiannually. The brushes are replaced when they are worn to 0.25 inch (0.635 cm) in length. Access to the brushes is obtained by removing the two Bakelite cap screws located directly opposite each other in the motor housing. The blower is fastened to the frame by a felt-lined steel clamp around the motor housing. The blower is connected to the plenum chambers by a tube. The connections at both ends of the tube consist of flexible hose, 2.25 inches (5.7 cm) in inside diameter, and hose clamps.

The filter, which is sandwiched between the plenum chambers, has a 12 inch (30 cm) by 12 inch (30 cm) face area and is 5.875 inches (15 cm) deep. The minimum filter efficiency and maximum pressure drop at 135 ft^3/min (3820 lpm) (for 0.3-diameter smoke particles) are 99.97% and 0.9 inches (2.3 cm) of water, respectively. This filter is replaced when periodic visual inspection indicates a reduction in efficiency due to the buildup of impurities on the filter. The two plenum chambers are made of steel. Two rods with wing nuts hold the filter sandwiched between the chambers.

The rabbit travels only through 1.25 inch (3.18 cm) tubing in passing to and from the reactor core. However, 2.25 inch (5.72 cm) tubing is utilized in some of the return air lines of long rabbit runs as a method of cutting down the total system pressure drop. Each tubing joint in the 2.25 inch (5.72 cm) tubing and most of those in the 1.25 inch (3.18 cm) tubing are formed by butting the ends of the tubes together and attaching a slightly larger-diameter sleeve over the top of the joint with epoxy.

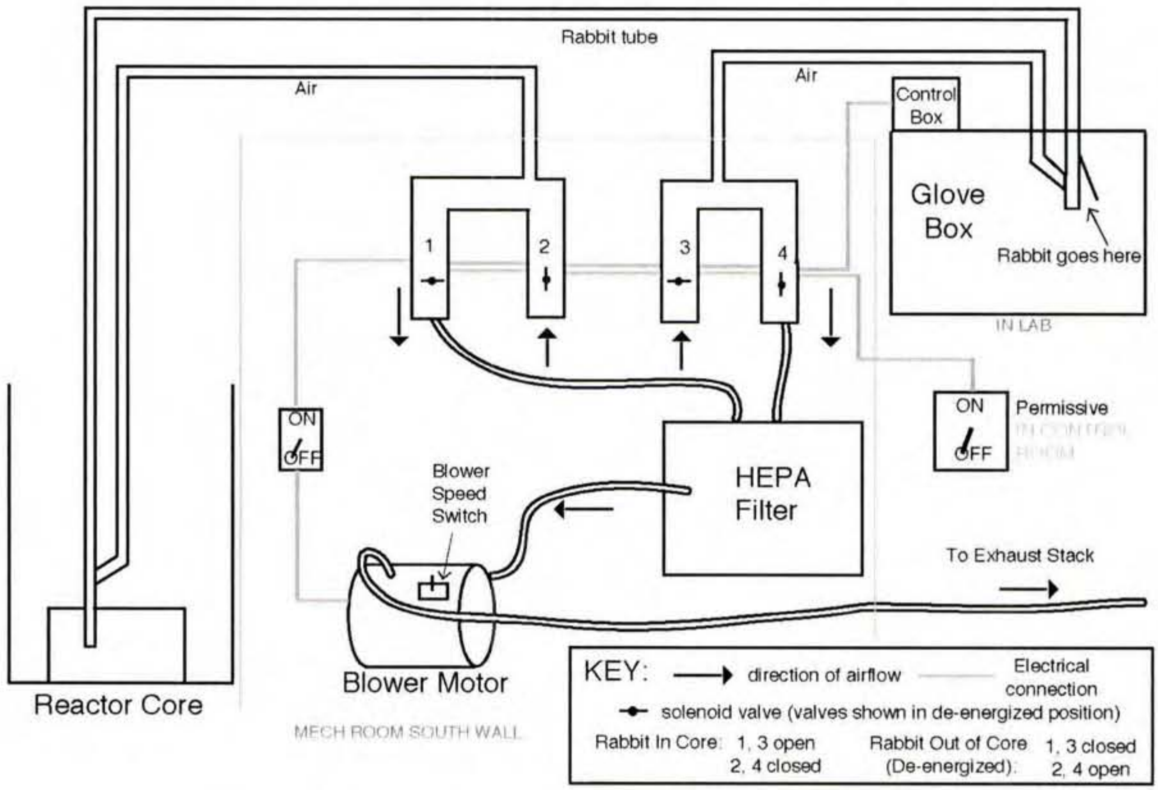

Figure 10.4 Diagram of the pneumatic transfer system

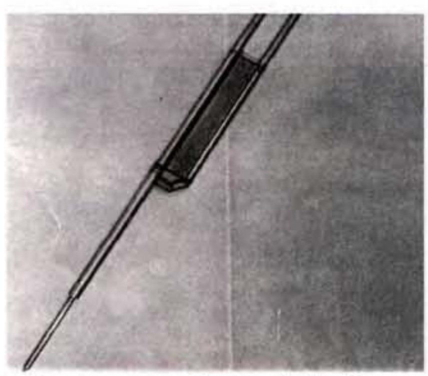

Figure 10.5 In-tank and in-core portions of the Pneumatic Transfer System

Figure 10.6 Pneumatic Transfer System Terminus

Figure 10.7 Pneumatic Transfer System Controller

Figure 10.8 Pneumatic Transfer System Blower-and-Filter Assembly

10.2.4 Single-Element Replacement Facilities

Experiments may be inserted in spaces designed for fuel elements using a special "dummy element" consisting of two threaded aluminum sections (Figure 10.9). When assembled, the dummy element has external dimensions matching a fuel element and an inner cavity [redacted] diameter and [redacted] long at the vertical center. This dummy element may be inserted in any position in the core with the standard fuel-handling tool, or by means of an attached ring.

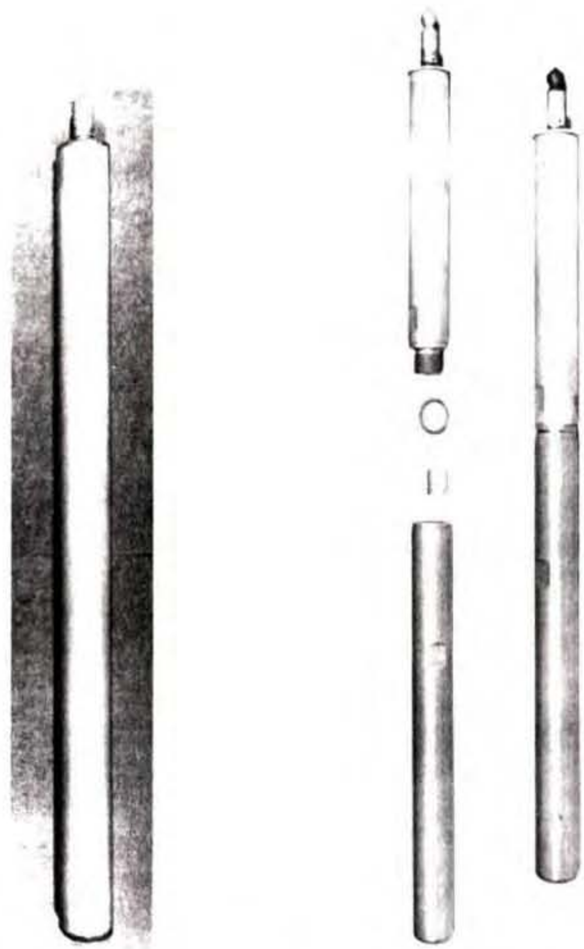

Figure 10.9 Source Holder

10.2.5 Gamma, In Tank, and Ex Core Facilities

The gamma irradiation facility is a standing fuel rack located approximately five feet from the reactor on the bottom of the pool. The gamma source is an iridium source activated by the reactor as a single-element replacement experiment. The geometry of the facility allows varying intensities of gamma irradiation without any substantial neutron dose from the reactor.

Experimental procedures also authorize irradiations adjacent to the radial reflector.

10.3 Experiment Review

A wide array of experiments has been documented and approved for execution in the operational history of the facility. The experiment review and approval process is the proposed experiment shall include (as a minimum, not limited to) that the likelihood of occurrences listed below are minimal or acceptable in both normal and failure modes:

- Breach of fission product barriers (which could occur through reactivity effects, thermal effects, mechanical forces, and/or chemical attack)
- Interference with reactor control system functions (which could occur through local flux perturbations or mechanical forces that can affect shielding or confinement)
- Introduction or exacerbation of radiological hazards (which could occur through irradiation of dispersible material, mechanical instability, inadequate shielding and/or inadequate controls for safe handling)
- Interferences with other experiments or operations activities (which could occur through reactivity effects from more than one source, degradation of performance of shared systems, e.g., electrical, potable water, etc., physical interruption of operational activities, or egress of toxic or noxious industrial hazards. Note this evaluation should also consider potential for fire or personnel exposure to toxic/noxious material)
- Determination that the proposed activity is in compliance with Technical Specifications

If an event or new information challenges the original evaluation, the RRC shall review the experiment approval and determine if the original approval is still valid prior to a continuation of the experimental program. When container failure is discovered that has released material with potential to damage the reactor fuel or structure (by corrosion or other means), physical inspection shall be performed to determine the consequences and need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by the RRC and determined to be satisfactory before operation of the reactor is resumed.

conducted in accordance with approved facility administrative procedures. If an experiment falls within the scope of a previously approved experiment, a request for operation is submitted to the Director. The Director verifies that operation is within the scope of previous experiments, and approves the request by signature so that the experiment may be scheduled. If it is determined that the proposed experiment does not fall within the scope of previously approved experiments or if the experiment involves an unreviewed safety question, the experiment is considered a new experiment.

10.3.1 Planning and Scheduling of New Experiments

New experiments require approval of the Reactor Review Committee (RRC) prior to implementation. To support RRC review, a written description of each proposed new experiment must be prepared, with sufficient detail to enable evaluation of experiment safety. The RRC shall evaluate whether new contemplated experiments, procedures, facility modifications (and/or changes thereto) meet review criteria, and either approve experimental operations (with or without changes or additional constraints) or prohibit the experiment from being performed. The following information is the minimum for a proposed experiment:

- Purpose of the experiment
- Background (if appropriate)
- Procedure - to include a description of the experimental methods to be used and a description of the equipment to be used. A sketch of the physical layout and a tabular list of equipment necessary for the experiment are recommended if appropriate
- A summary of various effects that the experiment could cause, or that could interact with the experiment, or including:
  - Reactivity Effects
  - Thermal-Hydraulic Effects
  - Mechanical Stress Effects
- References

The RRC may require additional information to determine that an experiment is acceptable; the experiment shall not be scheduled until the RRC has reviewed the proposed experiment, including any supplemental information requested by the RRC.

10.3.2 Review Criteria

The RRC shall consider new experiments in terms of effect on reactor operation and the possibility and consequences of failure, including, where significant, consideration of chemical reactions, physical integrity, design life, proper cooling, interaction with core components, and reactivity effects. Before approval, the RRC shall conclude that in their judgment the experiment, by virtue of its nature and/or design, will not constitute a significant hazard to the integrity of the core or to the safety of personnel. Evaluation of

Chapter 11

Radiation Protection and Waste Management

Reed Research Reactor Safety Analysis Report

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11 RADIATION PROTECTION AND WASTE MANAGEMENT

This Chapter deals with the overall radiation protection program for the Reed Research Reactor (RRR) and the associated practices for management of radioactive wastes. The chapter identifies radiation sources that may be present during normal operation of the reactor and the various procedures followed to monitor and control these sources. The chapter also identifies expected personnel radiation exposures due to normal operations.

11.1 Radiation Protection

The Radiation Protection Program for the RRR was prepared to meet the requirements of Title 10, Part 20.1101, Code of Federal Regulations and the requirements of the State of Oregon. The Program seeks to control radiation exposures and radioactivity releases to a level that is As Low As Reasonably Achievable (ALARA) without unnecessarily restricting operation of the reactor for purposes of education and research. The Program is executed in coordination with the Environmental Health and Safety office of Reed College. The Program is reviewed and approved by the Reactor Review Committee (RRC).

Certain aspects of the Program deal with radioactive materials regulated by the State of Oregon (an Agreement state) under license ORE-90010. Therefore, the Reed College Radioactive Materials Committee (responsible for administration of the State license) reviewed the Program. The Radiation Protection Program was developed following the guidance of the American National Standard *Radiation Protection at Research Reactor Facilities* [1] and Regulatory Guides issued by the NRC [2-7].

11.1.1 Radiation Sources

Radiation sources present in the reactor facility may be in gaseous (airborne), liquid, or solid form. These forms are treated individually in successive subsections.

11.1.1.1 Airborne Radiation Sources

Normal operation of the Reed Research Reactor (RRR) results in two potential source terms for radioactive gaseous effluent at significant levels, argon-41 and nitrogen-16. There are variations in experimental configuration and possible scenarios where the production of argon-41 may be different than the routine operations; these scenarios do not produce long term, routine radioactive effluent, but are assessed to determine if the amount of radioactive effluent is so high as to impact the annual exposure that might result from routine operations.

The nuclide argon-41 is produced by thermal neutron absorption by natural argon-40 in the atmosphere and in air dissolved in the reactor cooling water. The activation product appears in the reactor bay and is subsequently released to the atmosphere through the ventilation system.

The nuclide nitrogen-16 is produced by fast neutron interactions with oxygen. The only source of nitrogen-16 in the reactor that requires consideration results from interactions of fast neutrons with oxygen in the cooling water as it passes through the reactor core. Any interaction with oxygen in the atmosphere is relatively insignificant and is neglected in this analysis.

A portion of the nitrogen-16 produced in the core is eventually released from the top of the reactor tank into the reactor bay. The half-life of nitrogen-16 is only 7.13 seconds, so its radiological consequences outside the reactor bay are insignificant.

The cladding of a fuel element could fail during normal operations as a result of corrosion or manufacturing defect. If a failure occurs, a fraction of the fission products, essentially the noble gases and halogens, would be released to the reactor tank and, in part, ultimately become airborne and released to the atmosphere via building ventilation. This operational occurrence, taking place in air, is addressed in Chapter 13, Accident Analysis as the maximum hypothetical accident (MHA) for the RRR.

Neutron interactions with structural and control materials, including cladding, as well as materials irradiated for experimental purposes, result in the formation of activation products. These products are in the nature of fixed sources and are mainly a source of occupational radiation exposure. Administrative controls preclude the significant formation of airborne activation products, other than the aforementioned argon-41.

11.1.1.1.1 Radiological Standards

Appendix B of 10 CFR Part 20 lists the allowable Derived Air Concentration (DAC) for argon-41 as 3E-6 $\mu Ci/cm^3$. For 2000 hours exposure this will produce the 50 mSv (5 rem) maximum permissible annual exposure. Appendix B of 10 CFR Part 20 lists the allowable Effluent Concentration (EC) for argon-41 as 1E-8 $\mu Ci/cm^3$. For 8760 hours exposure this will produce the 0.5 mSv (0.05 rem) annual exposure for a member of the public.

11.1.1.1.2 System Parameters

The calculations for argon-41 and nitrogen-16 releases during normal operations are based on the following system parameters.

Table 11.1 General System Parameters for Normal Operations at 500 kW

Parameter	Symbol	Value
Reactor steady power	P	500,000 W
Core coolant mass flow rate	w	0.150 kg/s
Core coolant density	ρ	1.0 g/cm^3
Core avg. thermal neutron flux at full power	ϕ_{th}	6.8E12 n/cm^{2}s
Core avg. fast neutron flux at full power	ϕ_f	1.00E13 n/cm^{2}s
Thermal neutron flux in RSR at full power	ϕ_{RSR}	3.00E12 n/cm^{2}s
Fuel element heated length	L	0.381 m
Flow cross sectional area per fuel element	A	6.2 cm^2
Mass flow rate per fuel element	$\dot{m}$	108 g/s
Reactor tank width		3 m (10 ft)
Reactor tank length		4.6 m (15 ft)
Reactor tank depth		7.6 m (25 ft)
Reactor tank water depth above core		4.88 m (16 ft)
Coolant volume in reactor tank	V_c	1.05E8 cm^3
Reactor bay width		8.38 m (27.5 ft)
Reactor bay length		10.36 m (34 ft)
Reactor bay height		4.11 m (13.5 ft)
Reactor bay air volume	V_{bay}	3.57E8 cm^3 (12,600 ft^3)
Air volume in rotary specimen rack	V_{RSR}	3.75E4 cm^3
Decay constant for argon-41	λ_γ	0.379 h^{-1} = 1.05E-4 s^{-1}
Thermal absorption cross section for argon-40	σ_{40}	0.66 barns
Reactor bay air changes per hour	λ_v	6.33 h^{-1} = 1.76E-3 s^{-1}

11.1.1.1.3 Reactor Core Parameters

Modeling of the reactor core for radiation transport calculations is based on the following approximations. For purposes of radiation shielding calculations the TRIGA® reactor core may be approximated as a right circular cylinder [redacted]) in diameter [redacted]. The fuel region is [redacted] On each end axially is a graphite zone 3.94 inches (10.0 cm) high and an aluminum grid plate 0.75 inches (1.91 cm) thick. In [redacted] core locations, there are [redacted] fuel elements, 3 control rods, 1 central thimble (void), 1 source (assume void), 1 in-core irradiation site (assume void), and 1 pneumatic transfer site (assume void). The fuel region may be treated as a homogeneous zone, as may be the axial graphite zones and the grid plates.

Fuel elements are [redacted] and [redacted], clad with type 304 stainless steel. Fuel density is 5996 kg/m^3. Fuel composition is 8.5% uranium, 91.4% $ZrH_{1.65}$. The uranium is 20% uranium-235 and 80% uranium-238. Steel density is 7900 kg/m^3. Control rods are 1.25 inches (3.175 cm) OD and clad with 30-mil thick aluminum (2700 kg/m^3 density). The control material may be approximated as pure graphite, with density 1700 kg/m^3.

In radiation transport calculations, the core is modeled conservatively as a central homogenous fuel zone (air density neglected) bounded on either end by a homogeneous axial reflector zone, and by a 0.75 inches (1.9 cm) thick aluminum grid plate, treated as a homogeneous solid. Composition of the three zones, by weight fraction, is given in Table 11.2. Densities of the homogenous zones are in Table 11.3.

Table 11.2 Compositions of Homogenized Core Zones

Element	Mass Fraction
Fuel Zone	
C	0.0617
Al	0.0010
H	0.0139
Zr	0.7841
Mn	0.0013
Cr	0.0117
Ni	0.0052
Fe	0.0469
U	0.0741
Axial Reflector Zone	
C	0.7920
Al	0.0033
Mn	0.0041
Cr	0.0368
Ni	0.0164
Fe	0.1474
Grid Plate	
Al	1.0000

Table 11.3 Density Homogenized Zones

Zone	Density (kg/m^3)
Core Area	3602
Reflector	1147
Grid Plate	2700

11.1.1.1.4 Reactor Bay Parameters

For purposes of radiation dose calculations within the reactor bay, the reactor bay is approximated as a rectangle 13.5 feet (4.11 m) high, 34 feet (10.36 m) long, and 27.5 feet (8.38 m) wide. The free volume is 12,600 ft^3 (357 m^3). The air exhaust rate is 1330 cfm. The site boundary is 250 feet (76 m) from the center of the reactor.

11.1.1.1.5 Radiological Assessment Radiological Assessment of Argon-41 in Rotary Specimen Rack

The air volume in the rotary specimen rack (RSR) does not freely exchange with the air in the reactor bay; there is no motive force for circulation and the rotary specimen rack opening is routinely covered during operation. If the rotary specimen rack were to flood, water would force the air volume in the RSR into the reactor bay. The air volume of the RSR can be approximated as a section of a cylindrical annulus, with 28-inches (71 cm) OD, 24-inches (61 cm) ID, and 14-inches (35.6 cm) height. The volume of the rotary specimen rack, V_{RSR}, is therefore 3.75E4 cm^3. The thermal neutron flux density in the RSR is ϕ_{RSR} = 3.0E12 n/cm^{2}s at 500 kW. The microscopic cross section for thermal neutron absorption in argon-40 is 0.66 barns. The macroscopic cross section, μ, for thermal neutron absorption in argon-40 in air (0.0129 weight fraction) is the product of the atomic density of argon-40 and the microscopic cross section and is equal to μ = 1.54E-7 cm^{-1}. After sustained operation at full power, the equilibrium argon-41 activity (Bq) in the RSR volume is given by

$$A_0 = \mu\phi_{RSR}V_{RSR} = 1.7\text{E}10\ \text{Bq}. \qquad (1)$$

This is 0.47 Ci in conventional units. If this activity were flushed into the reactor bay atmosphere as a result of a water leak into the RSR, the initial activity concentration would be

$$A_0/V_{Bay} = 1.3\text{E-}3\ \mu\text{Ci/cm}^3. \qquad (2)$$

This would instantaneously be well above the occupational DAC for argon-41. However, with radioactive decay and ventilation, the concentration would decline in time according to

$$A(t) = A_0 e^{-(\lambda_\gamma + \lambda_v)t}. \qquad (3)$$

If a worker were exposed to the full course of the decay, cumulative concentration (μCi-h/cm^3) in the reactor bay would be

$$\frac{1}{V_{bay}}\int_0^\infty A(t)dt = \frac{A_0}{V_{bay}(\lambda_\gamma + \lambda_v)} = 1.91\text{E-}4\mu\text{Ci-h/cm}^3. \qquad (4)$$

Assuming an occupational exposure of 2000 hours per year, the value of 1.91E-4 μCi-h/cm^3 above produces

$$1.91\text{E-}4\ \mu\text{Ci-h/cm}^3/2000\ \text{hours} = 9.55\text{E-}8\ \mu\text{Ci/cm}^3. \qquad (5)$$

This is well below the 2000 hour annual limit of 3E-6 μCi/cm^3 specified in Appendix B of 10 CFR Part 20.

11.1.1.1.6 Radiological Assessment of Argon-41 from Coolant Water

The reactor tank water surface is open to the reactor bay. Radioactive argon-41 is circulated in the pool by convection heating, and exchanges with the reactor bay atmosphere during normal operation. The argon-41 activity in the reactor tank water results from irradiation of the air dissolved in the water. The following calculations evaluate the rate at which argon-41 escapes from the water into the reactor bay. The following variables, plus those in Table 11.1, are used in the calculations of argon-41 concentrations in the core region, in the reactor tank outside the core, and in the reactor bay air.

V_{core} = volume of the active core
= $\pi(38.1\text{ cm})(45.7/2\text{ cm})^2$
= 6.25E4 cm^3

V_{fuel} = volume of the fuel elements
= $84\pi(38.1\text{ cm})(3.73/2\text{ cm})^2$
= 3.48E4 cm^3

V_{water} = volume of water in active region
= $V_{core} - V_{fuel}$
= 2.76E4 cm^3

C_{40} = argon-40 atomic density (cm^{-3}) in coolant

v = volumetric flow rate through core (cm^3/s)

τ = residence time for coolant in core at full power (s)

T = out-of-core cycle time for coolant

Since the volume of the active core region is 6.25E4 cm^3 and the volume of the fuel is 3.48E4 cm^3, the active region of the core contains 2.76E4 cm^3 of water.

The saturated concentration of argon in water at the coolant inlet temperature of 20°C and 1 atm is approximately 6.7E-5 g/cm^3 [17]. If it is assumed that air is saturated with water vapor above the water tank (17.5 mm Hg vapor pressure at 20°C) and that the mole fraction of argon in dry air is 0.0094, the partial pressure of argon in air above the tank is

$$0.0094(760 - 17.5) = 7.0\text{ mm Hg.} \quad (6)$$

By Henry's law, the concentration of argon in water at the inlet temperature is

$$6.7E\text{-}5(7.0/760) = 6.2E\text{-}7\ g/cm^3 \quad (7)$$

Using Avogadro's constant and the GAW of argon-40,

$$C_{40} = 9.3E15\text{ atoms/cm}^3. \quad (8)$$

The number of atoms per second of argon-41 produced in the core is

$$C_{40}V_{water}\sigma_{40}\phi_{th} = 1.13E9. \tag{9}$$

Activity is calculated as the product of the isotope concentration and the mean lifetime. If it were assumed that 100% of the atoms escape to the reactor bay free air volume, the steady-state activity concentrations in the reactor bay atmosphere would be:

$$\frac{Bq}{cm^3} = \frac{C_{40}V_{water}\sigma_{40}\phi_{th}\lambda_\gamma}{V_{bay}(\lambda_\gamma + \lambda_v)} \tag{10}$$

The equilibrium argon-41 concentration during full power steady state operation at 500 kW in the reactor bay would be 0.178 Bq/cm^3 (4.8E-6 μCi/cm^3) with ventilation, and 3.165 Bq/cm^3 (8.56E-5 μCi/cm^3) without ventilation.

Appendix B of 10 CFR Part 20 lists the DAC for argon-41 as 3E-6 μCi/cm^3. Therefore, under these extremely conservative calculations, equilibrium argon-41 concentration during full power steady state operation at 500 kW is slightly than DAC. Since personnel do not stay in the reactor bay for extended periods when at power, this does not present a restriction.

Actual measurements over the past few years indicate that the DAC never exceeds 1E-9 μCi/cm^3 at 250 kW during normal operations.

11.1.1.1.7 Radiological Assessment of Argon-41 Outside the Operations Boundary

The argon-41 produced in the reactor bay during normal operations is released to the atmosphere via an exhaust fan at approximately height h = 12 feet (3.66 m) above grade. The flow rate is 1330 cfm (6.28E5 cm^3/s). At the steady state concentration (with ventilation) computed in the previous section, the release rate would be Q = 3.01 μCi/s. The maximum downwind concentration (μCi/m^3), at grade, may be computed using the Sutton formula [15]:

$$C_{\max} = \frac{2Q}{e\pi\bar{u}h^2}\frac{C_z}{C_y}, \tag{11}$$

in which $\bar{u}$ is the mean wind speed (m/s), e = 2.718, and C_y and C_z are diffusion parameters in the crosswind and vertical directions respectively. The maximum concentration downwind occurs at distance d (m) given by

$$d = (h / C_z)^{\frac{2}{2-n}}, \tag{12}$$

in which the parameter n is associated with the wind stability condition. In this calculation, we adopt the values of n and C_z used in the McClellan AFB SAR [13]. Calculations are shown in Table 11.5.

Table 11.5 Atmospheric Dispersion Calculations

Pasquill stability class	u (m/s)	n	C_y ($m^{n/2}$)	C_z ($m^{n/2}$)	d (m)	C_{max} ($\mu Ci/cm^3$)
Extremely unstable (A)	1.6	0.2	0.31	0.31	15.53	3.29E-8
Slightly unstable (C)	4.0	0.25	0.15	0.15	38.51	1.32E-8
Slightly stable (E)	3.5	0.33	$4C_z$	0.075	105.21	3.76E-9
Extremely stable (G)	0.77	0.5	$8C_z$	0.035	492.66	8.54E-9

Appendix B of 10 CFR Part 20 lists the EC for argon-41 as 1E-8 $\mu Ci/cm^3$ for 50 mrem to the public exposed for a full year of 8760 hours. This gives a conversion factor of 5.7E5 mrem/hr per $\mu Ci/cm^{-3}$. Using the highest maximum concentration of Table 11.5 (3.29E-8 $\mu Ci/cm^3$) at steady state full power operation continuously for a full year (8760 hours) and assuming a constant frequency of class A stability results in a dose to the public in excess of 50 mrem.

The assumed 24 hours per day, 7 days per week operating history is not feasible for the RRR, which has an average operating time for two decades of about 8 hours per week, which amounts to less than 5% of the 8760 hours of a year. Additionally, a full power, continuous operation would require a significant quantity of new fuel.

Note that over the full range of conditions examined in Table 11.5, the peak downwind concentration is substantially below the DAC of 3E-3 $\mu Ci/cm^3$ established in Appendix B of 10 CFR Part 20, and less than the permissible effluent concentration of 1E-8 $\mu Ci/cm^3$ for all meteorological conditions except the set of conditions with the lowest frequency of occurrence; for that stability classification, the instantaneous effluent concentration is slightly higher than the DAC.

11.1.1.1.8 Radiological Assessment of Nitrogen-16 Sources

Nitrogen-16 is generated by the reaction of fast neutrons with oxygen and the only significant source results from reactions with oxygen in the liquid coolant of the reactor. The nuclide has a half-life of 7.13 seconds (decay constant λ_{16} = 0.0972 s^{-1} = 350 h^{-1}) and emits, predominantly, 6.13 MeV gamma rays. The effective cross section, σ_{np}, for the oxygen-16 to nitrogen-16 reaction, averaged over the fast-neutron energy spectrum in the TRIGA® or over the fission-neutron spectrum is σ_{np} = 2E–5 barns = 2E-29 cm^2. [16]

The atomic density C_N (cm^{-3}) of the nuclide as it leaves the reactor core is given in terms of the oxygen density in water, C_O = 3.34E22 cm^{-3}, as

$$C_N = \frac{\phi_f C_O \sigma_{n,p}}{\lambda_{16}}\left(1 - e^{-\lambda t}\right) \tag{13}$$

where time in the core is represented by t. Fast-neutron flux, ϕ_f, varies linearly with reactor power.

Time in core is a function of convection flow rate, a function of reactor power (see Chapter 4). As power increases, the rate of production increase from increased neutron flux is mitigated by a reduced time in the core from the increase in core cooling flow rate.

As the warmed coolant leaves the core, it passes through 1.5 inch (3.8 cm) diameter channels (with area A_{gp} = 11.4 cm^2) in the upper grid plate, but the upper end fixture of the fuel element restricts the flow. This leaves a flow area, A_0, for each element of

$$A_0 \cong A_{gp} * \left[1 - \left(\frac{3}{\pi}\right) * \sin 30^\circ * \cos 30^\circ\right] = 6.69 cm^2, \tag{14}$$

Operation at power requires primary cooling; primary coolant enters the pool through a flow diffuser approximately 2 feet (61 cm) above the core exit, 14 feet (427 cm) below the pool surface. Core exit is at 16 feet (488 cm) below the pool surface. The flow diffuser induces mixing and avoids the direct rise from the core to the pool surface (which could otherwise occur through a chimney effect from core heating). A rough estimate of hydraulic diameter of the core exit (based on total flow area) is about 5.1 inches (13 cm); calculations show the contributions to total dose rates at the pool surface are negligible at 160 to 200 cm below the surface of the pool, 22 to 25 times the hydraulic diameter of the exit into the pool. Exit flows are a small fraction of mixing flow, and under these conditions it is considered adequate to use a nuclide concentration reduced by the ratio of the total core exit surface area (approximately 555 cm^2 for 83 elements) and the pool (with a total surface area of approximately 1.2E5 cm^2); mixing reduces the concentration of nitrogen-16 from the core exit by 0.0046. Therefore, concentration of the radionuclide used in calculation is reduced from core exit by dilution.

Because of the short half-life, the concentration of nitrogen-16 is also reduced by decay during transit. Since it is difficult to characterize flow velocity field from core exit to total mixing, flow rate from the core to the surface is conservatively assumed as core exit flow rate for dose rate calculations.

Dose rate calculations were modeled as a set of disk sources, each disk containing the appropriate volume source term multiplied by the difference between the disk locations. The appropriate volume source strength for each disk source calculation was modified by exponential decay of nitrogen-16, with the time element calculated from core exit surface area, flow rate, and distance form the core exit. Dose rate calculations were based on the two major emissions, 6.13 MeV (69%) and 7.11 MeV (5%). Total dose rate at each disk (where x is the distance from the disk to the pool surface) was therefore calculated as

$$\dot{D} = \sum_E \left[\frac{k(E) E S_V \Delta d}{2} \sum_i Ai\left(E_1(\mu_i x) - \left(E_1(\mu_i x \sec\theta\right)\right] \qquad (15)$$

Where:

$$k(E): \frac{R \cdot cm \cdot s}{MeV \cdot h}$$

$$S_v = S_0(flow)e^{\left(-\lambda \frac{xA_{channel}}{\dot{m}_{H_2O}\rho_{H_2O}^{-1}}\right)}$$

A_i = Taylor buildup factor

μ_i = linear attenuation coefficient, modified by Taylor buildup factor

$$\theta = \arctan\left(\frac{x}{R_{pool}}\right)$$

Parametric variation on the distance between the disk sources showed little improvement in convergence for separations smaller than 2 cm, and essentially no improvement below 1 cm; therefore 0.5 cm was used for final calculations. Locations of interest for dose calculations include waist high (approximately 51 inches (130 cm) above the pool, 39 inches (100 cm) above the bridge) and at the ceiling over the pool 15 feet (381 cm) above the pool).

Table 11.6 Dose Rate (mR/h) Above Pool

kW	30 cm	130 cm	381 cm
50	0.1	0.1	0.0
100	0.9	0.3	0.4
200	4.0	1.2	0.1
300	8.9	2.6	0.3
400	15.0	4.4	0.5
500	21.8	6.3	0.8

Only a small proportion of the nitrogen-16 atoms present near the tank surface are actually transferred to the air of the reactor bay. Upon its formation, the nitrogen-16 recoil atom has various degrees of ionization. According to Mittl and Theys [14], practically all nitrogen-16 combines with oxygen and hydrogen atoms in high purity water, and most combines in an anion form, which has a tendency to remain in the water. In this consideration, and in consideration of the very short half-life of the nuclide, the occupational consequences of any airborne nitrogen-16 are deemed negligible in comparison to consequences from the shine from the reactor tank. Similarly, off-site radiological consequences from airborne nitrogen-16 are deemed negligible in comparison to those of argon-41.

Actual measurements at 250 kW indicate that the reading on the Radiation Area Monitor never exceeds 0.2 mrem/hour during normal operations with the primary system running, and never exceed 1 mrem per hour with the primary system off. The only exception is during the extremely infrequent use of the central thimble for beam irradiations.

11.1.1.2 Liquid Radioactive Sources

During normal operation of the RRR, the primary production of liquid radioactive materials occurs through neutron activation of impurities in the primary coolant. Most of this material is captured in mechanical filtration or ion exchange in demineralizer resin; therefore, these materials are dealt with as solid waste. Non-routine liquid radioactive waste is generated from decontamination or maintenance activities; however, based on past experience, the quantity and radioactive concentrations would be small. It is Reed College policy not to release liquid radioactive waste as an effluent.

Analysis of semiannual liquid scintillation counts of the primary coolant, secondary coolant, and environs detect no measurable quantity of tritium in the water, and thus tritium is not a concern in this analysis.

Liquid samples are normally mixed with absorbent and handled as solid waste. Table 11.1 shows the normal, measured activity of the reactor pool water.

Table 11.7 Normal Reactor Pool Activity

	Per Gallon	Total Pool	Isotopes
After a long run	0.04 μCi	1 mCi	50% argon-41; 25% manganese-56; 25% sodium-24
1 day after scram	0.003 μCi	75 μCi	Mostly sodium-24
1 week after scram	Below detection limits	0.3 μCi	Mostly cobalt-60 and europium-154

11.1.1.3 Solid Radioactive Sources

Solid sources consist of reactor fuel, a startup neutron source, and fixed radioisotope sources such as those used for instrumentation calibration. Solid wastes include: ion-exchange resin used in reactor-water cleanup, irradiated samples, labware and anti-contamination clothing associated with reactor experiments and surveillance or maintenance operations.

The solid radioactive sources associated with reactor operations are summarized in Table 11.2. Because the actual inventory of fuel and other sources continuously changes in normal operation, the information in the table is to be considered representative rather than an exact inventory. Solid and liquid wastes are not included in Tables 11.1, 11.2, and 11.3. These sources are addressed in Section 11.2.

[redacted]

11.1.2 Radiation Protection Program

The Radiation Protection Program was prepared by personnel of the RRR and the Reed College office of Environmental Health and Safety (EHS) in response to the requirements of 10 CFR Part 20. The goal of the Program is the limitation of radiation exposures and radioactivity releases to a level that is As Low As Reasonably Achievable without seriously restricting operation of the Facility for purposes of education and research. The Program is executed in coordination with the EHS office. It has been reviewed and approved by the RRC for the Facility. Certain aspects of the Program deal with radioactive materials regulated by the State of Oregon (an Agreement state) under license ORE-90010 and the Program has been reviewed by the Reed College Radioactive Materials Committee, which is responsible for administration of that license. The Program is designed to meet requirements of 10 CFR Part 20. It has been developed following the guidance of the American National Standard *Radiation Protection at Research Reactor Facilities* [1] and Regulatory Guides issued by the NRC [2-7].

[redacted]

11.1.2.1 Management and Administration

Preparation, audit, and review of the Radiation Protection Program are the responsibility of the Director of the RRR. The Reactor Review Committee (RRC) reviews the activities of the Director and audits the Program.

Surveillance and record-keeping are the responsibility of the Reactor Supervisor who reports to the Director. ALARA activities, for which record keeping is the particular responsibility of the Reactor Supervisor, are incumbent upon all radiation workers associated with the Reactor Facility.

Substantive changes in the Radiation Protection Program require approval of the RRC. Editorial changes, or changes to appendices, may be made on the authority of the Director. Changes made to the Radiation Protection Program apply automatically to operating or emergency procedures; corresponding Program changes may be made without further consideration by the RRC.

11.1.2.2 Training

Implementation of training for radiation protection is the responsibility of the Reactor Supervisor. Personnel who need access to the facility, but are not reactor staff, are either escorted by trained personnel or provided facility access training. Radiation training for licensed operators and staff is integrated with the training and requalification program.

The goal of facility access training is to provide knowledge and skills necessary to control personnel exposure to radiation associated with the operation of the nuclear reactor. Specific training requirements of 10 CFR Part

19, 10 CFR Part 20, the Radiation Protection Plan, and the Emergency Plan are explicitly addressed. A facility walkthrough is incorporated.

All persons granted unescorted access to the reactor bay must receive the training and must complete without assistance a written examination over radiation safety and emergency preparedness. Examinations must be retained on file for audit purposes for at least three years.

The reactor staff accomplishes health physics functions at the reactor following approved procedures. Therefore, procedure training for the licensed reactor staff training includes additional radiological training. Examinations for reactor staff training are prepared and implemented in accordance with the Requalification Plan.

11.1.3 ALARA Program

11.1.3.1 Policy and Objectives

Management of the Facility is committed to keeping both occupational and public radiation exposure as low as is reasonably achievable (ALARA). The specific goal of the ALARA program is to assure that actual exposures are no greater than 10 percent of the occupational limits and 50 percent of the public limits prescribed by 10 CFR Part 20.

11.1.3.2 Implementation of the ALARA Program

Planning and scheduling of operations and experiments, education and training are the responsibilities of the Reactor Supervisor and/or the Reactor Director. Any action that, in either of their opinions, might lead to as a dose of 5 mrem to any one individual requires a formal Radiation Work Permit.

11.1.3.3 Review and Audit

Implementation of the ALARA Program is audited annually by the Director as part of the general audit of the Radiation Protection Program.

11.1.4 Radiation Monitoring and Surveillance

The radiation monitoring program for the reactor is structured to ensure that all three categories of radiation sources—air, liquid, and solid—are detected and assessed in a timely manner.

11.1.4.1 Surveillances

Radiation monitoring surveillance requirements are imposed by the RRC through the Radiation Protection Program (independent of the Emergency Plan) for:

Biweekly: Wipe test reactor bay, control room, and facility

Semi-annually: Source inventory and leak test. Environmental surveillance.

11.1.4.2 Radiation Monitoring Equipment

Radiation monitoring equipment used in the reactor program is summarized in Table 11.4. Because equipment is updated and replaced as technology and performance requires, the equipment in the table should be considered as representative rather than exact.

11.1.4.3 Instrument Calibration

Radiation monitoring instrumentation is calibrated according to written procedures. NIST traceable sources are used for the calibration. The Director is responsible for calibration of the Table 11.4 instruments on site. Calibration records are maintained by the facility staff and audited annually by the RRC. Calibration stickers containing pertinent information are affixed to instruments.

Table 11.10 Typical Radiation Monitoring and Surveillance Equipment

Item	Location	Function
Air monitors (3)	Reactor bay	Measure particulates in room air
Continuous air monitor Air particulate monitor	Release stack	Measure particulates released to the public
Gaseous stack monitor	Release stack	Measure radioactive gases released to the public
Area radiation monitors (3)	Reactor bay	Measure gamma-ray exposure rates in the reactor bay
Hand and shoe monitor Walkthrough monitor	Control room	Measure removable contamination upon leaving reactor bay
Portable ion chamber meters (4)	Reactor bay and control room	Measure gamma-ray exposure rate, sense beta particles
Portable survey meters (6)	Reactor bay and control room	Measure gamma-ray exposure rate, sense beta particles
Fixed alpha/beta counter	Counting room	Wipe-test assay
Liquid scintillation spectrometer	Chemistry building	Counts liquid samples
Gamma-spectroscopy systems (2)	Counting room	Gamma-ray assay
Direct reading pocket dosimeters	Control room	Personnel gamma dose

11.1.5 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 in Section 11.1.2. This section deals with design features such as shielding, ventilation, containment, and entry control for high radiation areas, protective equipment, personnel exposure, and estimates of annual radiation exposures for specific locations within the facility. Dosimetry records and trends are also included.

11.1.5.1 Shielding

The water around the Reed TRIGA® reactor is the principal design feature for control of radiation exposure during operation. The shielding is based on TRIGA® shield designs used successfully at many other similar reactors.

The reactor is designed so that radiation from the core area can be extracted via the central thimble for research and educational purposes. When the water is removed from the central thimble, additional measures are required to control radiation exposure by restricting access to areas of elevated radiation fields. Written procedures are required for any work to be done in the vicinity of the open beam.

11.1.5.2 Personnel Exposure

Regulation 10CFR20.1502 requires monitoring of workers likely to receive, in one year from sources external to the body, a dose in excess of 10 percent of the limits prescribed in 10CFR20.1201.The regulation also requires monitoring of any individuals entering a high or very high radiation area within which an individual could receive a dose equivalent of 0.1 rem in one hour.

Table 11.5 lists results of a 7-year survey of deep dose equivalent (DDE) and shallow dose equivalent (SDE) occupational exposures at the RRR. There have been no instances of any exposures in excess of 10 percent of the applicable limits.

Monitoring of workers and members of the public for radiation exposure required by the RRC and is described in the Program.

Table 11.11 Representative Occupational Exposures

	Numbers of persons in annual-dose categories				
Year	< 100 mrem DDE	> 100 mrem DDE	< 100 mrem SDE	100-500 mrem SDE	> 500 mrem SDE
2006	51	0	50	1	0
2005	47	0	46	1	0
2004	38	0	37	1	0
2003	31	0	31	0	0
2002	30	0	29	1	0
2001	26	0	25	1	0
2000	29	0	29	0	0

11.1.5.3 Authorization for Access

Personnel who enter the control room or the reactor bay will either hold authorization for unescorted access, or be under direct supervision of an escort (i.e., escorted individuals can be observed by the escort) who holds authorization for unescorted access.

11.1.5.4 Access Control During Operation

When the reactor is operating, the licensed reactor operator (or senior reactor operator) at the controls shall be responsible for controlling access to the control room and the reactor bay.

11.1.5.5 Exposure Records for Access

Personnel who enter the reactor bay shall have a record of accumulated dose measured by a gamma sensitive individual monitoring device, either a personal dosimeter or a self-reading dosimeter. Normally no less than two individual monitoring devices may be used for a group of visitors all spending the same amount of time in the bay.

11.1.5.6 Record Keeping

Although the RRR is likely exempt from federally required record keeping requirements of 10CFR20.2106(a), certain records are required in confirmation that personnel exposures are less than 10 percent of applicable limits.

11.1.5.7 Records of Prior Occupational Exposures

These records (NRC Form 4) are initially obtained, and then maintained permanently by the Office of Environmental Health and Safety. This is not normally done for students since they typically do not have any prior occupational exposure.

11.1.5.8 Records of Occupational Personnel Monitoring

The Office of Environmental Health and Safety permanently maintains exposure records.

11.1.5.9 Records of Doses to Individual Members of the Public

Self-reading dosimeter records are kept in a logbook maintained by the RRR. Such records are kept permanently. Results of measurements or calculations used to assess accidental releases of radioactive effluents to the environment are to be retained on file permanently at the RRR.

11.1.6 Contamination Control

Potential contamination is controlled at the RRR by trained personnel following written procedures to control radioactive contamination, and by a monitoring program designed to detect contamination in a timely manner.

There are no areas within the reactor facility with continuing removable contamination. The most likely sites of contamination are the sample port at the rotary specimen rack (Lazy Susan, or LS) and at a sample-handling fume hood for receiving irradiated samples. These sites are covered by removable absorbent paper pads with plastic backing, and are routinely monitored on a periodic basis. If contaminated, pads are removed and treated as solid radioactive waste. While working at this or other potentially contaminated sites, workers wear protective gloves, and, if necessary, protective clothing and footwear. Workers are required to perform surveys to assure that no contamination is present on hands, clothing, shoes, etc., before leaving workstations where contamination is likely to occur. If contamination is detected, then a check of the exposed areas of the body and clothing is required, with monitoring control points established for this purpose. Materials, tools, and equipment are monitored for contamination before removal from contaminated areas or from restricted areas likely to be contaminated. Upon leaving the reactor bay, hands and feet are monitored for removable contamination.

RRR staff and visiting researchers are trained on the risks of contamination and on techniques for avoiding, limiting, and controlling contamination.

Table 11.6 lists sample locations for routine monitoring of surface contamination control measures. On a biweekly basis, 100 cm^2 swipe tests are analyzed for contamination. Acceptable surface contamination levels for unconditional release are no more than 1000 dpm/100 cm^2 beta-gamma radiation.

Table 11.12 Representative Contamination Sampling Locations.

Reactor bay
Clean sample-preparation fume hood
Floor between LS removal port and contaminated fume hood
Floor near entrance to reactor bay
Mechanical room
Floor in NW corner of reactor bay
Outside reactor bay
Control room
Exit hallway floor
Table for rabbit sample preparation
Stairway to Psychology building

11.1.7 Environmental Monitoring

Environmental monitoring is required to assure compliance with Subpart F of 10 CFR Part 20 and with Technical Specifications. Installed monitoring systems include area radiation monitors and airborne contamination monitors.

11.1.7.1 Radiation Area Monitors

A radiation area monitor is required for reactor operation. Radiation area monitor calibration is accomplished as required by Technical Specifications in accordance with facility procedures.

11.1.7.2 Airborne Contamination Monitors

The facility has one required air monitoring system in the reactor bay. Two additional systems monitor air from the exhaust stack. Airborne contamination monitor calibration is accomplished as required by Technical Specifications in accordance with facility procedures.

11.1.7.3 Additional Monitoring

The RRC may impose additional requirements through the Radiation Protection Program.

11.1.7.4 Contamination Surveys

Contamination monitoring requirements and surveillances addressed in 11.1.6 prevent track-out of radioactive contamination from the reactor facilities to the environment.

As required by 10CFR20.1501, contamination surveys are conducted to ensure compliance with regulations reasonable under the circumstances to evaluate the magnitude and extent of radiation levels, concentrations or quantities of radioactive material, and potential radiological hazards.

11.1.7.5 Radiation Surveys

Quarterly environmental monitoring is conducted, involving measurement of both gamma-ray doses within the facility and exterior to the facility over the course of the quarter using fixed area dosimetry.

Weekly surveys are conducted daily before operation at the RRR for radiation levels.

Gamma-ray exposure-rate data, based on quarterly measurements over the most resent 7-year period is indicated in Table 11.6. Source terms are related to reactor power levels; therefore maximum radiation levels during operation at 500 kW should not exceed twice the maximum historical values in Table 11.7.

Table 11.13 Representative Environmental Exposures

Year	Highest Annual Dose Inside Reactor Bay	Highest Annual Dose Outside the Facility
2006	197	81
2005	136	17
2004	3875	81
2003	688	102
2002	659	25
2001	204	20
2000	21	1

11.1.7.6 Monitoring for Conditions Requiring Evacuation

An evacuation alarm is required in the reactor. Response testing of the alarm is performed in accordance with facility procedures.

11.2 Radioactive Waste Management

The reactor generates very small quantities of radioactive waste, as indicated in Section 11.1.1. Training for waste management functions are incorporated in operator license training and requalification program.

11.2.1 Radioactive Waste Management Program

Liquid wastes are not customarily released from the RRR. Solid wastes are either allowed to decay in storage to background, or are shipped for burial.

11.2.2 Radioactive Waste Controls

Radioactive solid waste is generally considered to be any item or substance no longer of use to the facility, which contains or is suspected of containing radioactivity above background levels. Volume of waste at the RRR is small, and the nature of the waste items is limited and of known characterization. Consumable supplies such as absorbent materials or protective clothing are declared radioactive waste if radioactivity above background is found to be present.

When possible, solid radioactive waste is initially segregated at the point of origin from items that are not considered waste. Screening is based on the presence of detectable radioactivity using appropriate monitoring and detection techniques and on the future need for the items and materials involved. Oregon is an "agreement state," so radioactive materials generated for research and experiments under the federal byproduct material license of the Reactor Facility are transferred to the State of Oregon license for conduct of the activities.

Although argon-41 is released from the RRR, this release is not considered to be waste in the same sense as liquid and solid wastes. Rather, it is an effluent, which is routine part of the operation of the facility. A complete description of argon-41 production and dispersal is provided in section 11.1.1.

11.2.3 Release of Radioactive Waste

The RRR does not have a policy of releasing radioactive waste to the environment as effluent. If contaminated liquids are produced, they are contained locally, added to absorbent, and transferred to a waste barrel in preparation for transfer to a licensed burial facility. Solid waste is likewise routinely contained on-site.

11.3 References

1) *ANSI/ANS/L5.11 (Final Draft), Radiation Protection at Research Facilities, American Nuclear Society, La Grange Park, Illinois, October, 1992.*

2) *Instructions for Recording and Reporting Occupational Radiation Exposure Data, Regulatory Guide 8.7 (Rev. 1), U.S. Nuclear Regulatory Commission, Washington, D.C., 1992.*

3) *Monitoring Criteria and Methods to Calculate Occupational Radiation Doses, Regulatory Guide 8.34, U.S. Nuclear Regulatory Commission, Washington, D.C., 1992.*

4) *Air Sampling in the Workplace, Regulatory Guide 8.25 (Rev. 1), U.S. Nuclear Regulatory Commission, Washington, D.C., 1992.*

5) *Planned Special Exposures, Regulatory Guide 8.35, U.S. Nuclear Regulatory Commission, Washington, D.C., 1992.*

6) *Radiation Dose to the Embryo/Fetus, Regulatory Guide 8.36, U.S. Nuclear Regulatory Commission, Washington, D.C., 1992.*

7) *Interpretation of Bioassay Measurements, Draft Regulatory Guide 8.9 (DG-8009), U.S. Nuclear Regulatory Commission, Washington, D.C., 1992.*

8) *Data for Use in Protection Against External Radiation, Publication 51, International Commission on Radiological Protection, 1987.*

9) *Limits for Intakes of Radionuclides by Workers, Publication 30, International Commission on Radiological Protection, 1979.*

10) *1990 Recommendations of the International Commission on Radiological Protection, Publication 60, International Commission on Radiological Protection, 1991.*

11) *Limits for Intakes of Radionuclides by Workers Based on 1990 Recommendations of the International Commission on Radiological Protection, Publication 61, International Commission on Radiological Protection, 1991.*

12) *Reed Reactor Facility Safety Analysis Report, License R-1128, Docket 50-288, 1968.*

13) *Facility Safety Analysis Report, Rev. 2, McClellan Nuclear Radiation Center Reactor, April 1998.*

14) *Mittl, R.L. and M.H. Theys, "N-16 Concentrations in EBWR," Nucleonics, March 1961, p. 81.*

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17) *Gevantman, LH. Solubility of Selected Gases in Water. In: Lide, DR, editor. CRC Handbook of Chemistry and Physics, 84th ed. Boca Raton: CRC Press, 2003. p 8-88.*

Chapter 12

Conduct of Operations

Reed Research Reactor Safety Analysis Report

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Chapter 12

Conduct of Operations

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12 CONDUCT OF OPERATIONS

This chapter describes the conduct of operations at the Reed Research Reactor (RRR). The conduct of operations involves the administrative aspects of facility operations, the facility emergency plan, the physical security plan, and the requalification plan. This chapter of the Safety Analysis Report (SAR) forms the basis of Section 6 of the Technical Specifications (Chapter 14).

12.1 Organization

The operating license R-112, Docket 50-288, for the reactor is held by Reed College. The chief administrating officer for Reed College is the President. The Reactor Director reports to the Vice President, Dean of the Faculty, who in turn reports directly to the President of Reed College. The Director is responsible for licensing and reporting information to the NRC.

12.1.1 Structure

As indicated on Figure 12.1, the President of Reed College is the licensee for the Reed Reactor Facility. The reactor is under the direct control of the Reed Reactor Facility Administration, consisting of the facility Director and Associate Director, who report to the college Dean of the Faculty and President.

Environmental, safety, and health oversight functions are administered through the Vice President, Treasurer, while reactor line management functions are through the Vice President, Dean of the Faculty. Radiation protection functions are divided between the Radiation Safety Officer (RSO) and the reactor staff and management, with management and authority for the RSO separate from line management and authority for facility operations. Day-to-day radiation protection functions implemented by facility staff and management are guided by approved administrative controls (Radiation Protection Program or RPP, operating and experiment procedures). These controls are reviewed and approved by the RSO as part of the Reactor Review Committee (RRC). The Reactor Health Physicist reports to the RSO and has specific oversight functions assigned though the Administrative Procedures. The Reactor Health Physicist provides routine support for personnel monitoring, radiological analysis, and radioactive material inventory control. The Reactor Health Physicist provides guidance on request for non-routine operations such as transportation and implementation of new experiments.

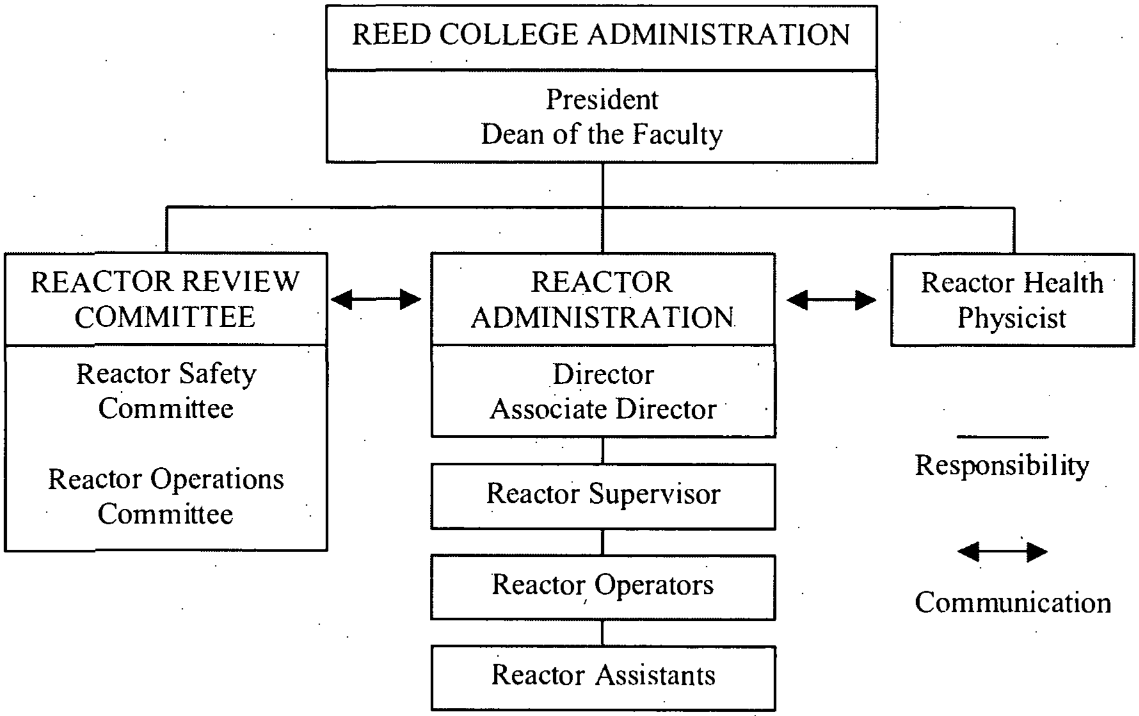

Figure 12.1 Organization and Management Structure for the Reed Reactor

12.1.2 Responsibility

The following describes the individuals and groups that appear in the organizational structure chart and their associated duties and responsibilities.

12.1.2.1 Reed College Administration

The Reed College administration is responsible for establishing the facility budget, and for appointing the Director, Associate Director, Health Physicist, and members of the Reactor Review Committee.

12.1.2.2 Reactor Review Committee

The Reactor Review Committee serves as an oversight committee. The RRC is responsible for reviewing reactor operations to assure that the reactor facility is operated and used in a manner within the terms of the facility license and consistent with the safety of the public. Duties of the Reactor Review Committee are enumerated in Technical Specifications, Section 6.2.b.

12.1.2.3 Reactor Health Physicist

The Reactor Health Physicist provides routine support for personnel monitoring, radiological analysis, and radioactive material inventory control. The Reactor Health Physicist reports to the College Radiation Safety Officer. Duties of the Reactor Health Physicist are enumerated in Technical Specifications, Section 6.1.2.g.

12.1.2.4 Reed Reactor Facility Administration

The Reactor Director has the ultimate responsibility for the safe and competent operation of the RRR. The Associate Director acts as an assistant to the Director, and can act on the behalf of the Director in some instances. Duties of the Director and Associate Director are enumerated in Technical Specifications, Section 6.1.2.b and 6.1.2.c.

12.1.2.5 Reactor Supervisor

The Reactor Supervisor is responsible to the Director for directing the activities of Reactor Operators and for the day-to-day operation and maintenance of the reactor. The Supervisor shall be a NRC licensed Senior Reactor Operator for the facility. Specific duties are enumerated in Technical Specifications, Section 6.1.2.d.

12.1.2.6 Reactor Operators

Reactor Operators are appointed by the Director, and must hold an NRC Reactor Operator or Senior Reactor Operator license. They are responsible for the safe and competent operation and maintenance of the reactor and associated equipment. Specific duties of Reactor Operators are enumerated in Technical Specifications, Section 6.1.2.h.

12.1.2.7 Reactor Assistants

Reactor Assistants are appointed by the Director to work at the Facility under the direct supervision of a licensed Reactor Operator or the Health Physicist. Assistants shall be trained in the safe use of radioactive materials, radiation safety, and emergency procedures.

12.1.3 Staffing

Whenever the reactor is not in the secured mode, the reactor shall be under the direction of a US NRC licensed Senior Operator. The Senior Operator shall be easily reachable, such as by phone or pager, on campus, and within five minutes travel time of the facility.

Whenever the reactor is not secured, a US NRC licensed Reactor Operator (or Senior Reactor Operator) who meets requirements of the Operator Requalification Program shall be at the reactor control console, and directly responsible for reactivity manipulations.

Whenever the reactor is not secured, a second person shall be in the facility. This person may leave the facility briefly to take readings or conduct inspections.

In addition to the above requirements, during fuel movement a Senior Operator inside the reactor bay directing fuel operations.

Only the Reactor Operator at the controls or personnel authorized by, and under direct supervision of, the Reactor Operator at the controls shall manipulate the controls. Whenever the reactor is not secured, operation of equipment that has the potential to affect reactivity or power level shall be manipulated only with the knowledge and consent of the Reactor Operator at the controls. The Reactor Operator at the controls may authorize persons to manipulate reactivity controls who are training either as (1) a student making use of the reactor, (2) to qualify for an operator license, or (3) in accordance the approved Reactor Operator requalification program.

12.1.4 Selection and Training of Personnel

The Director of the RRR shall select individuals with the requisite experience and qualifications recommended in ANSI/ANS 15.4, *Selection and Training of Personnel for Research Reactors*. All personnel shall have a combination of academic training, experience, health, and skills commensurate with their responsibility and duties. Training for new personnel includes emergency preparedness and radiation safety.

12.1.5 Radiation Safety

The radiation safety program is discussed in Chapter 11.

12.2 Review and Audit Activities

It is the responsibility of the Reactor Review Committee (RRC) to review reactor operations to assure that the reactor facility is operated and used in a manner within the terms of the facility license and consistent with the safety of the public and of persons.

12.2.1 Composition and Qualifications

The RRC shall be composed of:

1) One or more persons proficient in reactor and nuclear science or engineering,
2) One or more persons proficient in chemistry, geology, or chemical engineering,
3) One person proficient in biological effects of radiation,
4) The Reactor Director, *ex officio*,
5) The Radiation Safety Officer, *ex officio*, and,
6) The Vice President, Dean of Faculty, *ex officio*, or a designated deputy.

The same individual may serve under more than one category above, but the minimum membership shall be seven. At least two members shall be Reed College faculty members. Except for ex-officio members, the President of Reed College appoints committee members and chairs. The Reactor Director, Associate Director, and Supervisor serve as non-voting members of all committees and subcommittees. The Reactor Supervisor shall attend and participate in RRC meetings, but shall not be a voting member. No limit shall exist on the overlap of personnel, including chairperson(s), between the subcommittees.

12.2.2 Charter and Rules

The RRC shall have a written statement defining its authority and responsibilities, the subjects within its purview, and other such administrative provisions as are required for its effective functioning. Minutes of all meetings and records of all formal actions of the RRC shall be kept by the Director.

The RRC shall meet a minimum of two times each academic year. Additional meetings may be called by the chair, and the RRC may be polled in lieu of a meeting. Such a poll shall constitute RRC action subject to the same requirements as for an actual meeting. A quorum shall consist of not less than a majority of the voting RRC members. Any action of the RRC requires a majority vote of the members present.

The Reactor Review Committee may be divided into two subcommittees (Reactor Operations Committee and Reactor Safety Committee).

A written report of the findings of any audit shall be submitted to the Director after the audit has been completed.

12.2.3 Review Function

The responsibilities of the RRC shall include, but are not limited to, the following:

1) Review and approval of rules, procedures, and proposed Technical Specifications;

2) Review and approval of all proposed changes in the facility that could have a significant effect on safety and of all proposed changes in rules, procedures, and Technical Specifications, in accordance with procedures in Technical Specifications, Section 6.3;
3) Review and approval of experiments using the reactor in accordance with procedures and criteria in Technical Specifications, Section 6.4;
4) Determination of whether a proposed change, test, or experiment would constitute an unreviewed safety question or change in the Technical Specifications (Ref. 10 CFR 50.59);
5) Review of abnormal performance of equipment and operating anomalies;
6) Review of unusual or abnormal occurrences and incidents which are reportable under 10 CFR Part 20 and 10 CFR Part 50; and
7) Inspection of the facility, review of safety measures, and audit of operations at a frequency not less than once a year, including operation and operations records of the facility. Standard Operating Procedures shall be audited biennially.

12.2.4 Audit Function

The RRC shall audit the reactor operations, including but not limited to, operation and operations records of the facility, annually.

Members of the Reactor Review Committee who are assigned responsibility for audits shall perform or arrange for examination of operating records, logs, and other documents. Discussions with cognizant personnel and observation of operations shall be used as appropriate. In no case shall the individual immediately responsible for an aspect of facility operation audit that area.

The purpose of audits is to determine if activities since the last audit were conducted safely and in accordance with regulatory requirements and applicable procedures. In addition to checking the controlling document or procedure, the audit should verify that the records are completed and retrievable, that the procedures are clear, that deficiencies in previous audits have been addressed, and that the procedure fulfills the intended function.

The status of the reviews and audits shall be a standing agenda item for all committee meetings. Deficiencies uncovered in audits that affect reactor safety shall immediately be reported to the President of Reed College by the chairperson of the Committee. A written report of the findings of the audit shall be submitted to the Director after the audit has been completed.

The Radiation Protection Plan, Requalification Plan and Emergency Plan are audited each academic year. A list of items to be audited and their frequency is provided in the RRR Administrative Procedures, Section 2.3

12.3 Procedures

Written procedures, reviewed and approved by the RRC, shall be followed for the activities listed below. The procedures shall be adequate to assure the safety of the reactor, persons within the facility, and the public, but should not preclude the use of independent judgment and action should the situation require it. The activities are:

1) Startup, operation, and shutdown of the reactor, including

   (a) Startup procedures to test the reactor instrumentation and safety systems, area monitors, and continuous air monitors, and

   (b) Shutdown procedures to assure that the reactor is secured before the end of the day.

2) Installation or removal of fuel elements, control rods, and other core components that significantly affect reactivity or reactor safety.
3) Preventive or corrective maintenance activities that could have a significant effect on the safety of the reactor or personnel.
4) Periodic inspection, testing, or calibration of systems or instrumentation that relate to reactor operation.

Substantive changes in the above procedures shall be made only with the approval of the RRC, and shall be issued to the personnel in written form. The Reactor Director may make temporary changes that do not change the original intent. The change and the reasons thereof shall be reviewed by the RRC.

12.4 Required Actions

This is addressed in the RRR Technical Specifications.

12.5 Reports to the Nuclear Regulatory Commission

This is addressed in the RRR Technical Specifications.

12.6 Record Retention

This is addressed in the RRR Technical Specifications.

12.7 Emergency Planning

The RRR Emergency Plan contains detailed information regarding the RRR response to emergency situations. The RRR Emergency Plan is written to be in accordance with ANSI/ANS 15.16, *Emergency Planning for Research Reactors.*

12.8 Security Planning

The RRR Physical Security Plan contains detailed information concerning the RRR security measures. The plan provides the RRR with criteria and actions for protecting the facility.

Primary responsibility for the plan and facility security rest with the Director. Implementation of the plan on a day-to-day basis is also the responsibility of the Director.

12.9 Quality Assurance

Quality Assurance can be found through the operating procedures. It is not called out as a separate document.

12.10 Operator Training and Requalification

The RRR Requalification Plan is designed to satisfy the requirements of 10 CFR 55. The Requalification Plan is provided as an attachment to this Safety Analysis Report.

12.11 Startup Plan

This is not applicable.

12.12 Environmental Reports

The Environmental Report for the RRR is attached to this Safety Analysis Report.

Chapter 13

Accident Analysis

Reed Research Reactor Safety Analysis Report

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Chapter 13

Accident Analysis

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13 ACCIDENT ANALYSIS

This chapter provides information and analysis to demonstrate that the health and safety of the public and workers are protected in the event of equipment malfunctions or other abnormalities in reactor behavior. The analysis demonstrates that facility design features, limiting safety system settings, and limiting conditions for operation ensure that no credible accident could lead to unacceptable radiological consequences to people or the environment.

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* .[1] The information developed by the TRIGA® experience base, plus appropriate information from NUREG/CR-2387, serve as a basis for some of the information presented in this chapter.

The reactor physics and thermal-hydraulic conditions in the Reed Research Reactor (RRR) at power levels of 250 kW and 500 kW are established in Chapter 4. In this chapter, it was assumed that two different TRIGA® fuel types are in use in the RRR: 8.5 weight percent fuel with 20% enrichment with either aluminum or stainless steel cladding.

The fuel temperature is the limit on operation of the RRR. This limit stems from the out-gassing of hydrogen from U-ZrH fuel and the subsequent stress produced in the fuel element cladding material. Calculations performed by General Atomics and confirmed by experiments indicate that no cladding damage occurs at peak fuel temperatures as high as approximately 530°C (986°F) for low-hydride-type (U-$ZrH_{1.0}$), aluminum-clad elements, [3] and 1175°C (2150°F) for high-hydride-type (U-$ZrH_{1.65}$), stainless-steel-clad elements. [2,3] Cladding damage in the high-hydride-type, stainless-steel fuel is caused by a pressure buildup in the element as a result of the evolution of hydrogen produced by dehydriding of the fuel with increasing temperature. The pressure internal to the fuel element reaches the point where the cladding fails. Cladding damage in the low-hydride-type, aluminum-clad fuel is caused by a phase change in the fuel matrix that occurs at about 530°C (986°F). The phase change causes the fuel to swell that causes the cladding to fail. For a core containing only aluminum clad fuel or containing both aluminum and stainless steel clad fuel, a fuel temperature limit of 500°C (932°F) is determined by the cladding damage threshold temperature of the low-hydride-type, aluminum-clad fuel elements. For a future core with only stainless steel clad fuel, fuel temperature limits of 1,100°C (2012°F) (with clad < 500°C) and 930°C (1706°F) (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.

Nine credible accidents for research reactors were identified in NUREG-1537 [4] as follows:

1. The maximum hypothetical accident (MHA);
2. Insertion of excess reactivity;
3. Loss of coolant accident (LOCA);
4. Loss of coolant flow;
5. Mishandling or malfunction of fuel;
6. Experiment malfunction;
7. Loss of normal electrical power;
8. External events; and
9. 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 that appeared applicable or credible for the RRR, 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 not result in the release of radioactive materials from the 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 MHA for TRIGA® reactors is the cladding failure of a single irradiated fuel element in air with no radioactive decay of the contained fission products taking place prior to the release.

13.2 Accident Initiating Events and Scenarios, Accident Analysis, and Determination of Consequences

13.2.1 Maximum Hypothetical Accident (MHA)

The failure of the encapsulation of one fuel element, in air, resulting in the release of gaseous fission products to the atmosphere is considered to be the Maximum Hypothetical Accident (MHA). Administrative controls prevent removal of fuel from the reactor pool during fuel handling, but one could postulate fuel failure in air during fuel transfer circumstances. Potential consequences of fuel failure in air including inhalation by the public are considered in this scenario.

13.2.1.1 Accident Initiating Events and Scenarios

A single fuel element could fail at any time during normal reactor operation or while the reactor is in a shutdown condition, due to a manufacturing defect, corrosion, or handling damage. This type of accident is very infrequent, based on many years of operating experience with TRIGA® fuel, and such a failure would not normally incorporate all of the necessary operating assumptions required to obtain a worst-case fuel-failure scenario.

For the RRR, the MHA has been defined as the cladding rupture of one highly irradiated fuel element with no radioactive decay followed by the instantaneous release of the noble gas and halogen fission products into the air. For this accident, the fuel cladding type makes no difference as either element would contain the same amount of uranium-235 and, hence, the same inventory of fission products. The following assumptions and approximations were applied to this calculation.

1. For long-lived radionuclides, calculations of radionuclide inventory in fuel are based on continuous operation prior to fuel failure for 40 years at the average thermal power of 3.71 kW, the actual value for the RRR over the past 40 years.
2. For short-lived radionuclides, calculations of radionuclide inventory in fuel are based on operation at the full thermal power of 500 kW for eight hours per day, for five successive days prior to fuel failure, an average of 12.5 kW-hr/day.
3. Radionuclide inventory in one "worst-case" fuel element is based on a 64 element core for the historical period and 83 elements for full power operation of a future core, 39 grams of uranium-235 per element, and a value of 2.0 as a very conservative value of the ratio of the maximum power in the core to the average power. Thus, for the historical period, the worst case element has operated at a thermal power of 2(3710/64) = 115.9 W, and for the one-week full-power operation, 2(500/83) = 12.05 kW.
4. The fraction of noble gases and iodine contained within the fuel that is actually released is 1.0E-4. This is a very conservative value prescribed in NUREG 2387 [1] and may be compared to the value of 1.5E-5 measured at General Atomics [3] and used in SARs for other reactor facilities. [5]
5. The fractional release of particulates (radionuclides other than noble gases and iodine) is 1.0E-6, a very conservative estimate used in NUREG-2387. [1]

13.2.1.2 Accident Analysis and Determination of Consequences

Radionuclide Inventory Buildup and Decay

Consider a mass of uranium-235 yielding thermal power P (kW) due to thermal-neutron induced fission. The fission rate is related to the thermal power by the factor k = 3.12E13 fissions per second per kW.[1] Consider also a fission product radionuclide, which is produced with yield Y, and which decays with rate constant λ. It is easily shown that the equilibrium activity A_∞ of the fission product, which exists when the rate of creation by fission is equal to the rate of loss by decay, is given by $A_\infty = kPY$. Here it should be noted that the power must be small enough or the uranium mass large enough that the depletion of the uranium-235 is negligible.[2] Starting at time $t = 0$, the buildup of activity is given by

$$A(t) = A_\infty\left(1 - e^{-\lambda t}\right). \qquad (1)$$

For times much greater than the half-life of the radionuclide, $A \approx A_\infty$, and for times much less than the half-life, $A(t) = A_\infty\lambda t$. If the fission process ceases at time t_1, the specific activity at later time t is given by

$$A(t) = A_\infty\left(1 - e^{-\lambda t_1}\right)e^{-\lambda(t-t_1)}. \qquad (2)$$

Consider the fission product iodine-131, which has a half-life of 8.04 days (λ = 0.00359 h^{-1}) and a chain (cumulative) fission product yield of about 0.031. At a thermal power of 1 kW, the equilibrium activity is about A_∞ = 9.67E11 Bq (26.1 Ci). After only four hours of operation, though, the activity is only about 0.37 Ci. For equilibrium operation at 3.5 kW, distributed over 81 fuel elements, the average activity per element would be (26.1)(3.5)/81 = 1.13 Ci per fuel element. The worst case element would contain twice this activity. With a release fraction of 1.0E-4, the activity available for release would be about (1.13)(2)(1.0E-4) = 2.26E-4 Ci. This type of calculation is performed by the ORIGEN code [CCC-371] for hundreds of fission products and for arbitrary times and power levels of operation as well as arbitrary times of decay after conclusion of reactor operation. The code accounts for branched decay chains. It also may account for depletion of uranium-235 and in-growth of plutonium-239, although those features were not invoked in the calculations reported here because there is minimal depletion in TRIGA® fuel elements.

Data From ORIGEN Calculations

ORIGEN-2.1 calculation output files are included as Appendices A and B. Appendix A contains data for the buildup of long-lived radionuclides over the 40-year entire operating history of the RRR, which is modeled as 115.9 W continuous thermal power. Appendix B contains data for the buildup of relatively

[1] Note that the product of k and yield Y may be stated as $3.12E^{13} \times Y$ Bq/kW or $843 \times Y$ Ci/kW.

[2] Negligible burnup is modeled in ORIGEN calculations by setting the fuel mass very large (1 tonne) and the thermal power very low (1 kW or less).

short-lived radionuclides during a worst-case scenario modeled as 8-hours/day operations at 12.05 kW thermal power for five consecutive days. Tabulated results for Appendices A and B are μCi activities, by nuclide, immediately after reactor shutdown, and at 1, 2, 3, 7, and 14 days after shutdown. In Appendix A, which deals with relatively long-lived radionuclides, data are provided only for those nuclides present at activities greater than 100 mCi in a single fuel element at 1 day after reactor shutdown. In Appendix B, which deals with relatively short-lived radionuclides, data are provided only for those nuclides present at activities greater than 100 mCi in a single fuel element immediately after reactor shutdown. In both Appendices A and B, the activities per element are multiplied by the release fractions previously cited, thus yielding maximum activities available for release in a maximum hypothetical accident.

Reference Case Source Terms

Appendices A and B data for worst case TRIGA® fuel element are compared; greater values for any one isotope are selected as reference case source terms for maximum hypothetical accident. Data are presented in Table 13.1 for halogens/noble gases, and Table 13.2 for particulate radionuclides.

Derived Quantities

The raw data of Tables 13.1 and 13.2 are activities potentially released from a single worst-case fuel element that has experienced a cladding failure. This activity may itself be compared to the annual limit of intake (ALI) to gauge the potential risk to an individual worker. By dividing the activity by a conservative value of 11,000 ft^3 free volume of the reactor bay to allow for equipment present in the room, one obtains an air concentration (specific activity) that may be compared to the derived air concentration (DAC) for occupational exposure as given in 10 CFR Part 20 or in EPA federal guidance. [6]

Comparison with the DAC and the ALI

The ALI is the activity that, if ingested or inhaled, would lead to either (a) the maximum permissible committed effective dose equivalent incurred annually in the workplace, nominally 5 rem, or (b) the maximum permissible dose to any one organ or tissue, nominally 50 rem. The DAC is the air concentration that, if breathed by reference man for one work year (2,000 hours), would result in the intake of the ALI. ALI does not apply to noble-gas radionuclides.

Potential activity releases are compared to ALIs, and air concentrations in the reactor bay are compared to DACs in Tables 13.3 and 13.4. Only for radioiodine does the available activity significantly exceed the ALI. However, there is no credible scenario for accidental inhalation or ingestion of the undiluted radioiodine released from a fuel element.

Table 13.1 Reference Case Halogen & Noble Gas Activities Potentially Released in Maximum Hypothetical Accident at the RRR

		Available activity (μCi) at time in days after reactor shutdown						
Element	Nuclide	0	1	2	3	7	14	28
Br	83	12110	13	0	0	0	0	0
Br	84	25388	0	0	0	0	0	0
I	131	8490	8245	7695	7130	5125	2810	840
I	132	23280	20943	16930	13685	5838	1318	68
I	133	69950	33530	15073	6773	278	0	0
I	134	192228	0	0	0	0	0	0
I	135	98750	7980	645	53	0	0	0
Kr	85	45	45	45	45	45	45	45
Kr	87	64498	0	0	0	0	0	0
Kr	88	79048	225	0	0	0	0	0
Kr	83M	9953	48	0	0	0	0	0
Kr	85M	23368	580	15	0	0	0	0
Xe	133	20363	24105	24010	22390	14110	5673	895
Xe	135	58955	30518	6595	1195	0	0	0
Xe	138	158548	0	0	0	0	0	0
Xe	133M	35	33	28	20	8	0	0
Xe	135M	205	15	0	0	0	0	0

NOTE: Available activity (> 10 μCi) is from a single worst-case fuel element as a function of time after reactor operation. Data are derived from ORIGEN 2.1 calculations [7] *as summarized in* Appendices A and B. *Data are raw computational results and the number of significant figures exceeds the precision of the calculation.*

Table 13.2 Reference Case Particulate Activities Potentially Released in a Maximum Hypothetical Accident at the RRR

		Available activity (μCi) at time in days after reactor shutdown						
Element	Nuclide	0	1	2	3	7	14	28
Ba	139	1605	0	0	0	0	0	0
Ba	140	128	120	115	108	88	60	28
Ba	141	1480	0	0	0	0	0	0
Ba	142	1470	0	0	0	0	0	0
Ce	141	48	53	53	50	45	40	30
Ce	143	553	340	205	125	18	0	0
Cs	138	1713	0	0	0	0	0	0
La	140	68	88	98	103	95	68	33
La	141	1140	18	0	0	0	0	0
La	142	1455	0	0	0	0	0	0
La	143	1493	0	0	0	0	0	0
Mo	99	395	308	240	185	68	13	0
Mo	101	1273	0	0	0	0	0	0
Nb	97	598	248	93	35	0	0	0
Nb	98	1463	0	0	0	0	0	0
Nd	147	55	50	48	45	35	23	10
Nd	149	263	0	0	0	0	0	0
Nd	151	105	0	0	0	0	0	0
Pm	151	40	23	13	8	0	0	0
Pr	143	70	88	98	100	90	65	33
Pr	145	638	40	3	0	0	0	0
Pr	147	573	0	0	0	0	0	0
Rb	88	800	3	0	0	0	0	0
Rb	89	1218	0	0	0	0	0	0
Rh	105	78	65	40	25	5	0	0
Rh	107	40	0	0	0	0	0	0
Ru	105	188	5	0	0	0	0	0
Se	81	53	0	0	0	0	0	0
Se	83	50	0	0	0	0	0	0
Sn	128	83	0	0	0	0	0	0
Sr	89	28	28	28	28	25	23	20
Sr	91	795	138	25	5	0	0	0
Sr	92	1325	3	0	0	0	0	0
Tc	101	1273	0	0	0	0	0	0
Tc	104	460	0	0	0	0	0	0
Te	129	95	3	0	0	0	0	0
Te	131	628	5	3	3	0	0	0
Te	132	250	203	165	133	58	13	0
Te	133	965	0	0	0	0	0	0
Te	134	1713	0	0	0	0	0	0
Y	92	848	38	0	0	0	0	0
Y	93	863	170	33	8	0	0	0
Y	94	1583	0	0	0	0	0	0
Y	95	1613	0	0	0	0	0	0
Y	91M	423	88	15	3	0	0	0
Zr	95	30	30	30	28	28	25	23
Zr	97	658	245	93	35	0	0	0

Available activity (> 10 μCi) is for a single worst-case fuel element as a function of time after reactor operation. Data are derived from ORIGEN 2.1 calculations [7] as summarized in Appendices A and B. The table includes only those nuclides with activities in excess of 10 μCi. Data are raw computational results and the number of significant figures exceeds the precision of the calculation.

When one compares with DACs the potential airborne concentration of radionuclides in the reactor bay, only the iodine-131, iodine-133, and iodine-135 isotopes plus krypton-87 and krypton-88 are of potential consequence. However, annual dose limits could be attained only with a constant air concentration over a long period of time. The iodine-131 released in the failure of a single element, for example, would decay with a half-life of 8.04 days. Thus, even the undetected failure of a fuel element would not be expected to lead to violations of the occupational dose limits expressed in 10 CFR Part 20 or in other federal guidance.

Comparison with the Effluent Concentration

Effluent concentration, listed in the last columns of Tables 13.3 and 13.4, are defined in continuous exposure (8,760 hours per year) rather than 2,000 hours per year occupational exposure. Exposure to a constant airborne concentration equal to the effluent concentration for one full year results in the annual dose limit of 100 mrem to members of the public. As is apparent from Tables 13.3 and 13.4, the reactor bay average concentrations immediately after fuel element failure significantly exceed the effluent concentrations for several radionuclides. Thus, only for these radionuclides is it necessary to consider radioactive decay and atmospheric dispersal after release in estimating potential risk to members of the public. For posting purposes, concentrations relative to DACs are additive. For dosimetry purposes, products of concentrations and times, relative to DAC-hours, are additive.

Potential downwind dose to a member of the public

In this dose assessment, it is assumed that the available activity in a failed fuel element is released instantaneously and immediately after reactor shutdown. It is further assumed that a member of the public is positioned directly downwind from the RRR and remains in place during the entire passage of the airborne radioactivity. The very conservative approximations of Hawley and Kathren [1] are adopted in the assessment, namely, that the atmospheric dispersion (χ/Q) factor is 0.01 s/m^3 (2.78E-6 h/m^3) and the breathing rate V is 1.2 m^3/h. No credit is taken for partial containment, plateout, or other potential mitigating mechanisms, however realistic and probable.

Let the activity of nuclide i released be A (μCi) as given in Tables 13.3 and 13.4. If one neglects radioactive decay, the activity inhaled during passage of the airborne activity is $AV(\chi/Q)$. [8] The product of the activity inhaled and the dose conversion factor $\Re$ (mrem/μCi) [6] yields D_i (mrem), the more critical of the organ dose or the effective dose equivalent to the total body. Results of such calculations are presented in Table 13.5. As is apparent from the table, individual organ doses as well as the total committed effective dose equivalent are well below any regulatory limits. Entries are shown only for doses of 0.001 mrem or greater.

Table 13.3 Comparison of Halogen and Noble Gas Available Activities Immediately After Reactor Shutdown with ALIs and Reactor bay Concentrations with DACs and Effluent Concentrations

Element	Nuclide	Half-life	Available activity (μCi)	Inhalation ALI (μCi)	DAC (μCi/cm^3)	Effluent Conc. Limit (μCi/cm^3)	Reactor bay conc. (μCi/cm^3)	Ratio to DAC	Ratio to Effluent Conc. Limit
Br	83	2.39 h	12100	6E+04	3E-05	9E-08	3.88E-5	1.29	423.0
Br	84	31.8 m	25375	6E+04	2E-05	8E-08	8.15E-5	4.07	1020.0
I	131	8.04 d	8500	5E+01	2E-08	2E-10	2.73E-5	1360.0	136000.0
I	132	2.30 h	23275	8E+03	3E-06	2E-08	7.47E-5	24.9	3740.0
I	133	20.8 h	69950	3E+02	1E-07	1E-09	2.25E-4	2250.0	225000.0
I	134	52.6 m	192225	5E+04	2E-05	6E-08	6.17E-4	30.9	10300.0
I	135	6.61 h	98750	2E+03	7E-07	6E-09	3.17E-4	453.0	52800.0
Kr	83m	1.83 h	9950		1E-02	5E-05	3.19E-5	0.0	0.6
Kr	85m	4.48 h	23375		2E-05	1E-07	7.50E-5	3.75	750.0
Kr	85	10.7 y	50		1E-04	7E-07	1.61E-7	0.0	0.2
Kr	87	76.3 m	64500		5E-06	2E-08	2.07E-4	41.4	10400.0
Kr	88	2.84 h	79050		2E-06	9E-09	2.54E-4	127.0	28200.0
Xe	133m	2.19 d	0		1E-04	6E-07	0.0	0.0	0.0
Xe	133	5.25 d	20350		1E-04	5E-07	6.53E-5	0.6	131.0
Xe	135m	15.3 m	25		9E-06	4E-08	8.03E-8	0.0	2.0
Xe	135	9.09 h	58950		1E-05	7E-08	1.89E-4	18.9	2700.0

[a]Room concentration exceeds DAC.
[b]Room concentration exceeds effluent concentration.

Table 13.4 Comparison of Particulate Available Activities (> 100 μCi) with ALIs and Reactor bay Concentrations with DACs and Effluent Concentrations

Element	Nuclide	Half-life	Available activity (μCi)	Inhalation ALI (μCi)	DAC (μCi/cm^3)	Effluent Conc. Limit (μCi/cm^3)	Reactor bay conc. (μCi/cm^3)	Ratio to DAC	Ratio to Effluent Conc. Limit
Ba	139	82.7 m	1600	7.50E+04	1E-05	2E-09	4E-07	0.51	2570.00
Ba	140	12.7 d	125	2.50E+03	6E-07	2E-09	3E-08	0.67	201.00
Ba	141		1475	1.75E+05	3E-05	1E-07	3.5E-07	0.16	47.40
Ce	141		50	1.50E+03	2E-07	8E-10	1.23E-08	0.80	201.00
Ce	143	33.0 h	550	5.00E+03	7E-07	2E-09	1.35E-07	2.52	883.00
Cs	138	32.2 m	1700	1.50E+05	2E-05	8E-08	4.25E-07	0.27	68.20
La	140		75	2.50E+03	5E-07	2E-09	1.85E-08	0.48	120.00
La	141	3.93 h	1150	2.25E+04	4E-06	1E-08	2.75E-07	0.92	369.00
La	142	92.5 m	1450	5.00E+04	9E-06	3E-08	3.5E-07	0.52	155.00
La	143		1500	2.25E+05	4E-05	1E-07	3.75E-07	0.12	48.20
Mo	99	66.0 h	400	2.50E+03	6E-07	2E-09	9.75E-08	2.14	642.00
Nb	98		1475	1.25E+05	2E-05	7E-08	3.5E-07	0.24	67.60
Nd	147	1.73 h	50	2.00E+03	4E-07	1E-09	1.23E-08	0.40	161.00
Pm	151		50	7.50E+03	1E-06	4E-09	1.23E-08	0.16	40.10
Pr	143		75	1.75E+03	3E-07	9E-10	1.85E-08	0.80	268.00
Pr	145	5.98 h	650	2.00E+04	3E-06	1E-08	1.6E-07	0.70	209.00
Rb	88	17.8 m	800	1.50E+05	3E-05	9E-08	1.95E-07	0.09	28.50
Rb	89		1225	2.50E+05	6E-05	2E-07	3E-07	0.07	19.70
Ru	105	4.44 h	200	2.50E+04	5E-06	2E-08	5E-08	0.13	32.10
Se	81	18.5 m	50	5.00E+05	9E-05	3E-07	1.23E-08	0.00	0.54
Sn	128	59.1 m	75	7.50E+04	1E-05	4E-08	1.85E-08	0.02	6.02
Sr	89		25	2.50E+02	6E-08	2E-10	6.25E-09	1.34	401.00
Sr	91	9.5 h	800	1.00E+04	1E-06	5E-09	1.95E-07	2.57	514.00
Sr	92	2.71 h	1325	1.75E+04	3E-06	9E-09	3.25E-07	1.42	473.00
Te	129	69.6 m	100	1.50E+05	3E-05	9E-08	2.45E-08	0.01	3.57
Te	131	25.0 m	625	1.25E+04	2E-06	1E-09	1.53E-07	1.00	2010.00
Te	132	78.2 h	250	5.00E+02	9E-08	9E-10	6.25E-08	8.92	892.00
Te	133	12.5 m	975	5.00E+04	9E-06	8E-08	2.4E-07	0.35	39.10
Te	134	41.8 m	1725	5.00E+04	1E-05	7E-08	4.25E-07	0.55	79.10
Y	91m	49.7 m	425	5.00E+05	7E-05	2E-07	1.05E-07	0.02	6.82
Sr	91	9.5 h	800	1.00E+04	1E-06	5E-09	1.95E-07	2.57	514.00
Sr	92	2.71 h	1325	1.75E+04	3E-06	9E-09	3.25E-07	1.42	473.00
Te	129	69.6 m	100	1.50E+05	3E-05	9E-08	2.45E-08	0.01	3.57
Te	131	25.0 m	625	1.25E+04	2E-06	1E-09	1.53E-07	1.00	2010.00
Te	132	78.2 h	250	5.00E+02	9E-08	9E-10	6.25E-08	8.92	892.00
Te	133	12.5 m	975	5.00E+04	9E-06	8E-08	2.4E-07	0.35	39.10
Te	134	41.8 m	1725	5.00E+04	1E-05	7E-08	4.25E-07	0.55	79.10
Y	91m	49.7 m	425	5.00E+05	7E-05	2E-07	1.05E-07	0.02	6.82
Y	92	3.54 h	850	2.00E+04	3E-06	1E-08	2.08E-07	0.91	273.00
Y	93	10.1 h	850	5.00E+03	1E-06	3E-09	2.08E-07	2.73	910.00
Zr	95		25	2.50E+02	5E-08	4E-10	6.25E-09	1.61	201.00
Zr	97	16.9 h	650	2.50E+03	5E-07	2E-09	1.6E-07	4.17	1040.00

Table 13.5 Maximum downwind 50-year committed dose equivalents to members of the public from selected radionuclides assumed to be released in the maximum hypothetical accident

Element	Nuclide	A(μCi)	Tissue at risk	ℜ (mrem/μCi)	D (mrem)
I	133	27980	Thyroid	179.8	16.808
I	131	3400	Thyroid	1080.4	12.272
I	135	39500	Thyroid	31.3	4.131
I	132	9310	Thyroid	6.4	0.200
Te	132	100	Thyroid	232.4	0.078
I	134	76890	Whole Body	0.1	0.034
Te	131	250	Thyroid	9.8	0.008
Te	134	690	Thyroid	2.1	0.005
Zr	97	260	Whole Body	4.3	0.004
Br	84	10150	Whole Body	0.1	0.003
Te	133	390	Thyroid	2.2	0.003
Ce	143	220	Whole Body	3.4	0.002
Y	93	340	Whole Body	2.2	0.002
Mo	99	160	Whole Body	4.0	0.002
Sr	91	320	Whole Body	1.7	0.002
Br	83	4840	Whole Body	0.1	0.001
Sr	92	530	Whole Body	0.8	0.001
La	141	460	Whole Body	0.6	0.001
Y	92	340	Whole Body	0.8	0.001
Pr	143	30	Whole Body	8.1	0.001
Zr	95	10	Whole Body	23.3	0.001
Ba	140	50	Whole Body	3.7	0.001
Ce	141	20	Whole Body	9.0	0.001
Pr	145	260	Whole Body	0.7	0.001
				Total	33.563

Conclusions

Fission product inventories in TRIGA® fuel elements were calculated with the ORIGEN code, using very conservative approximations. Then, potential radionuclide releases from worst-case fuel elements were computed, again using very conservative approximations. Even if it were assumed that releases took place immediately after reactor operation, and that radionuclides were immediately dispersed inside the reactor bay workplace, some radionuclide concentrations would be in excess of occupational derived air concentrations, but only for a matter of hours or days. Only for certain nuclides of iodine would the potential release be significantly in excess of the annual limit of intake. However, since evacuation of the reactor bay would occur reasonably within 5 minutes of an indication of the increased activity, there is no credible scenario for accidental inhalation or ingestion of the undiluted radioiodine that might be released from a damaged fuel element.

As far as potential consequences to the general public are concerned, only for the few radionuclides listed in Table 13.5 are maximum concentrations inside the reactor facility in excess of effluent concentrations listed in 10 CFR Part 20 and potential doses 0.001 mrem or greater outside the RRR. However, even in the extremely unlikely event that radionuclides released from a damaged fuel element were immediately released to the outside atmosphere, very conservative calculations reveal that radionuclides inhaled by persons downwind from the release would lead to organ doses or effective doses very far below regulatory limits.

13.2.2 Insertion of Excess Reactivity

13.2.2.1 Accident Initiating Events and Scenarios

Insertion of excess reactivity at the RRR would involve the rapid removal of one or more control rods or the insertion of an experiment with a high positive reactivity.

Rapid compensation of a reactivity insertion is the distinguishing design feature of the TRIGA® reactor. Characteristics of a slow (ramp) reactivity insertion are less severe than a rapid transient since temperature feedback will occur rapidly enough to limit the maximum power achieved during the transient. Analyses of plausible accident scenarios reveal no challenges to safety limits for the TRIGA®. The fuel-integrity safety limit, according to Simnad, [9] may be stated as follows:

> *Fuel-moderator temperature is the basic limit of TRIGA® reactor operation. This limit stems from the out-gassing of hydrogen from the ZrH_x and the subsequent stress produced in the fuel element clad material. The strength of the clad as a function of temperature can set the upper limit on the fuel temperature. A fuel temperature safety limit of 1150 °C for pulsing, stainless steel $U\text{-}ZrH_{1.65}$ ... fuel is used as a design value to preclude the loss of clad integrity when the clad temperature is below 500 °C. When clad temperatures can equal the fuel temperature, the fuel temperature limit is 950 °C.*

The RRR has no transient control rods, so the sudden removal of a control rod could only happen if a control rod motor was physically removed and the rod pulled manually. Fuel loading and core configuration limits excess reactivity to $3.00, and experiments with test GA reactors indicate that while fuel warping occurs when standard TRIGA® fuel is subjected to a pulse, no cladding damage is sustained and the prompt negative temperature coefficient (PNTC) quickly lowers temperatures to acceptable levels. GA-7882 [10] reports that for a pulse of $3.00, the temperature increase is no more than 250°C, reaching a maximum fuel temperature of less than 400°C. This is well below the safety limit on either aluminum-clad or stainless steel-clad fuel.

13.2.3 Loss of Coolant Accident

Although total loss of reactor pool water is considered to be an extremely improbable event, RRR has considered such a failure. Limiting design basis parameters and values are addressed by Simnad [9] as follows:

> *Fuel-moderator temperature is the basic limit of TRIGA® reactor operation. This limit stems from the out-gassing of hydrogen from the ZrH_x and the subsequent stress produced in the fuel element clad material. The strength of the clad as a function of temperature can set the upper limit on the fuel temperature. A fuel temperature safety limit of 1150°C for pulsing, stainless steel U-$ZrH_{1.65}$ ... fuel is used as a design value to preclude the loss of clad integrity when the clad temperature is below 500°C. When clad temperatures can equal the fuel temperature, the fuel temperature limit is 950°C. There is also a steady-state operational fuel temperature design limit of 750°C based on consideration of irradiation- and fission-product-induced fuel growth and deformation.*

The RRR original SAR from 1968 discussed this issue in-depth for a maximum power of 250 kW and aluminum-clad fuel. The calculations demonstrated that the maximum fuel temperature reached is 150°C under very conservative estimations, and dose rates from the core are summarized in Table 13.6.

Table 13.6 Radiation Dose Rates After Extended 250 kW Operation and Loss of All Shielding Water

Time from Shutdown	Direct Radiation (R/hr)	Scattered Radiation (R/hr)
10 seconds	2.7E4	15
1 day	1.4E3	0.8
1 week	9.3E2	0.5
1 month	4.5E2	0.2

The radiation levels from scattered radiation are low enough that preventative action could be taken to restore shielding to the reactor.

For running the RRR at 500 kW, stainless steel fuel is required. Several other TRIGA® reactors have calculated the maximum hypothetical fuel temperatures and radiation fields caused by a loss of coolant accident. An analysis was performed for a TRIGA® Mk II reactor running at 1,250 kW for both fuel temperature and scattered radiation. The fuel temperature analysis results are found in Table 13.7 while the scattered radiation is given in Table 13.8.

Table 13.7 Maximum Post-Accident Fuel and Cladding Temperatures

T_{air} (°K)	T_{clad} (°K)	T_{fuel} (°K)
561	561	567

Table 13.8 Reed TRIGA® Gamma-Ray Ambient (Deep) Dose Rates (R/h) at Selected Locations for Times Following Loss of Coolant After Operation for One Year at 1,250 kW Thermal Power

	Time post accident				
	0	1 h	24 h	30 d	180 d
On-site (elev.)					
22 ft. (center)	3.75E+06	1.38E+06	4.25E+05	1.03E+05	1.98E+04
12 ft. (boundary)	1.08E+03	4.00E+02	1.43E+02	3.25E+01	7.00E+00
0 ft. (boundary	9.75E+02	3.75E+02	1.23E+02	3.00E+01	6.25E+00
Off-site (radius from center of reactor bay)					
13 m	6.75E+02	2.75E+02	8.75E+01	2.15E+01	4.50E+00
15 m	5.00E+02	2.00E+02	6.75E+01	1.60E+01	3.25E+00
20 m	2.50E+02	1.00E+02	3.50E+01	8.75E+00	1.73E+00
30 m	8.25E+01	3.25E+01	1.10E+01	2.75E+00	5.50E-01
40 m	3.75E+01	1.40E+01	5.25E+00	1.18E+00	2.33E-01
50 m	1.93E+01	7.25E+00	2.43E+00	6.25E-01	1.23E-01
70 m	6.50E+00	2.75E+00	9.00E-01	2.18E-01	4.00E-02
100 m	2.35E+00	9.50E-01	2.75E-01	6.75E-02	1.33E-02

The RRR is installed in a below-ground tank, and as such the numbers most relevant are the 22 foot level (above the center of the pool) and the off-site levels, although the latter are much higher than similar expected off-site values for the RRR due to the aforementioned difference in location. In any case, Reed College has control of access to the structures surrounding the reactor to a distance of greater than 100 meters, and access can be controlled while mitigating action is taken if necessary.

Similarly, the SAR calculates dose rates for the direct and scattered radiation in their building after a loss of coolant accident after a full year of operation at 1 MW. The results are summarized in Table 13.9.

Table 13.9 Radiation Dose Rates After Extended 1 MW Operation and Loss of All Shielding Water

Time from Shutdown	Direct Radiation (rem/hr)	Scattered Radiation (rem/hr)
10 sec	1.00E4	0.327
1 hour	1.52E3	0.050
1 day	1.19E3	0.039
1 week	6.39E2	0.021
1 month	3.51E2	0.011

Similar reactors running at higher maximum powers have calculated maximum dose rates after a severe, sudden loss of coolant accident that are still low enough to allow preventative measures to be taken to protect the public against exposure. As any of these loss of coolant accident calculations are designed with the utmost conservatism in their base assumptions, it is fair to conclude that the RRR, running at under half the power of these calculated values, will not pose a significant threat to the public welfare under even severe accident conditions.

13.2.3.1 Loss of Coolant Flow

As the RRR uses natural rather than forced convection cooling, a loss of coolant flow accident is not analyzed in this document.

13.2.4 Mishandling or Malfunction of Fuel

13.2.4.1 Accident Initiating Events and Scenarios

Events which could cause accidents at the RRR in this category 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 the fuel with subsequent cladding failure during steady-state or pulsing operations.

13.2.4.2 Accident Analysis and Determination of Consequences

All three scenarios mentioned in Section 13.2.5.1 result in a single fuel element failure in water. In the unlikely event that this failure occurred in air, this is the MHA analyzed in Section 13.2.1.

At various points in the lifetime of the RRR, fuel elements are moved to new positions or removed from the core. Fuel elements are moved only during periods when the reactor is subcritical.

Assumptions for this accident are almost exactly the same as those used for the MHA, except for one thing: the presence of the pool water contains most of the halogens and, thereby, reduces the halogen dose contribution.

The results of this accident show that the general public are well below the annual limits in 10 CFR Part 20, with the maximum dose being less than 33 mrem TEDE at 250 feet (76 m) (the site boundary). The occupational radiation doses to workers in the reactor bay are also well below the occupational annual limits in 10 CFR Part 20, with the maximum dose being less than 1 mrem TEDE for a 5-minute exposure. Five minutes is very ample time for workers to evacuate the reactor bay if such an accident were to occur.

13.2.5 Experiment Malfunction

13.2.5.1 Accident Initiating Events and Scenarios

Improperly controlled experiments involving the RRR 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 the creation of unplanned pressures in irradiated materials. These materials could subsequently vent into the irradiation facilities or into the reactor bay causing damage from the pressure release or an uncontrolled release of radioactivity. Other mechanisms for damage, such as large reactivity changes are also possible.

13.2.5.2 Accident Analysis and Determination of Consequences

There are two main sets of procedural and regulatory requirements that relate to experiment review and approval. These are the RRR Procedures and the RRR Technical Specifications. These requirements are focused on ensuring that experiments will not fail, and 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, the RRR Procedures contain detailed procedures for the safety review and approval of all reactor experiments.

Safety related reviews of proposed experiments require the performance of specific safety analyses of proposed activities to assess such things as generation of radionuclides and fission products (e.g., 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, this process is expected to be an effective measure in assuring experiment safety at the RRR.

In the RRR Technical Specifications, a limit of \$1.00 has been placed on the reactivity worth of any experiment. The RRR Procedures require that the reactor be shut down before any experiments are moved. The Technical Specifications require that the RRR shutdown margin be at least \$0.50 with the most reactive rod withdrawn. The safety rod is the most reactive rod, with a worth of \$4.00. Therefore with all the rods inserted, the reactor is shutdown by at least \$4.50. If the experiment were removed, the reactor would still be shutdown by at least \$3.50.

Limiting the generation of certain fission products in fueled experiments also helps to assure that occupational radiation doses as well as doses to the general public, due to experiment failure with subsequent fission product release, will be within the limits prescribed in 10 CFR Part 20. A limit of 5 millicuries of iodine-131 to iodine-135 is specified in the RRR Technical Specifications. This amount of iodine isotopes is very small compared to the approximately 4,000 curies 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 Part 20 limits. Therefore, limiting experiments to 5 millicuries of radioiodine will result in projected doses will within the 10 CFR Part 20 limits.

Experiments involving explosives are not allowed in the RRR.

13.2.6 Loss of Normal Electrical Power

Loss of electrical power to the RRR could occur due to many events and scenarios that routinely affect commercial power.

13.2.6.1 Accident Initiating Events and Scenarios

Since the RRR does not require emergency backup systems 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 RRR that mainly provides conditioned power to the instrumentation. The system will provide emergency power immediately after the loss of regular electrical power and will continue to supply power for a period of several hours. Battery-powered emergency lights are also located throughout the facility to allow for inspection of the reactor and for an orderly evacuation of the facility.

Loss of normal electrical power to the RRR facility during reactor operations will initiate a reactor scram. Loss of power is addressed in the RRR Procedures, which require that, upon loss of normal power, the operator on duty should verify the reactor is shutdown. This can be done without any electrically powered indications. The backup power supply would allow enhanced monitoring.

13.2.7 External Events

13.2.7.1 Accident Events and Scenarios

Hurricanes, tornadoes, and floods are virtually nonexistent in the area around the RRR. Therefore, these events are not considered to be viable causes of accidents for the reactor facility. In addition, seismic activity in the Portland area is relatively low compared to other areas in the Pacific Northwest.

13.2.7.2 Accident Analysis and Determinations of Consequences.

There are no accidents in this category that would have more on-site or off-site consequences than the MHA analyzed in Section 13.2.1, and, therefore, no additional specific accidents are analyzed in this section.

13.2.8 Mishandling or Malfunction of Equipment

13.2.8.1 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, assure safe reactor response, including in some cases reactor shutdown.

Malfunction of confinement or containment systems would have the greatest impact during the MHA, if 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 RRR. Furthermore, no safety considerations at the RRR depend on confinement or containment systems.

Rapid leaks of the coolant have been addressed in Section 13.2.3. Although no damage to the reactor occurs as a result of these leaks, the details of the previous analyses provide a more comprehensive explanation.

13.2.8.2 Accident Analysis and Determination of Consequences

Since there were no credible initiating events identified, no accident analysis was performed for this section and no consequences were identified.

13.3 References

1) NUREG/CR-2387 (PNL-4028), Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors, *Report Pacific Northwest Laboratory, Richland Washington, 1982., Hawley, S.C., and R.L. Kathren.*

2) *GA-4314, General Atomics Company, M. T. Simnad, " The U-ZrH$_x$ Alloy: Its Properties and use in TRIGA Fuel." (1974), February 1980.*

3) "Fuel Elements for Pulsed TRIGA Research Reactors," *Nuclear Technology 28, 31-56 (1976), Simnad, M.T., F.C. Faushee, and G.B. West*

4) NUREG-1537, Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Format and Content, *Report NUREG-1537 Part 1, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, 1996.*

5) NUREG-1390, Safety Evaluation Report Relating to the Renewal of the Operating License for the TRIGA Training and Research Reactor at the University of Arizona, *Report NUREG-1390, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, 1990.*

6) *Federal Guidance Report No. 11, Report EPA-5201/1-88-020, U.S. Environmental Protection Agency,* "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," *Washington, DC, (1988), Eckerman, K.F., A.B. Wolbarst, and A.C.B. Richardson.*

7) CCC-371, "ORIGEN 2.1 Isotope Generation and Depletion Code: Matrix Exponential Method," *Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1991.*

8) "Radiological Assessment," *Prentice Hall, Englewood Cliffs, N.J., 1993, Faw, R.E., and J.K. Shultis.*

9) "The U-Zr-Hx Alloy: Its Properties and Use in TRIGA Fuel," *Report E-117-833, Simnad, M.T, General Atomics Corp., 1980.*

10) "Kinetic Behavior of TRIGA Reactors," Report GA-7882, *West, G.B., W.L. Whittemore, J.R. Shoptaugh, Jr., J.B. Dee, and C.O. Coffer, , General Atomics Corp., 1967.*

13.4 Appendices

A ORIGEN2.1 input file for uranium-235 fission at 1 W thermal power for 40 years.

B ORIGEN2.1 input file for uranium-235 fission at 1 W thermal power 8 hours per day for 5 days.

C ORIGEN2.1 output file extracts for uranium-235 fission at 1 W thermal power for 40 years.

D ORIGEN2.1 output file extracts for uranium-235 fission at 1 W thermal power 8 hours per day for 5 Days.

E Maximum activity available for release from a single TRIGA® fuel element as a function of time after shutdown for a uranium-235-fueled thermal reactor operating at 3.71 kW thermal power for 40 years, based on one element of [redacted] at 115.9 W.

F Maximum activity available for release from a single TRIGA® fuel element as a function of time after shutdown for a uranium-235-fueled thermal reactor operating at 500 kW thermal power for 8 hours per day for 5 days, based on one element of [redacted] at 12.05 kW.

CHAPTER 13 APPENDIX A

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 ORIGEN Input file for 1 tonne U-235 at 1 watt for 40 years
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-1
-1
 RDA ORIGEN2, VERSION 2.1 (8-1-91) TRIGA REACTOR REFERENCE PROBLEM
 RDA UPDATED BY: Richard E. Faw, Kansas State University
 BAS ONE TONNE OF U-235
 RDA Continuous operation for 40 years at 1 watt
 RDA WARNING: VECTORS ARE OFTEN CHANGED WITH RESPECT TO THEIR CONTENT.
 RDA THESE CHANGES WILL BE NOTED ON RDA CARDS.
 CUT -1
 RDA LIBRARY PRINT (1=PRINT,0=DON'T PRINT)
 LIP 0 0 0
 RDA DECAY LIBRARY CHOICES (0=PRINT; 1 2 3 DECAY LIBRARIES; 601 ...
 RDA CROSS SECTIONS; ETC, SEE P. 47)
 LIB 0 1 2 3 201 202 203 9 3 0 1 38
 PHO 101 102 103 10 <<< PHOTON LIBRARIES, P. 47
 TIT INITIAL COMPOSITIONS OF UNIT AMOUNTS OF FUEL AND STRUCT MAT'LS
 RDA READ FUEL COMPOSITION INCLUDING IMPURITIES (1 G)
 INP -1 1 -1 -1 1 1
 TIT IRRADIATION OF ONE TONNE U-235
 MOV -1 1 0 1.0
 HED 1 CHARGE
 BUP
 IRP  5 .000001 1 2 5 2 1 W/Tonne FOR 5 YEARS
 IRP 10 .000001 2 1 5 0 1 W/Tonne FOR 5 YEARS
 IRP 15 .000001 1 2 5 0 1 W/Tonne FOR 5 YEARS
 IRP 20 .000001 2 1 5 0 1 W/Tonne FOR 5 YEARS
 IRP 25 .000001 1 2 5 0 1 W/Tonne FOR 5 YEARS
 IRP 30 .000001 2 1 5 0 1 W/Tonne FOR 5 YEARS
 IRP 35 .000001 1 2 5 0 1 W/Tonne FOR 5 YEARS
 IRP 40 .000001 2 1 5 0 1 W/Tonne FOR 5 YEARS
 BUP
 OPTL 24*8 ACTIVATION PRODUCT OUTPUT OPTS P. 56
 OPTA 24*8 ACTINIDE OUTPUT OPTIONS P. 59
 OPTF 6*8 5 17*8 FISSION PRODUCT OUTPUT OPTIONS P. 59
 RDA DECAY TO 28 DAYS
 DEC  1 1 2 4 2
 DEC  2 2 3 4 0
 DEC  3 3 4 4 0
 DEC  7 4 5 4 0
 DEC 14 5 6 4 0
 DEC 28 6 7 4 0
 OUT -7 1 -1 0
 OUT  7 1 -1 0
 END
 2 922340 0.0 922350 1.E06 922380 0. 0 0.0 PURE U-235
 0
```

CHAPTER 13 APPENDIX B

```
 ORIGEN Input File for 1 tonne U-235 at 1 watt
 8 hours per day for 5 days
-1
-1
-1
 RDA ORIGEN2, VERSION 2.1 (8-1-91) TRIGA REACTOR REFERENCE PROBLEM
 RDA UPDATED BY: Richard E. Faw, Kansas State University
 BAS One tonne U-235
 RDA -1 = Fuel composition
 CUT -1
 RDA LIBRARY PRINT (1=PRINT,0=DON'T PRINT)
 LIP 0 0 0
 RDA DECAY LIBRARY CHOICES (0=PRINT; 1 2 3 DECAY LIBRARIES; 601 ...
 RDA CROSS SECTIONS; ETC, SEE P. 47)
 LIB 0 1 2 3 201 202 203 9 3 0 1 38
 PHO 101 102 103 10 <<< PHOTON LIBRARIES, P. 47
 TIT INITIAL COMPOSITIONS OF UNIT AMOUNTS OF FUEL AND STRUCT MAT'LS
 RDA READ FUEL COMPOSITION INCLUDING IMPURITIES
 INP -1 1 -1 -1 1 1
 TIT One tonne U-235 8 h/d for 5 days at 1 kWt
 MOV -1 1 0 1.0
 HED 1 CHARGE
 BUP
 IRP 8.0 0.001 1 2 3 2 OPERATE FOR 8 HR AT 1 kW
 DEC 24.0 2 3 3 0 COOL FOR 16 HOURS
 IRP 32.0 0.001 3 4 3 0 OPERATE FOR 8 HR
 DEC 48.0 4 5 3 0 COOL FOR 16 HOURS
 IRP 56.0 0.001 5 6 3 0 OPERATE FOR 8 HR
 DEC 72.0 6 7 3 0 COOL FOR 16 HOURS
 IRP 80.0 0.001 7 8 3 0 OPERATE FOR 8 HR
 DEC 96.0 8 9 3 0 COOL FOR 16 HOURS
 IRP 104.0 0.001 9 10 3 0 OPERATE FOR 8 HR
 OPTL 24*8 ACTIVATION PRODUCT OUTPUT OPTS P. 56
 OPTA 24*8 ACTINIDE OUTPUT OPTIONS P. 59
 OPTF 6*8 5 17*8 FISSION PRODUCT OUTPUT OPTIONS P. 59
 RDA MOVE COMPOSITION VECTOR FROM 10 TO 1
 MOV 10 1 0 1.0
 RDA DECAY TO 0.1 UNITS (2=MINUTES) FROM COMP VEC 1 TO VEC 3
 DEC 1 1 2 4 2
 DEC 2 2 3 4 0
 DEC 3 3 4 4 0
 DEC 7 4 5 4 0
 DEC 14 5 6 4 0
 DEC 28 6 7 4 0
 OUT -7 1 -1 0
 OUT 7 1 -1 0
 END
 2 922340 0.0 922350 1.E6 922380 0.00 0 0.0 1 g U-235
 0
```

CHAPTER 13 APPENDIX C

ORIGEN Output File Extracts for 1 tonne U-235 at 1 watt for 40 Years
NUCLIDE TABLE: RADIOACTIVITY, CURIES

Time post discharge

	0	1.0 D	2.0 D	3.0 D	7.0 D	14.0 D	28.0 D
AG 111	1.65E-04	1.51E-04	1.37E-04	1.25E-04	8.62E-05	4.50E-05	1.22E-05
BA 140	5.23E-02	4.96E-02	4.70E-02	4.45E-02	3.58E-02	2.45E-02	1.15E-02
BA 137	2.93E-02	2.93E-02	2.93E-02	2.93E-02	2.93E-02	2.93E-02	2.92E-02
CE 141	4.93E-02	4.85E-02	4.75E-02	4.65E-02	4.27E-02	3.68E-02	2.73E-02
CE 143	4.98E-02	3.03E-02	1.83E-02	1.11E-02	1.47E-03	4.32E-05	3.72E-08
CE 144	4.56E-02	4.55E-02	4.54E-02	4.53E-02	4.48E-02	4.41E-02	4.26E-02
CS 137	3.10E-02	3.10E-02	3.09E-02	3.09E-02	3.09E-02	3.09E-02	3.09E-02
EU 155	2.75E-04	2.75E-04	2.75E-04	2.75E-04	2.75E-04	2.74E-04	2.72E-04
EU 156	1.13E-04	1.10E-04	1.06E-04	1.01E-04	8.43E-05	6.13E-05	3.23E-05
I 131	2.37E-02	2.20E-02	2.03E-02	1.87E-02	1.33E-02	7.28E-03	2.18E-03
I 132	3.56E-02	2.95E-02	2.39E-02	1.93E-02	8.23E-03	1.86E-03	9.45E-05
I 133	5.68E-02	2.62E-02	1.18E-02	5.29E-03	2.16E-04	8.00E-07	1.10E-11
I 135	5.31E-02	4.29E-03	3.46E-04	2.80E-05	1.19E-09	2.66E-17	1.34E-32
KR 85	2.10E-03	2.10E-03	2.10E-03	2.10E-03	2.10E-03	2.10E-03	2.09E-03
KR 85M	1.07E-02	2.64E-04	6.44E-06	1.57E-07	5.58E-14	2.88E-25	0.00E+00
LA 140	5.24E-02	5.19E-02	5.06E-02	4.89E-02	4.08E-02	2.82E-02	1.32E-02
LA 141	4.93E-02	7.77E-04	1.13E-05	1.64E-07	7.27E-15	9.89E-28	0.00E+00
MO 99	5.06E-02	3.94E-02	3.06E-02	2.38E-02	8.68E-03	1.49E-03	4.36E-05
NB 95	5.38E-02	5.38E-02	5.38E-02	5.37E-02	5.35E-02	5.28E-02	5.04E-02
NB 97	4.93E-02	1.85E-02	6.90E-03	2.58E-03	5.03E-05	5.14E-08	5.70E-14
NB 95M	3.78E-04	3.77E-04	3.76E-04	3.74E-04	3.64E-04	3.41E-04	2.95E-04
NB 97M	4.66E-02	1.74E-02	6.51E-03	2.43E-03	4.74E-05	4.84E-08	5.01E-14
ND 147	1.90E-02	1.79E-02	1.68E-02	1.58E-02	1.23E-02	7.91E-03	3.29E-03
PM 147	1.91E-02	1.91E-02	1.91E-02	1.91E-02	1.91E-02	1.90E-02	1.89E-02
PM 149	9.11E-03	6.89E-03	5.04E-03	3.68E-03	1.05E-03	1.17E-04	1.46E-06
PM 151	3.52E-03	1.97E-03	1.10E-03	6.11E-04	5.86E-05	9.70E-07	2.65E-10
PR 143	4.98E-02	4.93E-02	4.80E-02	4.64E-02	3.86E-02	2.71E-02	1.33E-02
PR 144	4.56E-02	4.55E-02	4.54E-02	4.53E-02	4.48E-02	4.41E-02	4.26E-02
PR 145	3.29E-02	2.06E-03	1.27E-04	7.89E-06	1.16E-10	4.06E-19	5.14E-36
PR 144	5.48E-04	5.46E-04	5.45E-04	5.43E-04	5.38E-04	5.29E-04	5.11E-04
RH 105	8.53E-03	6.09E-03	3.83E-03	2.39E-03	3.64E-04	1.35E-05	1.86E-08
RH 106	3.27E-03	3.26E-03	3.26E-03	3.25E-03	3.23E-03	3.18E-03	3.10E-03
RH 103	2.37E-02	2.33E-02	2.29E-02	2.25E-02	2.09E-02	1.85E-02	1.45E-02
RU 103	2.63E-02	2.58E-02	2.54E-02	2.49E-02	2.32E-02	2.05E-02	1.60E-02
RU 105	8.53E-03	2.08E-04	4.90E-06	1.16E-07	3.57E-14	1.44E-25	0.00E+00
RU 106	3.27E-03	3.26E-03	3.26E-03	3.25E-03	3.23E-03	3.18E-03	3.10E-03
SB 125	2.49E-04	2.49E-04	2.49E-04	2.49E-04	2.48E-04	2.47E-04	2.45E-04
SB 127	1.10E-03	9.28E-04	7.75E-04	6.47E-04	3.15E-04	8.93E-05	7.18E-06
SB 129	5.33E-03	1.15E-04	2.45E-06	5.20E-08	1.06E-14	2.08E-26	0.00E+00
SM 151	9.34E-04	9.34E-04	9.34E-04	9.34E-04	9.34E-04	9.33E-04	9.33E-04
SM 153	1.36E-03	9.55E-04	6.69E-04	4.68E-04	1.13E-04	9.30E-06	6.34E-08
SN 125	1.13E-04	1.05E-04	9.75E-05	9.07E-05	6.80E-05	4.11E-05	1.50E-05
SR 89	4.05E-02	3.99E-02	3.94E-02	3.89E-02	3.68E-02	3.34E-02	2.76E-02
SR 90	2.97E-02	2.97E-02	2.97E-02	2.97E-02	2.97E-02	2.97E-02	2.97E-02
SR 91	4.94E-02	8.59E-03	1.49E-03	2.59E-04	2.35E-07	1.12E-12	2.51E-23
SR 92	5.04E-02	1.09E-04	2.35E-07	5.07E-10	1.10E-20	0.00E+00	0.00E+00
TC 99M	4.43E-02	3.76E-02	2.95E-02	2.29E-02	8.36E-03	1.43E-03	4.20E-05

TE 127	1.09E-03	1.02E-03	8.88E-04	7.68E-04	4.49E-04	2.27E-04	1.37E-04
TE 129	5.25E-03	6.43E-04	4.99E-04	4.86E-04	4.48E-04	3.88E-04	2.90E-04
TE 131	2.13E-02	3.95E-04	2.27E-04	1.30E-04	1.42E-05	2.92E-07	1.24E-10
TE 132	3.54E-02	2.86E-02	2.32E-02	1.87E-02	7.99E-03	1.80E-03	9.17E-05
TE 127	1.53E-04	1.53E-04	1.52E-04	1.52E-04	1.50E-04	1.44E-04	1.33E-04
TE 129	7.91E-04	7.78E-04	7.63E-04	7.47E-04	6.88E-04	5.95E-04	4.46E-04
TE 131	3.04E-03	1.75E-03	1.01E-03	5.79E-04	6.30E-05	1.30E-06	5.52E-10
XE 133	5.68E-02	5.47E-02	5.03E-02	4.52E-02	2.74E-02	1.09E-02	1.72E-03
XE 135	5.52E-02	2.02E-02	4.15E-03	7.39E-04	5.35E-07	1.47E-12	1.09E-23
XE 131	2.63E-04	2.62E-04	2.61E-04	2.58E-04	2.41E-04	1.97E-04	1.11E-04
XE 133	1.65E-03	1.51E-03	1.24E-03	9.65E-04	2.96E-04	3.27E-05	3.90E-07
XE 135	9.52E-03	6.87E-04	5.55E-05	4.48E-06	1.91E-10	4.27E-18	2.14E-33
Y 90	2.97E-02	2.97E-02	2.97E-02	2.97E-02	2.97E-02	2.97E-02	2.97E-02
Y 91	4.94E-02	4.91E-02	4.86E-02	4.80E-02	4.58E-02	4.21E-02	3.57E-02
Y 92	5.05E-02	1.60E-03	1.70E-05	1.60E-07	1.10E-15	5.65E-30	0.00E+00
Y 93	5.44E-02	1.06E-02	2.04E-03	3.93E-04	5.41E-07	5.32E-12	5.15E-22
Y 91M	2.86E-02	5.46E-03	9.47E-04	1.64E-04	1.49E-07	7.09E-13	1.60E-23
ZR 95	5.38E-02	5.32E-02	5.26E-02	5.21E-02	4.99E-02	4.62E-02	3.97E-02
ZR 97	4.92E-02	1.84E-02	6.87E-03	2.57E-03	5.00E-05	5.11E-08	5.29E-14

CHAPTER 13 APPENDIX D

ORIGEN Output File Extracts for 1 tonne U-235 at 1 W for 8 h/d, 5 Days
NUCLIDE TABLE: RADIOACTIVITY, CURIES

Time post discharge

	0	1.0 D	2.0 D	3.0 D	7.0 D	14.0 D	28.0 D
AS 78	1.24E-01	2.17E-05	5.66E-10	1.16E-14	1.11E-33	0.00E+00	0.00E+00
BA 139	5.33E+01	3.47E-04	1.99E-09	1.14E-14	1.28E-35	0.00E+00	0.00E+00
BA 140	4.22E+00	4.00E+00	3.79E+00	3.59E+00	2.89E+00	1.98E+00	9.27E-01
BA 141	4.91E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
BA 142	4.88E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
BR 83	4.02E+00	4.13E-03	3.92E-06	3.72E-09	3.01E-21	0.00E+00	0.00E+00
BR 84	8.43E+00	2.19E-13	5.13E-27	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CE 141	1.57E+00	1.74E+00	1.71E+00	1.67E+00	1.54E+00	1.32E+00	9.81E-01
CE 143	1.84E+01	1.13E+01	6.83E+00	4.13E+00	5.49E-01	1.61E-02	1.39E-05
CE 144	1.85E-01	1.84E-01	1.84E-01	1.83E-01	1.82E-01	1.79E-01	1.73E-01
CS 138	5.68E+01	3.40E-12	1.17E-25	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I 131	2.82E+00	2.74E+00	2.55E+00	2.37E+00	1.70E+00	9.33E-01	2.79E-01
I 132	7.73E+00	6.95E+00	5.62E+00	4.54E+00	1.94E+00	4.37E-01	2.22E-02
I 133	2.32E+01	1.11E+01	5.00E+00	2.25E+00	9.17E-02	3.40E-04	4.66E-09
I 134	6.38E+01	1.62E-06	9.36E-15	5.37E-23	0.00E+00	0.00E+00	0.00E+00
I 135	3.28E+01	2.65E+00	2.14E-01	1.73E-02	7.35E-07	1.65E-14	8.26E-30
KR 87	2.14E+01	4.51E-05	9.39E-11	1.96E-16	0.00E+00	0.00E+00	0.00E+00
KR 88	2.62E+01	7.49E-02	2.14E-04	6.09E-07	4.03E-17	6.19E-35	0.00E+00
KR 83M	3.30E+00	1.59E-02	1.65E-05	1.58E-08	1.29E-20	0.00E+00	0.00E+00
KR 85M	7.76E+00	1.93E-01	4.70E-03	1.15E-04	4.07E-11	2.10E-22	0.00E+00
LA 140	2.25E+00	2.88E+00	3.22E+00	3.38E+00	3.19E+00	2.27E+00	1.07E+00
LA 141	3.79E+01	6.11E-01	8.86E-03	1.29E-04	5.72E-12	7.77E-25	0.00E+00
LA 142	4.83E+01	1.15E-03	2.43E-08	5.12E-13	1.01E-31	0.00E+00	0.00E+00
LA 143	4.96E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MO 99	1.31E+01	1.02E+01	7.93E+00	6.17E+00	2.25E+00	3.85E-01	1.13E-02
MO 101	4.23E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NB 97	1.98E+01	8.19E+00	3.06E+00	1.14E+00	2.23E-02	2.28E-05	2.53E-11
NB 98	4.86E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ND 147	1.79E+00	1.69E+00	1.59E+00	1.49E+00	1.16E+00	7.50E-01	3.12E-01
ND 149	8.74E+00	5.96E-04	3.98E-08	2.65E-12	5.23E-29	0.00E+00	0.00E+00
ND 151	3.50E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
PD 109	1.19E-01	3.50E-02	1.02E-02	2.95E-03	2.11E-05	3.69E-09	1.13E-16
PM 151	1.36E+00	7.72E-01	4.30E-01	2.39E-01	2.30E-02	3.80E-04	1.04E-07
PR 143	2.31E+00	2.92E+00	3.22E+00	3.33E+00	3.03E+00	2.16E+00	1.06E+00
PR 144	1.86E-01	1.84E-01	1.84E-01	1.83E-01	1.82E-01	1.79E-01	1.73E-01
PR 145	2.12E+01	1.33E+00	8.25E-02	5.11E-03	7.52E-08	2.63E-16	3.22E-33
PR 147	1.90E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RB 88	2.65E+01	8.37E-02	2.39E-04	6.80E-07	4.50E-17	7.40E-35	0.00E+00
RH 105	2.57E+00	2.17E+00	1.37E+00	8.55E-01	1.30E-01	4.84E-03	6.68E-06
RH 107	1.37E+00	1.84E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RU 103	7.45E-01	7.33E-01	7.20E-01	7.08E-01	6.59E-01	5.83E-01	4.55E-01
RU 105	6.23E+00	1.54E-01	3.62E-03	8.54E-05	2.64E-11	1.07E-22	0.00E+00
SB 127	2.19E-01	1.96E-01	1.63E-01	1.36E-01	6.64E-02	1.88E-02	1.51E-03
SE 81	1.76E+00	2.49E-09	6.77E-17	1.84E-24	0.00E+00	0.00E+00	0.00E+00
SE 83	1.64E+00	8.98E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SM 153	4.27E-01	3.01E-01	2.11E-01	1.48E-01	3.55E-02	2.93E-03	2.00E-05
SM 155	2.76E-01	8.53E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

SN 127	6.52E-01	2.37E-04	8.59E-08	3.12E-11	5.40E-25	0.00E+00	0.00E+00
SN 128	2.78E+00	1.25E-07	5.62E-15	2.53E-22	0.00E+00	0.00E+00	0.00E+00
SR 89	9.30E-01	9.27E-01	9.15E-01	9.02E-01	8.54E-01	7.76E-01	6.40E-01
SR 91	2.64E+01	4.60E+00	7.99E-01	1.39E-01	1.26E-04	5.98E-10	1.35E-20
SR 92	4.40E+01	9.50E-02	2.05E-04	4.42E-07	9.59E-18	1.62E-36	0.00E+00
TC 101	4.23E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TC 104	1.53E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TE 127	1.61E-01	1.77E-01	1.55E-01	1.31E-01	6.49E-02	1.94E-02	2.81E-03
TE 129	3.11E+00	1.18E-01	1.87E-02	1.62E-02	1.49E-02	1.29E-02	9.65E-03
TE 131	2.09E+01	1.51E-01	8.68E-02	4.99E-02	5.43E-03	1.12E-04	4.76E-08
TE 132	8.32E+00	6.74E+00	5.45E+00	4.41E+00	1.88E+00	4.24E-01	2.16E-02
TE 133	3.20E+01	6.42E-08	9.61E-16	1.44E-23	0.00E+00	0.00E+00	0.00E+00
TE 134	5.69E+01	2.42E-09	1.03E-19	4.40E-30	0.00E+00	0.00E+00	0.00E+00
XE 133	6.76E+00	8.00E+00	7.97E+00	7.43E+00	4.68E+00	1.88E+00	2.97E-01
XE 135	1.96E+01	1.01E+01	2.19E+00	3.97E-01	2.91E-04	7.99E-10	5.94E-21
XE 138	5.26E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 91	7.71E-01	9.16E-01	9.32E-01	9.25E-01	8.84E-01	8.13E-01	6.89E-01
Y 92	2.81E+01	1.25E+00	1.35E-02	1.28E-04	8.81E-13	4.51E-27	0.00E+00
Y 93	2.86E+01	5.64E+00	1.09E+00	2.09E-01	2.88E-04	2.83E-09	2.74E-19
Y 94	5.26E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 95	5.35E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 91M	1.41E+01	2.93E+00	5.08E-01	8.81E-02	8.00E-05	3.80E-10	8.56E-21
ZR 95	9.72E-01	9.68E-01	9.58E-01	9.47E-01	9.07E-01	8.41E-01	7.23E-01
ZR 97	2.18E+01	8.15E+00	3.05E+00	1.14E+00	2.22E-02	2.27E-05	2.34E-11

CHAPTER 13 APPENDIX E

Maximum Activity Available for Release
One TRIGA Element at 86.42 W for 40 Years
(Release fractions: 1E-04 for halogens and noble gases, 1E-06 for particulates)
Potential Activity Release (μCi) Inhalation Initial Bay

	Time post discharge (days)							ALI		DAC Concentration
	0	1	2	3	7	14	28	μCi	μCi/cm³	μCi/cm³
AG 111	0.0	0.0	0.0	0.0	0.0	0.0	0.0	9.0E+02	4.0E-07	4.7E-12
BA 140	6.0	5.8	5.5	5.1	4.2	2.8	1.3	1.0E+03	6.0E-07	1.5E-09
BA 137	3.4	3.4	3.4	3.4	3.4	3.4	3.4	na	na	8.3E-10
CE 141	5.8	5.6	5.5	5.4	5.0	4.3	3.2	8.0E+02	2.0E-07	1.3E-09
CE 143	5.8	3.5	2.1	1.3	0.1	0.0	0.0	2.0E+03	7.0E-07	1.5E-09
CE 144	5.2	5.2	5.2	5.2	5.2	5.1	5.0	1.0E+01	6.0E-09	1.3E-09
CS 137	3.6	3.6	3.6	3.6	3.6	3.6	3.6	2.0E+02	6.0E-08	8.8E-10
EU 155	0.0	0.0	0.0	0.0	0.0	0.0	0.0	9.0E+01	4.0E-08	7.8E-12
EU 156	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0E+02	2.0E-07	3.2E-12
I 131	274.4	254.9	235.1	216.5	154.1	84.4	25.2	5.0E+01	2.0E-08	6.7E-08
I 132	412.1	342.4	276.8	223.7	95.5	21.5	1.1	8.0E+03	3.0E-06	1.0E-07
I 133	657.8	303.8	136.5	61.3	2.5	0.0	0.0	3.0E+02	1.0E-07	1.6E-07
I 135	615.0	49.6	4.0	0.3	0.0	0.0	0.0	2.0E+03	7.0E-07	1.5E-07
KR 85	24.3	24.3	24.3	24.3	24.3	24.3	24.3	1.0E-04	4.5E-09	
KR 85M	123.6	3.1	0.1	0.0	0.0	0.0	0.0	2.0E-05	2.3E-08	
LA 140	6.0	6.0	5.9	5.6	4.7	3.2	1.5	1.0E+03	5.0E-07	1.5E-09
LA 141	5.8	0.1	0.0	0.0	0.0	0.0	0.0	9.0E+03	4.0E-06	1.3E-09
MO 99	5.9	4.6	3.5	2.8	0.9	0.1	0.0	1.0E+03	6.0E-07	1.5E-09
NB 95	6.2	6.2	6.2	6.2	6.2	6.2	6.2	1.0E+03	5.0E-07	1.5E-09
NB 97	508	2.1	0.8	0.3	0.0	0.0	0.0	7.0E+04	3.0E-05	1.3E-09
NB 95M	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.0E+03	9.0E-07	1.0E-11
NB 97M	5.4	2.0	0.8	0.3	0.0	0.0	0.0	na	na	1.3E-09
ND 147	2.1	2.0	1.9	1.9	1.5	0.9	0.4	8.0E+02	4.0E-07	5.4E-10
PM 147	2.1	2.1	2.1	2.1	2.1	2.1	2.1	1.0E+02	5.0E-08	5.4E-10
PM 149	1.1	0.8	0.5	0.4	0.4	0.0	0.0	2.0E+03	8.0E-07	2.6E-10
PM 151	0.4	0.3	0.1	0.1	0.0	0.0	0.0	3.0E+03	1.0E-06	1.0E-10
PR 143	5.8	5.8	5.6	5.4	4.4	3.1	1.5	7.0E+02	3.0E-07	1.5E-09
PR 144	5.2	5.2	5.2	5.2	5.2	5.1	5.0	1.0E+05	5.0E-05	1.3E-09
PR 145	3.8	0.3	0.0	0.0	0.0	0.0	0.0	8.0E+03	3.0E-06	9.4E-10
PR 144	0.0	0.0	0.0	0.0	0.0	0.0	0.0	na	na	1.6E-11
RH 105	0.9	0.7	0.4	0.3	0.0	0.0	0.0	6.0E+03	2.0E-06	2.4E-10
RH 106	0.4	0.4	0.4	0.4	0.4	0.4	0.4	na	na	9.2E-11
RH 103	2.7	2.7	2.7	2.5	2.4	2.1	1.6	1.0E+06	5.0E-04	6.7E-10
RU 103	3.1	3.0	3.0	3.0	2.7	2.4	1.9	6.0E+02	3.0E-07	7.5E-10
RU 105	0.9	0.0	0.0	0.0	0.0	0.0	0.0	1.0E+04	5.0E-06	2.4E-10
RU 106	0.4	0.4	0.4	0.4	0.4	0.4	0.4	1.0E+01	5.0E-09	9.2E-11
SB 125	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0E+02	2.0E-07	7.1E-12
SB 127	0.1	0.1	0.1	0.1	0.0	0.0	0.0	9.0E+02	4.0E-07	3.1E-11
SB 129	0.7	0.0	0.0	0.0	0.0	0.0	0.0	9.0E+03	4.0E-06	1.5E-10
SM 151	0.1	0.1	0.1	0.1	0.1	0.1	0.1	1.0E+02	4.0E-08	2.7E-11
SM 153	0.1	0.1	0.1	0.0	0.0	0.0	0.0	3.0E+03	1.0E-06	3.9E-11
SN 125	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4.0E+02	1.0E-07	3.2E-12
SR 89	4.7	4.7	4.6	4.6	4.3	3.9	3.2	1.0E+02	6.0E-08	1.2E-09
SR 90	3.5	3.5	3.5	3.5	3.5	3.5	3.5	4.0E+00	2.0E-09	8.4E-10

SR 91	5.8	0.9	0.1	0.0	0.0	0.0	0.0	4.0E+03	1.0E-06	1.3E-09
SR 92	5.9	0.0	0.0	0.0	0.0	0.0	0.0	7.0E+03	3.0E-06	1.5E-09
TC 99M	5.1	4.4	3.4	2.7	0.9	0.1	0.0	2.0E+05	6.0E-05	1.3E-09
TE 127	0.1	0.1	0.1	0.1	0.0	0.0	0.0	2.0E+04	7.0E-06	3.1E-11
TE 129	0.7	0.1	0.0	0.0	0.0	0.0	0.0	6.0E+04	3.0E-05	1.5E-10
TE 131	2.4	0.0	0.0	0.0	0.0	0.0	0.0	5.0E+03	2.0E-06	6.0E-10
TE 132	4.2	3.4	2.7	2.1	0.9	0.3	0.0	2.0E+02	9.0E-08	1.0E-09
TE 127	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.0E+02	1.0E-07	4.3E-12
TE 129	0.1	0.1	0.1	0.1	0.1	0.1	0.0	2.0E+02	1.0E-07	2.3E-11
TE 131	0.4	0.3	0.1	0.1	0.0	0.0	0.0	4.0E+02	2.0E-07	8.6E-11
XE 133	658.1	634.5	582.7	523.4	317.6	126.7	20.0	1.0E-04	1.2E-07	
XE 135	639.7	233.6	48.0	8.6	0.0	0.0	0.0	1.0E-05	1.2E-07	
XE 131	3.1	3.1	3.1	3.0	2.8	2.3	1.3	4.0E-04	5.6E-10	
XE 133	19.2	17.6	14.3	11.1	3.5	0.4	0.0	1.0E-04	3.5E-09	
XE 135	110.4	7.9	0.7	0.0	0.0	0.0	0.0	9.0E-06	2.0E-08	
Y 90 2	0.8	3.5	3.5	3.5	3.5	3.5	3.5	1.0E+02	5.0E-08	8.4E-10
Y 91	5.8	5.6	5.6	5.5	5.4	4.8	4.2	8.0E+03	3.0E-06	1.3E-09
Y 92	5.9	0.1	0.0	0.0	0.0	0.0	0.0	2.0E+03	1.0E-06	1.5E-09
Y 93	6.3	1.2	0.3	0.0	0.0	0.0	0.0	2.0E+03	1.0E-06	1.6E-09
Y 91M	3.4	0.7	0.1	0.0	0.0	0.0	0.0	2.0E+05	7.0E-05	8.2E-10
ZR 95	6.2	6.2	6.0	6.0	5.8	5.4	4.6	1.0E+02	5.0E-08	1.5E-09
ZR 97	5.8	2.1	0.8	0.3	0.0	0.0	0.0	1.0E+03	5.0E-07	1.3E-09

CHAPTER 13 APPENDIX F

Maximum Activity Available for Release
One TRIGA Element at 31.125 kW, 8 h/d, 5 Days

(Release fractions: 1E-04 for halogens and noble gases, 1E-06 for particulates)

	Potential Activity Release (µCi) Time post discharge							ALI	DAC	Initial Bay Concentration	
	0	1 d	2 d	3 d	7 d	14 d	28 d	µCi	µCi/cm	µCi/cm	issue
AS 78	3.75	0	0	0	0	0	0	2.E+04	9.E-06	9.25E-10	NA
BA 139	1604.3	0	0	0	0	0	0	3.E+04	1.E-05	4.00E-07	NA
BA 140	127.25	120.5	114.3	108.3	87.25	59.5	28	1.E+03	6.E-07	3.00E-08	NA
BA 141	1480.3	0	0	0	0	0	0	7.E+04	3.E-05	3.75E-07	NA
BA 142	1470.5	0	0	0	0	0	0	1.E+05	6.E-05	3.50E-07	NA
BR 83	12110	12.5	0	0	0	0	0	6.E+04	3.E-05	3.00E-06	NA
BR 84	25386	0	0	0	0	0	0	6.E+04	2.E-05	6.25E-06	NA
CE 141	47.25	52.5	51.5	50.25	46.25	39.75	29.5	6.E+02	2.E-07	1.15E-08	NA
CE 143	553.5	340.75	205.8	124.3	16.5	0.5	0	2.E+03	7.E-07	1.35E-07	NA
CE 144	5.5	5.5	5.5	5.5	5.5	5.5	5.25	1.E+01	6.E-09	1.38E-09	NA
CS 138	1711.8	0	0	0	0	0	0	6.E+04	2.E-05	4.25E-07	NA
I 131	8489.3	8245.3	7694	7131	5124	2810	840.5	5.E+01	2.E-08	2.08E-06	103.8
I 132	23281	20943	16930	13686	5838	1317	67	8.E+03	3.E-06	5.75E-06	1.9
I 133	69950	33529	15072	6772	276.3	1	0	3.E+02	1.E-07	1.73E-05	172.5
I 134	192228	0	0	0	0	0	0	5.E+04	2.E-05	4.75E-05	2.4
I 135	98750	7980	644.3	52	0	0	0	2.E+03	7.E-07	2.43E-05	34.6
KR 87	64498	0.25	0	0	0	0	0	na	5.E-06	1.58E-05	3.2
KR 88	79048	225.75	0.75	0	0	0	0	na	2.E-06	1.95E-05	9.8
KR 83M	9953.3	47.75	0	0	0	0	0	na	1.E-02	2.45E-06	NA
KR 85M	23368	580.25	14.25	0.25	0	0	0	na	2.E-05	5.75E-06	NA
LA 140	67.75	86.75	97	101.8	96	68.5	32.25	1.E+03	5.E-07	1.68E-08	NA
LA 141	1140.3	18.5	0.25	0	0	0	0	9.E+03	4.E-06	2.75E-07	NA
LA 142	1454.5	0	0	0	0	0	0	2.E+04	9.E-06	3.50E-07	NA
LA 143	1493.3	0	0	0	0	0	0	9.E+04	4.E-05	3.75E-07	NA
MO 99	395.5	307.5	239	185.8	67.75	11.5	0.25	1.E+03	6.E-07	9.75E-08	NA
MO 101	1272.8	0	0	0	0	0	0	1.E+05	6.E-05	3.00E-07	NA
NB 97	596.75	246.75	92.25	34.5	0.75	0	0	7.E+04	3.E-05	1.48E-07	NA
NB 98	1463.5	0	0	0	0	0	0	5.E+04	2.E-05	3.50E-07	NA
ND 147	53.75	51	48	45	35	22.5	9.5	8.E+02	4.E-07	1.33E-08	NA
ND 149	263.25	0	0	0	0	0	0	2.E+04	1.E-05	6.50E-08	NA
ND 151	105.5	0	0	0	0	0	0	2.E+05	8.E-05	2.50E-08	NA
PD 109	3.5	1	0.25	0	0	0	0	5.E+03	2.E-06	8.75E-10	NA
PM 151	41	23.25	13	7.25	0.75	0	0	3.E+03	1.E-06	1.00E-08	NA
PR 143	69.5	88	97	100.3	91.25	65	31.75	7.E+02	3.E-07	1.70E-08	NA
PR 144	5.5	5.5	5.5	5.5	5.5	5.5	5.25	1.E+05	5.E-05	1.38E-09	NA
PR 145	638	40.25	2.5	0.25	0	0	0	8.E+03	3.E-06	1.58E-07	NA
PR 147	572	0	0	0	0	0	0	2.E+05	8.E-05	1.40E-07	NA
RB 88	799	2.5	0	0	0	0	0	6.E+04	3.E-05	1.95E-07	NA
RH 105	77.5	65.25	41.25	25.75	4	0.25	0	6.E+03	2.E-06	1.90E-08	NA
RH 107	41	0	0	0	0	0	0	2.E+05	1.E-04	1.00E-08	NA
RU 103	22.5	22	21.75	21.25	19.75	17.5	13.75	6.E+02	3.E-07	5.50E-09	NA
RU 105	187.5	4.75	0	0	0	0	0	1.E+04	5.E-06	4.50E-08	NA
SB 127	6.5	6	5	4	2	0.5	0	9.E+02	4.E-07	1.63E-09	NA

	Potential Activity Release (μCi) Time post discharge						ALI	DAC	Initial Bay Concentration	
	0	1 d	2 d	3 d	7 d	14 d	28 d	μCi	μCi/cm	μCi/cm	issue
SE 81	53	0	0	0	0	0	0	2.E+05	1.E-04	1.30E-08	NA
SE 83	49.5	0	0	0	0	0	0	1.E+05	5.E-05	1.23E-08	NA
SM 153	12.75	9	6.25	4.5	1	0	0	3.E+03	1.E-06	3.25E-09	NA
SM 155	8.25	0	0	0	0	0	0	2.E+05	9.E-05	2.05E-09	NA
SN 127	19.75	0	0	0	0	0	0	2.E+04	8.E-06	4.75E-09	NA
SN 128	83.75	0	0	0	0	0	0	3.E+04	1.E-05	2.05E-08	NA
SR 89	28	28	27.5	27.25	25.75	23.25	19.25	1.E+02	6.E-08	6.75E-09	NA
SR 91	796.25	138.75	24	4.25	0	0	0	4.E+03	1.E-06	1.95E-07	NA
SR 92	1325.8	2.75	0	0	0	0	0	7.E+03	3.E-06	3.25E-07	NA
TC 101	1273	0	0	0	0	0	0	3.E+05	1.E-04	3.00E-07	NA
TC 104	459.75	0	0	0	0	0	0	7.E+04	3.E-05	1.13E-07	NA
TE 127	4.75	5.25	4.75	4	2	0.5	0	2.E+04	7.E-06	1.18E-09	NA
TE 129	93.75	3.5	0.5	0.5	0.5	0.5	0.25	6.E+04	3.E-05	2.30E-08	NA
TE 131	628.75	4.5	2.5	1.5	0.25	0	0	5.E+03	2.E-06	1.55E-07	NA
TE 132	250.75	203	164.3	132.8	56.75	12.75	0.75	2.E+02	9.E-08	6.25E-08	NA
TE 133	964.25	0	0	0	0	0	0	2.E+04	9.E-06	2.38E-07	NA
TE 134	1712.5	0	0	0	0	0	0	2.E+04	1.E-05	4.25E-07	NA
XE 133	20362	24106	24010	22389	14111	5673	894.75	na	1.E-04	5.00E-06	NA
XE 135	58955	30517	6594	1195	1	0	0	na	1.E-05	1.45E-05	1.5
XE 138	158548	0	0	0	0	0	0	na	4.E-06	4.00E-05	10.0
Y 91	23.25	27.5	28	28	26.5	24.5	20.75	1.E+02	5.E-08	5.75E-09	NA
Y 92	846.75	37.75	0.5	0	0	0	0	8.E+03	3.E-06	2.08E-07	NA
Y 93	861.25	169.75	32.75	6.25	0	0	0	2.E+03	1.E-06	2.10E-07	NA
Y 94	1583.8	0	0	0	0	0	0	8.E+04	3.E-05	4.00E-07	NA
Y 95	1613	0	0	0	0	0	0	1.E+05	6.E-05	4.00E-07	NA
Y 91M	423.5	88	15.25	2.75	0	0	0	2.E+05	7.E-05	1.05E-07	NA
ZR 95	29.25	29.25	28.75	28.5	27.25	25.25	21.75	1.E+02	5.E-08	7.25E-09	NA
ZR 97	657	245.5	91.75	34.25	0.75	0	0	1.E+03	5.E-07	1.60E-07	NA

Chapter 14

Technical Specifications

Reed Research Reactor Safety Analysis Report

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Technical Specifications
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1.0 DEFINITIONS

The following frequently used terms are defined to aid in the uniform interpretation of these specifications. Capitalization is used in the body of the Technical Specifications to identify defined terms.

ACTION	Tasks to be accomplished in the event a required condition identified in a Specification section is not met, as stated in the Condition column of Actions for that section. In using Action Statements, the following guidance applies: • Where multiple conditions exist in a Limiting Condition for Operations, actions are linked to the Condition by letter and number. • Where multiple action steps are required to address a condition, completion time for each action is linked to the action by letter and number. • AND in an Action Statement means all steps need to be performed to complete the action; OR indicates options and alternatives, only one of which needs to be performed to complete the action. • If a Condition exists, the Action consists of completing all steps associated with the selected option except where the Condition is corrected prior to completion of the steps.
ANNUAL	Every 12 months, not to exceed a 15-month interval.
BIENNIAL	Every two years, not to exceed a 28-month interval.
CHANNEL CALIBRATION	An adjustment of the channel so that its output responds, with acceptable range and accuracy, to known values of the parameter that the channel measures.
CHANNEL CHECK	A qualitative verification of acceptable performance by observation of channel behavior. This verification shall include comparison of the channel with expected values, other independent channels, or other methods of measuring the same variable.
CHANNEL TEST	The introduction of an input signal into a channel to verify that it is operable. A channel test is a functional test of operability.
DAILY	Prior to initial operation each day (when the reactor is operated), or before an operation extending more than 1 day.
ENSURE	Verify existence of specified condition or (if condition does not meet criteria) take action necessary to meet condition.
EXPERIMENT	An EXPERIMENT is (1) any apparatus, device, or material placed in the reactor core region (in an EXPERIMENTAL FACILITY associated with the reactor, or in line with a beam of radiation emanating from the reactor) or (2) any in-core operation designed to measure reactor characteristics.

EXPERIMENTAL FACILITY	The rotary specimen rack, pneumatic transfer systems, central thimble, in-core element replacement positions, and near-core irradiation facilities.
IMMEDIATE	Without delay, and not exceeding one hour. *NOTE: IMMEDIATE permits activities to restore required conditions for up to one hour; this does not permit or imply deferring or postponing action.*
LIMITING CONDITION FOR OPERATION (LCO)	The lowest functional capability or performance levels of equipment required for safe operation of the facility.
LIMITING SAFETY SYSTEM SETTING (LSSS)	Settings for automatic protective devices related to those variables having significant safety functions.
MEASURED VALUE	The value as it appears at the output of a MEASURING CHANNEL.
MEASURING CHANNEL	The combination of sensor, lines, amplifiers, and output devices that are connected for the purpose of measuring the value of a process variable.
OPERABLE	A system or component is OPERABLE when it is capable of performing its intended function in a normal manner.
OPERATING	A system or component is OPERATING when it is performing its intended function in a normal manner.
OPERATING MODE	The reactor is in the OPERATING MODE when the key switch is in the ON position.
REFERENCE CORE CONDITION	The condition of the core when it is at 20°C and the reactivity worth of xenon is zero.
RING	One of the five concentric bands of fuel elements surrounding the central thimble of the core. The rings are designate by letters B through F, with B the innermost ring.
SAFETY CHANNEL	A MEASURING CHANNEL in the SAFETY SYSTEM.
SAFETY SYSTEM	That combination of MEASURING CHANNELS and associated circuitry that is designed to initiate reactor scram or that provides information that requires manual protective action to be initiated.

SECURED MODE	The reactor is in the SECURED MODE when either item (1) or item (2) is satisfied: (1) There is insufficient moderator or insufficient fissile material in the reactor to attain criticality under optimum available conditions of moderation and reflection. (2) All of the following: a. The console key is it the OFF position and the key is removed from the console and under the control of a licensed operator or stored in a locked storage area. b. No work is in progress involving core fuel, core structure, control rods, or control rod drives (unless the drive is physically decoupled from the control rod). c. No EXPERIMENTS are being moved or serviced that have a reactivity worth greater than \$1.00.
SEMIANNUAL	Every six months, not to exceed an eight-month interval.
SHALL (SHALL NOT)	Indicates specified action is required/(prohibited).
SHUTDOWN	The reactor is SHUTDOWN if it is subcritical by at least \$1.00 both in the REFERENCE CORE CONDITION and for all allowable ambient conditions with the reactivity worth of all EXPERIMENTS included.
SHUTDOWN MARGIN	The minimum reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control rods starting from any permissible operating condition, and that the reactor will remain subcritical without further operator action.
STARTUP CHANNEL	The core neutron MEASURING CHANNEL used for the interlock preventing rod withdrawal if no neutron-induced signal is present.
TECHNICAL SPECIFICATION VIOLATION	A violation of a Safety Limit occurs when the Safety Limit value is exceeded. A violation of a Limiting Safety System Setting or Limiting Condition for Operation occurs when a Condition exists which does not meet a Specification and the corresponding Action has not been met within the required Completion Time. If the Action statement of an LSSS or LCO is completed or the Specification is restored within the prescribed Completion Time, a violation has not occurred. *NOTE: Condition, Specification, Action, and Completion Time refer to applicable titles of sections in individual Technical Specifications.*

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT

2.1.1 Applicability

This specification applies when the reactor in the OPERATING MODE.

2.1.2 Objective

This SAFETY LIMIT ensures fuel element cladding integrity.

2.1.3 Specifications

(1)	Power level SHALL NOT exceed 700 kW.
(2)	Power level SHALL NOT exceed 300 kW with any aluminum clad fuel elements in the core.

2.1.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Power level exceeds 700 kW OR With aluminum clad fuel elements in the core, power level exceeds 300 kW	A.1.a Establish SHUTDOWN condition AND A.1.b Establish SECURED mode AND A.2 Report per Section 6.8	A.1.a IMMEDIATE A.1.b IMMEDIATE A.2 Within 24 hours

2.1.5 Bases

Safety Analysis Report, Section 3.5.1 (Fuel System) identifies design and operating constraints for TRIGA® fuel that will ensure cladding integrity is not challenged.

NUREG 1282 identifies the safety limit for the high-hydride ($ZrH_{1.7}$) fuel elements with stainless steel cladding based on the stress in the cladding (resulting from the hydrogen pressure from the dissociation of the zirconium hydride). This stress will remain below the yield strength of the stainless steel cladding with fuel temperatures below 1,150°C. A change in yield strength occurs for stainless steel cladding temperatures of 500°C, but there is no scenario for fuel cladding to achieve 500°C while submerged; consequently the safety limit during reactor operations is 1,150°C.

Therefore, the important process variable for a TRIGA® reactor is the fuel element temperature. NUREG 1537 Appendix 14.1 allows for reactors without instrumented fuel to establish a power level that limits fuel cladding maximum temperatures below safety limits.

During operation, fission product gases and dissociation of the hydrogen and zirconium builds up gas inventory in internal components and spaces of the fuel elements. Fuel temperature acting on these gases controls fuel element internal pressure. Limiting the maximum temperature prevents excessive internal pressures that could be generated by heating these gases.

The temperature at which phase transitions may lead to cladding failure in aluminum-clad low-hydride fuel elements is reported to be 530°C; references: Technical Foundations of TRIGA®, GA-471 (1958), pp. 63-72; also in "Hazards Analysis for the Oregon State University 250 kW TRIGA® Mark II Reactor," (June 1965), Section 4.7. There is also extensive operating experience with aluminum-clad, low-hydride fuel.

Fuel growth and deformation can occur during normal operations, as described in General Atomics technical report E-117-833. Damage mechanisms include fission recoils and fission gases, strongly influenced by thermal gradients. Operating with maximum long-term, OPERATING fuel temperature of 750°C does not have significant time- and temperature-dependent fuel growth.

2.2 *LIMITING SAFETY SYSTEM SETTINGS (LSSS)*

2.2.1 Applicability

This specification applies in the OPERATING MODE.

2.2.2 Objective

The objective of this specification is to ensure the safety limit is not exceeded.

2.2.3 Specifications

(1)	Power level SHALL NOT exceed 600 kW.
(2)	Power level SHALL NOT exceed 300 kW with any aluminum clad fuel elements in the core.
(3)	High voltage to required reactor power detectors is at least 90% of nominal voltage.

2.2.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Power level exceeds 600 kW OR Power level exceeds 300 kW with aluminum clad fuel	A.1 Reduce power to the limit OR A.2 Establish SHUTDOWN condition	A.1 IMMEDIATE A.2 IMMEDIATE
B. High voltage to reactor power level detector less 90% of nominal voltage	B. Establish SHUTDOWN condition	B. IMMEDIATE

2.2.5 Bases

Analysis in the Safety Analysis Report, 4.5.3, demonstrates fuel centerline temperature does not exceed 600°C at power levels approximately 1.25 MW with bulk pool water temperature at approximately 100°C. Using an LSSS of 600 kW provides an adequate margin to the safety limit while allowing maximum flexibility for operations and maintenance. The LSS is reduced to 300 kW if there are any aluminum clad fuel elements in the core.

According to General Atomics, detector voltages less than 90% of required operating value do not provide reliable, accurate nuclear instrumentation.

3.0 LIMITING CONDITIONS FOR OPERATION (LCO)

3.1 CORE REACTIVITY

3.1.1 Applicability

These specifications are required prior to entering OPERATING MODE; reactivity limits on EXPERIMENTS are specified in Section 3.8.

3.1.2 Objective

This LCO ensures the reactivity control system is OPERABLE, and that a power excursion does not result in exceeding the safety limit.

3.1.3 Specifications

(1)	The reactor is capable of being made subcritical by a SHUTDOWN MARGIN more than $0.50 under REFERENCE CORE CONDITIONS and under the following conditions: 1. The highest worth control rod is fully withdrawn. 2. The highest worth EXPERIMENT is in its most positive reactive state.
(2)	The maximum available core reactivity (excess reactivity) with all control rods fully withdrawn is less than $3.00 when: 1. REFERENCE CORE CONDITIONS exists. 2. No EXPERIMENTS with net negative reactivity worth are in place.

3.1.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The reactor is not subcritical by more than $0.50 under specified conditions	A.1.a ENSURE control rods fully inserted AND A.1.b Secure electrical power to the control rod circuits AND A.1.c Secure all work on in-core EXPERIMENTS or installed control rod drives AND A.2 Configure reactor to meet LCO	A.1.a IMMEDIATE A.1.b IMMEDIATE A.1.c IMMEDIATE A.2 Prior to continued operations
B. Reactivity with all control rods fully withdrawn exceeds $3.00	B.1 ENSURE SHUTDOWN condition AND B.2 Configure reactor to meet LCO	B.1 IMMEDIATE B.2 Prior to continued operations

3.1.5 Bases

The limiting SHUTDOWN MARGIN is necessary so that the reactor can be shut down from any operating condition, and will remain shut down after cool down and xenon decay, even if one control rod should remain in the fully withdrawn position.

The value for excess reactivity was used in establishing core conditions for calculations that demonstrate fuel temperature limits are met during potential accident scenarios under extremely conservative conditions of analysis. Since the fundamental protection for the RRR is the maximum power level that can be achieved with the available positive core reactivity, EXPERIMENTS with positive reactivity are included in determining excess reactivity. Since EXPERIMENTS with negative reactivity will increase available reactivity if they are removed during operation, they are not credited in determining excess reactivity.

Safety Analysis Report Section 13.2 demonstrates that a $3.00 reactivity insertion from critical, zero power conditions leads to maximum fuel temperature of 250°C, well below the limit.

3.2 OPERATING MODE

3.2.1 Applicability

This specification applies when the reactor is in the OPERATING MODE.

3.2.2 Objectives

The objective is to prevent the SAFETY LIMIT from being exceeded during operations.

3.2.3 Specification

The reactor SHALL NOT be operated at steady-state power levels above:

(a) 500 kW.

(b) 250 kW if the core contains aluminum clad fuel elements.

3.2.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Power level is greater than the LCO	A. Reduce reactor power to LCO	A. IMMEDIATE

3.2.5 Bases

The Safety Analysis is based on power levels up to 500 kW. The Reed College reactor was licensed in 1968 for operation at 250 kW with aluminum clad fuel elements.

Calculations in Chapter 4 assuming 500 kW operation and 83 fuel elements demonstrate fuel temperature limits are met.

A value of 500 kW for maximum power level with stainless steel clad fuel was used to establish core conditions for calculations (Table 13.4) that demonstrate fuel temperature limits are met during potential accident scenarios under extremely conservative conditions of analysis.

A 500 kW operating history is assumed to determine maximum fission product inventory available for release. The unrealistically conservative assumptions for maximum hypothetical release of fission products from fuel assume a complete release of all available inventory. Analysis in Chapter 13 demonstrates that even with these unrealistically conservative assumptions, limits of 10 CFR Part 20 for releases to unrestricted areas are not challenged, and although instantaneous releases to the reactor bay exceed limits for ALI for a few radionuclides if trapped within the reactor bay and not released, time averaged values are within limits.

3.3 *MEASURING CHANNELS*

3.3.1 Applicability

This specification applies to the reactor MEASURING CHANNELS during OPERATING MODE.

3.3.2 Objective

The objective is to require that sufficient information is available to the operator to ensure safe operation of the reactor.

3.3.3 Specifications

(1)	The MEASURING CHANNELS specified in TABLE 1 SHALL be OPERATING.
(2)	There is a neutron-induced signal on the STARTUP CHANNEL.

TABLE 1: MINIMUM MEASURING CHANNEL COMPLEMENT

MEASURING CHANNEL	Minimum Number Operable
Reactor Power Level	2
Primary Pool Water Temperature	1
Radiation Area Monitor	1
Continuous Air Monitor	1

3.3.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fewer than two reactor power channels OPERATING	A.1 Restore two channels to OPERATION OR A.2 ENSURE reactor is SHUTDOWN	A.1 IMMEDIATE A.2 IMMEDIATE
B. Primary water temperature channel not OPERATING	B.1 Restore channel to OPERATION OR B.2 ENSURE reactor is SHUTDOWN	B.1 IMMEDIATE B.2 IMMEDIATE
C. Radiation Area Monitor is not OPERATING	C.1 Restore MEASURING CHANNEL OR C.2 ENSURE reactor is shutdown OR C.3 ENSURE personnel are not in the reactor bay OR C.4.a ENSURE an equivalent monitor measuring the same area is OPERATING AND C.4.b Restore MEASURING CHANNEL	C.1 IMMEDIATE C.2 IMMEDIATE C.3 IMMEDIATE C.4.a IMMEDIATE C.4.b Within 30 days
D. Continuous Air Monitor is not OPERATING	D.1 Restore MEASURING CHANNEL OR D.2 ENSURE reactor is shutdown OR D.3.a ENSURE an equivalent monitor measuring the same airflow is OPERATING AND D.3.b Restore MEASURING CHANNEL	D.1 IMMEDIATE D.2 IMMEDIATE D.3.a IMMEDIATE D.3.b Within 30 days
E. STARTUP CHANNEL is not OPERATING	E.1 Do not perform a reactor startup OR E.2 Terminate reactor startup	E.1 IMMEDIATE E.2 IMMEDIATE

3.3.5 Bases

Maximum OPERATING power level is 500 kW. The neutron detectors ensure measurement of the reactor power level. Chapter 4 and 13 discuss heat removal capabilities in normal and accident scenarios. Chapter 7 discusses neutron and power level detection systems.

Primary water temperature indication is required to assure water temperature limits are met, protecting the primary cleanup resin integrity.

The radiation area monitor provides information about radiation hazards in the reactor bay. A loss of reactor pool water (Chapter 13), changes in shielding effectiveness (Chapter 11), and releases of radioactive material to the restricted area (Chapter 11) could cause changes in radiation levels within the reactor bay detectable by this monitor. Portable survey instruments will detect changes in radiation levels. Chapter 7 discusses radiation detection and monitoring systems.

The continuous air monitor provides indication of airborne contaminants in the reactor bay prior to discharge of gaseous effluent.

Chapter 13 discusses inventories and releases of radioactive material from fuel element failure into the reactor bay and to the environment. Particulate and noble gas channels monitor more routine discharges. Chapter 11 discusses routine discharges of radioactive gasses generated from normal operations into the reactor bay and into the environment. Chapter 3 identifies design bases for the confinement and ventilation system. Chapter 7 discusses air-monitoring systems.

3.4 Safety Channel and Control Rod Operability

3.4.1 Applicability

This specification applies to the reactor MEASURING CHANNELS during OPERATING MODE.

3.4.2 Objective

The objectives are to require the minimum number of SAFETY SYSTEM channels that must be OPERABLE in order to ensure the safety limit is not exceeded, and to ensure prompt shutdown in the event of a scram signal.

3.4.3 Specifications

(1)	The SAFETY SYSTEM CHANNELS specified in TABLE 2 are OPERABLE.
(2)	Control rods are capable of 90% of full reactivity insertion from the fully withdrawn position in less than 1 second.

TABLE 2: REQURIED SAFETY SYSTEM CHANNELS

Safety System Channel or Interlock	Minimum Number OPERABLE	Function	Required for OPERATING MODE
Reactor power level	2	Scram	YES
Manual scram bar	1	Scram	YES
STARTUP CHANNEL interlock	1	Prevent control rod withdrawal when there is no neutron-induced signal	YES (Startup)
Control rod withdrawal interlock	1	Prevent manual withdrawal of more than one control rod at a time	YES

3.4.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any required SAFETY SYSTEM CHANNEL function is not OPERABLE	A.1 Restore channel to operation OR A.2 ENSURE reactor is SHUTDOWN	A.1 IMMEDIATE A.2 IMMEDIATE
B. STARTUP CHANNEL interlock is not OPERABLE	B.1 Do not perform a reactor startup OR B.2 Terminate reactor startup	B.1 IMMEDIATE B.2 IMMEDIATE
C. Control Rod interlock is not OPERABLE	C.1 Restore interlock function OR C.2 ENSURE reactor is SHUTDOWN	C.1 IMMEDIATE C.2 IMMEDIATE

3.4.5 Bases

The power level scram is provided as added protection to ensure that reactor operation stays within the licensed limits of 500 kW, preventing abnormally high fuel temperature. The power level scram is not credited in analysis, but provides assurance that the reactor is not operated in conditions beyond the assumptions used in analysis (Table 13.2.1.4).

The manual scram allows the operator to shut down the reactor if an unsafe or abnormal condition occurs.

The interlock ensures a neutron detection channel is operating prior to startup by preventing startup of the reactor without a neutron-induced signal indicated on the STARTUP CHANNEL.

The control rod interlock will prevent accidental insertion of an reactivity at an excessive rate.

3.5 *GASEOUS EFFLUENT CONTROL*

3.5.1 Applicability

This specification applies to gaseous effluent in the OPERATING MODE.

3.5.2 Objective

The objective is to ensure that exposures to the public resulting from gaseous effluents released during normal operations and accident conditions are within limits and keeping with the principle of ALARA.

3.5.3 Specification

(1)	The reactor bay ventilation exhaust system SHALL maintain in-leakage to the reactor bay.

3.5.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The reactor bay ventilation exhaust system is not OPERABLE	A.1 ENSURE reactor is SHUTDOWN AND A.2.a Secure operations for EXPERIMENTS with failure modes that could result in the release of radioactive gases or aerosols AND A.2.b ENSURE no irradiated fuel handing AND A.2.c Restore the reactor bay ventilation exhaust system to OPEABLE	A.1 IMMEDIATE A.2.a IMMEDIATE A.2.b IMMEDIATE A.2.c Within 30 days

3.5.5 Bases

The confinement and ventilation system is described in Section 3.5.4. Routine operations produce radioactive gas, principally argon-41, in the reactor bay. The ventilation system is not taken credit for in the SAR. Consequently, the ventilation system can be secured without causing significant personnel hazard from normal operations. Thirty days for a confinement and ventilation system outage is selected as a reasonable interval to allow major repairs and work to be accomplished, if required. During this interval, experiment activities that might cause airborne radionuclide levels to be elevated are prohibited.

It is shown in Section 13.2.2 of the Safety Analysis Report that, if the reactor were to be operating at full power, fuel element failure would not occur even if all the reactor tank water were to be lost instantaneously.

Section 13.2.4 addresses the maximum hypothetical fission product inventory release. Using unrealistically conservative assumptions, concentrations for a few nuclides of iodine would be in excess of occupational derived air concentrations for a matter of hours or days. Strontium-90 activity available for release from fuel rods previously used at other facilities is estimated to be at most about 4 times the ALI. In either case (radio-iodine or radio-strontium), there is no credible scenario for accidental inhalation or ingestion of the undiluted nuclides that might be released from a damaged fuel element. Finally, fuel element failure during a fuel handling accident is likely to be observed and mitigated immediately.

3.6 LIMITATIONS ON EXPERIMENTS

3.6.1 Applicability

This specification applies to the OPERATING MODE.

3.6.2 Objectives

These Limiting Conditions for Operation prevent reactivity excursions that might cause the fuel to exceed the safety limit (with possible resultant damage to the reactor), and the excessive release of radioactive materials in the event of an EXPERIMENT failure.

3.6.3 Specifications

(1)	The reactivity worth of any individual EXPERIMENT SHALL NOT exceed $1.00.
(2)	If two or more EXPERIMENTS in the reactor are interrelated so that operation or failure of one can induce reactivity-affecting change in the other(s), the sum of the absolute reactivity of such EXPERIMENTS SHALL NOT exceed $1.00.
(3)	Irradiation holders and vials SHALL prevent release of encapsulated material in the reactor pool and core area.

3.6.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. An EXPERIMENT worth is greater than $1.00	A.1 ENSURE the reactor is SHUTDOWN AND A.2 Remove the EXPERIMENT	A.1 IMMEDIATE A.2 Prior to continued operations
B. Operation or failure of EXPERIMENT can induce a reactivity change in a second EXPERIMENT such that the sum of their reactivities is greater than $1.00	B.1 ENSURE the reactor is SHUTDOWN AND B.2 Remove the experiment, or ENSURE that failure cannot cause a reactivity change	B.1 IMMEDIATE B.2 Prior to continued operations
C. An irradiation holder or vial releases material into the pool or core area that is capable of causing damage to the reactor fuel or structure	C.1 ENSURE the reactor is SHUTDOWN AND C.2 Inspect the affected area AND C.3 Obtain RRC approval	C.1 IMMEDIATE C.2 Prior to continued operation C.3 Prior to continued operation

3.6.5 Bases

Specifications 3.6(1) and 3.6(2) are conservatively chosen to limit reactivity additions to maximum values that are less than an addition that could cause the fuel temperature to rise above the limiting safety system set point (LSSS) value. The temperature rise for a $1.00 insertion is known from previous license conditions and operations and is known not to exceed the LSSS.

EXPERIMENTS are approved with expectations that there is reasonable assurance the facility will not be damaged during normal or failure conditions. If an irradiation capsule which contains material with potential for challenging the fuel cladding or pool wall the facility will be inspected to ensure that continued operation is acceptable.

3.7 *FUEL INTEGRITY*

3.7.1 Applicability

This specification applies to the OPERATING MODE.

3.7.2 Objective

The objective is to prevent the use of damaged fuel in the reactor.

3.7.3 Specifications

(1)	Fuel elements in the reactor core SHALL NOT be elongated more than 0.32 mm over manufactured length.
(2)	Fuel elements in the reactor core SHALL NOT be laterally bent more than 0.32 mm.

3.7.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any fuel element is elongated greater than 0.32 mm over manufactured length, or bent laterally greater than 0.32 mm	A. Do not insert the fuel element into the core	A. IMMEDIATE

3.7.5 Bases

The above limits on the allowable distortion of a fuel element have been shown to correspond to strains that are considerably lower than the strain expected to cause rupture of a fuel element and have been successfully applied at TRIGA® installations. Fuel cladding integrity is important since it represents the only fission product release barrier for the TRIGA® reactor.

3.8 *Reactor Pool Water*

3.8.1 Applicability

This specification applies to the OPERATING MODE and SECURED MODE.

3.8.2 Objective

The objective is to set acceptable limits on the water quality, temperature, conductivity, and level in the reactor pool.

3.8.3 Specifications

(1)	Water temperature at the exit of the reactor pool SHALL NOT exceed 55°C with flow through the primary cleanup loop.
(2)	Water conductivity SHALL be less than 2 μSiemens/cm.
(3)	Water level above the core SHALL be at least 5 meters above the top of the core.

3.8.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Water temperature at the exit of the reactor pool exceeds 55°C	A.1 ENSURE the reactor is SHUTDOWN AND A.2 Secure flow through demineralizer AND A.3 Reduce water temperature to less than 55°C	A.1 IMMEDIATE A.2 IMMEDIATE A.3 Before allowing flow through demineralizer
B. Water conductivity greater than 2 μSi/cm	B.1 ENSURE the reactor is SHUTDOWN AND B.2 Restore conductivity to less than 2 μSi/cm	B.1 IMMEDIATE B.2 Within 4 weeks
C. Water level less than 5 meters above the top of the core	C.1 ENSURE the reactor is SHUTDOWN AND C.2 Restore water level	C.1 IMMEDIATE C.2 Before resuming operation

3.8.5 Bases

The resin used in the deionizer limits the water temperature of the reactor pool. Resin in use (as described in Section 5.4) maintains mechanical and chemical integrity at temperatures below 60°C.

Maintaining low water conductivity over a prolonged period prevents possible corrosion, deionizer degradation, or slow leakage of fission products from degraded cladding.

Although fuel degradation does not occur over short time intervals, long-term integrity of the fuel is important, and a 4-week interval was selected as an appropriate maximum time for high conductivity.

The top of the core is 6.4 meters below the top of the primary coolant tank. The lowest suction of primary cooling flow into the forced cooling loop is 1 meter below the top of the primary coolant tank; if the water level is less than 5.4 meters above the core, primary cooling system suction is lost. The principal contributor to radiation dose rates at the pool surface is nitrogen-16 generated in the reactor core and dispersed in the pool. A minimum pool level of 5 meters above the core is adequate to provide shielding and support the cooling system.

3.9 *MAINTENANCE RETEST REQUIREMENTS*

3.9.1 Applicability

This specification applies to the OPERATING MODE.

3.9.2 Objective

The objective is to ensure Technical Specification requirements are met following maintenance that occurs within surveillance test intervals.

3.9.3 Specification

(1)	Maintenance activities SHALL NOT change, defeat, or alter equipment or systems in a way that prevents the systems or equipment from being OPERABLE or otherwise prevent the systems or equipment from fulfilling the safety basis.

3.9.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Maintenance is performed that has the potential to change a setpoint, calibration, or other parameter that is measured or verified in meeting a surveillance or operability requirement	A.1 Perform surveillance OR A.2 Operate only to perform retest	A.1 Prior to continued, normal operation in the OPERATING MODE A.2 Prior to continued, normal operation in the OPERATING MODE

3.9.5 Bases

Operation of the RRR will comply with the requirements of Technical Specifications. This specification ensures that if maintenance might challenge a Technical Specifications requirement, the requirement shall be verified prior to resumption of normal operations.

4.0 SURVEILLANCE REQUIREMENTS

4.1 CORE REACTIVITY

4.1.1 Objective

This surveillance ensures that the minimum SHUTDOWN MARGIN requirements and maximum excess reactivity limits of Section 3.1 are met.

4.1.2 Specification

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SHUTDOWN MARGIN Determination	SEMIANNUAL
Excess Reactivity Determination	SEMIANNUAL
	Following insertion of EXPERIMENTS with measurable positive reactivity
Control Rod Reactivity Worth Determination	SEMIANNUAL

4.1.3 Basis

Experience at the RRR has shown verification of the minimum allowed SHUTDOWN MARGIN at the specified frequency is adequate to assure that the limiting safety system setting is met.

When core reactivity parameters are affected by operations or maintenance, additional activity is required to ensure changes are incorporated in reactivity evaluations.

4.2 OPERATING MODE

4.2.1 Objectives

This surveillance assures that the high power level trips function at the required setpoint values.

4.2.2 Specification

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
CHANNEL TEST of Percent Power Safety Circuit scram	SEMIANNUAL
CHANNEL TEST of Linear Power Safety Circuit scram	SEMIANNUAL

4.2.3 Basis

The histories of the reactor power level instruments at the Reed College reactor are exceptionally stable over time. The SEMIANNUAL test of power level scram are adequate to ensure the scram set points meet requirements.

4.3 MEASURING CHANNELS

4.3.1 Objectives

Surveillances on MEASURING CHANNELS at specified frequencies ensure instrument problems are identified and corrected before they can affect operations.

4.3.2 Specification

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Reactor power level MEASURING CHANNEL	
CHANNEL TEST	DAILY
Calorimetric calibration	ANNUAL
Primary pool water temperature CHANNEL CALIBRATION	ANNUAL
Radiation Area Monitor	
CHANNEL CHECK	DAILY
CHANNEL CALIBRATION	ANNUAL
Continuous Air Monitor	
CHANNEL CHECK	DAILY
CHANNEL CALIBRATION	ANNUAL
STARTUP CHANNEL Interlock	DAILY

4.3.3 Basis

The DAILY CHANNEL CHECKS will ensure that the SAFETY SYSTEM and MEASURING CHANNELS are operable. The required periodic calibrations and verifications will permit any long-term drift of the channels to be corrected.

4.4 *Safety Channel and Control Rod Operability*

4.4.1 Objective

The objectives of these surveillance requirements are to ensure the SAFETY SYSTEM will function as required. Surveillances related to safety system MEASURING CHANNELS ENSURE appropriate signals are reliably transmitted to the shutdown system; the surveillances in this section ensure the control rod system is capable of providing the necessary actions to respond to these signals.

4.4.2 Specifications

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Manual scram SHALL be tested by releasing partially withdrawn control rod(s)	DAILY
STARTUP CHANNEL interlock test	DAILY
Control rod withdrawal interlock test	DAILY
Control rod drop times SHALL be measured to have a drop time from the fully withdrawn position of less than 1 second	ANNUAL
The control rods SHALL be visually inspected for corrosion and mechanical damage	BIENNIAL

4.4.3 Basis

Manual and automatic scrams are not credited in accident analysis. The systems do function to assure long-term safe shutdown conditions. The manual scram and control rod drop timing surveillances are intended to monitor for potential degradation that might interfere with the operation of the control rod systems.

The control rod inspections are similarly intended to identify potential degradation that may lead to control rod degradation or inoperability.

The functional checks of the control rod drive system assure the control rod drive system operates as intended for any operations.

4.5 GASEOUS EFFLUENT CONTROL

4.5.1 Objectives

These surveillances ensure that routine releases are normal, and (in conjunction with MEASURING CHANNEL surveillances) that instruments will alert the facility if conditions indicate abnormal releases.

4.5.2 Specification

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL TEST of air monitor	ANNUAL

4.5.3 Basis

The continuous air monitor provides indication that levels of radioactive airborne contamination in the reactor bay are normal.

4.6 LIMITATIONS ON EXPERIMENTS

4.6.1 Objectives

This surveillance ensures that EXPERIMENTS do not have significant negative impact on safety of the public, personnel or the facility.

4.6.2 Specification

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Experiments SHALL be evaluated and approved per section 6.4 prior to implementation	Prior to inserting a new EXPERIMENT for purposes other than determination of reactivity worth
Measure and record experiment worth of the EXPERIMENT (where estimated worth is greater than $0.40)	Initial insertion of a new EXPERIMENT where estimated worth is greater than $0.40

4.6.3 Basis

These surveillances allow determination that the limits of 3.6 are met.

Experiments with an estimated significant reactivity worth (greater than $0.40) will be measured to assure that maximum experiment reactivity worths are met. If an estimate indicates less than $0.40 reactivity worth, even a error with a factor of 2 will result in actual reactivity less than the assumptions used in analysis for a power excursion in the Safety Analysis Report.

4.7 FUEL INTEGRITY

4.7.1 Objective

The objective is to ensure that the dimensions of the fuel elements remain within acceptable limits.

4.7.2 Applicability

This specification applies to the surveillance requirements for the fuel elements in the reactor core.

4.7.3 Specification

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
The fuel elements SHALL be visually inspected for corrosion and mechanical damage, and measured for length and bend	Following the exceeding of a limited safety system set point with potential for causing degradation
Fuel elements in B, C, D, E, and F RINGS comprising approximately 1/5 of the core SHALL be visually inspected for corrosion and mechanical damage such that every element in the core SHALL be inspected at 10-year intervals, but not to exceed 122 months	BIENNIAL

4.7.4 Basis

Biennial visual inspection of fuel elements is considered adequate to identify potential degradation of fuel prior to catastrophic fuel element failure.

4.8 REACTOR POOL WATER

This specification applies to the water contained in the reactor pool.

4.8.1 Objective

The objective is to provide surveillance of reactor primary coolant water quality, pool level, temperature and (in conjunction with MEASURING CHANNEL surveillances) conductivity.

4.8.2 Specification

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify reactor pool water level above the inlet line vacuum breaker	DAILY
Verify reactor pool water temperature CHANNEL operable	DAILY
Measure reactor pool water conductivity	DAILY At least once every 4 weeks

4.8.3 Bases

Surveillance of the reactor pool will ensure that the water level is adequate before reactor operation. Evaporation occurs over longer periods of time, and daily checks are adequate to identify the need for water replacement.

Water temperature must be monitored to ensure that the limit of the ion exchange resin will not be exceeded. A daily check on the channel prior to reactor operation is adequate to ensure the channel is operable when it will be needed.

Water conductivity must be checked to ensure that the deionizer is performing properly and to detect any increase in water impurities. A daily check is adequate to verify water quality is appropriate and also to provide data useful in trend analysis. If the reactor is not operated for long periods of time, the requirement for checks at least every 4 weeks will ENSURE water quality is maintained in a manner that does not permit fuel degradation.

4.9 *MAINTENANCE RETEST REQUIREMENTS*

4.9.1 Objective

The objective is to ensure that a system is OPERABLE within specified limits before being used after maintenance has been performed.

4.9.2 Specification

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Evaluate potential for maintenance activities to affect operability and function of equipment required by Technical Specifications	Following maintenance of systems of equipment required by Technical Specifications
Perform surveillance to assure affected function meets requirements	Prior to resumption of normal operations

4.9.3 Bases

This specification ensures that work on the system or component has been properly carried out and that the system or component has been properly reinstalled or reconnected before reliance for safety is placed on it.

5.0 *DESIGN FEATURES*

5.1 *REACTOR FUEL*

5.1.1 Applicability

This specification applies to the fuel elements used in the reactor core.

5.1.2 Objective

The objective is to ENSURE that the fuel elements are of such a design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their mechanical integrity.

5.1.3 Specifications

(1) The fuel element shall contain uranium-zirconium hydride, clad in 0.020 in. of aluminum or 304 stainless steel. It shall contain a maximum of 9.0 weight % uranium which has a maximum enrichment of 20% in uranium-235. There shall be 1.55 to 1.80 hydrogen atoms to 1.0 zirconium atom.

(2) For the loading process, the elements shall be placed in a close packed array except for EXPERIMENTAL FACILITIES or for single positions occupied by control rods and a neutron startup source.

5.1.4 Bases

These types of fuel elements have a long history of successful use in TRIGA® reactors.

5.2 *REACTOR BUILDING*

5.2.1 Applicability

This specification applies to the building that houses the TRIGA® reactor facility.

5.2.2 Objective

The objective is to ENSURE that provisions are made to restrict the amount of release of radioactivity into the environment.

5.2.3 Specifications

(1) The reactor SHALL be housed in a closed room designed to restrict gaseous leakage when the reactor is in operation or when spent fuel is being handled exterior to a cask.

(2) The minimum free volume of the reactor room shall be approximately 10,000 cubic feet (280,000 liters).

(3) The building shall be equipped with a ventilation system capable of exhausting air or other gases from the reactor room at a minimum of 3.5 meters above ground level.

5.2.4 Bases

To control the escape of gaseous effluent, the reactor room contains no windows that can be opened. The room air is exhausted through an independent exhaust system, and discharged at roof level to provide dilution.

5.3 EXPERIMENTS

5.3.1 Applicability

This specification applies to the design of EXPERIMENTS.

5.3.2 Objective

The objective is to ensure that EXPERIMENTS are designed to meet criteria.

5.3.3 Specifications

(1) EXPERIMENTS with a design reactivity worth greater than $1.00 SHALL NOT be placed in the reactor.

(2) Design shall ENSURE that failure of an EXPERIMENT SHALL NOT lead to a direct failure of a fuel element or of other EXPERIMENTS that could result in a measurable increase in reactivity or a measurable release of radioactivity due to the associated failure.

(3) EXPERIMENTS SHALL be designed so they do not cause bulk boiling of core water.

(4) EXPERIMENT design SHALL ENSURE no interference with control rods or shadowing of reactor control instrumentation.

(5) EXPERIMENT design SHALL minimize the potential for industrial hazards, such as fire or the release of hazardous and toxic materials.

(6) Each fueled EXPERIMENT SHALL be limited such that the total inventory of iodine isotopes 131 through 135 in the experiment is not greater than 5 millicuries.

(7) No explosive EXPERIMENTS are allowed.

(8) Where the possibility exists that the failure of an EXPERIMENT (except fueled EXPERIMENTS) could release radioactive gases or aerosols to the reactor bay or atmosphere, the quantity and type of material shall be limited such that the airborne concentration of radioactivity averaged over a year will not exceed the limits of Table II of Appendix B of 10 CFR Part 20 assuming 100% of the gases or aerosols escape.

(9) The following assumptions shall be used in EXPERIMENT design:

1. If effluents from an experimental facility exhaust through a hold-up tank which closes automatically at a high radiation level, at least 10% of the gaseous activity or aerosols produced will escape.
2. If effluents from an experimental facility exhaust through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of the aerosols produced will escape.
3. For materials whose boiling point is above 55°C and where vapors formed by boiling this material could escape only through an undisturbed column of water above the core, at least 10% of these vapors will escape.

5.3.4 Basis

Designing the EXPERIMENT to reactivity and thermal-hydraulic conditions ENSURE that the EXPERIMENT is not capable of breaching fission product barriers or interfering with the control systems (interferences from the control and safety systems are also prohibited). Design constraints on industrial hazards ensure personnel safety and continuity of operations. Design constraints limiting the release of radioactive gasses prevent unacceptable personnel exposure during off-normal experiment conditions.

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION AND RESPONSIBILITIES OF PERSONNEL

6.1.1 Structure

a) Reed College holds the license for the Reed Research Reactor, located on the Reed College campus in Portland, Oregon. The chief administrating officer for Reed College is the President. Environmental, safety, and health oversight functions are administered through the Vice President, Treasurer, while reactor line management functions are through the Vice President, Dean of the Faculty.

b) Radiation protection functions are divided between the Radiation Safety Officer (RSO) and the reactor staff and management, with management and authority for the RSO separate from line management and authority for facility operations. Day-to-day radiation protection functions implemented by facility staff and management are guided by approved administrative controls (Radiation Protection Program or RPP, operating and experiment procedures). These controls are reviewed and approved by the RSO as part of the Reactor Review Committee (RRC). The Reactor Health Physicist reports to the RSO and has specific oversight functions assigned though the RPP. The Reactor Health Physicist provides routine support for personnel monitoring, radiological analysis, and radioactive material inventory control. The Reactor Health Physicist provides guidance on request for non-routine operations such as transportation and implementation of new EXPERIMENTS .

c) The reactor organization is related to the college structure as shown in SAR Figure 12.1.

6.1.2 Responsibility

a) The Reed College administration is responsible for establishing the budget of the facility and for appointing the Director, Associate Director, Health Physicist, and all members of the RRC, except for the Reactor Supervisor.

b) The Director is the chief administrator with the ultimate responsibility for the safe and competent operation of the RRR. This responsibility manifests itself in:

   1) The selection of responsible and competent personnel as Reactor Supervisor, Reactor Operators, and Reactor Assistants.
   2) The establishment of administrative controls consistent with the NRC and other (college, state, or local government) licenses and regulations.
   3) Adding individuals to the security access levels.
   4) The initial approval authority for all reactor experiments.
   5) The enforcement of controls and regulations.
   6) Serving as a non-voting member of the RRC.

7) Authorizing all reactor operation.
8) Interacting with federal, state, and local officials at the operational level.
9) Interacting with reactor users and with other interested parties regarding the program of the reactor.
10) Authorizing all transfers of radioactive materials in and out of the facility.

c) The Associate Director acts as assistant to the Director, and acts on behalf of the latter in some instances. Specifically, the Associate Director is responsible for:

1) Assisting the Director in manners designated by the latter.
2) Acting for the Director, in the absence of the latter, in carrying out numbers (4) through (10) above.
3) Serving as a non-voting member of the RRC.

d) The Reactor Supervisor has responsibility for the operation of the facility and the reactor. The Supervisor is directly responsible to the Director. Specifically, the Reactor Supervisor is responsible for:

1) Assigning Operators and Assistants to, and scheduling of, previously authorized reactor operations.
2) Compliance with facility licenses and applicable regulations.
3) Limiting exposure of personnel and dispersal of radioactive material to the limits set forth in the NRC regulations contained in Title 10, Chapter 1, Part 20, Code of Federal Regulations (10 CFR Part 20), Standards for Protection Against Radiation.
4) Supervising instruction of reactor license candidates.
5) Maintaining all logs and records involving the reactor.
6) Maintaining all Standard Operating Procedures and other administrative directives involving the reactor.
7) Assignment of operators and assistants to perform all periodic inspections and surveys of the facility.
8) Serving as a non-voting member of the RRC.
9) Developing and conducting an annual emergency drill.

e) The Reactor Supervisor shall be an NRC licensed Senior Reactor Operator for the facility. When the Reactor Supervisor is absent, a designated Senior Reactor Operator shall assume the duties of the Reactor Supervisor. The Director or Associate Director may always act in the place of the Supervisor.

f) The Director may choose to delegate some of the Supervisor's duties to other personnel (e.g., operations supervisor, training supervisor, special projects supervisor, etc.). In such a case one supervisor must be assigned the authority of reactor supervisor in case of an emergency or other circumstance. Unless otherwise indicated, the operations supervisor will fill this role.

g) The Reactor Health Physicist has the responsibility to supervise or assist personnel with radiation and contamination control problems. The Reactor

Health Physicist shall notify the Reactor Supervisor, the RRC, and the Director of any unsafe conditions and departures from the approved procedures, licenses, and policies. The Reactor Health Physicist shall be responsible for:

1) Making periodic radiation surveys and reviews of operating practices. Any hazardous conditions are to be reported as above.
2) Supervising decontamination operations when necessary.
3) Supervising the periodic reading, calibration, and evaluation of radiation-measuring devices, including personnel dosimetry devices.
4) Recommending the availability of protective clothing and other safety devices, as required for the protection of personnel working at the facility, and instructing personnel in their use.
5) Serving as a non-voting member of the RRC.
6) Reviewing personnel exposure records and recommending procedural modifications to reduce exposures. Investigating any overexposures.
7) Reviewing and advising on emergency procedures. Recommending the availability of protective clothing and other safety devices for use in emergencies and instructing operators in their use.
8) Reviewing unusual levels of radioactivity released or discharged to the environment.
9) Reviewing and advising on environmental impacts and calculations of off-site dose rates from standard operations and emergencies.
10) Supervising the radiation safety aspects of special experiments as required by the RRC.

h) Senior Reactor Operators and Reactor Operators for the reactor are appointed by the Director and shall hold the corresponding license issued by the Nuclear Regulatory Commission. Specifically, each licensed operator is responsible for:

1) Proper shielding and storage of radioactive materials removed from the reactor, until they are turned over to a person authorized by the Reactor Director to receive them.
2) Participation in a required requalification program.
3) Provide training for license candidates and other groups who use the reactor for educational purposes.
4) Preparing the logs and records of reactor operations.
5) Operating the reactor in accordance with the administrative and operating procedures approved by the RRC and within the limitations of the Facility License and Technical Specifications.
6) The radiation safety of all personnel inside the facility during operation of the reactor in accordance with 10 CFR Part 20 and Oregon Regulations for Control of Radiation.
7) Insertion and removal of experiments as instructed by the Reactor Supervisor.

8) Reporting all unusual conditions and events pertaining to the reactor and its operation to the Reactor Supervisor.

i) The licensed operators may direct the activities of Assistants. If assigned to a reactor operation, an Assistant shall work under a licensed operator's direct supervision.

j) Reactor Assistants are appointed by the Director to work at the Facility under the supervision of the latter, the Health Physicist, or as directed by the Reactor Supervisor. Training for Assistants shall include radiation safety and emergency procedures.

6.1.3 Staffing

a) Whenever the reactor is not in the secured mode, the reactor shall be under the direction of a US NRC licensed Senior Operator. The Senior Operator shall be reachable by phone or pager, on campus, and within five minutes travel time to the facility.

b) Whenever the reactor is not in the secured mode, a US NRC licensed Reactor Operator (or Senior Reactor Operator) who meets requirements of the Operator Requalification Program shall be at the reactor control console, and directly responsible for reactivity manipulations.

c) Whenever the reactor is not secured, a second person shall be in the facility. The second person may leave the facility briefly to take readings or conduct inspections.

d) In addition to the above requirements, during fuel movement a Senior Operator shall be inside the reactor bay directing fuel operations. This may be the same Senior Operator required by 6.1.3.a.

6.2 REVIEW AND AUDIT

a) There will be a Reactor Review Committee (RRC) which shall review reactor operations to assure that the reactor facility is operated and used in a manner within the terms of the facility license and consistent with the safety of the public and of persons.

b) The responsibilities of the RRC include, but are not limited to, the following:

1) Review and approval of rules, procedures, and proposed Technical Specifications.

2) Review and approval of all proposed changes in the facility that could have a significant effect on safety and of all proposed changes in rules, procedures, and Technical Specifications, in accordance with procedures in Section 6.3.

3) Review and approval of experiments using the reactor in accordance with procedures and criteria in Section 6.4.

4) Determination of whether a proposed change, test, or experiment would constitute an unreviewed safety question or change in the Technical Specifications (Ref. 10 CFR 50.59).
5) Review of abnormal performance of equipment and operating anomalies.
6) Review of unusual or abnormal occurrences and incidents which are reportable under 10 CFR Part 20 and 10 CFR Part 50.
7) Inspection of the facility, review of safety measures, and audit of operations at a frequency not less than once a year, including operation and operations records of the facility. Standard Operating Procedures shall be audited biennially.

c) The RRC shall be composed of:
1) One or more persons proficient in reactor and nuclear science or engineering,
2) One or more persons proficient in chemistry, geology, or chemical engineering,
3) One person proficient in biological effects of radiation,
4) The Reactor Director, *ex officio*,
5) The Radiation Safety Officer, *ex officio*, and,
6) The Vice President, Dean of Faculty, *ex officio*, or a designated deputy.

d) The same individual may serve under more than one category above, but the minimum voting membership shall be seven. At least two members shall be Reed College faculty members. The Reactor Supervisor shall attend and participate in RRC meetings, but shall not be a voting member.

e) The RRC shall have a written statement defining its authority and responsibilities, the subjects within its purview, and other such administrative provisions as are required for its effective functioning. Minutes of all meetings and records of all formal actions of the RRC shall be kept by the Director.

f) A quorum shall consist of not less than a majority of the voting RRC members.

g) Any action of the RRC requires a majority vote of the members present.

h) The RRC shall meet a minimum of two times each academic year. Additional meetings may be called by the chair, and the RRC may be polled in lieu of a meeting. Such a poll shall constitute RRC action subject to the same requirements as for an actual meeting.

i) The Reactor Review Committee may be divided into two subcommittees (Reactor Operations Committee and Reactor Safety Committee).

6.3 *PROCEDURES*

a) Written procedures, reviewed and approved by the RRC, shall be followed for the activities listed below. The procedures shall be adequate to assure the safety of the reactor, persons within the facility, and the public, but should not preclude the use of independent judgment and action should the situation require it. The activities are:

1) Startup, operation, and shutdown of the reactor, including

(a) Startup procedures to test the reactor instrumentation and safety systems, area monitors, and continuous air monitors, and

(b) Shutdown procedures to assure that the reactor is secured before the end of the day.

2) Installation or removal of fuel elements, control rods, and other core components that significantly affect reactivity or reactor safety.

3) Preventive or corrective maintenance activities that could have a significant effect on the safety of the reactor or personnel.

4) Periodic inspection, testing, or calibration of systems or instrumentation that relate to reactor operation.

b) Substantive changes in the above procedures shall be made only with the approval of the RRC, and shall be issued to the personnel in written form. The Reactor Director may make temporary changes that do not change the original intent. The change and the reasons thereof shall be reviewed by the RRC.

6.4 *REVIEW OF PROPOSALS FOR EXPERIMENTS*

a) All proposals for new experiments involving the reactor shall be reviewed with respect to safety in accordance with the procedures in (b) below and on the basis of criteria in (c) below.

b) Procedures:

1) Proposed reactor operations by an experimenter are reviewed by the Reactor Director, who may determine that the operation is described by a previously approved experiment or procedure. If the Reactor Director determines that the proposed operation has not been approved by the RRC, the experimenter shall describe the proposed experiment in written form in sufficient detail for consideration of safety aspects. If potentially hazardous operations are involved, proposed procedures and safety measures including protective and monitoring equipment shall be described.

2) The proposal is then to be submitted to the RRC for consideration and approval.

3) The scope of the experiment and the procedures and safety measures as described in the approved proposal, including any amendments or conditions added by those reviewing and approving it, shall be binding

on the experimenter and the operating personnel. Minor deviations shall be allowed only in the manner described in Section 6.3b above.

4) Recorded affirmative votes on proposed new or revised experiments or procedures must indicate that the RRC determines that proposed actions do not involve unreviewed safety questions, changes in the facility as designed, or changes in Technical Specifications, and could be taken without endangering the health and safety of workers or the public or constituting a significant hazard to the integrity of the reactor core.

c) Criteria that shall be met before approval can be granted shall include:

1) The experiment must fall within the limitations given in Section 3.6.
2) The experiment must not involve violation of any condition of the facility license or of Federal, State, College, or Facility regulations and procedures. The possibility of an unreviewed safety question (10 CFR 50.59) must be examined.
3) In the safety review the basic criterion is that there shall be no hazard to the reactor, personnel, or public due to the experiment. The review shall determine that there is reasonable assurance that the experiment can be performed with no significant risk to the safety of the reactor, personnel or the public.

6.5 EMERGENCY PLAN AND PROCEDURES

An emergency plan shall be established and followed in accordance with NRC regulations. The plan shall be reviewed and approved by the RRC prior to its submission to the NRC. In addition, emergency procedures that have been reviewed and approved by the RRC shall be established to cover all foreseeable emergency conditions potentially hazardous to persons within the Facility or to the public, including, but not limited to, those involving an uncontrolled reactor excursion or an uncontrolled release of radioactivity.

6.6 OPERATOR REQUALIFICATION

An operator requalification program shall be established and followed in accordance with NRC regulations.

6.7 PHYSICAL SECURITY PLAN

Administrative controls for protection of the reactor shall be established and followed in accordance with NRC regulations.

6.8 ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED

In the event a safety limit is exceeded:

a) The reactor shall be shut down and reactor operation shall not be resumed until authorized by the Director, Division of Reactor Licensing, NRC.

b) An immediate report of the occurrence shall be made to the Chair of the RRC, and reports shall be made to the NRC in accordance with Section 6.11 of these specifications.

c) A report shall be made to include an analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to RRC for review, and a suitable similar report submitted to the NRC when authorization to resume operation of the reactor is sought.

6.9 ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE OCCURRENCE

a) A reportable occurrence is any of the following conditions:

1) Any actual safety system setting less conservative than specified in Section 2.2, Limiting Safety System Settings;
2) Violation of a Safety Limit, Limiting Safety System Setting, or Limiting Condition for Operation;
    - Violation of an LSSS or LCO occurs through failure to comply with an Action statement when Specification is not met; failure to comply with the Specification is not by itself a violation.
    - Surveillance Requirements must be met for all equipment, components, and conditions to be considered operable.
    - Failure to perform a surveillance within the required time interval or failure of a surveillance test shall result in the equipment, component, or condition being inoperable.
3) Incidents or conditions that prevented or could have prevented the performance of the intended safety functions of a SAFETY SYSTEM;
4) Release of fission products from the fuel that cause airborne contamination levels in the reactor bay to exceed 10 CFR Part 20 limits for releases to unrestricted areas;
5) An uncontrolled or unanticipated change in reactivity greater than \$1.00;
6) An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy has caused the existence or development of an unsafe condition in connection with the operation of the reactor; and

7) An uncontrolled or unanticipated release of radioactivity above permitted levels.

b) In the event of a reportable occurrence, the following actions shall be taken:

1) The reactor shall be shut down immediately. The Director shall be notified and corrective action taken before operations are resumed; the decision to resume shall require approval following the procedures in Section 6.3.
2) 2A report shall be made to include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to the RRC for review.
3) A report shall be submitted to the NRC in accordance with Section 6.11 of these specifications.

6.10 OPERATING RECORDS

a) In addition to the requirements of applicable regulations in 10 CFR Part 20 and 10 CFR Part 50 records and logs shall be prepared and retained for a period of at least 5 years for the following items as a minimum:

1) Normal operation, including power levels;
2) Principal maintenance activities;
3) Reportable occurrences;
4) Equipment and component surveillance activities;
5) Experiments performed with the reactor; and
6) All emergency reactor scrams, including reasons for emergency shutdowns.

b) The following records shall be maintained for the life of the facility:

1) Gaseous and liquid radioactive effluents released to the environs;
2) Offsite environmental monitoring surveys;
3) Fuel inventories and transfers;
4) Facility radiation and contamination surveys;
5) Radiation exposures for all personnel; and
6) Updated, corrected, and as-built drawings of the facility.

6.11 REPORTING REQUIREMENTS

All written reports shall be sent within the prescribed interval to the US Nuclear Regulatory Commission, Washington, DC 20555, Attn: Document Control Desk.

In addition to the requirements of applicable regulations reports shall be made to the NRC as follows:

a) A report within 24 hours by telephone and either fax or electronic mail to the NRC Operation Center of:

1) Any accidental release of radioactivity above permissible limits in unrestricted areas, whether or not the release resulted in property damage, personal injury, or exposure;
2) Any violation of a safety limit; and
3) Any reportable occurrences as defined in Section 6.9 of these specifications.

b) A report within 10 days in writing to the NRC Operation Center:

1) Any accidental release of radioactivity above permissible limits in unrestricted areas, whether or not the release resulted in property damage, personal injury or exposure; the written report (and, to the extent possible, the preliminary telephone report) shall describe, analyze, and evaluate safety implications, and outline the corrective measures taken or planned to prevent recurrence of the event;
2) Any violation of a safety limit; and
3) Any reportable occurrence as defined in Section 6.9 of these specifications.

c) A report within 30 days in writing to the Director, Non-Power Reactors and Decommissioning Project Directorate, US Nuclear Regulatory Commission, Washington, DC 20555 of:

1) Any significant variation of a measured value from a corresponding predicted or previously measured value of safety-connected operating characteristics occurring during operation of the reactor; and
2) Any significant change in the accident analysis as described in the Safety Analysis Report.

d) A report within 60 days after criticality of the reactor in writing to the NRC Operation Center, resulting from a receipt of a new facility license or an amendment to the license authorizing an increase in reactor power level or the installation of a new core, describing the measured value of the operating conditions or characteristics of the reactor under the new conditions.

e) A routine report in writing to the US Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, within 60 days after completion of the first calendar year of operating and at intervals not to exceed 14 months, thereafter, providing the following information:

1) A brief narrative summary of operating experience (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;
2) A tabulation showing the energy generated by the reactor (in megawatt-hours);
3) The number of emergency shutdowns and inadvertent scrams, including the reasons thereof and corrective action, if any, taken;

4) Discussion of the major maintenance operations performed during the period, including the effects, if any, on the safe operation of the reactor, and the reasons for any corrective maintenance required;
5) A summary of each change to the facility or procedures, tests, and experiments carried out under the conditions of 10 CFR 50.59;
6) A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or before the point of such release or discharge;
7) A description of any environmental surveys performed outside the facility; and
8) A summary of radiation exposures received by facility personnel and visitors, including the dates and time of significant exposure, and a brief summary of the results of radiation and contamination surveys performed within the facility.

Chapter 15

Financial Considerations

Reed Research Reactor Safety Analysis Report

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Chapter 15

Financial Considerations

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15 FINANCIAL QUALIFICATIONS

15.1 Financial Ability to Construct a Non-Power Reactor

This is not applicable for a renewal application.

15.2 Financial Ability to Operate a Non-Power Reactor

Table 15.1 shows the Reed Research Reactor (RRR) total budget for the five most recent years. Table 15.2 shows the projected budget for the next five years.

Table 15.1 RRR Budget History

Account	2003-04	2004-05	2005-06	2006-07	2007-08
Staff Salaries	92,000	93,800	95,800	97,600	99,500
Student Wages	10,000	10,500	14,000	17,000	15,000
Student Wages: Work-study	3,000	3,100	3,300	3,300	3,000
Supplies	8,700	8,960	9,700	13,900	13,500
Postage-Departmental	350	500	500	500	500
Telephone & Fax	100	100	100	100	1,300
Fees & Services	6,556	8,860	24,800	24,000	27,000
Duplicating/Printing	800	800	800	800	700
Conferences/Travel	4,200	4,200	6,200	6,000	5,000
Insurance	8,000	8,000	7,400	7,400	7,400
TOTAL	133,706	138,820	162,600	170,600	172,900

Table 15.2 RRR Projected Budget

Account	2008-09	2009-10	2010-11	2011-12	2012-13
Staff Salaries	101,500	103,500	105,600	107,700	109,900
Student Wages	15,800	16,600	17,400	18,300	19,200
Student Wages: Work-study	3,200	3,400	3,600	3,800	4,000
Supplies	13,900	14,300	14,700	15,100	15,600
Postage-Departmental	500	500	500	500	500
Telephone & Fax	1,300	1,300	1,300	1,300	1,300
Fees & Services	27,800	28,600	29,500	30,400	31,300
Duplicating/Printing	700	700	700	700	700
Conferences/Travel	5,200	5,400	5,600	5,800	6,000
Insurance	7,600	7,800	8,000	8,200	8,400
TOTAL	177,500	182,100	186,900	191,800	196,900

Funding is approved by the Board of Trustees of Reed Institute. The administration of the college has been very supportive. This application for renewal indicates this support.

While the RRR does perform some commercial irradiation, this represents only a very small percentage of either expense or income, less than 0.5% of the cost of operating the facility. In accordance with 10 CFR 50.21, the RRR should therefore be licensed as a Class 104 facility.

15.3 Financial Ability to Decommission the Facility

Reed College is a private institution. By letter dated September 30, 1999, the NRC indicated their acceptance that Reed College has met the requirement to self-guarantee funds for decommissioning by non-profit colleges. [1] By letters dated August 22, 2007, Reed College updated its commitments to provide self-guarantee of decommissioning costs, [1] and to carry out the decommissioning if and when the reactor is decommissioned. [2] The most recent audited financial report is included. [3]

By the letter dated March 9, 1992, the consultants to the RRR estimated the cost of decommissioning at $500,000. [4] The 1992 estimate has been updated as required by 10 CFR 50.75(g)(3) using data and methodology supplied in NUREG-1307 [5]. The current estimated cost of decommissioning is estimated at between $1 million and $1.5 million.

15.4 References

1) Letter from Marvin Mendonca, U.S. Nuclear Regulatory Commission, to Reed College dated September 20, 1999.
2) Letter from Colin Diver, Reed College, to U.S. Nuclear Regulatory Commission dated August 22, 2007.
3) Letter from Edwin McFarlane, Reed College, to U.S. Nuclear Regulatory Commission dated August 22, 2007.
4) Letter from Ronald Katheren, Washington State University and George Miller, University of California, to Reed College dated March 9, 1992.
5) U.S. Nuclear Regulatory Commission, "Report on Waste Burial Charges: Changes in Decommissioning Waste Disposal Costs at Low-Level Waste Burial Facilities," NUREG-1307, Rev. 10.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 20, 1999

Mr, Stephen G. Frantz, Director
Reed Reactor Facility
Reed College
3203 SE Woodstock Boulevard
Portland, OR 97202

SUBJECT: SELF GUARANTEE OF DECOMMISSIONING FUNDS FOR THE REED COLLEGE RESEARCH REACTOR FACILITY IN ACCORDANCE WITH APPENDIX E TO 10 CFR PART 30

Dear Mr. Frantz:

We have completed our review of the Reed College letter dated March 10, 1999, as supplemented on August 27, 1999, relating to use of financial tests and self-guarantee for providing reasonable assurance of funds for decommissioning by non-profit colleges, universities and hospitals. Reed College showed that they have an unrestricted endowment consisting of assets located in the United States valued at $213 Million. This unrestricted endowment is well above the $50 Million minimum required. Further, Reed College has secured an independent auditor's report from KPMG Peat Marwick, LLP, verifying this endowment. Finally, Mr. Steven Koblik, President of Reed College, has committed that funds will be provided to carry out the required decommissioning activities, at the time of decommissioning. Therefore, Reed College has met requirement to self-guarantee funds for decommissioning by non-profit colleges.

Sincerely,

Marvin M. Mendonca, Senior Project Manager
Events Assessment, Generic Communications
and Non-Power Reactors Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No. 50-288

cc: See next page

REED COLLEGE

OFFICE OF
THE PRESIDENT

3203 Southeast
Woodstock Boulevard
Portland, Oregon
97202-8199

telephone
503/777-7500
fax
503/777-7701

August 22, 2007

Mr. Marvin M. Mendonca, Sr. Project Manager
Non-Power Reactors and Decommissioning
Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation
Mail Stop 011D19
Washington, D.C. 20555-0001

Dear Mr. Mendonca:

The purpose of this letter is to provide the Nuclear Regulatory Commission with assurance that Reed College will fund and carry out the required decommissioning activities for its TRIGA Mark I Research Reactor if and when this reactor is decommissioned.

Currently, we have no plans to decommission the reactor. Therefore, for purposes of planning, we are assuming that the reactor will continue to operate under a renewed license that will not expire until October 3, 2027. The College will continue to provide adequate annual funding for the safe operation of the reactor.

We have provided your office with documentation as required in accordance with Appendix E to Part 30, Items I, IIA, and IIC, indicating Reed College's ability to provide self-guarantee of decommissioning costs.

Sincerely,

Colin S. Diver
President

cc: Stephen Frantz
Peter Steinberger
Edwin O. McFarlane

REED COLLEGE

August 22, 2007

OFFICE OF
THE TREASURER

3203 Southeast
Woodstock Boulevard
Portland, Oregon
97202-8199

telephone
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emcfarl@reed.edu

Mr. Marvin M. Mendonca, Sr. Project Manager
Non-Power Reactors and Decommissioning
Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation
Mail Stop 011D19
Washington, D.C. 20555-0001

Dear Mr. Mendonca:

Thank you for researching the matter regarding the ability of colleges to self-guarantee decommissioning costs. After reviewing the criteria in Appendix E to Part 30, Items I, IIA, IIC and IIID, it appears that Reed College does meet the requirements necessary to provide the self-guarantee.

Therefore, enclosed are the following documents in support of our request that Reed College be approved as a college providing self-guarantee of decommissioning costs:

- A written guarantee signed by Reed's president that Reed College will fund and carry out the required decommissioning activities if and when the Reed College Reactor is decommissioned;
- The June 12, 2007, letter from Standard and Poor's indicating that Reed College has been rated at AA-; and,
- The latest audited financial statement for the year ended June 30, 2006, which shows that Reed has unrestricted endowment funds of approximately $300,000,000. Please see Footnote 10 as presented on page 19.

I believe the above documentation should be sufficient to allow you to grant approval to Reed to provide self-guarantee of decommissioning cost.

Thank you for your consideration of this request and I look forward to your reply.

Sincerely,

Edwin O. McFarlane
Vice President/Treasurer

cc: Stephen Frantz

STANDARD
&POOR'S

One Market
Steuart Tower, 15th Floor
San Francisco, CA 94105-1000
tel 415 371-5004
reference no.: 838690

June 12, 2007

Reed College
3203 SE Woodstock Boulevard
Portland, OR 97202-8199
Attention: Mr. Edwin O. McFarlane, Vice President / Treasurer

Re: ***US$32,100,000 Oregon Facility Authority, Oregon, Auction Rate Certificates, (Reed College), Series 2007, dated: Date of Delivery, due: June 1, 2038***

Dear Mr. McFarlane:

Pursuant to your request for a Standard & Poor's rating on the above-referenced obligations, we have reviewed the information submitted to us and, subject to the enclosed *Terms and Conditions*, have assigned a rating of "AA-". Standard & Poor's views the outlook for this rating as positive. A copy of the rationale supporting the rating is enclosed.

The rating is not investment, financial, or other advice and you should not and cannot rely upon the rating as such. The rating is based on information supplied to us by you or by your agents but does not represent an audit. We undertake no duty of due diligence or independent verification of any information. The assignment of a rating does not create a fiduciary relationship between us and you or between us and other recipients of the rating. We have not consented to and will not consent to being named an "expert" under the applicable securities laws, including without limitation, Section 7 of the Securities Act of 1933. The rating is not a "market rating" nor is it a recommendation to buy, hold, or sell the obligations.

This letter constitutes Standard & Poor's permission to you to disseminate the above-assigned rating to interested parties. Standard & Poor's reserves the right to inform its own clients, subscribers, and the public of the rating.

Standard & Poor's relies on the issuer/obligor and its counsel, accountants, and other experts for the accuracy and completeness of the information submitted in connection with the rating. This rating is based on financial information and documents we received prior to the issuance of this letter. Standard & Poor's assumes that the documents you have provided to us are final. If any subsequent changes were made in the final documents, you must notify us of such changes by sending us the revised final documents with the changes clearly marked.

To maintain the rating, Standard & Poor's must receive all relevant financial information as soon as such information is available. Placing us on a distribution list for this information would facilitate the process. You must promptly notify us of all material changes in the financial information and the documents. Standard & Poor's may change, suspend, withdraw, or place on

CreditWatch the rating as a result of changes in, or unavailability of, such information. Standard & Poor's reserves the right to request additional information if necessary to maintain the rating.

Please send all information to:

Standard & Poor's Ratings Services
Public Finance Department
55 Water Street
New York, NY 10041-0003

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Sincerely yours,

Standard & Poor's Ratings Services
a division of The McGraw-Hill Companies, Inc.

Standard + Poor's/ap

zt
enclosures
cc: Mr. C. Harper Watters
Mr. Gwendolyn Griffith

THE REED INSTITUTE

Financial Statements

June 30, 2006 and 2005

(With Independent Auditors' Report Thereon)

KPMG

KPMG LLP
Suite 3800
1300 South West Fifth Avenue
Portland, OR 97201

Independent Auditors' Report

The Board of Trustees
The Reed Institute:

We have audited the accompanying statements of financial position of The Reed Institute as of June 30, 2006 and 2005, and the related statements of activities and changes in net assets, and cash flows for the years then ended. These financial statements are the responsibility of The Reed Institute's management. Our responsibility is to express an opinion on these financial statements based on our audits.

We conducted our audits in accordance with auditing standards generally accepted in the United States of America. Those standards require that we plan and perform the audit to obtain reasonable assurance about whether the financial statements are free of material misstatement. An audit includes consideration of internal control over financial reporting as a basis for designing audit procedures that are appropriate in the circumstances, but not for the purpose of expressing an opinion on the effectiveness of the College's internal control over financial reporting. Accordingly, we express no such opinion. An audit also includes examining, on a test basis, evidence supporting the amounts and disclosures in the financial statements, assessing the accounting principles used and significant estimates made by management, as well as evaluating the overall financial statement presentation. We believe that our audits provide a reasonable basis for our opinion.

As discussed in note 3, the Reed Institute has approximately 65% of its total investments, or approximately $277.1 million, in investments where readily determinable fair values do not exist.

In our opinion, the financial statements referred to above present fairly, in all material respects, the financial position of The Reed Institute as of June 30, 2006 and 2005, and the changes in its net assets and its cash flows for the years then ended in conformity with accounting principles generally accepted in the United States of America.

As discussed in Notes 2 and 4 to the financial statements, the Reed Institute changed its method of accounting for conditional asset retirement obligations in 2006.

KPMG LLP

December 1, 2006

THE REED INSTITUTE

Statements of Financial Position

June 30, 2006 and 2005

Assets		2006	2005
Current assets:			
Cash and cash equivalents (note 5)	$	18,737,562	2,413,132
Accounts receivable – student and other (note 9)		506,865	664,299
Investments (note 3)		185,710	—
Contributions receivable, net of allowance $10,000 in 2006 and $9,000 in 2005 (note 9)		192,628	172,961
Prepaid expenses and other assets		1,369,643	1,295,763
Total current assets		20,992,408	4,546,155
Noncurrent assets:			
Cash and cash equivalents whose use is limited		1,766,088	—
Accounts receivable noncurrent – student and other, net of allowance of $60,239 in 2006 and 2005 (note 9)		4,569,214	4,342,580
Property, plant, and equipment, net (note 4)		87,797,945	90,875,650
Contributions receivable – noncurrent net of allowance of $251,000 in 2006 and $125,000 in 2005 (note 9)		4,735,418	2,338,871
Funds held in trust by others (note 8)		13,228,229	10,518,495
Long-term investments (note 3)		423,766,192	376,110,009
Other assets		743,660	1,394,626
Total noncurrent assets		536,606,746	485,580,231
Total assets	$	557,599,154	490,126,386
Liabilities and Net Assets			
Current liabilities:			
Accounts payable and accrued liabilities	$	5,398,070	4,314,313
Debt and capital leases, net of discount costs (note 5)		17,403,133	707,666
Deferred revenue		4,484,301	1,391,437
Total current liabilities		27,285,504	6,413,416
Long-term liabilities:			
Accrued liabilities		217,732	292,606
Liability for split interest agreements		9,186,547	7,172,936
Postretirement benefits payable (note 7)		16,524,029	18,710,870
Refundable loan programs		3,160,397	3,238,581
Asset retirement obligations (note 4)		2,149,760	—
Debt and capital leases, net of discount costs (note 5)		37,664,734	37,151,852
Total long-term liabilities		68,903,199	66,566,845
Total liabilities		96,188,703	72,980,261
Net assets (note 10):			
Unrestricted		363,525,924	333,247,026
Temporarily restricted		25,285,709	18,223,114
Permanently restricted		72,598,818	65,675,985
Total net assets		461,410,451	417,146,125
Total liabilities and net assets	$	557,599,154	490,126,386

See accompanying notes to financial statements.

THE REED INSTITUTE

Statement of Activities and Changes in Net Assets

Year ended June 30, 2006

	Unrestricted	Temporarily restricted	Permanently restricted	Total 2006
Revenues, gains, and other support:				
Tuition and fees	$ 39,825,893	—	—	39,825,893
Less college-funded scholarships	(14,296,440)	—	—	(14,296,440)
Net tuition and fees	25,529,453	—	—	25,529,453
Auxiliary enterprises	9,350,356	—	—	9,350,356
Gifts and private grants	3,758,001	6,133,774	6,533,669	16,425,444
Government grants, contracts, and student aid	1,250,618	—	—	1,250,618
Endowment investment income	3,634,407	—	—	3,634,407
Realized and unrealized gains and losses	43,360,754	—	—	43,360,754
Other investment income	1,599,120	—	—	1,599,120
Other revenues and additions	629,611	—	7,703	637,314
Subtotal	63,582,867	6,133,774	6,541,372	76,258,013
Net assets released from restrictions	2,360,958	(2,360,958)		—
Total revenues, gifts, and other support	91,473,278	3,772,816	6,541,372	101,787,466
Expenses:				
Educational and general:				
Instruction	20,633,174	—	—	20,633,174
Research	1,074,595	—	—	1,074,595
Academic support	7,072,410	—	—	7,072,410
General institutional support	8,587,554	—	—	8,587,554
Student services	4,498,232	—	—	4,498,232
Public affairs	3,853,703	—	—	3,853,703
Total educational and general	45,719,668	—	—	45,719,668
Auxiliary enterprises	12,933,123			12,933,123
Total expenses	58,652,791	—	—	58,652,791
Increase from operations	32,820,487	3,772,816	6,541,372	43,134,675
Nonoperating activity:				
Other interest expense	(365,249)	—	—	(365,249)
Change in value of split interest agreements	—	3,289,779	381,461	3,671,240
Other deductions	(456,371)	—	—	(456,371)
Total nonoperating activity	(821,620)	3,289,779	381,461	2,849,620
Increase in net assets before cumulative effect of change in accounting principle	31,998,867	7,062,595	6,922,833	45,984,295
Cumulative effect of change in accounting principle (note 4)	(1,719,969)	—	—	(1,719,969)
Increase in net assets	30,278,898	7,062,595	6,922,833	44,264,326
Net assets, beginning of year	333,247,026	18,223,114	65,675,985	417,146,125
Net assets, end of year	$ 363,525,924	25,285,709	72,598,818	461,410,451

See accompanying notes to financial statements.

THE REED INSTITUTE

Statement of Activities and Changes in Net Assets

Year ended June 30, 2005

	Unrestricted	Temporarily restricted	Permanently restricted	Total 2005
Revenues, gains, and other support:				
Tuition and fees	$ 37,504,104	—	—	37,504,104
Less college-funded scholarships	(13,569,575)	—	—	(13,569,575)
Net tuition and fees	23,934,529	—	—	23,934,529
Auxiliary enterprises	8,357,454	—	—	8,357,454
Gifts and private grants	2,905,442	1,303,190	7,458,733	11,667,365
Government grants, contracts, and student aid	1,432,964	—	—	1,432,964
Endowment investment income	1,532,276	—	—	1,532,276
Realized and unrealized gains and losses	23,630,526	—	—	23,630,526
Other investment income	1,671,301	—	—	1,671,301
Other revenues and additions	447,088	—	5,000	452,088
Subtotal	39,977,051	1,303,190	7,463,733	48,743,974
Net assets released from restrictions	2,322,274	(2,322,274)	—	—
Total revenues, gifts, and other support	66,233,854	(1,019,084)	7,463,733	72,678,503
Expenses:				
Educational and general:				
Instruction	22,048,799	—	—	22,048,799
Research	1,148,940	—	—	1,148,940
Academic support	6,382,357	—	—	6,382,357
General institutional support	12,696,940	—	—	12,696,940
Student services	4,136,347	—	—	4,136,347
Public affairs	3,359,187	—	—	3,359,187
Total educational and general	49,772,570	—	—	49,772,570
Auxiliary enterprises	11,725,175	—	—	11,725,175
Total expenses	61,497,745	—	—	61,497,745
Increase (decrease) from operations	4,736,109	(1,019,084)	7,463,733	11,180,758
Nonoperating activity:				
Other interest expense	(212,550)	—	—	(212,550)
Change in value of split interest agreements	—	(891,243)	381,480	(509,763)
Other deductions	(74,156)	—	—	(74,156)
Reclassification of net assets	(2,926,887)	2,926,887	—	—
Total nonoperating activity	(3,213,593)	2,035,644	381,480	(796,469)
Increase in net assets	1,522,516	1,016,560	7,845,213	10,384,289
Net assets, beginning of year	331,724,510	17,206,554	57,830,772	406,761,836
Net assets, end of year	$ 333,247,026	18,223,114	65,675,985	417,146,125

See accompanying notes to financial statements.

THE REED INSTITUTE

Statements of Cash Flows

Years ended June 30, 2006 and 2005

		2006	2005
Cash flows from operating activities:			
Increase in net assets	$	44,264,326	10,384,289
Adjustments to reconcile change in net assets to net cash used in operating activities:			
Depreciation and amortization costs		4,443,368	4,281,534
Gain on sale of equipment		—	(268,819)
Contributions restricted for long-term investment		(11,200,266)	(8,826,643)
Noncash contributions		(4,026,494)	(5,426,151)
Net realized and unrealized gain on investments		(43,891,691)	(21,594,196)
Change in value of split interest agreements		(3,671,240)	509,763
Change in fair value of derivative instruments		687,519	(1,767,511)
Cumulative effect of change in accounting principle		1,719,969	—
Changes in operating assets and liabilities:			
Increase in cash and cash equivalents whose use is limited		(1,766,088)	—
Increase in accounts receivable – students and other		(69,200)	(629,164)
(Increase) decrease in contributions receivable		(2,416,214)	1,110,422
Decrease (increase) in prepaid expenses and other assets		(78,206)	5,214
Increase in accounts payable and accrued liabilities		1,008,883	430,081
Increase in obligations for split interest agreements		3,718,814	1,458,114
Decrease in Grayco environmental cleanup accrual		—	(525,000)
(Decrease) increase in postretirement benefits payable		(2,186,841)	4,932,998
Increase in deferred revenue		3,092,864	307,104
Net cash used in operating activities		(10,370,497)	(15,617,965)
Cash flows from investing activities:			
Proceeds from maturities/sales of investments		134,721,192	120,905,096
Purchases of investments		(134,893,542)	(111,958,375)
Contracts receivable advanced		(37,000)	(53,517)
Contracts receivable collected		29,290	21,398
Proceeds from sale of equipment		520,000	953,911
Purchase of property, plant, and equipment		(282,640)	(2,171,950)
Net cash provided by investing activities		57,300	7,696,563
Cash flows from financing activities:			
Contributions restricted for long-term investment		11,200,266	8,826,643
Issuance of new debt		16,650,000	—
Payment of debt principal and capital lease obligations		(119,400)	(709,133)
Payments of obligations for split interest agreements		(1,015,055)	(825,979)
Changes in governmental loan funds		(78,184)	15,648
Net cash provided by financing activities		26,637,627	7,307,179
Net increase (decrease) in cash and cash equivalents		16,324,430	(614,223)
Cash and cash equivalents, beginning of year		2,413,132	3,027,355
Cash and cash equivalents, end of year	$	18,737,562	2,413,132
Supplemental disclosures:			
Noncash gifts expensed	$	19,863	16,676
Interest paid		2,394,717	2,234,086
Assets acquired under capital leases		382,357	22,000

See accompanying notes to financial statements.

(1) Background

The Reed Institute (Reed College), founded in 1908 by Simeon and Amanda Reed, is today one of the nation's preeminent institutions of the liberal arts and sciences. The Reed College educational program pays particular attention to a balance between broad study in the various areas of human knowledge and close, in-depth study in a recognized academic discipline.

(2) Summary of Significant Accounting Policies

(a) Accrual Basis

The financial statements of Reed College have been prepared on the accrual basis of accounting in accordance with accounting principles generally accepted in the United States of America.

(b) Basis of Presentation

Net assets, revenues, expenses, gains and losses are classified based on the existence or absence of donor-imposed restrictions. The definitions used to classify and report net assets are as follows:

- Unrestricted net assets – net assets that are not subject to donor-imposed restrictions.
- Temporarily restricted net assets – net assets subject to donor-imposed restrictions that will be met either by actions of Reed College or the passage of time.
- Permanently restricted net assets – net assets subject to donor-imposed restrictions that they be permanently maintained by Reed College. Generally, the donors of these assets permit Reed College to use all or part of the income earned on related investments for general or specific purposes.

Revenues are reported as increases in unrestricted net assets unless their use is limited by donor-imposed restrictions. All expenses are reported as decreases in unrestricted net assets with the exception of activity related to split interest agreements. Gains and losses on investments and other assets or liabilities are reported as increases or decreases in unrestricted net assets unless their use is restricted either by donor stipulation or by law. Expirations of temporary restrictions are reported as reclassifications between the applicable classes of net assets and are reported as net assets released from restrictions in the statement of activities and changes in net assets.

Income and net gains on investments of endowment and similar funds are reported as follows:

- Increases in permanently restricted net assets if the terms of the gift or Reed College's interpretation of relevant state law require they be added to the principal of a permanently restricted net asset.
- Increases in temporarily restricted net assets if the terms of the gift impose restrictions on the use of the income.
- Increases in unrestricted net assets in all other cases.

(c) ***Use of Estimates***

The preparation of financial statements in conformity with accounting principles generally accepted in the United States of America requires management to make estimates and assumptions that affect reported amounts of assets and liabilities at the date of the financial statements and the reported amounts of revenues and expenses during the reporting period. Actual results could differ from those estimates.

(d) Revenues and Revenue Distribution

The principal sources of revenue, consisting of tuition, room and board, various other educational fees, unrestricted income from funds functioning as endowments, unrestricted gifts, and net assets released from restrictions, are accounted for as increases to unrestricted net assets. Unrestricted net assets also include revenue from grants, auxiliary enterprises, endowment gains and gains on disposal of assets.

Prepayments of student tuition and fees related to future academic years are deferred and recognized as revenues in the appropriate year.

The following assets have become available for general operating purposes from release from donor restrictions through the passage of time and through the maturation of various planned giving agreements for the years ended June 30, 2006 and 2005, respectively:

		2006	2005
Maturation of planned giving agreements	$	261,400	36,133
Passage of time		2,099,558	2,286,141
Total net assets released from restrictions	$	2,360,958	2,322,274

With a few exceptions, the monies in the endowment and similar funds are invested as a pool, and the related income of the pool is distributed to each participating fund based upon a spending formula and its relative proportion of the pool.

Reed College utilizes the "total return" method of pooled investment management. This technique considers both realized and unrealized appreciation or depreciation in the market value of investments, in addition to conventional income sources such as dividends, interest, and rents, net of investment fees as being part of current return. Based on this method, a predetermined percentage of the total return of the endowment funds (computed on a thirteen-quarter moving average market value) is made available each year for operating purposes.

In addition, monies that are not required to meet short-term demands are combined and invested. The income earned on these intermediate investments is allocated to each participating fund based upon its relative proportion of the combined investment.

(e) ***Investments***

Investments are stated at fair value. The fair value of all debt and equity securities with a readily determinable fair value are based on quotations obtained from national securities exchanges. The alternative investments, which are not readily marketable, are carried at estimated fair values as provided by the investment managers. Reed College reviews and evaluates the values provided by the investment managers and agrees with the valuation methods and assumptions used in determining the fair value of the alternative investments. Those estimated fair values may differ significantly from the values that would have been used had a ready market for those securities existed. Reed College has certain investments in real estate and related assets that are recorded at cost when purchased or fair value on the date of gift, as appropriate.

With the exception of split interest agreements, gains and losses arising from the sale, collection or other disposition of investments and other noncash assets are accounted for in unrestricted net assets.

(f) ***Split Interest Agreements***

Reed College has been named as a remainder beneficiary for various split interest agreements. Each agreement provides for contractual payments to stated beneficiaries for their lifetimes, after which the remaining principal and interest revert to Reed College. Assets contributed are recorded at fair value. In addition, Reed College has recognized the present value of estimated future payments to be made to beneficiaries over their expected lifetimes as a liability. The present values of these estimated payments are determined on the basis of published actuarial factors for ages of the respective beneficiaries discounted using various rate tables. Annual adjustments are made between the liability and net assets to record actuarial gains or losses. Differences between the assets contributed and expected payments to be made to beneficiaries are recorded as donations in the year established. These donations are either temporarily restricted on the basis of time or permanently restricted based on the intent of the donor.

(g) ***Grants and Contracts***

Revenues and reimbursements receivable from research and instructional grants and contracts are recorded at the time when reimbursable costs are incurred. Indirect cost support for these grants and contracts is generally based upon a standard rate negotiated with the U.S. Department of Health and Human Services.

(h) ***Contributions Receivable***

Unconditional promises to give (contributions) are recorded as gifts and private grant income and contributions receivable. Conditional promises to give are not recognized until they become unconditional, that is, when the donor-imposed restrictions are substantially met. Contributions other than cash are recorded at their estimated fair value. Management estimates an allowance for uncollectible contributions based on risk factors such as prior collection history, type of contribution, and the nature of the fund-raising activity. Contributions are generally receivable within five years of the date the commitment was made and are discounted to present value using a discount rate commensurate with the risk involved.

(Continued)

(i) Derivative Instruments

Reed College accounts for derivatives of interest rate swaps in accordance with Financial Accounting Standards Board (FASB) Statement No. 133, *Accounting for Derivative Instruments and Certain Hedging Activities*, as amended, which requires that all derivative instruments be recorded on the statement of financial position at their respective fair values.

(j) Property, Plant, and Equipment, Net

Property, plant, and equipment are stated at cost at the date of acquisition, if purchased, or at fair market value, at the date of receipt, if acquired by donation. Depreciation is computed on a straight-line basis over the estimated useful lives of buildings (twenty to fifty years) and equipment and furniture (five years). Routine repair and maintenance expenses and equipment replacement costs are expensed as incurred.

(k) Donated Materials

Donated materials are included in the statement of activities and changes in net assets as gifts and private grants at their estimated values at date of receipt when such values are communicated to Reed College by the donor. These materials are subsequently expensed.

(l) Income Tax Status

As a qualified educational institution under the provisions of Section 501(c)(3) of the Internal Revenue Code, Reed College is exempt from federal and state income taxes on related activities. No tax provision has been made in the accompanying financial statements.

(m) Cash and Cash Equivalents

Cash and cash equivalents represent cash and other liquid investments with original maturities of three months or less. Cash and cash equivalents whose use is limited are restricted for the Federal Perkins Loan program.

(n) Deferred Revenue

Deferred revenue consists primarily of prepayments of tuition and fees related to future academic years.

(o) Concentration of Risk

Reed College's standard financial instruments include commercial paper, U.S. Government and agency securities, corporate obligations, equity securities, mutual funds, insurance contracts and real estate. These financial instruments may subject Reed College to risk as cash balances may exceed amounts insured by the Federal Deposit Insurance Corporation and the value of securities is dependent on the ability of the issuer to honor its contractual commitments. The investments are subject to fluctuations in fair value (see note 3).

(Continued)

(p) ***New Accounting Pronouncements***

In March 2005, the Financial Accounting Standards Board (FASB) issued Interpretation No. 47 (FIN 47), *Accounting for Conditional Asset Retirement Obligations, an interpretation of FASB Statement No. 143* (SFAS 143). The summary to FIN 47 states that this interpretation was issued to address diverse accounting practices with respect to the timing of liability recognition for legal obligations associated with the retirement of a tangible long-lived asset when the timing and/or method of settlement of the obligation are conditional on a future event. FIN 47 clarifies that the obligation to perform the asset retirement activity is unconditional even though uncertainty exists about the timing and/or method of settlement. Accordingly, an entity is required to recognize a liability for the fair value of a conditional asset retirement obligation if the fair value can be reasonably estimated. Reed College adopted FIN 47 as of June 30, 2006. See Note 4 for additional information.

(q) ***Reclassifications***

Certain reclassifications have been made to prior year amounts to conform with current year presentation.

(3) Investments

The fair values of investments at June 30, 2006 and 2005 are as follows:

	Fair value	
	2006	2005
Investments:		
Short-term investments	$ 185,710	—
Mutual funds	141,932,344	181,978,559
Government fixed income	1,732,350	1,049,954
Corporate fixed income	—	1,048,795
Hedge funds	240,269,330	167,569,882
Private equity	36,788,632	18,398,518
Real estate	1,720,981	3,141,972
Money market and other	1,322,555	2,922,329
Total investments	$ 423,951,902	376,110,009

At June 30, 2006, Reed College has approximately $277.1 million in investments which are not readily marketable. These investments represent 65% of total investments and 60% of net assets at June 30, 2006. These investment instruments may contain elements of both credit and market risk. Such risks include, but are not limited to, limited liquidity, absence of regulatory oversight, dependence upon key individuals, emphasis on speculative investments (both derivatives and non-marketable investments), and nondisclosure of portfolio composition. Because these investments are not readily marketable, their estimated value is subject to uncertainty and therefore may differ from the value that would have been used had a ready market for such investments existed. Such difference could be material.

Reed College has funds invested in fourteen limited partnerships with ownership interests ranging from 0.41% to 11.86% at June 30, 2006 and 1.3% to 15.3% at June 30, 2005. Ten of these partnerships are private equity funds and four are hedge funds. Included in the assets of the various partnerships is a small portion of derivative instruments. Overall, these derivative instruments represent 2% of the College's total investments. The majority are acquired through exchange related transactions with the counter parties being the respective exchanges on which they are traded.

Total investment income and realized and unrealized gains (losses) on investments which are not readily marketable was $25,126,938 and $16,003,099 for the years ended June 30, 2006 and 2005, respectively.

(4) Property, Plant, and Equipment, Net

Property, plant, and equipment, net at June 30, 2006 and 2005, consists of the following:

		2006	2005
Land and land improvements	$	9,858,121	9,858,121
Buildings and improvements		113,210,257	112,413,818
Construction in progress		—	10,000
Equipment, furniture, and fixtures		16,495,034	16,125,533
		139,563,412	138,407,472
Less accumulated depreciation		(51,765,467)	(47,531,822)
Net property, plant, and equipment	$	87,797,945	90,875,650

Reed College has identified asbestos abatement as a conditional asset retirement obligation. Asbestos abatement costs were estimated using a per square foot estimate of the areas containing asbestos. As of June 30, 2006, Reed College recorded site improvements of $796,439, accumulated depreciation of $366,648, an asset retirement obligation of $2,149,760, and a cumulative effect of change in accounting principle of $1,719,969.

(5) Long-Term Debt

(a) Capital Lease Obligations

Reed College leases copiers over various terms. During the year ended June 30, 2006, Reed College recorded one new lease as a capital lease obligation, replacing two previous leases. The new lease is included in the accompanying financial statements under current and long-term debt net of amortization costs, and also in equipment, furniture, and fixtures in the amount of $389,971. Amortization costs of $10,773 and $75,104 are included in accumulated depreciation for the years ended June 30, 2006 and 2005, respectively.

The payment schedule for the capital lease obligation is as follows:

2007	$	85,434
2008		85,434
2009		85,434
2010		83,047
2011		67,324
		406,673
Less amount representing interest		14,556
	$	392,117

(b) Notes Payable

Reed College borrowed $20,000,000 from the State of Oregon on May 1, 2000. The purpose of the issuance was to finance the construction of certain renovations, additions, alterations, and improvements to the premises and educational facilities of Reed College, and the equipping, furnishing and landscaping thereof. The full amount borrowed, net of unamortized discount and issuance costs, was expended on projects during the year ended June 30, 2002. The notes bear interest from 5.00% to 5.75% and mature in varying amounts from 2005 to 2032.

Reed College borrowed $14,825,000 from the State of Oregon on May 1, 1991. Effective December 1, 1995, Reed College refinanced all but $1,565,000 of the 1991 State of Oregon notes payable and borrowed an additional $7,105,000.

Effective June 7, 2006, Reed College refinanced the callable portion of the 1995 State of Oregon notes payable in the amount of $16,650,000. The 2006 State of Oregon Notes mature July 1, 2025 and bear interest at a variable rate set on a weekly basis by a dutch auction process (3.85% at June 30, 2006). As of June 30, 2006, the College had not remitted the refinanced portion to the bank and $15,935,000 is included in cash and cash equivalents and current portion of the debt and capital leases, net of discount costs. Subsequent to year-end, in July 2006, the College paid the refinanced portion.

(Continued)

Principal payments on the notes payable and bonds become due as follows:

	2000 State of Oregon notes	1995 State of Oregon notes	2006 State of Oregon notes	Total
2007	$ 230,000	17,085,000	50,000	17,365,000
2008	120,000	620,000	75,000	815,000
2009	130,000	650,000	75,000	855,000
2010	135,000	685,000	75,000	895,000
2011	140,000	—	800,000	940,000
Thereafter	18,940,000	—	15,575,000	34,515,000
	$ 19,695,000	19,040,000	16,650,000	55,385,000

Interest on the State of Oregon notes payable bonds and amortization of discount and issuance costs are as follows:

	2006	2005
Interest	$ 2,173,758	2,171,276
Amortization of discount and issuance costs	648,659	62,810
Total interest expensed	$ 2,822,417	2,234,086

Notes payable discount, net of amortization was $709,250 and $429,861 at June 30, 2006 and 2005, respectively. Issuance cost, which is included in other assets, net of amortization was $1,012,607 and $378,864 at June 30, 2006 and 2005, respectively. Amortization is calculated over the life of the notes.

(c) Interest Rate Risk Management

In order to take advantage of low interest rates in long-term interest rates, Reed College entered into interest rate swap agreements, which allow the College to change the long-term fixed interest rate to a variable rate on the State of Oregon notes payable. In May 2000, Reed College entered into a fixed to variable interest rate swap with an investment bank (effective June 1, 2000 and maturing June 1, 2030) with an option granting the investment bank the ability to cancel the interest rate swap (effective June 1, 2007 and maturing June 1, 2030) for a $20 million notional amount. In May 2003, Reed College entered into an additional fixed to variable interest rate swap with an investment bank (effective June 1, 2003 and maturing June 1, 2007) for a $20 million notional amount. In May 2006, Reed College refunded the callable portion of its 1995 State of Oregon notes by issuing $16.65 million of auction rate debt through the Oregon Facilities Authority. The College entered into an interest rate swap of like term, amortization and notional amount with an investment bank to hedge this underlying variable rate debt. Pursuant to this swap, the College is receiving a variable interest rate and paying a fixed interest rate. Reed College works with a consulting firm to aid in monitoring changes in interest rates and the impact they may have on long-term debt.

During the years ended June 30, 2006 and 2005, $655,193 and $1,192,223 was received, respectively, and is recorded in the statement of activities and changes in net assets as other investment income. The change in unrealized gain and loss on the swap agreements and option for the years ended June 30, 2006 and 2005 was a $687,520 loss and $1,767,511 gain, respectively, and are recorded in the statement of activities and changes in net assets as realized and unrealized gains and losses. The fair value of the swap agreements and option as of June 30, 2006 and 2005 was $236,257 and $923,777, respectively, which are recorded in other assets in the statement of financial position.

(6) Operating Leases

Reed College leases copiers over various terms. Future minimum payments pertaining to operating leases are as follows:

2007	$	29,432
2008		6,684
2009		822

Expenses incurred for operating leases were $98,784 and $115,344 for the years ended June 30, 2006 and 2005, respectively.

(7) Pension and Postretirement Benefits

Reed College has a defined contribution noncontributory pension plan administered through Teachers Insurance and Annuity Association – College Retirement Equities Fund. Certain employees are eligible to participate and must be employed one year and have attained the age of twenty-one. All contributions vest immediately with the employee at the rate of 10% of the participating employees' monthly compensation. Reed College's policy is to fund pension expenses as incurred. Expenses relating to this plan were $2,050,916 and $1,956,314 for the years ended June 30, 2006 and 2005, respectively.

Reed College maintains a defined benefit retiree medical insurance plan, which is administered by Pioneer Educators Health Trust (PEHT), and is not funded. In order to participate, employees hired prior to September 2, 2001 must retire from Reed College at or after age fifty-five with at least ten years of continuous service. Employees hired after September 1, 2001 must retire from Reed College at or after age fifty-five with twenty years of continuous service. Employees are covered for the lowest premium Reed College plan for his or her lifetime and spouses/domestic partners are covered at the rate of 50% of the lowest premium College plan for his or her lifetime. Employer premium expenses were $498,848 and $482,922 for the years ended June 30, 2006 and 2005, respectively.

The following table sets forth the status of the plan as of June 30, 2006 and 2005:

		2006	2005
Accumulated postretirement benefit of obligation (APBO):			
Retirees	$	6,297,153	7,154,114
Active employees		10,226,876	11,556,756
Total APBO	$	16,524,029	18,710,870

The components of net periodic postretirement benefit cost for the years ended June 30, 2006 and 2005 are as follows:

		2006	2005
Interest cost	$	1,054,578	967,549
Service cost		374,039	526,188
Actuarial (gain) loss recognized		(3,615,458)	3,439,261
Net periodic postretirement benefit (gain) cost	$	(2,186,841)	4,932,998

Assumptions used in determining the postretirement benefit obligation and net periodic benefit cost using a measurement date of June 30, 2006 and 2005 were:

	2006	2005
Benefit obligation:		
Weighted average discount rate	6.50%	5.25%
Rate of increase in per capita cost of covered healthcare benefits	9% trending to 5% in 2012	10% trending to 5% in 2011
Net periodic benefit cost:		
Weighted average discount rate	5.25%	6.50%
Rate of increase in per capita cost of covered healthcare benefits	10% trending to 5% in 2012	11% trending to 5% in 2011

Benefits expected to be paid in each of the next five fiscal years are:

Year:		
2006-07	$	599,383
2007-08		715,792
2008-09		815,625
2009-10		912,079
2010-11		998,054

Aggregate benefits expected to be paid for the five fiscal years beginning with 2011-2015 are $6,114,321.

The Medicare Prescription Drug, Improvement and Modernization Act of 2003 provides an employer subsidy of 28% of gross annual prescription drug costs between $250 and $5,000 for actuarially equivalent plans. FASB Staff Position 106-2 requires that the estimated impact of this subsidy be reflected in the APBO for periods beginning after June 15, 2004. This reduction in APBO reduces the net periodic postretirement benefit cost due to corresponding reductions in the service cost and interest cost.

Actuaries have determined that the Reed College Postretirement Medical Plans are actuarially equivalent to the Medicare Part D plan and Reed College is applying for the employer subsidy and Reed College will be receiving the first subsidy in the fall of 2006.

(8) Funds Held in Trust By Others

Reed College has been named beneficiary of a portion of the remainder of seven trusts maturing at specified dates in the future. These trusts are administered by other entities. Reed College revalues the receivables associated with these trusts annually based on published actuarial factors for the respective beneficiaries' ages, discounted at 6.0%. At June 30, 2006 and 2005, the trusts receivable was $13,228,229 and $10,518,495, respectively, and was included under funds held in trust by others – noncurrent.

(9) Contributions and Accounts Receivable

Contributions receivable consist of the following:

	2006	2005
Annual fund	$ 325,788	185,241
Campaign fund	196,015	151,900
Endowment fund	4,426,508	1,790,323
Campus Center	760,250	536,200
Gross contributions receivable	$ 5,708,561	2,663,664

Contributions receivable reported on the statements of financial position were as follows:

		2006	2005
Current:			
Gross contributions receivable	$	202,628	181,961
Less allowance for doubtful accounts		(10,000)	(9,000)
Total current net contributions receivable		192,628	172,961
Long-term (one to five years):			
Gross contributions receivable		5,505,963	2,481,703
Less allowance for doubtful accounts		(251,000)	(125,000)
Long-term contributions receivable, net		5,254,963	2,356,703
Less discount to present value		(519,545)	(17,832)
Total long-term contributions receivable, net		4,735,418	2,338,871
Total contributions receivable, net	$	4,928,046	2,511,832

Contributions receivable due in excess of one year were discounted at 3.3% to 8.0% and 3.1% to 3.3% for the years ended June 30, 2006 and 2005, respectively.

Of the net unconditional promises to give included above, $5,241,299 represents an unconditional promise to give from 9 members of the Reed College board of trustees due in one to five years.

Accounts receivable consist of the following at June 30, 2006:

		Unrestricted	Restricted	Loan fund	Endowment	Total
Current:						
Student accounts receivable	$	87,759	—	—	—	87,759
Related parties		—	28,936	—	—	28,936
Grants and contracts receivable		—	184,871	—	—	184,871
Endowment		—	—	—	57,186	57,186
Other receivables		142,113	6,000	—	—	148,113
		229,872	219,807	—	57,186	506,865
Noncurrent:						
Student accounts receivable		—	—	22,054	—	22,054
Reed loans		—	—	739,476	—	739,476
Related parties		—	—	(4,847)	—	(4,847)
Federal Perkins loans		—	—	3,872,770	—	3,872,770
		—	—	4,629,453	—	4,629,453
		229,872	219,807	4,629,453	57,186	5,136,318
Less allowance for doubtful accounts		—	—	(60,239)	—	(60,239)
	$	229,872	219,807	4,569,214	57,186	5,076,079

Accounts receivable consist of the following at June 30, 2005:

	Unrestricted	Restricted	Loan Fund	Endowment	Total
Current:					
Student accounts receivable	$ 114,962	—	—	—	114,962
Related parties	—	43,526	—	—	43,526
Grants and contracts receivable	—	184,516	—	—	184,516
Endowment	—	—	—	189,041	189,041
Other receivables	132,254	—	—	—	132,254
	247,216	228,042	—	189,041	664,299
Noncurrent:					
Student accounts receivable	—	—	28,458	—	28,458
Reed loans	—	—	672,343	—	672,343
Related parties	—	—	(1,236)	—	(1,236)
Federal Perkins loans	—	—	3,703,254	—	3,703,254
	—	—	4,402,819	—	4,402,819
Less allowance for doubtful accounts	—	—	(60,239)	—	(60,239)
	$ 247,216	228,042	4,342,580	189,041	5,006,879

The Federal Perkins loans and Reed loans are generally payable at interest rates of 3%, 4%, and 5% over approximately ten years. Repayment begins after a designated grace period following college attendance. Principal payments, interest, and losses due to cancellation are shared by Reed College and the U.S. Government in proportion to their share of funds provided. The Federal Perkins loan program provides for cancellation of loans if the student is employed in certain occupations following graduation (employment cancellations). Such employment cancellations are absorbed in full by the U.S. Government.

(10) Net Assets

At June 30, 2006 and 2005, net assets consisted of the following:

		2006	2005
Unrestricted:			
Operating	$	1,963,183	(587,989)
Designated for special programs		21,970,437	21,107,185
Institutional loan programs		3,161,188	2,909,352
Funds functioning as endowment		82,973,988	84,725,362
Accumulated endowment gains		217,613,265	189,419,092
Net investment in plant		35,843,863	35,674,024
Total unrestricted	$	363,525,924	333,247,026
Temporarily restricted:			
Educational and general programs	$	89,199	143,898
Annuity and life income funds		20,182,964	15,884,244
Other temporarily restricted net assets		5,013,546	2,194,972
Total temporarily restricted	$	25,285,709	18,223,114
Permanently restricted:			
True endowment funds	$	68,259,308	62,424,180
Annuity and life income funds		4,339,510	3,251,805
Total permanently restricted	$	72,598,818	65,675,985

(11) Commitments and Contingencies

Reed College has placed certain of its medical and dental insurance coverage with Pioneer Educators Health Trust (PEHT). PEHT was formed by seven similar western colleges and universities for the purpose of providing medical and dental insurance to higher education institutions. Under the agreement, member institutions are required to make contributions to the fund at such times and in an amount as determined by the Trustees' for the various benefit programs sufficient to provide the benefits, pay the administrative expenses of the Plan which are not otherwise paid by Reed College directly, and to establish and maintain a minimum reserve as determined by the Trustees. In the event that losses of PEHT exceed its capital and secondary coverages, the maximum contingent liability exposure to Reed College is approximately $406,232. This exposure fluctuates based on changes in actuarial assumptions, medical trend rates, and reinsurance amounts. The level of reinsurance is not expected to fluctuate significantly in the future.

On July 1, 1988 Reed College elected to place its liability insurance coverage with the College Liability Insurance Company, Ltd. (CLIC). CLIC was formed by seven similar western colleges and universities for the purpose of providing liability insurance to higher education institutions. As a portion of its capital, CLIC has placed a $2,000,000 standby letter of credit of which Reed College is contingently liable for a pro rata portion based upon premium contributions from covered institutions. In the event the losses of CLIC exceed its capital and secondary coverages, the maximum contingent liability exposure to Reed College is approximately $154,680. As of June 30, 2006 and 2005, there were no amounts outstanding against the standby letter of credit.

From time to time, Reed College is involved in various claims and legal actions arising in the ordinary course of business. In the opinion of management, most of these claims and legal actions are covered by insurance and the ultimate disposition of these matters will not have a material effect on Reed College's financial position, statement of activities or cash flow.

(12) Fair Value of Financial Instruments

The estimated fair values of financial instruments have been determined by Reed College using available market information and appropriate valuation methodologies. At June 30, 2006 and 2005, the carrying values of cash, and accounts and notes receivable approximate fair value due to the short-term nature of these instruments. The fair value of investments is estimated based on quoted market prices for those investments that are actively traded securities. For other investments for which there are no quoted market prices, an estimate of the amount that could be realized in a sale was made by management. As of June 30, 2006, management believes the fair value of all investments for which there are not quoted market prices approximates the carrying value. The fair value of interest rate swaps are estimated based on estimates from the holders of the instrument and represent the estimated amount that Reed College would expect to receive or pay to terminate the agreement.

The fair value of Reed College's long-term debt is estimated based on the current rates available to the College for debt of the same remaining maturities. Taking into account current borrowing rates as of June 30, 2006, the fair value of Reed College's bonds approximates $55,241,022 as compared to its carrying value of $54,675,750.

(13) Fund Raising Expense

Reed College expended $2,038,987 and $1,670,761 for the years ended June 30, 2006 and 2005, respectively, for payroll and benefits, informational materials, travel and special events relating to fund raising activities. These expenses are all classified as public affairs in the statement of activities and changes in net assets.

Douglas C. Bennett, Provost
Reed College
3203 SE Woodstock Boulevard
Portland, OR 97202-8199

March 9, 1992

Dear Dr Bennett:

We are pleased to provide this written report of our review of the Reed Reactor Facility (RRF) which was carried out on February 6 and 7. Our efforts were largely directed towards answering the questions posed in your letter to us of January 17 regarding the future of the facility. To that end we reviewed the draft Mission Statement, annual reports, and other documents provided us, inspected the physical facilities, and interviewed a number of Reed faculty and students involved with the facility as well as outside users and interested individuals. We are grateful to all those we talked with for their courtesy and frankness in responding to our questions.

The RRF is a potentially valuable educational, research, analytical, and radionuclide production resource that is currently at a crossroads with respect to its continued operation. Over the years, the facility has existed at a marginal subsistence level through the heroic efforts of a number of dedicated individuals, maintaining a low profile with Reed faculty, students and administrators through benign neglect. Recent events, including the unusual event of November 23, 1991, and the need to make more permanent staffing arrangements, have served to focus attention on the facility. Our opinion is that RRF should not be permitted to continue as an out-of-sight, out-of-mind, low profile stepchild, expected to make it on its own resources. We are pleased that the College administration has seen fit to seek external advice and to squarely face the issue of the future of the facility. While this report has been kept brief, we have attempted to provide details of our notions in a number of areas that we hope will be useful in your deliberations.

The decision facing the College is simple: either continue operation of the facility under revised circumstances, or decommission the facility. The decision should be made swiftly and without equivocation and should be implemented rapidly. Should the decision be made to decommission, then a plan should be immediately drawn up to put in place the necessary staffing and other financial resources to initiate and complete the task expeditiously and with as little fanfare as possible. We would estimate the decommissioning would take at least two years and an expenditure of at least $500,000.

If the decision is made to continue operation, which is our personal recommendation, a solid commitment from the administration must be made to guarantee the necessary funds, personnel, and administrative support to refurbish this long-neglected facility and ensure its operation as a first class educational facility and ancillary resource for at least the next 10 to 20 years. The situation that currently exists must not be allowed to occur again, where a severely understaffed and underfunded facility is having difficulty recovering from a situation that it should be able to take in its stride. We expand considerably on this recommendation in addressing the questions you raised for us below.

1. Mission Statement. The draft Mission Statement given us contains several excellent analyses and suggestions, and is very comprehensive in scope. From discussions during our visit, it was not clear that this has yet had extensive review and input from many Reed individuals, including faculty, students, alumni, or community advisors and thus mainly represents the view of one hard-working and enthusiastic individual. As such it is extremely commendable.

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As part of the primary teaching mission, RRF provides unique opportunities for thesis research. With greater support, more Reed students could avail themselves of this opportunity. Secondarily, RRF can also serve the greater Portland educational community not only for similar student coursework and research, but as a resource for teacher education and general public education. It should be possible for courses initiated at Reed to be credited for students at other institutions to enhance these opportunities and the utilization of the Reed facility. With more staff time (*vide infra*) available to develop such relationships, creative uses of RRF in education will increase.

2. Physical Plant Upgrading. The radiochemistry laboratory adjacent to the reactor requires renovation and needs to be provided with suitable instrumentation to meet the educational mission described above as well as to support ancillary uses of the facility.

We were somewhat dismayed to observe that RRF is physically separated from the new Arthur F. Scott Chemistry Center, which has psychological and perhaps symbolic overtones as well. Serious, urgent planning consideration should be given to making the north entrance to the facility into the main entrance to the facility and connecting it to the new building by a covered walkway. If the old chemistry building is to become the psychology department, we believe it will cause unacceptable friction between academic units, and inhibit future uses of the facility (for tours, students at all hours, etc), as well as raise real safety concerns, for the entrance to be through the psychology department. The present entrance can remain as an emergency exit. To our inexperienced architectural eyes, it would seem that the north entrance could rather easily be remodelled to include a small entry lobby to serve as the security and safety checkpoint. Visitors and personnel responding to emergencies can then view the reactor through the hallway window before entering the facility itself. The purely experimental facilities will then be more towards the "rear" and impact tour use less, for example. Consideration might be given to renovation of the smaller laboratory rooms and office space to provide office space for staff. (*vide infra*) The facility should be refurbished and future general maintenance scheduled with the goal of maintaining a clean, smart, and professional appearance to attract confidence from regulators and potential users and supporters. It is likely the proposed changes can be accomplished for less than the costs of decommissioning.

3. Instrumentation. The process of upgrading the facility control and safety systems, which has begun under DOE sponsorship, should continue at as rapid a pace as possible with the College providing necessary matching support. It is possible that local industry might support radiological safety monitoring instrumentation (CAM, ARM, Stack Monitor, etc) acquisition, or help to extend the life of existing instruments by providing resources for maintenance to keep such instruments in good condition. Staff can be encouraged to pursue these and other funding opportunities if not stretched to the limit to maintain daily operations. In addition to the instrumentation needed for the renovated radiochemistry laboratory, the facility needs modern counting equipment for both gamma spectroscopy and beta spectroscopy to support the primary educational mission and to enable better service to ancillary users. Such a facility should expect to serve as a general facility for all departments employing radioisotopes. More useful service to the outside community can be given if the facility is able to maintain measurement traceability to NIST standards (achievable at modest cost), and develops a formalized quality assurance program. These features would enhance the ability of RRF to attract contract work from regional industry and government.

4. Staffing. A major issue for the continuation of facility operations is staffing: this facility has, since its inception, been minimally staffed. Given increased regulatory requirements over several years, the need for greater public accountability, and the need to develop more effective utilization of the facility, basic level staffing must be increased. Two professional FTE is the minimum recommended, with at least 1.5 FTE devoted directly to facility operations, while 0.5 FTE could be devoted to teaching responsibilities. Both individuals would be expected to hold Senior Reactor Operator Licenses for the facility. We suggest that the one individual with a 0.5/0.5 assignment should be the facility Director with clear responsibility for overall management of the facility. The other would serve as full time Associate Director for Operations and have responsibility for all day-to-day operations including supervision and training of operators and meeting regulatory requirements. The Director would have primary responsibility for building off-campus (Consortium) and on-campus relations and would hold a regular faculty appointment (with the 1/2 time teaching load). Assistance with maintenance of the Consortium would be provided by the Associate Director who might also hold a faculty appointment, or at the least an adjunct appointment that would enable him or her to participate in the educational program, for example by supervising thesis research students, or by offering credit classes for teachers.

We strongly support the continuation of the student operator program. Even more training might be offered in the area of radiological health and safety. We might envision parallel programs leading to "reactor operator" or to "radiological safety associate" as better meeting the needs for the facility and the campus than a single track program. However, the facility must clearly establish responsibilities for scheduling operations and maintenance. Such responsibility should remain with the Associate Director in consultation with a student Reactor Supervisor in order that reliable services can be offered by RRF. Students can have priorities unrelated to their reactor position that do not always blend well with operation of the facility within the strict regulatory environment or with offering reliable service to either on-campus or off-campus users.

As noted above, it is important that the Reactor Facility Director be a full faculty member acceptable to an existing department. It is most likely that the appropriate fit to Reed and the RRF needs will be found with an individual with a background in radiochemistry or nuclear analytical chemistry, or use of these methods in related areas such as geochemistry. Such individuals will have had some experience in regulatory issues and in reactor utilization. Obviously, willingness to make a strong contribution to undergraduate education and some experience with internal and external development of resources are essential.

5. Financing. There are a number of avenues for support for the operation of the RRF. It is important to appreciate that none can develop without adequate staff time to work on them. It is suggested that past practices, which may have included expecting key staff to "raise a portion of their own salary" may not be fruitful in today's competitive environment. It is important for the administration to recognize that the ability to compete for external resources is, in many instances, dependent on being able to offer routine and reliable services from the RRF. It is also important for the College to accept that full self-support should not be a goal for RRF. If the facility is perceived to play a genuinely broad educational role, it should receive basic support for that role, much as an interdisciplinary department might.

Traditional sources of income include charges and/or recharges for isotope production or neutron activation analysis. Cost recovery may be possible for specialized courses such as teacher education, and TAG, and joint projects with other educational institutions where their students use facilities such as the radiochemistry laboratory at Reed. Consortium support in conjunction with PGE and other interested industries can probably be increased. It is important to include all items in any cost-recovery program. In the experience of one of the reviewers, it is easy to overlook "hidden" costs in creating a cost recovery system that ends up putting a heavy burden on existing staff, who compensate by "donating" time to the project, preventing them from having sufficient time left for administrative needs. For example, if neutron activation analysis "service" is to be performed, proper allowance must be made for all supplies needed, including vials, standards, rabbits, and liquid nitrogen, waste disposal and radiological control, a fair contribution to instrument maintenance, and for staff and/or student labor including time for training, sample and standard preparation, and for data processing including quality control checks. As a related issue, RRF should make sure it commits to establishing a reputation for high quality, reliable, service, rather than for sporadic, cost-cutting, lower quality performance designed simply to raise funds.

Some support for instrumentation improvement will likely continue to be available through the Department of Energy. Grants for undergraduate research should be possible through NSF and/or DOE. It is possible to look to these, as well as regional agencies, as sources of support for undergraduates from other institutions to do work at RRF. All such utilization, if fully meeting its fair cost share, will contribute to the overall fraction of cost recovery for the RRF budget.

It does not appear that the general Reed College supporters - alumni and supportive local community - have yet been asked to support RRF. If the reactor can be firmly placed as a showcase within the Reed mission, support contributions might be forthcoming. Former reactor operators appear to have strong positive feelings about the reactor and wish to see it succeed. A program which would solicit contributions towards specific aspects of facility needs (e.g: a fund to guarantee student operators a certain amount of support) might appeal to such individuals.

Clearly the potential income from private sources is limited. There are some positive signs within governmental agencies, based partly on manpower need projections[1,2], that might result in increased support opportunities for small reactor facilities. Private foundations may also be moving in directions more favorable to nuclear science education. In seeking support from outside, the unique aspects of RRF should be stressed, such as:

- Location of the reactor on the campus of one of the outstanding small liberal arts colleges in the U.S.
- Genuine integration of the reactor into the educational mission of the College (*vide supra*).
- Reactor operations designed to heavily involve undergraduates in training to manage the facility.
- RRF as a genuine community resource, providing unique education opportunities to the entire region including teachers, and TAG programs.
- Location of the reactor in Portland and adjacent to the Seattle-Puget Sound area can serve specific needs of the technical, educational and medical communities in this region.
- Genuine community acceptance as evidenced by the community response following the unusual occurrence of November 23, 1991.

1. University Research Reactors in the United States - their Role and Value. National Academy Press, 1988.
2. Training Requirements for Chemists in Nuclear Medicine, Nuclear Industry, and Related Areas. National Academy Press, 1988.

6. Administrative. The administrative structure at present includes two formal "oversight committees", one for operations and one for safety. This arrangement is unusual and seems unnecessarily complex. There also seems to be a problem in having committee members who lack interest in their assignment. We recommend that there be a single oversight committee responsible to the College administration for assuring that the facility operates safely and meets its State and Federal (NRC) license commitments. To do this the Reactor Safety Committee needs at least one member who has expertise in nuclear engineering or nuclear science, one who has professional radiological safety qualifications, one who represents Reed's academic community, and one who represents the local off-campus community. The Campus RSO and the Facility Director should serve as ex officio members. The committee should have a charter which allows it to exercise its audit and policy and procedure review functions effectively to meet the Technical Specification requirements, but which assumes that daily operations are not its direct concern. Licensed student senior operators should be invited to attend but will not vote.

To assist in education and training of student operators in management of the facility, an informal Operations Committee can meet to assist the Associate Director for Operations in his or her duties. This group, consisting of all student licensed operators, with auditing attendance of operators in training, could assist with review and scheduling of operations and maintenance. It would be clear that final responsibility for organizing operations and maintenance rests with the Associate Director. Only in this way can regular routine operations at the facility be assured.

Conclusion. In conclusion we stress again the unique nature of the RRF, and its great potential as an educational tool, evidenced by the high interest in and enthusiasm for the reactor expressed by at least one group of Reed students. The existence of the reactor does influence students to attend Reed. We note the high degree of community acceptance, and the opportunities for ancillary uses of the RRF for research, isotope production, and specialized analyses by neutron activation. Taken together with the growing recognition that nuclear science education is important and deserving of support, we believe the future portends well. If the decision is made to continue operation of the reactor, the RRF should be incorporated as a full and valued part of the overall Reed College educational mission.

Finally we note the need to proceed with haste to make a final decision and to commit the resources needed to pursue either continued operation or decommissioning. The present acting Director, J. Michael Pollack has single-handedly kept the facility operational, maintained both on-campus and off-campus relations, and tried to plan for its future. This task is simply too great for one person and he is at or near burn-out. His efforts are commendable and worthy of some recognition.

We thank all of those who assisted us with our review, and the excellent hospitality shown by all connected with the College. Should you have any questions or desire further amplification of our ideas, please do not hesitate to call on us.

Sincerely yours,

RL Kathren

George E. Miller

Ronald L. Kathren
Washington State University
at Tri-Cities

George E. Miller
University of California, Irvine

Chapter 16

Other License Conditions

Reed Research Reactor Safety Analysis Report

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Chapter 16

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16 OTHER LICENSE CONSIDERATIONS

16.1 Prior Use of Reactor Components

16.1.1 Fuel Elements

The fuel used in the Reed Research Reactor (RRR) is the same TRIGA® fuel installed by General Atomics in 1968. Over the course of 40 years of operation, only approximately 70 grams of the original licensed 2,500 grams of uranium-235 have been consumed. Two fuel elements have had pinhole leaks in their cladding and have were removed from service. One element was dropped during an inspection and its bottom pin was bent, so it was removed from service.

Nine unused stainless steel clad elements were received from the Berkeley TRIGA® reactor when it shutdown. They are all being used it the reactor core.

In order to increase to a licensed power of 500 kW thermal, RRR will need additional fuel elements.

16.1.2 Control Rods

Unused boron carbide control rods were received from the Cornell TRIGA® reactor when it shutdown. One has been installed in place of the original regulating rod.

Environmental Report

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Environmental Report

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1 General

This environmental report is prepared in accordance with 10 CFR Part 51 as part of the nuclear reactor license renewal at Reed College, Docket 50-288, License R-112. The document summarizes the environmental effects that are imposed by operation of the Reed Research Reactor (RRR). The RRR is a TRIGA® Mark I reactor, light-water cooled and moderated reactor using uranium fuel. The historical maximum steady-state power of the RRR is 250 kW. Licensing is being requested however for operation of the RRR at 500 kW once the current aluminum clad TRIGA® fuel is replaced with stainless steel clad TRIGA® fuel. A full description of the reactor is contained in the Safety Analysis Report, License R-112, Docket 50-288.

2 Site Description

The RRR is a TRIGA® Mark I reactor with zirconium/uranium hydride fuel elements in a circular grid array. The uranium fuel is enriched to 19.9% in uranium-235. The reactor is located on the campus of Reed College, in the City of Portland, in Multnomah County, Oregon. The reactor is located adjacent to the Psychology building. The operations boundary of the reactor facility encompasses the reactor room and control room. The site boundary encompasses the entire building and 250 feet from the center of the reactor, including the Psychology Building and Chemistry Building. The reactor is at the bottom of a 25-foot-deep tank of water and is surrounded by a graphite reflector. The RRR operates at various steady power levels. The reactor is brought up to a desired power level (up to the historical license ceiling of 250 kW-thermal or in the future the requested new license ceiling of 500 kW-thermal) and is kept at that power until the experiment or irradiation is completed. This power level is usually maintained for periods ranging from a few minutes to several hours. Repeated operation over several days is possible for long-term irradiations. The main uses of the RRR Facility are instruction and research, especially trace element analysis. In addition to providing student research opportunities, the reactor staff works to educate the surrounding community on the principles of nuclear energy and radiation safety.

3 Environmental Effects of Operation

3.1 Thermal Impact

The fission energy generated in the RRR core is transferred to a closed primary coolant system, and to a secondary coolant system through a heat exchanger. The heat is then dissipated to the environment by means of a cooling tower. Municipal water is used to replenish the secondary coolant that is lost through evaporation. The rate of heat dissipation is less than that associated with shopping malls, large office buildings, and local factories.

3.2 Radiological Impact During Normal Operations

3.2.1 Environmental Monitoring

Environmental monitoring is performed by dosimetry devices, direct dose rate measurements, and sampling. Soil and surface water are routinely sampled and measured. No activity above background has been measured in these evaluated samples. Dosimetry devices monitor ambient radiation levels inside and outside the reactor room. These values are presented in Table 1 and Table 2. Readings less than the minimum detectable are indicated with the letter M. Doses are in millirem per calendar year.

Table 1 Inside Facility Area Radiation Dosimeters

Year	North Low	North High	East	South	West	Counting Room	Control Room
1998	50	M	M	M	M	-	-
1999	25	M	M	M	M	-	-
2000	24	45	40	9	9	-	-
2001	69	109	522	22	22	-	-
2002	227	2155	726	85	45	-	-
2003	3928	240	663	181	149	-	-
2004	148	284	121	141	262	-	-
2005	205	90	153	92	176	26	165
2006	197	87	146	105	100	M	155

Table 2 Outside Facility Area Radiation Dosimeters

Year	North	East	South	Roof
1998	M	M	M	0
1999	M	M	M	0
2000	3	M	M	2
2001	31	M	M	10
2002	82	M	M	6
2003	105	M	M	10
2004	8	M	M	1
2005	74	M	M	M
2006	8	M	M	2

3.2.2 Personnel Exposure Monitoring

Each person who may use or handle radioactive materials must receive radiation safety training. Radiation exposures to reactor personnel are administratively controlled to meet ALARA (as low as reasonably achievable) criteria. Experience from RRR operations shows that all personnel exposures are well below the whole body dose limit of 5000 mrem per year, as specified in 10 CFR Part 20. Historical whole body and extremity exposures are presented in Tables 3 and 4.

Table 3 Historical Deep Dose and Shallow Dose Exposures

Numbers of persons in annual-dose categories					
Year	< 100 mrem DDE	> 100 mrem DDE	< 100 mrem SDE	100-500 mrem SDE	> 500 mrem SDE
2000	29	0	29	0	0
2001	26	0	25	1	0
2002	30	0	29	1	0
2003	31	0	31	0	0
2004	38	0	37	1	0
2005	47	0	46	1	0
2006	51	0	50	1	0

Table 4 Historical Maximum Deep Dose and Shallow Dose Exposures

Year	Deep Dose (mrem)	Shallow Dose (mrem)
2000	15	0
2001	12	51
2002	12	100
2003	11	30
2004	64	120
2005	13	240
2006	13	120

3.2.3 Solid Wastes

Solid wastes generated at the RRR are low-level wastes such as ion-exchange resins, filters, laboratory supplies and cleaning materials. Solid wastes are sent to an appropriate waste disposal facility when sufficient solid waste is on hand to make disposal economical. Historical records of recent solid waste disposals are presented in Table 5.

Table 5 Historical Solid Waste Generation and Shipments

Year	Quantity (cu ft)	Activity (mCi)
1999	0	0
2000	0	0
2001	0	0
2002	30	37
2003	0	0
2004	22	31.8
2005	0	0
2006	18	0.13

3.2.4 Liquid Wastes

No liquid waste is generated by operation of the RRR.

3.2.5 Radioactive Gas Effluent

The only routine release of gaseous radioactivity is from argon-41 (1.83-hour half-life) and nitrogen-16 (7.13-second half-life). These come from activation of pool water, air in the pool water, and air in the irradiation facilities. Table 6 shows the historical values for the average concentration of released radionuclides at the site boundary along with the calculated dose to a member of a hypothetical member of the public who was constantly residing at that point. Radioactive gas effluent is discharged from the building exhaust stack 12 feet (3.7 m) above the confinement building. All nuclides are well below the regulatory effluent concentration limits given in the RRR Technical Specifications, Appendix B, Table 2, of 10 CFR Part 20 and USNRC Regulatory Guide 4.20. The calculated dose to the public is well below regulatory guidelines and constraints.

Table 6 Historical Data for Gaseous Releases

Year	Average Concentration at Site Boundary (µCi/ml)	Dose to Member of Public at Site Boundary (mrem/yr)
1998	3.47E-12	0.02
1999	1.24E-11	0.06
2000	1.31E-11	0.07
2001	1.18E-10	0.60
2002	2.24E-10	1.12
2003	4.40E-11	0.22
2004	6.36E-11	0.32
2005	4.94E-11	0.25

3.2.6 Radiological Impact During Abnormal Operations

Chapter 13 of the RRR Safety Analysis Report provides accident analysis for the RRR. The Maximum Hypothetical Accident (MHA) for the RRR, as it is with virtually all TRIGA® type reactors, is postulated to be an instantaneous loss of coolant water, followed an instantaneous movement of volatile fission products from the fuel uniformly distributed into the reactor room air, and an instantaneous disappearance of the north wall of the confinement building. Results of these calculations for this scenario predict doses for the general public and occupational workers were all well below the annual dose limits specified in 10 CFR Part 20.

4 Benefits of Facility Operations

The reactor was obtained in 1968 through a grant from the United States Atomic Energy Commission and is currently operated under Nuclear Regulatory Commission License R-112 and the regulations of Chapter 1, Title 10, Code of Federal Regulations. The reactor supports education and training, research, and public service activities.

5 Alternatives to Increased Power Operation

Each U.S. university research reactor is a unique facility, with individual educational and research objectives. Alternatives to increased power operation of the RRR are (1) status quo at 250 kW, and thus a lesser operational and research capability, or (2) closure of the RRR. Closure of the reactor is not a consideration in the request for relicensing along with increased power operation. The loss of any of the remaining U.S. university reactors would constitute a significant weakening of the U.S. ability to operate and control nuclear-related facilities. Numerous recent studies, including studies by the National Academy, have confirmed this observation.

6 Analysis

The RRR is an integral part of the Reed College Campus. Based on the data presented here, the facility is currently operating with minimal radiation exposures and releases, well within regulatory limits. Personnel, environmental and area radiation monitoring confirm that all exposures are within ALARA expectations. Increase of licensed power to 500 kW operation is not expected to significantly increase these levels. The RRR is an existing facility that was originally built with expansion capability. No capital funds are required for increased power operation. The desirable and anticipated decision is that the license be renewed and the power upgrade be approved.

7 Long Term Effects on the Environment

At the eventual closure of RRR operations, all areas housing or impacted by the RRR and the affiliated laboratories will be decommissioned and returned to general university use. The reactor fuel (owned by DOE) will be shipped to a designated DOE facility. Upgrade of the RRR power level will not materially impact this outcome. Indeed, it is anticipated that the increased power level will significantly enhance the services and research potential of the facility. The environmental impact associated with renewing the RRR license and upgrading the power is deemed to be insignificant compared to the positive benefits resulting from enhanced educational and research opportunities offered by Reed College to the region.

Radiation Protection Plan

August 2007

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Radiation Protection Plan

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1 Management Policy Statement

The Reed Research Reactor (RRR) will strive to maintain a safe workplace and a clean environment. No activity will be undertaken at the facility unless it can be performed in a manner that protects the staff, the public, and the environment.

2 Administrative Procedures

The RRR Administrative Procedures and Standard Operations Procedures (SOPs) detail the administrative procedures for the Facility.

3 ALARA program

3.1 ALARA Policy Statement

It is the policy of the Reed Reactor facility to keep radiation exposures As Low As is Reasonably Achievable (ALARA). Four general principles govern the application of this policy to actual work conditions:

Minimize Time: The less time a worker spends near a radioactive source, the smaller the exposure will be. The most common way to minimize exposure time is to plan and practice the operation beforehand.

Maximize Distance: The greater the distance between a worker and a radioactive source, the smaller the exposure will be. This can often be accomplished by manipulating sources with tongs or other implements. When implements are impractical, sources should be held as far away from the body as is practical. Step away from the sources when not actually using them.

Use Proper Shielding: Placing an appropriate shield between a source and a worker reduces the workers exposure. In practice this means placing gamma sources in lead pigs to move them, and storing radioactive sources in lead caves. Goggles should be worn when working with beta emitters.

Control Contamination: Contamination can be controlled by:

a) Wearing lab coats, gloves, booties, safety glasses, goggles, and anti-contamination clothing where appropriate.
b) Changing gloves frequently.
c) No eating, drinking, or smoking in radioactive work areas.
d) No mouth pipetting.
e) Washing hands at completion of work with radioactive materials.
f) Frequent monitoring of hands, clothes, and work areas.

3.2 Management Responsibility

a) Ensure the that radiation dose to workers, visitors, and the public are kept ALARA.
b) Ensure that radiation workers receive appropriate training.

3.3 Worker Responsibility

a) Obey posted signs and instructions.
b) Follow standard operating procedures (SOPs).
c) Incorporate ALARA principles and apply concepts of time, distance, shielding, and contamination control.
d) Minimize radioactive wastes.
e) Receive training for their job assignments.
f) Inform their supervisor of radiation hazards or potential problems.
g) Stop work on anything that poses addition to the health or safety of the worker, other member of the staff, the public, or the environment.
h) Ask questions when in doubt.

3.4 Qualifications of personnel and adequacy of resources

Section 2.2 of the Administrative Procedures details the qualifications for staff positions at the facility. Section 2.3 addresses qualifications for members of the Reactor Review Committee. The management of Reed College has the financial responsibility for ensuring the facility has adequate resources for maintaining personnel and funding.

3.5 Adequacy of authority for responsible persons

Section 2.2 of the Administrative Procedures details the assigned authority of personnel.

3.6 New staff training and continuing education for all personnel

New staff and students are trained in ALARA and radiological work procedures. Everyone must pass the Radioactive Material (RAM) Handling Exam before handling radioactive materials.

3.7 Radiological design as an integral aspect of facility and experiment design

The facility is designed to limit the radiological doses the staff and public as described in the Safety Analysis Report. Section 4 of the Administrative Procedures details the authorizations and reviews required before an experiment may be performed.

3.8 Radiological planning as an integral aspect of operations planning

SOP 28, Radiation Work Permits, details the requirements for operations where a significant dose is likely.

3.9 Performance reviews of designs and operations (lessons learned)

SOP 28, Radiation Work Permits, requires a review of past operations before planning a new operation, as well as a final review of the operation after its completion.

3.10 Analysis of personal exposure patterns

The Radiation Safety Officer (RSO) is responsible for maintaining radiological dose records as well as analysis of personal dosimetry results. The Director also reviews the radiological dose records.

3.11 Periodic assessment and trend analysis of the radiological environment

The RSO maintains records of dosimetry results. Personnel dosimetry reports are made available quarterly for each worker. The doses have been historically been below the detection limit of dosimetry devices, so that any measurable dose attracts the attention of the RSO. Any deep dose above 50 mrem/quarter or shallow dose above 100 mrem/quarter is be investigated by the reactor management.

3.12 Periodic assessments and audits of the protection program

This Radiation Protection Plan is audited by the Reactor Review Committee annually. Records related to the dosage and contamination control are reviewed annually by the Reactor Review Committee.

4 Records

The RSO maintains records of personnel exposure. These records are designed to meet federal and State regulations.

5 Surveillance Activities

5.1 Personnel exposure (dosimetry)

The personnel dosimetry is administered by the RSO who monitors the badges so as to conform to State and Federal laws.

5.2 Radiation and contamination surveys

SOP 23, Wipe Tests, details the contamination surveys.

5.3 Environmental monitoring

SOP 24, Environmental Sampling, details the sampling of the environment.

5.4 Effluent Monitoring

The Gaseous Stack Monitor (GSM) and Air Particulate Monitor (APM) measure gaseous radioactive effluent to the environment.

5.5 Warning and active protection systems functionality

Warning and active protection systems functionality is verified by the SOPs.

5.6 Operational limitation compliance

The reactor staff are trained to recognize and respond to abnormal situations. The Reactor Review Committee reviews and audits all of the records associated with meeting our operational limits.

5.7 Engineered protective systems

The primary protective system is the ventilation system that is described in Section 4.4 of the Reed Research Reactor Safety Analysis Report. The water in the reactor tank provides shielding.

5.8 Instrumentation

Technical Specifications describe the required radiological monitoring instrumentation.

5.9 Radioactive material accountability

This is addressed by Section 5.2 of the Administrative Procedures.

6 Protective Equipment

The semi-annual checklist includes an inventory of the emergency grab bag. Survey meters are calibrated annually. Disposable gloves, booties, and other anticontamination clothing is used when contamination is possible.

7 Calibration and Q.A. programs

Calibration is covered by the following SOPs:

SOP 25 Radiation Monitor Calibration
SOP 26 Personal Dosimetry Calibration
SOP 30 Radiation Area Monitor Calibration
SOP 31 Continuous Area Monitor Calibration
SOP 32 Air Particulate Monitor Calibration
SOP 33 Gaseous Stack Monitor Calibration

8 Training

All personnel who handle radioactive material are trained as an authorized user. Proper ALARA concepts are part of the training.

9 Waste Program

Waste collection is administered by the reactor staff and management. Waste disposal is administered by the campus RSO who manages this program so as to conform to State and Federal laws, especially 10 CFR Part 20.

10 Emergency Plan

The Emergency Plan and Emergency Implementation Procedures are audited annually by the Reactor Review Committee.

11 Audit and review programs

The responsibilities of the various oversight committees are detailed in Sections 2.3, 2.3.1, and 2.3.2 of the Administrative Procedures. These responsibilities include the annual review of this radiation protection plan.

Requalification Plan

August 2007

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Requalification Plan

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1 Introduction

This Requalification Plan is developed in compliance with the requirements of 10CFR55.59.

The requalification requirements are the same for Reactor Operators and Senior Reactor Operators. The term operator is used to apply to both levels of US NRC licensing.

2 Schedule

The Requalification Program shall be continuous and operate on a two-year cycle. For purposes of this Plan, a year begins on July 1 and ends on June 30. Calendar quarters are as follows:

First Quarter:	July 1 to September 30
Second Quarter:	October 1 to December 31
Third Quarter:	January 1 to March 31
Fourth Quarter:	April 1 to June 30

3 Meetings

A minimum of five meetings shall be scheduled during each academic year (i.e., ten each requalification cycle), with a minimum of two each semester. The subjects of these meetings may be drawn from the topics listed below, but are not restricted to those topics. The specific content of the meetings is to be based on items identified as weaknesses in the training program or operator knowledge. These weakness may be identified using the results on NRC licensing exams, facility administered requalification exams, or observations of operator knowledge and performance by the Director. It is not required that all topics be covered in any given requalification cycle. A single meeting may address more than one topic.

a) Theory and principles of reactor operation

b) Reactor instrumentation, control, and safety systems

c) Radiation control and safety; use and handling of radioactive materials

d) Facility and Regulatory requirements including License, Technical Specifications, Requalification and Emergency Plans, Administrative Procedures, and applicable portions of Title 10 Code of Federal Regulations

e) Standard and Emergency Operating Procedures

f) General and specific plant operating characteristics

g) Fitness for duty requirements for operators

h) Harassment policies

i) Other topics identified by the reactor staff as relevant to the safe operation of the reactor in accordance with federal and state regulations

3.1 Reviews of Facility Changes

At least twice each year (i.e., four times each requalification cycle), a meeting of Operators shall be held to make them aware of facility design changes, and changes in procedures and administrative requirements. This meeting may be held in conjunction with a Requalification meeting. Written reviews of changes to the facility should be issued as a Procedure Change Notification.

3.2 Missed Meetings

Any operator who is absent from any required meeting shall review any notes, handouts, or other reference material from the meeting. This material shall be discussed with the Director (or his designate) to assure that the operator has understood the material. This discussion shall be documented by having the operators sign and date the attendance record from the meeting the operator missed.

4 On-The-Job Training

4.1 Hours per Quarter

Each operator must actively perform the functions for which they are licensed for a minimum of 4 hours per calendar quarter. The Director shall maintain a list of those activities that may and may not be counted towards this requirement.

4.2 Reactivity Manipulations

Each operator shall demonstrate an ability to conduct reactivity manipulations during the course of the year. A minimum of 10 reactivity manipulations shall be completed during each academic year (i.e., 20 each requalification cycle). The Director shall maintain a list of those operations that qualify towards meeting this requirement.

4.3 Records of Hours and Reactivity

Each operator shall be responsible for maintaining records, in a format approved by the Director, to verify compliance with Sections 4.1. and 4.2. Operator records will be reviewed by the Director (or designate) at regular intervals to ensure all operators comply with Sections 4.1 and 4.2.

Any operator who does not document 4 hours per quarter or 10 reactivity manipulations per year must spend six hours operating the reactor under the supervision of a licensed operator before resuming licensed duties.

5 Observation

Reactor administration, supervisory, training, and/or health physics personnel shall conduct periodic observations of the performance and competency of operators during routinely scheduled operations. These observations should normally be unannounced and shall be documented.

Documented observation of each operator shall occur at least once each academic year.

6 Evaluation

6.1 Written Exam

A comprehensive written exam shall be administered during each requalification cycle to each operator (one exam every two years). Passing criteria shall be 70% in each Section.

The written exam shall either use an on-line random exam bank or be prepared by the Director or a Senior Reactor Operator assigned by the Director. Any person who prepares the exam will be exempt from taking the exam during that requalification cycle; preferably this is someone who is exempt under Section 6.3 below. The same person shall not be exempted for two consecutive cycles.

6.2 Operating Exam

An operating exam shall be administered each academic year to each operator. Passing criteria shall be 70% overall.

The operating exam shall be administered by a Senior Reactor Operator designated by the Director. This may be in conjunction with the operator observation required by Section 5.0.

6.3 NRC SRO Exam Exemption

The Senior Reactor Operator exam administered by the NRC shall suffice to satisfy the requalification cycle written and operating exam requirement.

7 Accelerated Requalification Program

The Accelerated Requalification Program is designed to ensure an operator can safely perform their licensed duties. Operators enter Accelerated Requalification by failing to demonstrate competency in performing licensed duties.

An operator in Accelerated Requalification shall not perform duties that require a license except under the direct supervision of a Senior Reactor Operator, until completion of an Accelerated Requalification Program.

The content of any Accelerated Requalification Program shall be determined on a case-by-case basis by the Director. It may include, but need not be limited to, attending or presenting meetings on the areas of weakness, revising facility documentation, individual or group study sessions, or tutorials.

Any operator who in the opinion of the Director is deficient in operating knowledge or skills may be place in Accelerated Requalification by the sole discretion of the Director.

Any operator who achieves a score less than 70% on any Section of the written exam required by Section 6.1, or less than 70% on the operating exam required by Section 6.2 shall be placed in Accelerated Requalification.

Successful completion of the Accelerated Requalification Program shall be based on the passing of an additional exam. This exam shall be administered within six months of the date of entering the Accelerated Requalification Program. The extent of this additional exam will normally be limited to areas of identified weakness. The examination may be written, oral, or operational. The type of exam used should be determined at the start of the Accelerated Requalification program.

If an operator missed one of the requirements to stay in requalification, it is only necessary to fulfill the requirement that was missed. For example, if the operator failed a written exam, all that is need is to a pass another written exam (six hours of licensed duties under observation is not required).

8 Physicals

Each operator is required to satisfactorily complete a physical exam every 24 months, not to exceed 25 months. If the exam is not completed, the operator may not perform licensed duties until they pass a physical exam.

Operators whose NRC license specifies the need for corrective lenses, shall be wearing corrective lenses when performing any licensed duties.

If any operator has a significant permanent change in their physical condition that may affect their ability to perform licensed duties, they must inform the Director immediately. evaluate their license status.

The Director will notify the NRC in writing within 30 days of any permanent disability or illness that affects the operator's ability to perform licensed duties per 10CFR50.74.

9 Other Requirements

If any operator is convicted of a felony, they must inform the Director immediately.

If an operator us unable to participate in the Requalification Program for some reason, e.g., overseas for a year, the Director must terminate their license per 10CFR55.53(h).

Operator licenses expire six years from their effective date. To renew a license, an application (Form 398 and 396) must be submitted to the NRC per 10CFR55.57.

10 Records

Originals of all records relating to the requalification requirements in this Plan shall be retained by the facility. Copies shall be provided to the operator at their request.

Fire Plan

August 2007

Reed Research Reactor
3203 SE Woodstock Boulevard
Portland, Oregon 97202
(503) 777-7222

License R-112
Docket 50-288

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Fire Plan

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1 Introduction

The Reed Research Reactor (RRR) Fire Plan is a broad fire protection program that interfaces with the RRR Emergency Plan and Emergency Implementation Procedures for emergency situations and for emergency preparedness training. It is not the purpose of this document to duplicate the plans and procedures in those documents, and where appropriate, they will be referenced. The plans and procedures in the Emergency Plan and Emergency Implementation Procedures take precedence in emergency situations.

The objective of the RRR Fire Plan is to provide a fire protection program that provides reasonable assurance that safety-related systems can perform their required functions and that the defined loss criteria are met. The plan provides for three components:

a) Passive Fire Protection

b) Active Fire Protection

c) Fire Prevention

This plan is based on the criteria given in ANSI/ANS-15.17-1981, American National Standard Fire Protection Program Criteria for Research Reactors. This document was reaffirmed without change in 1987 and 2000.

2 Definitions

Loss Criteria	Those criteria established by facility management in accordance with all applicable regulations as limits for risk to personnel, radioactive or toxic contaminant release, property damage, and interruptions of operations that might occur from a fire of maximum credible proportions or effect. The following shall be the loss criteria for the adverse effects of fire: a) No injury to the personnel or persons in or about the facility under all conceivable circumstances. b) No radioactive release to the environment beyond those limits established by federal or state requirements. c) No injury or exposure to the general public. d) No dollar losses beyond acceptable limits as determined by the RRR Management.
Management	The Reactor Director and Associate Director.
Potential Fire Situation	A situation where a fire may occur and result in harm to life, property, or the environment.
Research Reactor	A device designed to support a self-sustaining neutron chain reaction for research, developmental, educational, training, or experimental purposes, and which may have provisions for the production of non-fissile radionuclides.
Risk	The risk associated with a potential fire situation is a compound measure which includes both the likelihood and the consequences of realizing that

	situation's potential for adverse effects.
Safety-Related Systems	Those systems, structures, and components that perform functions necessary to shut down the reactor and maintain it in a safe shutdown condition, and to minimize radioactive releases to the environment.
Shall, Should, and May	The word "shall" is used to denote a requirement; the word "should" to denote a recommendation; and the word "may" to denote permission, neither a requirement nor a recommendation.

3 Management and Organization

3.1 Management Commitment

The RRR Management is cognizant of the importance of an adequate fire protection program to ensure the health and safety of the operating staff and the physical plant, and to ensure the health and safety of the general public and the environment from any fire related events. To this end, the RRR Management supports this fire protection program and monitors implementation and performance through the appropriate review and audit functions reporting directly to management as described in the RRR Technical Specifications.

3.2 Organizational Structure

3.2.1 Normal Organizational Structure

The normal organizational structure of the RRR is given in the RRR Technical Specifications. The Director shall have responsibility for the fire plan and implementation procedures. The Reactor Safety Committee (RSC) shall have responsibility for the review and audit of this plan at intervals specified in this plan.

3.2.2 Emergency Organizational Structure

The emergency organizational structure of the RRR is given in the RRR Emergency Plan. Fire fighting is an emergency response function, and is handled by the Portland Fire Bureau through a written agreement on file.

The Director has functional responsibility for the Emergency Plan and emergency preparedness. The RSC has responsibility for the review and audit of emergency planning, training, and preparedness.

4 Safety Related Systems

The following components and systems shall be considered safety related systems for the purposes of the fire plan:

a) Reactor Core and Associated Support Structures

b) Reactor Pool

c) Bridge and Control Rod Support System

d) Control Rods
e) Reactor Instrumentation Channels
f) Cables from Bridge to Control Room
g) Control Console
h) Continuous Air Monitor
i) Cables from Continuous Air Monitor to Damper Control
j) Air Confinement System in the Loft
k) Radiation Area Monitor

5 Risk Assessment

The RRR TRIGA® Mark I Nuclear Reactor is located at the bottom of a [redacted] pool structure with the reactor internals supported by the pool structure. There is no conceivable series of events under which a fire can directly affect the reactor pool, associated support structures, or core components.

The following are potential fire situations at the RRR:

5.1 Combustible Material in the Control Room

The amount of such material is kept to a minimum consonant with operations of the RRR. The log books are normally stowed separate from the control console. A set of shelves containing references is on a wall away from the control console. The trash can is metal. No smoking is allowed in the entire facility. A fire from combustible materials in the control room may cause damage to the materials involved and smoke damage to other components in the room, but is not likely to directly involve the control console nor to spread out of the control room as the walls and ceiling are structural concrete.

5.2 Electrical Fire in the Control Console

In the event of electrical fire while the reactor is operating, failure of any one of many systems (e.g., 110 Volt AC power, Linear Power Channel, Percent Power Channel) will cause an immediate scram to a shutdown condition. When the reactor is shutdown, there is no conceivable series of events initiated by fire in the control console that could change the status of the reactor core to an unsafe condition.

5.3 Fire in the Cable Conduit from the Control Console to the Bridge

The cables are in a steel pipe set in concrete. There would be no danger of spread of the fire, nor any danger to the restoration of the reactor to a shutdown condition. Any interruption in the instrument voltage lines or in the magnet current lines immediately causes a scram of the reactor. A scram is initiated by the interruption of the magnet current causing the control rods to drop into the core under the force of gravity so any interruption of the 110 VAC power will scram the reactor.

5.4 Fire in the Reactor Bay

The reactor bay is a structural concrete room with little combustible material in it. The effects of fire in the reactor bay are minimal in terms of damage to safety related systems and to the facility in general. There are some wooden storage cabinets in the reactor bay and some wood is used for experimental facilities such as a prompt gamma facility. Solvents and flammables are kept in metal cabinets in the reactor bay.

Fire in the material handling area has the possibility of releasing radioactive materials in the form of an aerosol or attached to smoke particles. The material handling area is next to the north wall with no direct fire path to any other structure.

5.5 Fire in the Mechanical Room

Fire in the mechanical room may originate in electrical apparatus such as a pump or in the electrical distribution system in the room. In either case, the fire will be contained because the walls and ceiling are structural concrete, and its damage confined to the room. Any interruption of the service power will cause an immediate scram of the reactor if it is operating.

5.6 Fire in the Ventilation Loft

Fire in the ventilation loft may be caused by electrical equipment such as fan, blower, or pump motors or the associated electrical supplies. There is also the possibility of fires in the air filters (conventional and High Efficiency Particulate Air (HEPA)). The dampers are motor actuated. A fire in the loft may compromise the air confinement system and require shutting down the reactor. There is no radioactive material stored in the loft, and no danger of release of radioactive materials from such a fire alone (as a single failure event).

5.7 Fire in the Continuous Air Monitor

The Continuous Air Monitor (CAM) is located in the reactor bay and presents a small electrical fire risk. It has a radiation detection system with associated high and low voltage power supplies. It monitors the air in the reactor bay and generates an alarm signal and air confinement trip signal in the case of radioactivity exceeding a preset limit. A fire in the CAM will affect the air confinement system. A fire located in the CAM will not spread as it is a self contained unit in a steel case against a concrete wall.

5.8 Fire in the Radiation Area Monitor

The Radiation Area Monitor (RAM) is a self-contained unit operating on 110 VAC mounted on steel posts above the southwest corner of the reactor pool. A fire in the RAM would result in damage to the RAM only, and not be a threat to any other equipment.

5.9 Fire in Bridge Structure

The only places fire is possible on the bridge are in the control cables, the control rod drive motors, and the rotary specimen rack drive system. Of these, only the control cables and the control rod drive motors are part of the safety related systems. There is no sequence of credible events that could have a fire in these components result in an unsafe condition of the reactor core. Fire in the rotary specimen rack drive system could result in damage to an experiment, but equipment damage will be limited to the drive system.

5.10 Fire in the Radiochemistry Laboratory and Counting Rooms

This is an analytical chemistry laboratory with all attendant electrical, chemical, and combustible materials. While a fire in the Radiochemistry Laboratory and Counting Rooms does not present a direct threat to the integrity of the reactor facility, there are cables, phone lines, and the pneumatic transfer system passing through the laboratory, and fire-generated toxic gases could be a personnel hazard to operators in the facility. Personnel may be operating the pneumatic transfer system in which case there is a direct air path from the counting room to the control room including the Radiochemistry Laboratory.

5.11 Fire in the Adjacent Psychology Building

The major threats to the RRR from fire in the Psychology Building have been identified to include:

a) Water damage to the radiochemistry laboratory, counting rooms, and reactor from fire-fighting activities.

b) Toxic Gases entering the facility via the stairway to the Psychology Building or through air intakes on or near the Psychology Building.

c) Loss of power to the RRR.

d) Loss of phone lines to the RRR.

6 Program Components

The general overall program to reduce the possibility of fire at the RRR depends on elements of all three fire protection components:

6.1 Passive Protection

The design of the RRR incorporates passive protection into the basic structure of the building. The building is constructed of concrete, brick, and glass. The only wood structures in the facility are experimental facilities, laboratory sink, cabinets, and wood support structure in the southeast corner. The doors separating the facility from the outside are fire resistant doors. The door at the bottom of the stairs to the Psychology Building is a fire resistant door. The doors to the control room and the reactor bay are fire resistant, and have automatic closers on smoke alarm. Inside the reactor facility, the door to the mechanical room is a fire resistant door. The walls between the mechanical room and the reactor bay and control room are concrete. Ventilation for the control room and the mechanical room is separate from that for the reactor bay. All structural walls and ceilings in the facility, radiochemistry laboratory, and counting rooms are concrete.

6.2 Active Protection

6.2.1 Fire Detection

The RRR shall have an active fire detection system capable of detecting fire in the reactor bay, control room, mechanical room, and radiochemistry laboratory with local alarm and transmission of an alarm signal to a monitoring location with 24-hour coverage. The system shall have provisions for testing the transmission of alarm signals.

The reactor bay, mechanical equipment room, and control room have fire detection (but no automatic suppression).

The lab and classroom areas have fire detection and automatic wet-pipe sprinkler suppression.

There are four fire extinguishers in the facility.

There are three fire pull stations in the facility.

6.2.2 Emergency Response Actions

Emergency Response Actions are detailed in the Emergency Plan and Emergency Implementation Procedures. Personnel safety takes precedence in all incidents.

6.3 Fire Prevention

6.3.1 Effect of Facility Changes

All facility changes shall include a determination and evaluation of the effect of such changes on fire risks, and shall also include institution of the necessary compensatory changes in the fire protection program.

6.3.2 Written Procedures

Procedures shall be written for all activities related to fire protection, including those for inspection and test activities. These procedures shall supplement those in the Emergency Implementation Procedures.

6.3.3 Record Retention

All records relating to the fire prevention program including audit and review documents shall be retained for a period of five years.

6.3.4 Review System for Special Activities

A system for controlling open-flame work, such as welding or cutting, shall be established. All non-reactor equipment to be operated continuously in the facility shall be evaluated for fire hazard. These activities may be governed by procedures and require authorization prior to commencement.

6.3.5 Fire Protection Equipment Program

Active fire protection equipment shall be tested annually, and the transmission network checked semiannually. Appropriate items may be added to the annual and semiannual maintenance checklists. The maintenance of the fire extinguishers shall be the responsibility of Reed College Environmental Health and Safety or their contractors.

7 Fire Safety Assurance

7.1 Area Inspections

Area inspections shall be conducted at least bimonthly, with the interval between inspections not to exceed eleven weeks. These inspections shall consist of visual reviews of facility areas in order to detect the existence of new potential fire situations and may be coordinated with the inspections and tests of the fire protection or other equipment. Appropriate items may be added to the RRR Bimonthly Checklist. The Reactor Supervisor shall be responsible for the completion of these inspections.

7.2 Audits

Audits of the total Fire Protection Program shall be conducted on an annual basis, with the interval between audits not to exceed 15 months. The audits shall be conducted in accordance with prepared audit procedures by the Reactor Safety Committee. The audit shall determine the adequacy of the program including:

a) Policy Statements

b) Management Support

c) Size and Qualifications of Staff

d) Funding

e) Program Documentation

Administrative Procedures

August 2007

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License R-112
Docket 50-288

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Administrative Procedures

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1 Introducton

The administration of the Reed Research Reactor is described in Technical Specifications Section 6. That information is not repeated here. These Administrative Procedures provide additional guidelines and policies. The administrative organization is shown in Figure 1.

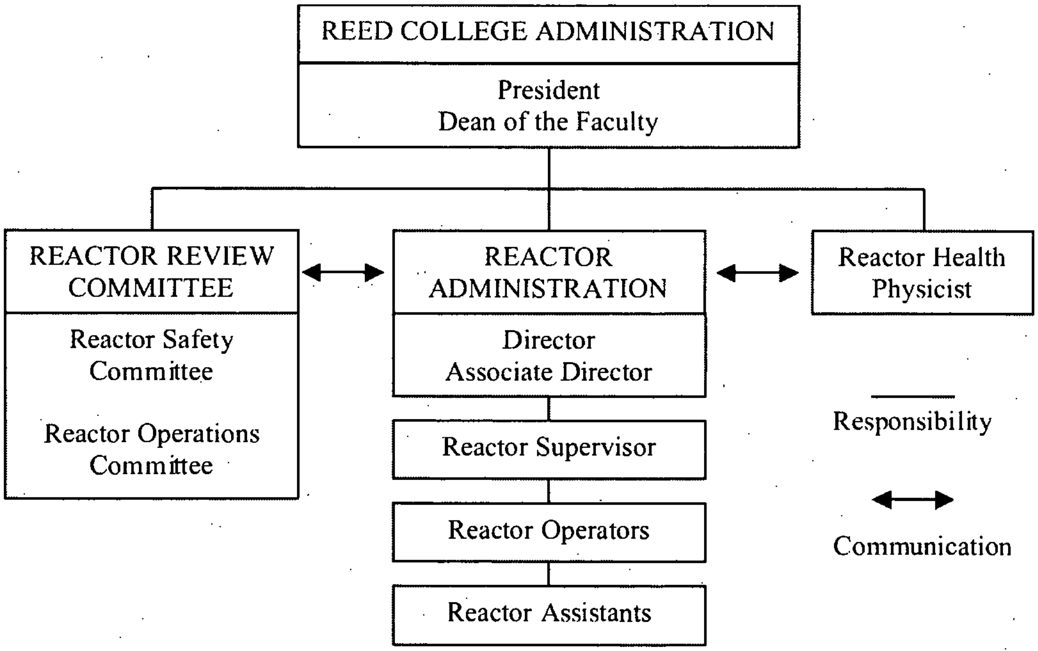

Figure 1 Reed Research Reactor organization chart

2 Reactor Review Committee

The Reactor Review Committee (RRC) is established to provide for the independent review and audit of reactor facility operations and to advise the President of Reed College regarding these matters. The RRC may meet either as a single committee or as subcommittees to be known as the Reactor Operations Committee (ROC) and the Reactor Safety Committee (RSC). Except for ex-officio members, the President of Reed College appoints committee members and chairs. The Reactor Director, Associate Director, and Supervisor serve as non-voting members of all committees and subcommittees. No limit shall exist on the overlap of personnel, including chairperson(s), between the subcommittees. Terms of office will normally run from September 1 through August 31.

Minutes shall be maintained of committee decisions. Each subcommittee shall meet at least twice each academic year.

2.1 The Reactor Operations Committee

The ROC shall deal with the day-to-day operations of the reactor, reactor maintenance, reactor safety, and operator training and requalification. Members of the ROC are expected to have a background in reactor, mechanical, or electrical engineering, nuclear physics, nuclear chemistry, or other similar technical fields.

The ROC is expected to guide the operations of the reactor from a technical standpoint, making certain that the technical concerns of federal, state, and private insurance agencies are responded to in a timely and technically correct manner. The ROC shall be composed of Reed faculty and others (according to their experience).

The Reactor Operations Committee shall review the following items:

a) Determinations that proposed changes in equipment, systems, tests, experiments, or procedures do not involve an unreviewed safety question as defined in 10CFR50.59.

b) New and modified Standard Operating Procedures as specified in Part VII of these Administrative Procedures.

c) All new experiments.

d) Proposed changes in the Facility License or Technical Specifications (with the RSC).

e) Violations of the facility License or Technical Specifications (with the RSC).

f) Violations of internal procedures or instructions (with the RSC).

g) Fuel movement or core configuration changes.

h) Any Reportable Occurrences to federal or state regulatory agencies.

i) Operator training program.

j) Operator requalification program.

k) Unexplained scrams prior to restart of the reactor and the written procedures to be followed for the restart.

2.2 Reactor Safety Committee

The RSC shall be concerned with emergency preparedness, health physics, radiation safety, physical security, environmental impact, and the interface between the Reed Research Reactor and the Reed College Campus and the surrounding Community. In addition, the RSC will be responsible for evaluating the yearly emergency drill. The members of the RSC are expected to have a background in emergency planning, health care, environmental issues, health physics, or be concerned with community issues. The RSC shall be composed, aside from ex-officio members, of individuals not connected with operation of the reactor.

The Reactor Safety Committee shall review the following items:

a) Radiation exposure records.

b) Radiation safety and ALARA program.

c) Physical security.

d) Personnel safety.

e) Emergency drills and scenarios.

f) Emergency Planning, Implementation, and Preparedness.

g) Radioactive waste disposal.

h) Radioactive material releases.

i) Community affairs.

j) Interface between the facility and Portland Police Bureau, Portland Fire Department, Oregon Energy Facilities Siting Council, and the Multnomah County Emergency Management Plan.

k) Proposed changes in the License or Technical Specifications (with the ROC).

l) Violations of the License or Technical Specifications (with the ROC).

m) Violations of internal procedures or instructions (with the ROC).

2.3 Audits

Members of the RRC who are assigned responsibility for audits shall perform or arrange for examination of operating records, logs, and other documents. Discussions with cognizant personnel and observation of operations shall be used as appropriate. In no case shall the individual immediately responsible for an aspect of facility operation audit that area.

The purpose of audits is to determine if activities since the last audit were conducted safely and in accordance with regulatory requirements and applicable procedures. In addition to checking the controlling document or procedure, the audit should verify that the records are completed and retrievable, that the procedures are clear, that deficiencies in previous audits have been addressed, and that the procedure fulfills the intended function.

The status of the reviews and audits shall be a standing agenda item for all committee meetings. Deficiencies uncovered in audits that affect reactor safety shall immediately be reported to the President of Reed College by the chairperson of the Committee. A written report of the findings of the audit shall be submitted to the Director after the audit has been completed.

The RSC shall audit the following items each academic year (except as noted):

a) Facility License (every four years).

b) Technical Specifications (every four years).

c) Administrative Procedures.

d) Major facility documents relating to reactor safety to identify changes needed.

e) Standard Operating Procedures relating to health physics, environmental monitoring, calibration of monitoring equipment, and security. Each procedure shall be audited at least once every two years.

f) The Emergency Plan and Emergency Implementation Procedures.

g) Facility operations and logs relating to safety.

h) The Radiation Protection Plan.

The ROC shall audit the following items each academic year (except as noted):

a) Major facility documents relating to facility operations.

b) The Requalification Plan and requalification records.

c) Standard Operating Procedures, except for those assigned to the RSC above. Each procedure shall be audited at least once every two years.

d) Facility operations and logs other than those assigned to the RSC above.

3 Reactor Operations

3.1 Staffing

When the reactor is not shutdown, as defined in the Technical Specification the following conditions must be met:

a) A licensed operator who is current in requalification shall be at the console able to observe and respond to alarms.

b) At least two persons shall be present within the facility. The second person may leave the facility briefly to take readings.

c) The Senior Reactor Operator of record must either be present within the facility, or must be located on the Reed campus in such a way that he or she is able to get to the facility within 5 minutes or less, is easily reachable at all times (such as by telephone or pager), and such that the operator on duty knows his/her location prior to beginning operation.

The SRO must be in the facility during the first core excess of the day, during a return to power following an inadvertent scram, and during fuel movements.

All reactivity changes shall be made by, or in the presence and under the direction of, an NRC-licensed operator.

3.2 Checklists and Logs

The operator must certify, except during continuous runs, the completion of the Reactor Startup Checklist before each day's reactor operations are begun. Completion of this checklist ensures that:

a) The mechanical and electrical components of the reactor have been tested and found to be in satisfactory working condition;

b) The radiological safety devices positioned around the reactor have been calibrated and tested for proper operation; and

c) The limits on operating conditions, e.g., scram circuits, interlocks, and alarms, have been tested and accurately set.

At the end of each operating day, except during continuous runs, a Shutdown Checklist shall be completed. This checklist constitutes a status report on the condition of the reactor at the end of each operating day. The Senior Reactor Operator shall sign it before the end of the calendar day.

The periodic surveillance checklists shall be completed as appropriate. It is not required that each item indicated on each checklist be done at one time; they may be spread out over several days.

The format for the various reactor checklists may be changed at the discretion of the Director with the concurrence of the ROC.

The reactor operating logs and all checklists are to be considered official records and must be kept on file.

3.3 Reactivity and Fuel

Changes in core loading, or insertion of new experiments shall be made only under the supervision of a Senior Reactor Operator.

The reactor shall not be operated for routine operations with fuel elements that are known to be damaged without specific approval from the ROC and RSC. If any evidence of fuel element damage exists, the Reactor Supervisor shall propose a program for locating the damage, which may include operating the reactor to locate the damage.

An NRC-licensed operator shall be present during routine maintenance. At least two persons, one of whom holds an NRC Senior Reactor Operator license, shall be present whenever maintenance is performed on a reactor control system.

The reactivity worth of samples containing fissionable material must be determined in position by operating the reactor at 5 W power and the result compared with the Technical Specifications before the sample can be activated in the reactor operating at higher power levels. The only exception shall be for pneumatic tube irradiations of naturally occurring fissionable nuclides to produce at most 2E10 fissions. All such experiments shall be treated as New experiments each time they are performed.

4 Reactor Experiments

4.1 Classes of experiments

There are two classes of experiments:

a) <u>Routine experiments</u> are those that involve operations under conditions which have been extensively examined in the course of the reactor test programs. Under the Facility License, routine operation within the limits of the Technical Specifications applicable to the reactor is permissible at the discretion of the Director and no further review is necessary.

b) New experiments are those that may be performed under the Technical Specifications and are not routine experiments. New experiments shall be authorized by the review procedure given below.

4.2 Procedure for Review of New Experiments

Proposals for the performance of new experiments and associated changes in operating procedures, administrative procedures, or reactor instrumentation are subject to the following review procedure in advance of reactor operation:

a) The Director must submit a complete written description of the proposed action to the ROC.

b) The ROC must review and approve the proposal. If approved, the experiment may be performed at the discretion of the Director.

c) If an experiment involves radiation safety questions, the ROC may request concurrent approval by the RSC.

The description of the proposed experiment or change must contain sufficient detail to enable the ROC to evaluate the safety of the experiment.

The following data must be included in the description:

a) Object of the experiment.

b) Description of the experiment. This will include a discussion of both the equipment and the experimental methods to be used. If the experiment involves making a change in the existing core, the maximum change in reactivity that can be introduced with this experiment should be estimated and should be stated in the proposal. The experiment shall be considered for its effect on reactor operation, and the possibility and consequences of its failure including any significant consideration of interaction with core components.

c) Equipment required. This is for the information of the operating staff.

d) Time required for the experiment (including setup and take down time).

e) Date on which the equipment and experiment will be ready.

f) Names of individuals who will perform the experiment.

A copy of the description of the new experiment, as finally approved, shall be filed in the Reactor Facility. Once a new experiment has been conducted, it can become a routine experiment at the discretion of the ROC.

4.3 Radionuclide Production for Campus and Off-Campus Users

Radionuclide production for other State or NRC licensees shall be limited by the terms of their specific NRC or state license. A copy of the license covering the particular radionuclide requested must accompany a request for the production of any radioactive materials.

Proper transportation of all radioactive materials from the facility shall be the sole responsibility of the requestor. Part of the approval for production of radioactive materials shall be the certification by the requestor that arrangements have been made which ensure that the transportation complies with all applicable regulations (NRC, Department of Transportation, State of Oregon, and if applicable, state of final destination of shipment).

The Director may authorize the reactor staff to transport radioactive materials to a common carrier that is qualified to accept such materials for shipment.

The Director may lease to a qualified recipient of radioactive materials, a DOT-approved shipping container to be used by the recipient and a common carrier, or only by the recipient, for the transportation of such materials.

No radioactive material shall be transferred to any person who has not been approved by the Director to receive such material.

5 Handling, Storage, And Disposal Of Radioactive Material

The operator shall keep a record of the radiation level of the specimen when removed from the reactor.

The operator shall record the transfer to an authorized person, the name of such person, the time of the transfer, and the dose rate at one foot (30 cm) from the surface of the container at the time of the transfer. A copy of the record after disposal of the specimen will be kept in the Reactor Facility.

Radioactive material remaining at the facility shall be stored in a properly labeled area in accordance with 10 CFR Part 20. There shall be no area in the Facility that is not labeled where the radiation level exceeds 5 mrem/hr.

No radioactive sample or specimen should be stored in the Facility in excess of one year, unless it is to be used at a later time. If the sample or specimen has not decayed to negligible levels within six months, it should be prepared for disposal.

The Health Physicist will dispose of all radioactive waste by shipment to a waste disposal area in accordance with all applicable regulations. The waste disposal area used by the facility is the Hanford Site (U.S. Ecology, Inc.).

6 Fuel and Special Nuclear Material

Special nuclear material is the property of the United States Department of Energy (DOE). It is on lease to the Reed Institute (Reed College) which is accountable to DOE for its location and proper handling.

The Director is responsible for all fissile and fertile material in the Facility.

The Director is accountable to the Reactor Operations Committee for any changes in the fuel configuration in the reactor core and for proper storage of used and spare fuel elements.

7 Access To The Reed Research Reactor

The only entrance to the Reactor Bay when the reactor is in operation shall be through the Control Room. The register for visitors and storage racks for self-reading dosimeters shall be located in this room.

Access to the Reactor Bay will be permitted only to persons who have been given the necessary authorization as set forth in the Security Plan.

8 Adoption And Revision Of Operating Procedures

Any changes to Standard and Emergency Operating Procedures which directly affect safety of the facility and/or personnel shall be approved in advance by one or both of the reactor committees, as designated in the Technical Specifications and dictated by the expertise of the committee members.

Those items that directly affect safety, and hence constitute safety standards as referred to in the Technical Specifications, include, but are not limited to:

a) Emergency Implementation Procedures
b) Establishment of radiation dose limits for employees, students, and visitors to the facility, including ALARA policies
c) Establishment of limits for operation including scram and interlock set points, area and release radiation levels and warning set points
d) Fuel handling, loading, or unloading
e) Control rod removal and replacement
f) Elimination, modification, or replacement of reactor monitoring systems

Changes in existing procedures (including procedures which may contain safety standards) and adoption of new Standard Operating Procedures may be instituted by the Director without prior committee approval provided the change itself does not fall into any of the categories described in paragraph 7.1. The Director shall inform the committee chairs of any such change implemented.

The Reactor Supervisor may make changes that are of a purely editorial nature, such as corrections of spelling errors, grammar, formatting, or wording clarifications, with notification to the Director.

Any reactor operator or staff member who believes that a proposed change affects facility or personnel safety may notify the Director and the Chair of the RRC. Such change shall then not take effect until the committee has considered and taken action on the change.