REED RESEARCH REACOR
LICENSE NO. R-112
DOCKET NO. 50-288

REED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION WITH A REVISED SAFETY ANALYSIS REPORT AND TECHNICAL SPECIFICATIONS

JULY 2010

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

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July 30, 2010

ATTN: Document Control Desk
U S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Docket: 50-288
License No: R-112
Subject: RAI TAC NO. ME1583

Attached are some of the answers to the subject RAI dated March 8, 2010. As noted under the individual RAIs, some of the information needed for response is to be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010. The response and attachments do not contain any sensitive information.

Please contact us if you have any questions. Thank you.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 7-30-10

Stephen G. Frantz
pDirector, Reed Research Reactor.

Attachments:

1. RRR RAIs 1-58
2. RRR SOP 34, Control Rods
3. RRR SOP 34B Control Rod Calibration January 2010
4. RRR SAR Chapaters that have changed since the 2007 submital

A020
NRR

1. *NUREG-1537, Part 1, Section 1.4, Shared Facilities and Equipment, states the applicant should consider whether the loss of any shared facilities or equipment could lead to a loss of function that would lead to an uncontrolled release of radioactive material, or if released, are analyzed and found to be acceptable. The 2007 SAR, Section 1.4 discusses this subject. However, the discussion is incomplete in that it does not include the loss of electricity and how it would affect the release of radiation should it coincide with the loss of fuel cladding integrity. Please provide this information including the loss of alarms, automatic isolation, operation of heating, ventilation, and air conditioning (HVAC) systems, etc. Please provide information concerning whether the analysis provided in Chapter 13 envelopes this condition.*

   See updated SAR Section 1.4.

   Although Chapter 13 will not be completed until November 2010, that analysis will envelope this condition.

2. *NUREG-1537, Part 1, Section 1.5, Comparison With Similar Facilities states the applicant should use pertinent information from other reactors and this information can be used to compare the safety envelope of Reed Research Reactor (RRR) and to support analysis in appropriate chapters of the SAR. The 2007 SAR discusses this, but the information is incomplete. Please provide a comparison of the RRR to other TRIGA facilities so as to characterize the degree to which generic information or operational experience from other reactor facilities is applicable.*

   See updated SAR Section 1.5.

3. *NUREG-1537, Part 1, Section 2.2, Nearby Industrial, Transportation, and Military Facilities, states information on nearby military facilities be included in the SAR. The2007 SAR, Section 2.2 discusses industrial and transportation facilities but does not discuss military installations. Please provide information concerning the nearby military installations.*

   See updated SAR Section 2.2.

4. *NUREG-1537, Part 1, Section 3.1, Design Criteria, states the applicant should identify the design criteria that are applicable to each structure, system and component that performs a safety function. The 2007 SAR, Section 3.1 briefly addresses this matter and states "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." Please provide the criteria applicable to the original design and construction and to subsequent modifications to the design and construction.*

   See updated SAR Section 3.1.

5. *NUREG-1537, Part 1, Section 3.3, Water Damage, requires the applicant identify the potential for flooding which could prevent structures, systems and components from performing their safety function. The 2007 SAR, Section 3.3 states "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." However, this information is incomplete. Please provide*

information that demonstrates that should flooding occur, it will not prevent operation of the RRR safety systems.

See updated SAR Section 3.3.

6. *NUREG-1537, Part 1, Section 3.5, Systems and Components, states the applicant should identify the bases or design features of the electromechanical systems that are used to ensure safe operation and shutdown of the reactor during all conditions. The2007 SAR, Sections 3.5 and 3.6 provide some information on this topic but do not provide information on the design features of the control rods (e.g., fail safe in the event of loss of power) or the systems associated with reactor operation and safety (e.g., power level scrams, interlocks to limit reactivity insertion). Please provide the design for electromechanical systems and components required for operation, shutdown and to maintain shutdown.*

   See updated SAR Section 3.5 and 3.6.

7. *NUREG-1537, Part 1, Section 4.2.1, Reactor Fuel, states the applicant should describe the fuel elements used in the reactor including detailed design information. References should be provided to demonstrate that the design basis assures that integrity of the fuel is maintained under all conditions assumed in the safety analysis. The description should also include information necessary to establish limiting conditions beyond which fuel integrity would be lost. The 2007 SAR, Sections 1.3.3, 4.2.4 and 4.2.5 which provide some information are incomplete. Please discuss the differences in fuel length for the aluminum and stainless steel clad fuel utilized in the core and the implications of these differences on analysis. In addition, please address mechanical forces and stresses, corrosion and erosion of cladding, hydraulic forces, thermal changes and temperature gradients, and internal pressures from fission products and the production of fission gas. Include in the analyses the impact of radiation effects, including the maximum fission densities and fission rates that the fuel is designed to accommodate.*

   This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

8. *NUREG-1537, Part 2, Section 4.2.2 states the control rods should be sufficient in number and reactivity worth to comply with the 'single stuck rod' criterion; that is, it should be possible to shut down the reactor and comply with the requirement of minimum shutdown margin with the highest worth scrammable control rod stuck out of the core. The control rods should also be sufficient to control the reactor in all designed operating modes and to shut down the reactor safely from any operational condition.*

   The control rods, blades, followers (if used), and support systems should be designed conservatively to withstand all anticipated stresses and challenges from mechanical, hydraulic, and thermal forces and the effects of their chemical and radiation environment.

   The control rods should be designed so that scramming them does not challenge their integrity or operation or the integrity or operation of other reactor systems.

   The 2007 SAR, Sections 4.2.7, 4.2.8 and 4.2.9, while providing some of this information, is incomplete. The SAR does not provide the worths of the 3 RRR control rods. Please provide calculated and measured control rod worths under all conditions of operation. Please determine if control rod withdrawal insertion limitation limits (rod position vs. power) are necessary to preserve assumptions in the departure from nucleate boiling ratio (DNBR) analysis.

This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

9. *NUREG-1537, Part 1, Section 4.2.3, Neutron Moderator and Reflector, states the applicant should describe reflectors and moderators designed into the core and their special features. The 2007 SAR, Sections 4.2.2 and 4.2.6 RRR provides a discussion of the radial reflector and the graphite reflector elements; however, it does not provide any information pertaining to the naturally circulating water which is also moderator/reflector. Please provide a description of the water moderator and reflector and an assessment of the function and importance of the moderator and the effect of loss of moderator on the behavior of the reactor core during operations.*

   This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

10. *NUREG-1537, Part 1, Section 4.2.4 Neutron Startup Source, states the applicant should describe the neutron source used for reactor startup. The 2007 SAR, Section 4.2.10 provides a description of the neutron source holder only. Please review the cited requirement and then supply a revised description of the neutron source in use at RRR including the following:*

    - *type of neutron source including information on neutron startup material*
    - *type of nuclear reaction*
    - *energy spectra of neutrons*
    - *source strength*
    - *interaction of the source and holder, while in use, with the chemical,*
    - *thermal, and radiation environment*
    - *design features that ensure the function, integrity, and availability of the source*

    See updated SAR Section 4.2.4.1 to be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

11. *NUREG-1537, Part 1, Section 4.2.5, Core Support Structure, states the applicant should describe structural performance of the core support structure under all reasonable conditions. Furthermore, it is required that the design basis, operational analysis and safety considerations should be provided for each reactor component placed on the grid plate. The 2007 SAR, Sections 4.21 and 4.23, while providing some of this information, are incomplete. Please provide information demonstrating the adequacy of the core support structure under flooded and empty tank conditions to support all required components under all operating conditions.*

    See updated SAR Section 4.1.1 to be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

12. *NUREG-1537, Part 1, Section 4.3, Reactor Tank or Pool, states the applicant should describe the reactor tank and associated components and provide assurances regarding those components to perform their intended function free from any problems associated with chemical interactions, failure of penetrations and welds that could lead to loss of coolant, and to propose TS that impose limiting conditions. In addition, the applicant should assess the possibility of uncontrolled leakage of contaminated coolant and should discuss detection, preventive and protective measures. The 2007 SAR, Section4.1, while providing some of this information (e.g., a physical description), is*

incomplete. Please provide information regarding loss of water through failure in the tank including detection methods and consequences, chemical compatibility of components; resistance to corrosion, suitability of penetrations below the normal coolant level, and propose TS applicable to these topics.

See updated SAR Section 4.3 to be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

13. *NUREG-1537, Part 1, Section 4.4, Biological Shield, states the applicant should describe the biological shied employed to ensure doses are in conformance with Title 10of the Code of Federal Regulations (10 CFR) Part 20. The 2007 SAR does not provide a characterization of the biological shield. Please provide a description of the biological shielding employed at RRR including consideration of concrete, tank structure and pool water.*

    See updated SAR Section 4.4 to be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

14. *NUREG-1537, Part 1, Section 4.5, Nuclear Design, states the applicant should discuss normal operating conditions, reactor core physics parameters and operating limits. The discussion should include a discussion of the complete, operable core; control rod worths; kinetic parameters; excess reactivities; shut down margins; and flux distribution or all planned configurations for the life of the core.*

    The 2007 SAR, Subsection 4.6.1 states:

    > *"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). ... 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 GA reactor cited was analyzed and licensed based on particular neutronic and thermal-hydraulic conditions pertinent to that reactor. RRR needs to establish the basis for incorporating the GA conclusions into the RRR SAR.

    In addition:

    The 2007 SAR, Subsection 4.6.2.1 (Excess Reactivity) discusses limiting RRR to +$3.00of core reactivity to prevent excessive fuel temperatures. However, the excess reactivity of RRR has not been established in the SAR.

    The 2007 SAR, Subsection 4.6.2.2 (Shutdown Margin) lists the shutdown margin Technical Specification requirement. However, the ability to meet this requirement is not presented in the SAR.

    The 2007 SAR, Subsection 4.6.2.3 (Reactivity Limits on Experiments) states that limiting reactivity insertions from experiments to -$1.00 will prevent sudden removals from causing excessive fuel temperatures. However, there is no analysis demonstrating this in the SAR.

The 2007 SAR, Subsection 4.6.3 (Stainless Steel Clad Fuel) assumes that there is no neutronic difference between the aluminum and the stainless steel clad fuel. However, there is no analysis establishing this and the stainless steel clad fuel meat is longer by 1inch and stainless steel is neutronically different from aluminum.

Please provide the information that addresses the stated points and that provides core physics parameters consistent with the NUREG citation.

This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

15. *NUREG-1537, Part 1, Section 4.5.2, Reactor Core Physics Parameters, states the applicant should describe reactor core physics parameters that determine operating characteristics as they are influenced by reactor design including:*

- *methods used to neutronically characterize the RRR,*
- *uncertainties required to apply calculated results to the RRR operation,*
- *methods to calculate kinetics parameters,*
- *coefficients of reactivity applicable to the RRR,*
- *comparisons with measurements to demonstrate the effectiveness of the methods employed, and*
- *changes in reactivity coefficients that result from changes to core configurations.*

The 2007 SAR, Section 4.6, does not provide this information. Please provide this information regarding methods, uncertainties, comparisons and all required technical parameters.

This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

16. *NUREG-1537, Part 1, Section 4.5.3, Operation Limits, states that the applicant should describe operating limits including those nuclear design features necessary to ensure safe operation and shutdown, namely:*

- *temperature coefficients or reactivity, void coefficients, Xe-Sm worths, power coefficients (if not otherwise accounted for), and the influence of experiments,*
- *minimum control rod worths and stuck rod worths for all allowed core conditions,*
- *transient analysis of an uncontrolled rod withdrawal,*
- *shutdown margin calculations for limiting core conditions, and*
- *technical specification implemented to ensure safe operation.*

The 2007 SAR, Section 4.6 describes some of these limits but is incomplete. Please provide information specific to the RRR regarding methods, uncertainties, comparisons and all technical parameters as identified in NUREG 1537.

This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

16b. *NUREG-1537, Part 1, Section 4.6, Thermal-Hydraulic Design states the applicant should describe operating limits on cooling conditions necessary to prevent fuel overheating and to ensure that fuel integrity will not be lost under any reactor conditions including accidents. Technical characteristics are that the DNBR limit of 2 is never violated and flow instability may not contribute to a loss of fuel cooling under any conditions. The 2007 SAR, Section 4 does not provide this information. Please provide information regarding methods, uncertainties, and results of a DNB analysis showing* *that the safety limits proposed will never violate the limits stated. Please provide information concerning restrictions on pool temperature, inlet temperature, adequacy of bottom grid geometry, spacer geometry, and nuclear issues such as peaking factors, rod insertion limits, delay times, and measurement uncertainties affecting DNB analysis.*

This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

17. *NUREG-1537, Part 1, Section 5, Reactor Coolant Systems, states the applicant should demonstrate that the system can remove the fission and decay heat from the fuel during reactor operation and decay heat during reactor shutdown. The 2007 SAR Section 5 describes the reactor coolant systems but is incomplete because it does not discuss the capability of the systems. Please provide a discussion of the capability of the cooling systems.*

    See updated SAR Section 5.1.

18. *NUREG-1537, Part 1, Section 5.2, Primary Coolant System, states the primary coolant should provide a chemical environment that limits corrosion of fuel cladding, control and safety rod surfaces, reactor vessels, and other essential components. The 2007 SAR, Section 5.2 describes components of a system to control coolant conductivity and pH without describing the objectives stated. Please provide and justify the value of electrical conductivity and pH that is used for controlling and maintaining chemical environment in the primary coolant system.*

    See updated SAR Section 5.4.

19. *NUREG-1537, Part 1, Section 5.3, Secondary Coolant System, states the applicant should discuss the secondary coolant system recognizing that some non-power reactors are designed with secondary coolant systems that will not support continuous reactor operation at full licensed power. This is acceptable, provided the capability and such limiting conditions as maximum pool temperature are analyzed in the SAR and included in the TS. The 2007 SAR, Section 5.3, while discussing the secondary coolant system, inadequately discusses the capabilities of the secondary coolant system and the bases of the TS on maximum pool temperature. Please provide information on heat load as it pertains to the secondary coolant system and review Technical Specification 3.8 which states that the basis for the pool temperature limit is protection of the resin beds and does not address the limits on pool temperature.*

    See updated SAR Section 5.3.

20. *NUREG-1537 guidance states in Section 5.4, Primary Coolant Cleanup System, the applicant needs to ensure that when operating the system, exposure and release of radioactivity do not exceed the requirements of 10 CFR Part 20 and are consistent with the facility ALARA program. The 2007 SAR, Section 5.2.4 does not address the consistency of the cleanup system with the ALARA program. Please provide information that operation of the cleanup system does not challenge the commitment of the ALARA program of RRR.*

    See updated SAR Section 5.4.

21. *NUREG 1537, Part 1, Section 5.5, Primary Coolant Makeup Water System, states the applicant needs to ensure that: the makeup water system or plan should include*

provisions for recording the use of makeup water to detect changes that indicate leakage or other malfunction of the primary coolant system. The 2007 SAR, Sections 5.26 and 5.4, while discussing aspects of the detection system, is incomplete in that it does not provide information concerning the provisions or plans to indicate leakage or other malfunction of the primary coolant system. Please provide information concerning provisions and plans to detect abnormal leakage in the primary system.

See updated SAR Section 5.5.

22. *NUREG-1537, Part 2, Section 5.6, Nitrogen-16 Control System, states the applicant should confirm the amount of nitrogen-16 (N^{16}) predicted by the SAR analysis at the proposed power level and the potential personnel exposure rates, including exposures from direct radiation and airborne N^{16}. The 2007 SAR, Section 5.5, describes the N^{16} control system but provides no information on confirmation of effectiveness and exposure rates. Please provide information concerning the amount of N^{16} produced during operation at full power and the resulting personnel exposures.*

    See updated SAR Section 5.6.

23. *NUREG-1537, Part 1, Section 9.1, Heating, Ventilation, and Air Conditioning Systems, states the applicant should consider modes of operation and features of the HVACs stem designed to control (contain or confine) reactor facility atmospheres, including damper closure or flow-diversion functions, during the full range of reactor operation. The 2007 SAR, Section 9.1, describes the general features of the HVAC system but does not describe how isolation is initiated, the set-points used, or the TS governing the use and testing of the system. Please provide information concerning the HVAC and address the above.*

    See updated SAR Section 9.1.

24. *NUREG-1537, Part 2, Section 9.2, Handling and Storage of Reactor Fuel, states the applicant should consider the methods, analyses, and systems for secure storage of new and irradiated fuel that will prevent criticality (k_{eff} not to exceed 0.80) under all conditions of moderation during storage and movement. The 2007 SAR, Section 9.2 states that the spacing in the rack is sufficiently far apart to prevent accidental criticalities. However, analysis supporting this statement is not provided or referenced. Please provide this information for the fuel rack design.*

    This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

25. *NUREG-1537, Part 1, Section 9.3, Fire Protection Systems and Programs, states the applicant needs to discuss fire protection systems and plans that would affect reactor safety systems. The 2007 SAR, Section 9.3, discusses this issue. However, there is no discussion of the sources of fire or expected outcomes that would affect safety systems. Fire barriers protecting safety systems are not discussed. Please provide information regarding fire sources and outcomes consistent with the guidance.*

    See updated SAR Section 9.3.

26. *NUREG-1537, Section 9.7, Other Auxiliary Systems, states the applicant should discuss auxiliary systems that are not fully described in other sections that are important to the safe operation and shutdown of the reactor, and to the protection of the health and safety of the public, the facility staff, and the environment. The 2007*

SAR, Section 9.7.1, discusses a reactor bay crane. However, there is no discussion regarding prohibiting the movement of heavy objects over the reactor core. Nor is there discussion regarding operating procedures, load testing and required maintenance and surveillances of the crane. Please provide information relating to crane limitations (if any) and procedures for using and parking the crane.

See updated SAR Section 9.7.1.

27. *NUREG-1537, Part 2, Section 10.1, Summary Description, states the applicant should discuss:*

    - *limiting experimental characteristics (e.g., reactivity, contents)*
    - *monitoring and control of the experiments and the interaction between the experiment and the reactor control and safety systems*
    - *design requirements for the experiment and the review and approval process.*

    The 2002 SAR, Section 10.1 presents a summary description. However, the information provided is not in sufficient detail to enable conclusions to be drawn regarding the safe operation of the experimental facilities. Please provide a description of the principal features of the experimental and irradiation facilities including experimental limitations.

    See updated SAR Section 10.2.

28. *NUREG-1537, Part 1, Section 10.2, Experimental Facilities, states the applicant should discuss the experiment safety system and the functional interface between the experimental safety system and the reactor protection system. The 2007 SAR, Section 10.2 discusses the experimental facilities. However, the discussion only addresses physical features and does not provide any information regarding safety, assurance of independence, or compliance with requirements. Please provide information regarding the interface between reactor safety systems and experiment safety systems. Provide information on design requirements and how the design requirements are met.*

    See updated SAR Section 10.2.

29. *NUREG-1537, Part 1, Section 10.3, Experiment Review, states the experiment review committee should have the appropriate scope of responsibility, including the review of procedures that pertain to the use of experimental facilities. The 2007 SAR, Section 10.3 does not state this authority for the Reactor Review Committee and the scope of the Committee's review appears to be limited. Please provide information on the Committee's authority to review and approve procedures including procedures for the experimental facilities.*

    See updated SAR Section 10.3.

30. *NUREG-1537, Part 1, Section 11.1.1, Radiation Sources, states that the applicant should present the best estimates of the maximum annual dose and the collective doses for major radiological activities during the full range of normal operations for facility staff and members of the public. The doses shall be shown to be within the applicable limits of 10 CFR Part 20. The 2007 SAR, Section 11.1.1.1 provides calculations, using maximizing assumptions, that result in values greater than the applicable limits in 10 CFR Part 20, Appendix B. Please provide the results of best estimate calculations that demonstrate compliance with 10 CFR Part 20, Appendix B.*

    See updated SAR Section 11.1.1.

31. *NUREG-1537, Part 1, Section 11.1.2, Radiation Protection Program, guidance states program procedures need to establish clear lines of responsibility and clear methods for radiation protection under normal and emergency conditions. Also, procedures should be organized and presented for convenient use by operators and technicians at appropriate locations, and should be free of extraneous material. The 2007 SAR, Section 11.1.2, provides a description of the program and, the attached Radiation Protection Plan, references procedures used for various activities concerning radiation protection. However, the other NUREG-1537, Part 1 attributes cannot be established from review of the SAR. Please provide information which shows that clear lines of responsibility and clear methods for radiation protection are established for normal and emergency conditions.*

    See updated SAR Section 11.1.2.

32. *NUREG-1537, Part 1, Section 11.1.2, Radiation Protection Program, and Section 11.1.5, Radiation Exposure Control and Dosimetry, guidance states the radiation protection program records management system should include records such as ALARA program records, individual occupational dose records, monitoring and area control records, monitoring methods records, and training records. The 2007 SAR, Section 11.1.2 provides a description of the program including management, administration, and training. Section 11.1.5 provides information regarding exposure records. However, the other required attributes have not been discussed. Please provide information concerning the maintenance of records and that demonstrate acceptance with the above criteria.*

    See updated SAR Section 11.1.2.

33. *NUREG-1537, Part 2, Section 11.1.4, Radiation Monitoring and Surveillance, states the bases of the methods and procedures used for detecting contaminated areas, materials, and components should be clearly stated. The 2007 SAR provides the surveillance frequency for contamination as biweekly for the reactor bay, control room, and facility. Section 5 of the RRR Administrative Procedures for Handling, Storage, and Disposal of Radioactive Material indicates that the operator shall keep a record of the radiation level of the specimen when removed from the reactor. However, the procedures do not address possible contamination of the sample. Please provide any additional bases or methods that are used for detecting contaminated materials and components, including the measures taken to ensure experimental samples being removed have not become contaminated.*

    See updated SAR Section 11.1.4.

34. *NUREG-1537, Part 1, Section 11.1.4, Radiation Monitoring and Surveillance, states the bases of the methods and procedures used for detecting contaminated areas, materials, and components should be clearly stated. The 2007 SAR, Section 11.1.4 provides a brief discussion of monitoring equipment and Table 11.10 of the 2002 SAR provides a listing of typical monitoring equipment. However, the methods and procedures used for detecting contaminated areas, materials, and components cannot be learned from the information provided. Please provide information on the methods and procedures for sampling and monitoring air, liquids, solids, and reactor radiation beams and effluents.*

    See updated SAR Section 11.1.4 and 11.1.16.

35. *NUREG-1537, Section 11.1.6, Contamination Control, states the contamination control*

program should include provisions to avoid, prevent and remedy the occurrence and spread of contamination. The 2007 SAR, Section 11.1.6 provides the most likely sites of contamination and the measures taken to minimize the spread of contamination. This section also states that staff and visiting researchers are trained on the risks of contamination and techniques for avoiding, limiting and controlling contamination. However, contamination of personnel is not addressed. Please describe the means for addressing personnel contamination, if it should occur.

See updated SAR Section 11.1.6.

36. *NUREG-1537, Part 1, Section 13.1.1, Maximum Hypothetical Accident, guidance states the applicant needs to present a methodology for reviewing the systems and operating characteristics of the reactor facility that could affect its safe operation or shutdown. The methodology should be used to identify limiting accidents, analyze the evolution of the scenarios, and evaluate the consequences. The 2007 SAR, Section 13.2.1 discusses the Maximum Hypothetical Accident (MHA) and provides the method and assumptions used to estimate potential consequences from an MHA and discusses compliance with 10 CFR Part 20. However, the discussion is not complete and requires further clarifications.*

Please provide the following information:

a. *Provide the approach used in determining the average thermal reactor power over 40 years.*

b. *Given a thermal power of 250 kw operating 8 hours per day, 5 successive days, provide the method to show the average utilization (kw-hr/day) indicated in the SAR.*

c. *In Chapter 4 Section 4.2.4 and in Figure 4.4 of the 2002 SAR, fuel rods with various Uranium 235 (U^{235}) contents have been described. In addition, the U^{235} content of fuel rods will vary because of burn-up. This would indicate the presence of different power level per rod, affecting the estimate for a peak rod power level. Provide clarification on the method used for assigning a peaking factor of 2.*

d. *Subsection 13.2.1.2, Radionuclide Inventory Buildup and Decay, describes a power level and number of fuel rods that is inconsistent with those provided in the preceding subsection. Please clarify.*

e. *Subsection 13.2.1.2 contains a subsection, Data from ORIGEN Calculations. The text refers to values in Appendices A and B where as there are Appendices A through F in this section. Please clarify.*

f. *In Chapter 13, Appendix B, the heading indicates an ORIGEN input for irradiation at "1 watt 8 hours per day for 5 days". Should this be irradiation at "1 kw 8 hours per day for 5 days?" Please clarify.*

g. *In Chapter 13, Appendix D, the heading indicates an ORIGEN input for irradiation at "1 watt 8 hours per day for 5 days". Should this be irradiation at "1 kw 8 hours per day for 5 days?"*

h. *In Chapter 13, Appendix E, there is confusion concerning the number of fuel rods. Please clarify.*

i. *In Chapter 13, Appendices E and F, it is not clear how the values are produced from those provided in Appendices C and D. Please provide an example of the method used. In addition, the headings for data presented in Appendices E and F do not appear to be correct. Please clarify.*

j. *Chapter 13, Table 13.5 provides values in the third column (A, activity (n-Ci)) of the released* *curies. Discuss the method used to determine these values. It appears that the values given in this column are 2.5 times less than those given in Tables 13.3 and 13.4. Please clarify.*

This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

37. *NUREG-1537, Part 1, Chapter 13, Accident Analysis, states the applicant needs to present a methodology for reviewing the systems and operating characteristics of the reactor facility that could affect its safe operation or shutdown. The methodology should be used to identify limiting accidents, analyze the evolution of the scenarios, and evaluate the consequences. The 2007 SAR, Section 13.2.3 presents an analysis of the LOCA and provides radiation dose rates in Tables 13.6 and 13.9 after extended operation at 250 kw and 1 MW respectively. The values in Table 13.9 at various times after shutdown are smaller than those in Table 13.7 for same times after shutdown. Please clarify this discrepancy.*

    This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

38. *NUREG-1537, Part 1, Chapter 13, Accident Analysis, states the applicant needs to describe the mathematical models and analytical methods employed, including assumptions, approximations, validation, and uncertainties. The 2007 SAR, Section13.2.5 provides a descriptive analysis involving control rod worths whose origins and relationship to the RRR have not been established, and whose worths are combined additively without justification. Section 13.2.5 discusses the Experiment Malfunction accident and assumes a $1.00 reactivity worth for the experiment. It should be established that the experiment reactivity worth is a negative value and failure in the experiment introduces positive reactivity. The means for combining the worths need to be clearly presented. Please provide a revised presentation of the information.*

    This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

39. *NUREG-1537, Part 1, Chapter 14, Technical Specifications, states the applicant needs to establish technical specifications (TS) that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff. NUREG-1537, Part 1 provides guidance regarding TS in Appendix 14.1. The 2007 SAR, Chapter 14 presents proposed TS for the operation of the RRR. However, they do not incorporate all of the guidance (e.g., required action, completion time). Please consider proposing TS following the guidance of Appendix 14.1.*

    This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

40. *NUREG-1537, Part 1, Chapter 14, Technical Specifications, states the applicant needs to establish TS that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff. The licensee shall select appropriate safety criteria, establish a Safety Limit (SL) and then establish an associated Limiting Safety System Setting (LSSS) that will ensure that the SL is not exceeded. The 2007 SAR, Chapter 14,TS 2.0, establishes the SL at 300 kw when operating with aluminum clad fuel elements in the core. The associated LSSS is also set at 300 kw which will not ensure*

that the SL is not exceeded. Please provide clarification and justification for setting both limits at the same value.

This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

41. *NUREG-1537, Part 1, Chapter 14, Technical Specifications states the applicant should establish TS that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff. The licensee shall select appropriate safety criteria, establish a SL and then establish an associated LSSS that will ensure that the SL is not exceeded. The important parameter for a TRIGA reactor is the fuel rod temperature. The SL should be established based on the maximum permissible temperature of the fuel rod. The LSSS should be set so that the SL will not be exceeded under all conditions of operation. The 2007 SAR, Chapter 14, TS 2.0 establishes the SL and the LSSS using reactor power with no correlation of this power to fuel temperature. Please provide fuel rod temperatures at the power levels established for the SL and LSSS.*

This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

42. *NUREG-1537, Part 1, Chapter 14, Technical Specifications, states the applicant needs to establish TS that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff. The 2007 SAR, Chapter 14 in several sections of the TS refers back to sections of the SAR that do not exist, do not have the stated information discussed, or do not provide the requisite analysis required to validate the information in the Technical Specification.*

    For example:

    Technical Specifications 2.1.5, "Bases" states:

    "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."

    Technical Specifications 2.2.5, "Bases" states:

    "Analysis in the Safety Analysis Report, 4.5.3, demonstrates fuel centerline temperature does not exceed 6O0OC at power levels approximately 1.25 MW with bulk pool water temperature at approximately 100 °C."

    Technical Specifications 3.1.5, "Bases" states:

    "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."

    Technical Specifications 3.2.5, "Bases" states:

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

These calculations or sections do not appear in the SAR. Please provide the information supporting these statements in the TS.

This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

43. *NUREG-1537, Chapter 14, Technical Specifications, states the applicant needs to establish TS that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff. The 2007 SAR, Chapter 14, TS 3.14, Actions, presents required actions for various TS violations. However, it is incomplete in that it does not include conditions, required actions and completion time for the rate of reactivity insertion by control rod motion (i.e. no greater than 0.12% delta k/k/second. In addition, it states the limitations on experiments are found in Section 3.8 which is incorrect. Please provide this additional information and corrections.*

See updated SAR Chapter 14.

44. *NUREG-1537, Part 1, Chapter 14, Technical Specifications states the applicant needs to establish TS that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff. NUREG 1537, Part 1 provides guidance regarding TS in Appendix 14.1.*

Appendix 14.1 suggests that the maximum scram time should be specified for each scrammable rod and the specification should ensure that the drop times are consistent with the SAR analysis of reactivity required as a function of time to terminate a reactivity addition event accounting for measurement and calculational uncertainties.

The 2007 SAR, Chapter 14, TS 3.4.3, Specification, there is the statement, "Control rods are capable of 90% of full reactivity insertion from the fully withdrawn position in less than 1 second" but an associated action statement has not been included if the control rods fail to meet the specification.

Please provide information concerning why this has not been included. Additionally, automatic scram conditions are usually established, with associated actions, for reactor operations outside of the normal operating mode or normal conditions, (e.g. scram at110% of full licensed power or reactor tank coolant level below a specified normal operating value).

Conditions such as those described above are not clearly stated in the TS section of the application with associated required Surveillance Requirements and Actions. Please provide the missing information.

See updated SAR Chapter 14.

45. *NUREG-1537, Part 1, Chapter 14, Technical Specifications, states the applicant needs to establish TS that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff. Water level monitors for the reactor tank would provide information concerning possible tank leakage. The 2007 SAR, Chapter 14 does not include TS (Limiting Conditions for Operations (LCO) and/or Surveillance Requirements(SR)) for monitoring the water level and water additions to the tank. Please propose TS on reactor tank water level and water addition monitors which would provide assurance for early detection of a possible leak in the reactor tank.*

See updated SAR Chapter 14.

46. *NUREG 1537, Part 1, Chapter14, Technical Specifications, states the applicant needs to establish TS that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff. The 2007 SAR Chapter 14, TS 4.4.5 (BASES), it is stated that "the power level scram in not credited in the analysis, but provides assurance that the reactor is not operated in conditions beyond the assumptions used in the analysis (Table 13.2.1.4)." Neither the Table nor the analysis referenced could be located in the 2007 SAR. In addition, Section 13 of the SAR discusses accident analysis and does not normally provide a basis for a TS on the required measuring channels during operation. Please correct the TS.*

See updated SAR Chapter 14.

47. *NUREG-1537, Part 1, Chapter 14, Technical Specifications, states the applicant needs to establish TS that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff. The 2007 SAR, Sections 1.3.5.2, 3.5 and 9.1 state that if radioactive material releases associated with reactor operations occur, a controlled ventilation system minimizes exposure to reactor personnel and the public. Ventilation exhaust from the reactor room will shift to a filtered exhaust upon a manual signal or on high radioactivity of the air in the room and the function shall be tested semi-annually. However, a SR has not been established for testing the Gaseous Effluent Control System to ensure that it functions correctly when needed. Please propose a SR or provide a justification as to why one is not necessary.*

See updated SAR Chapter 14.

48. *NUREG-1537, Part 1, Chapter 14, Technical Specifications, states the applicant needs to establish TS that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff. The 2007 SAR, Chapter 14, TS 3.5 provides a LCO for the reactor bay ventilation system. The objective stated is to ensure that exposures to the public resulting from gaseous effluents released during normal operation and accident conditions are within limits. However, the LCO is incomplete in that it does not establish the conditions under which the ventilation system operates in the various modes possible. In addition, the discussion in the bases is incomplete. Please propose a TS limiting the operation of the ventilation system for normal and accident conditions.*

See updated SAR Chapter 14.

49. *NUREG-1537, Part 1, Chapter 14, Technical Specifications states the applicant needs to establish TS that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff. In the 2007 SAR, Sections 14, TS 2.2.4 and 3.2.4, it is stated in the Actions-Required Action section that if the SL or LCO is exceeded then the operator has the option of reducing the power level to the SL or LCO limit. These TS are in direct conflict with TS 6.8 and 6.9 which specify the action to be taken in the event a safety limit is exceed and in the event of a reportable occurrence. Please correct thefts.*

See updated SAR Chapter 14.

50. *NUREG-1537, Part 1, Chapter 14, Technical Specifications, states the applicant needs to establish TS that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff. In the 2007 SAR, Section 14, TS 3.3 and 3.4.3, Measuring Channels and Safety Channel and Control Rod Operability there is a specification that states:*

   "(2) There is a neutron-induced signal on the STARTUP CHANNEL"

   Table 1 of the same Section lists the Minimum Measuring Channel Complement. However Table 1 does not list the "STARTUP CHANNEL" as one of the required measuring channels that must be operable prior to actual reactor startup. Please correct this omission.

   See updated SAR Chapter 14.

51. *NUREG-1537, Part 1, Technical Specifications, states the applicant needs to establishes that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff. The 2007 SAR, Chapter 14, does not provide TS concerning the requirement for interlocks. As an example, there is no TS requiring an interlock to prevent reactor startup if there is not a neutron induced signal on the start up channel. Please propose TS which include specifications for all the interlocks required for operation.*

   See updated SAR Chapter 14.

52. *NUREG-1537, Part 1, Chapter 14, Technical Specifications, states the applicant needs to establish TS that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff. The 2007 SAR Chapter 14, Section 14, TS 3.8.3 it is stated that:*

   (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 micro-siemens/cm

   (3) Water level above the core shall be at least 5 meters above the top of the core

   However, there is no discussion of where the parameters in (1) and (2), above are monitored and by whom. In addition, the surveillance frequency for parameter (2) is confusing because it states that it will be measured daily and at least once every four weeks. Technical Specification Amendment #8 states that the new criteria for reactor pool water temperature is 48 °C for Parameter (1). Clarify the discrepancies identified and provide the information requested above.

   See updated SAR Chapter 14.

53. *NUREG-1537, Part 1, Chapter 14, Technical Specifications, states the applicant needs to establish TS that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff. From the Tables, it appears that the "CHANNEL TEST of Percent Power Safety Circuit SCRAM" and the "Reactor power level MEASURINGCHANNEL, CHANNEL TEST" are the same thing with different*

surveillance frequencies(refer to TS 4.2.2, and 4.3.2). Please clarify the SR including what daily means (e.g., does daily mean each day before startup.

See updated SAR Chapter 14.

54. *NUREG-1537, Part 1, Chapter 14, Technical Specifications, states the applicant needs to establish TS that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff. There are many terms in the TS that refer to "TEST","CHECK", or "CALIBRATION", that are used interchangeably (refer to TS 4.5.2; and4.3.2). The terms are defined in Chapter 14, TS 1.0. However, the terms are not always consistently applied, leading to confusion. Please clarify the usage of the terms.*

See updated SAR Chapter 14.

55. *NUREG-1537, Part 1, Chapter 14, Technical Specifications, states the applicant needs to establish TS that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff.*

The 2007 SAR, Chapter 14, TS 3.6.5, it is stated that:

"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."

Please provide the documented analysis to support the statement.

This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted by November 2010.

56. *NUREG-1537, Part 1, Chapter 14, Technical Specifications, states the applicant needs to establish TS that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff. NUREG 1537 guidance states in Section 14, 4.1 that the shutdown margin needs to be determined semiannually (every 6 months). In the 2007 SAR, the licensee has not provided actual RRR core reactivity and control rod worths. It is therefore, difficult to understand how this requirement is being met. Please provide the procedure for determining shutdown margin and an example from RRR records showing how this procedure has been implemented.*

See attached SOP 34, Control Rods. Section 34.7.1 describes how we calibrate our control rods See attachment SOP 34A from 01/14/10 for our most recent calibration. Control Rod Worths were calculated as:

Safety Rod	\$3.31
Shim Rod	\$3.27
Regulating Rod	\$1.34

57. *NUREG-1537, Part 1, Section 16.1 states the applicant should consider how a component or system was used in the past and evaluate the continued serviceability*

considering aging, wear, etc. and also to consider the suitability of items procured from other facilities. The 2007 SAR, Section 16.1, Prior Use of Reactor Components, the licensee described the depletion of the original fuel, receipt of fuel assemblies from Berkley University, and some damage to the RRR fuel inventory. Also described is the receipt of control rods from Cornell University. However, there is no discussion of the aging of components or the effect of the used components upon the ability of RRR to continue to safely operate. Furthermore, there is no discussion regarding the suitability of items supplied from other universities for use by RRR. Please provide an analysis of component aging to ensure that systems and components important to safety continue to be appropriate for use. Please provide a discussion of the safety evaluations performed on the previously utilized fuel rods and control rods, before they were placed into service at Reed.

See updated SAR Section 16.1.

58. *NUREG-1537, Part 1, Section 12.9, Quality Assurance, provides guidance on Quality Assurance for research reactors. The 2007 SAR, Section 12.9 discusses quality assurance (QA). However the discussion is incomplete in that it does not include how QA will apply to replacements, modifications and changes to systems having a safety related function. Nor does it discuss how QA will be applied to the required audit function of the Reactor Review Committee. Please address these deficiencies.*

See updated SAR Section 12.9.

Revision History

04/14/10
Added Appendix C, Control Rod Inspection Form; switched Appendix A and B.
Clarified many steps.
Created troubleshooting section by moving the section on inoperable control rods.

09/30/09
Clarified maintenance logging instructions.
Clarified who is required to be present for maintenance.
Added items to the Schedule section.

08/31/09
Revised instructions for control rod reactivity worth measurement so that the target rod positions are preset rather than calculating them each time.

Contents

Attachment to RRR RAI Response 20100730

34.1 Scope

This procedure covers the calibration and maintenance of the control rods.

34.2 Schedule

34.2.1 Each of the three control rods shall be visually inspected at least once every two years, usually during the January annual maintenance. (Tech Spec F.2)

34.2.2 Control rod worth measurement shall be done semiannually and following any inspection or handling of fuel elements or control rods (SOP 34 or SOP 35). Worth measurements may be done at other times at the discretion of the Operations Supervisor.

34.2.3 Control rod drop times shall be measured semiannually. (Tech Spec F.9.a)

34.2.4 Adjusting the rod position indication may be necessary following maintenance on the limit switches or the position indication.

34.2.5 The console buttons are cleaned as part of the Annual Checklist.

34.2.6 Auto Demand is calibrated as part of the Semiannual Checklist.

34.2.7 The Reactor Operations Committee audits this procedure every two years.

34.3 Personnel

34.3.1 During control rod operation, removal, or calibration the reactor is not shutdown therefore:

- A licensed operator must be at the console
- A second person who can summon help must be in the facility.
- An SRO must be on duty.

34.3.2 During control rod maintenance the SRO of Record must be present in the facility.

34.4 Precautions

34.4.1 The circuit boards on the control rod motors are exposed and energized. Some maintenance requires the use of metal tools, so be careful to avoid electric shock.

34.5 OPERATION

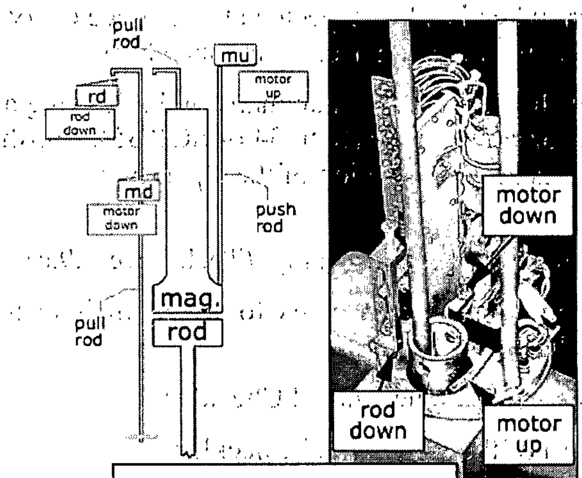
34.5.1 See SOP 1 (Reactor Operation) and SOP 20 (Startup Checklist).

OPERATION

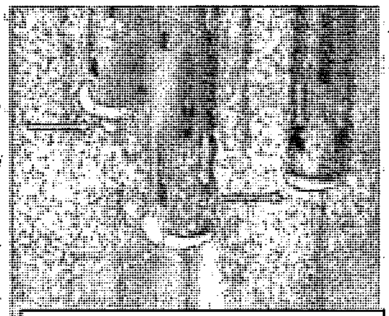
34.6 MAINTENANCE

34.6.1 Adjusting the Limit Switches

34.6.1.1 An SRO must be present for any control rod maintenance.



Limit Switch Schematic


Rings on Control Rods

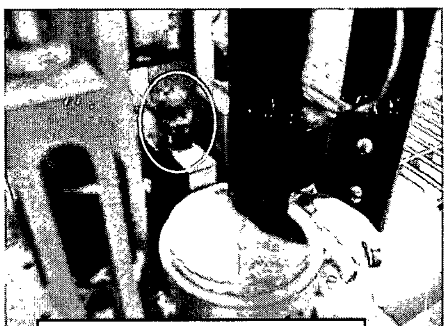
34.6.1.2 First ensure the motor moves in far enough to connect with the control rod armature; if there's a gap between the electromagnet and the control rod armature, the motor won't be able to raise the control rod.

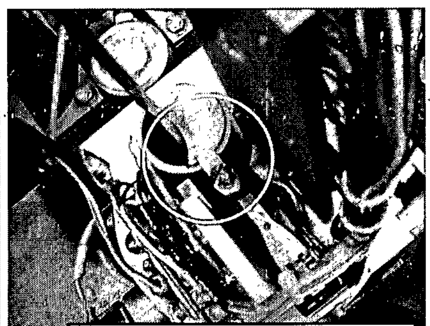
34.6.1.2.1 Manually raise the rod. Watch the actual control rod in the Reactor Bay to see if the light-colored ring under the water disappears as it moves up into the shroud. If the rod withdraws normally, go to §34.6.1.3.

34.6.1.2.2 If the rod does not withdraw, lower the motor until it stops. Turn the screw on the motor down limit switch (pictured below) counterclockwise until the motor down limit switch clears. You will hear a click.

34.6.1.2.3 Drive the motor down until it stops.

34.6.1.2.4 Repeat §34.6.1.2.1 to §34.6.1.2.3 until the rod withdraws freely.


Motor Down Limit Switch

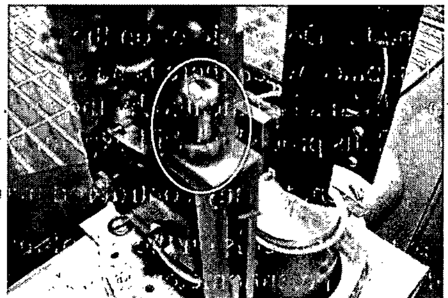

Motor Down Limit Switch

MAINTENANCE

34.6.1.3 Now ensure that the rod down limit switch works on a scram.

34.6.1.3.1 Raise and scram the rod. Ensure the motor drives in automatically after the scram.

34.6.1.3.2 If it does not drive in, use a 3/8" wrench to loosen the locking nut on the rod down limit switch. Be careful because the circuit board on the rod is exposed and is energized.



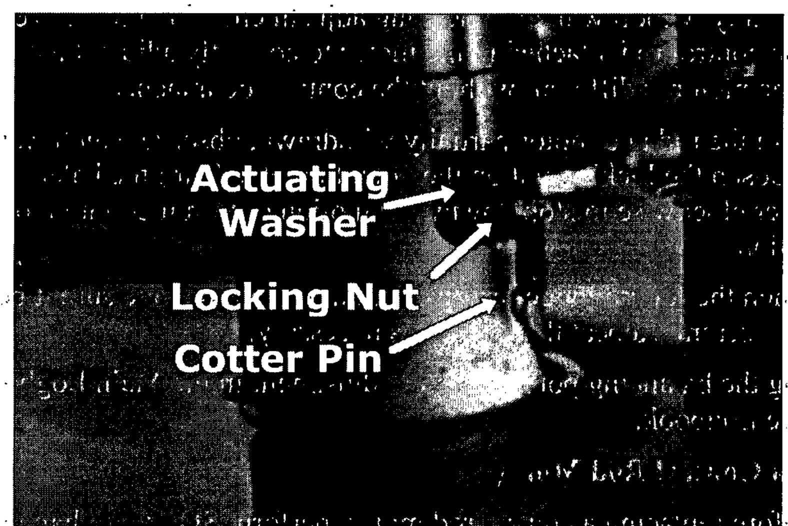
Rod Down Limit Switch

34.6.1.3.3 Turn the screw on the rod down limit switch clockwise a quarter turn.

34.6.1.3.4 Raise the rod and scram it again.

34.6.1.3.5 Repeat as necessary.

34.6.1.3.6 If you run out of room on the screw on the rod down limit switch, you may have to reach under the bridge and adjust the washers at the bottom of the pull rod. Loosen the nut and actuating washer (located a few inches above the water) on the pull rod. You will have to lean over the pull to do this. It may be necessary to tighten the actuating washer a bit in order to get at the nut. You should be able to do it all with your fingers. Adjust the height of the washers to provide more room on the rod down limit switch screw.



Pull Rod Bottom

MAINTENANCE

34.6.1.4 Now make sure the rod and motor will withdraw properly. Repeat these steps until the rod withdraws and scrams properly.

34.6.1.5 Tighten the Rod Down locking nut.

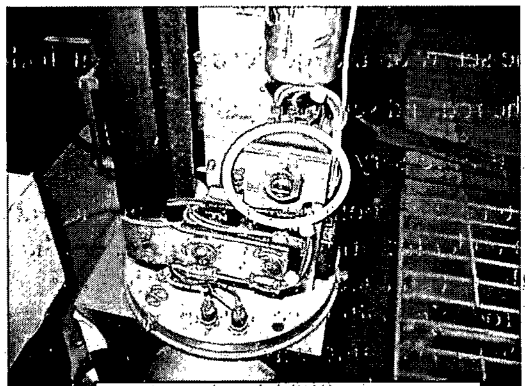
34.6.1.6 Log the limit switch adjustment in the Main Logbook and the Maintenance Logbook.

34.6.2 Adjusting the Balancing Potentiometer

34.6.2.1 The Up Motor and the Down Motor on the Safety and Shim Rods are balanced to hold the rod in place when there is no signal to move it. Sometimes the motor becomes unbalanced. This is indicated by the motor creeping up or down when there is no demand signal. This procedure will explain how to rebalance the motors.

34.6.2.2 An SRO must be present for any control rod maintenance.

34.6.2.3 The balancing potentiometer is on the top of the control rod motor. The Reg Rod does not have a balancing potentiometer since it has been replaced with a stepping motor.



Rod Balancing Potentiometer

34.6.2.4 Partially withdraw the rod needing adjustment. You must have the scrams reset and the control rod attached to the motor to correctly adjust the rod since the weight on the motor is different without the control rod attached.

34.6.2.5 With the rod and motor partially withdrawn, observe which way the motor is creeping. Loosen the locking nut on the potentiometer. Turn the balancing potentiometer clockwise to stop downward motion or turn it counterclockwise to stop upward motion.

34.6.2.6 When the rod no longer creeps, scram the rod to make sure it behaves properly on a scram. Repeat these steps as necessary.

34.6.2.7 Log the balancing potentiometer adjustment in the Main Logbook and the Maintenance Logbook.

34.6.3 Replacing a Control Rod Motor

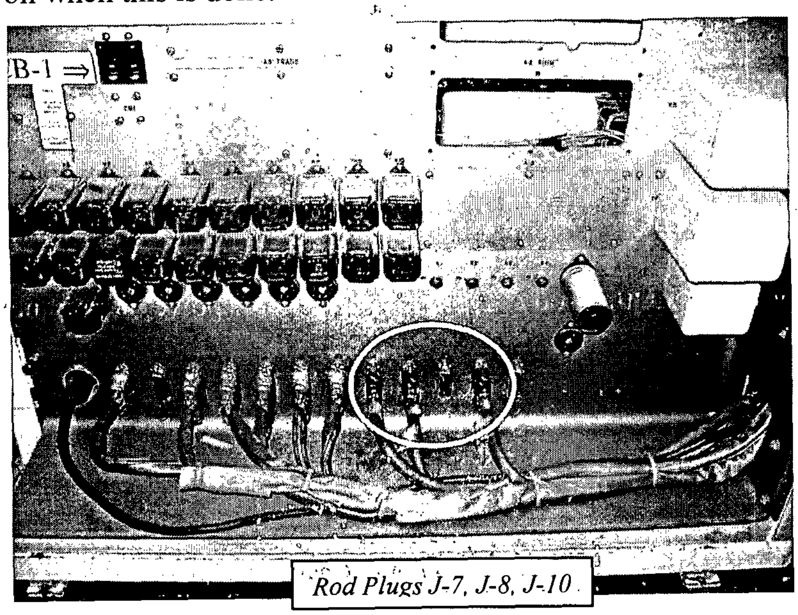
34.6.3.1 Before replacing a control rod motor, perform SOP 62 (Changes, Tests, and Experiments) to see if a safety evaluation is necessary.

MAINTENANCE

34.6.3.2 While removing the control rod motor, the reactor is not shutdown so a licensed operator must be at the console and a second person in the facility.

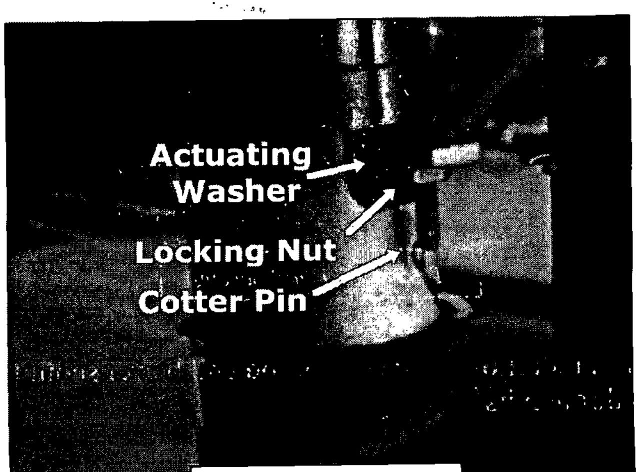
34.6.3.3 An SRO must be present for any control rod maintenance.

34.6.3.4 Unplug all three console-to-rod-drive motor plugs inside the console: J-7 (Reg Rod), J-8 (Shim Rod), and J-10 (Safety Rod). The rod UP and DOWN lights will come on when this is done.



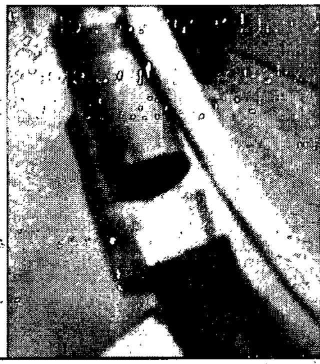
Rod Plugs J-7, J-8, J-10

34.6.3.5 Remove the rod-down actuator cotter pin, nut, and actuating washer (located a few inches above the water) from the pull rod. You will have to lean over the pull to do this. Be careful not to drop them. The cotter pin is really a paperclip that has to be unbent before removing it from the small hole at the bottom of the pull rod. The nut and actuating washer unscrew as normal. It may be necessary to tighten the actuating washer a bit in order to loosen and get at the nut. You should be able to do it all with your fingers.



Pull Rod Bottom

34.6.3.6 Unplug the position indication for the rod being removed. It is at the potentiometer at the side of the motor.

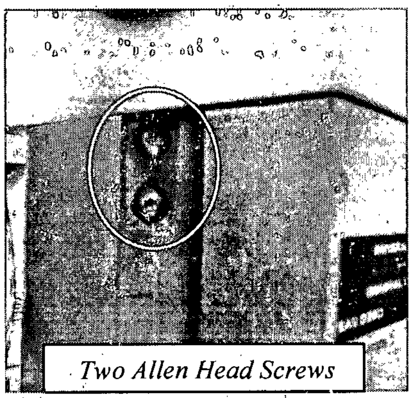


Rod Position Connection

34.6.3.7 For the regulating rod only, unplug the motor power supply next to the motor.

Reg Rod Power Connection

34.6.3.8 Remove the small Allen head 3/32-inch screws from the upper edge of each base mount. There are screws on four locations around the rod motor, approximately 90° apart. Two locations have two screws; two have only one screw. Note that you only have to remove the top screw of the pair. The bottom one holds the control rod housing (which is not being removed).



Two Allen Head Screws

34.6.3.9 Lift the rod motor off. The control rod and barrel should not move. The pull rod will come up through its opening.

34.6.3.10 Install the new rod motor on the rod motor mount. You'll need to guide the pull rod into its opening. (You may also have to transfer the pull rod from the old motor to the new motor if the new one doesn't have a pull rod.)

34.6.3.11 Install the small Allen head 3/32-inch screws in the upper edge of each base mount.

34.6.3.12 Reinstall the rod-down actuator cotter pin, nut, and actuating washer (located a few inches above the water) from the pull rod. You will have to lean over the pool to do this; be careful not to drop them.

34.6.3.13 Plug in the position indication for the rod.

34.6.3.14 For the regulating rod only, plug in the motor power supply.

34.6.3.15 Reconnect all three console to rod drive motor plugs inside the console: J-7, J-8, and J-10.

34.6.3.16 Adjust the screws on the limit switches until the micro-switches work properly when scramming the rod per §34.6.1.

34.6.3.17 Check the rod by raising it a short distance and then releasing it by a scram.

34.6.3.18 Check the top and bottom position indications per §34.7.2.

34.6.3.19 Check the rod travel time by raising and lowering the control rod full travel.

34.6.3.20 Log the control rod motor replacement in the Main Logbook and the Maintenance Logbook. Include the model and serial number of both the old motor and the new motor in the Maintenance Logbook entry.

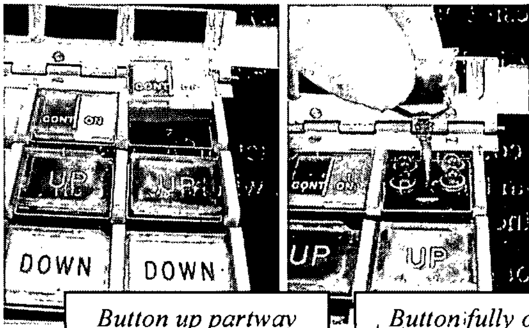
34.6.4 Replacing a Console Light Bulb

34.6.4.1 Anyone designated by the Operations Supervisor may perform this procedure.

- This does not count as maintenance of a reactor control system, so the presence of an SRO is not required.

34.6.4.2 Note that the magnet power supply is in series with the ON light for that rod. If you remove the CONT/ON button for a control rod, it will lose magnet power and drop into the core. Hence, do not do this with the rod withdrawn.

34.6.4.3 To replace a light bulb, start by lifting straight up on the console button. It come out partway and then stop.



or! Obj ated from editing field codes.

34.6.4.4 Turn the top of the button 90° clockwise to release it, and then pull it the rest of the way out.

34.6.4.5 Remove the burnt out bulb and replace it with a 328 bulb, normally kept in the "Frequent Light Bulb" container in storage cabinet F.

34.6.4.6 With the button top still at 90° to the base, reinsert it in the opening. Note that the small notch on the base must be on the left side to fit in.

34.6.4.7 Push down until the button stops, then turn it 90° counterclockwise and push it the rest of the way in.

34.6.4.8 Log the light bulb replacement in the Main Logbook and the Maintenance Logbook.

34.6.5 Cleaning Console Buttons

34.6.5.1 Anyone designated by the Operations Supervisor may perform this procedure.

- This does not count as maintenance of a reactor control system, so the presence of an SRO is not required.
- A licensed operator, SRO of record, and second person in the facility will be needed to insert the console key.

34.6.5.2 Open CB-1 inside the back of the console and log it in the Main Logbook.

34.6.5.3 Gently pry up the console POWER button until it comes out, and then rotate it by 90° counterclockwise so that the linkage to which it is attached, can also be withdrawn.

34.6.5.4 Use cotton swab to clean the button housing of all oil and debris. The cotton swab may be dipped in alcohol, but the housing must be left dry.

34.6.5.5 Use a dry lubricant on the linkage shaft where it passes through the bushing.

34.6.5.6 Inspect the spring for integrity.

34.6.5.7 Lock the shaft into the light-socket assembly, and re-install the entire mechanical linkage into its housing. Note that it will insert in only one orientation.

34.6.5.8 Test to see that the button depresses and returns easily.

34.6.5.9 Repeat §34.6.5.3 through §34.6.5.8 for the following buttons:

- Safety CONT/ON
- Safety UP
- Safety DOWN
- Shim CONT/ON
- Shim UP, Shim DOWN
- Reg CONT/ON
- Reg UP
- Reg DOWN.

34.6.5.10 Add a small amount of dry lubricant to the console key switch and rotate the key through all its positions a few times to ensure easy operation. Log this in the Main Logbook.

34.6.5.11 If the key does not rotate easily, remove the two screws that hold the control console in place and lubricate the internal workings of the key switch. Rotate the key through all its positions a few times to ensure easy operation. Screw the control console back into place.

34.6.5.12 Close CB-1 inside the back of the console and log it in the Main Logbook.

34.6.5.13 If this procedure was done for the Annual Checklist, record the date on the Checklist form. Otherwise, log this maintenance in the Main Logbook and the Maintenance Logbook.

34.7 CALIBRATION AND INSPECTION

34.7.1 Control Rod Calibration

34.7.1.1 This procedure describes how to perform the semiannual control rod calibration. The calibration includes:

- Adjusting the control rod position indications on the console and Multitrend.
- Measuring control rod drop times.
- Measuring control rod reactivity worth.

34.7.1.2 Use the date stamp (or print very neatly) to enter the date on SOP 34A, Control Rod Calibration Form.

34.7.1.3 It will take several hours to complete the calibration. Each person who acts as Operator of Record for part of the calibration should print his or her name at the top of SOP 34A.

34.7.1.4 Adjust the rod position indications per §34.7.2. This must be done before the reactivity worth measurement. Record the as found and as left data on SOP 34A.

34.7.1.5 Measure the control rod drop times per §34.7.3 and record the values on SOP 34A. This may be done before of after the worth measurement.

34.7.1.6 Measure the control rod reactivity worth per §34.7.4.

34.7.1.7 The operator who completes the calibration shall sign the form and leave it on the hanging Weekly and Other Clipboard for the Operations Supervisor to review.

34.7.1.8 If the calibration is being performed for the Semiannual Checklist, initial and date the appropriate line of the checklist.

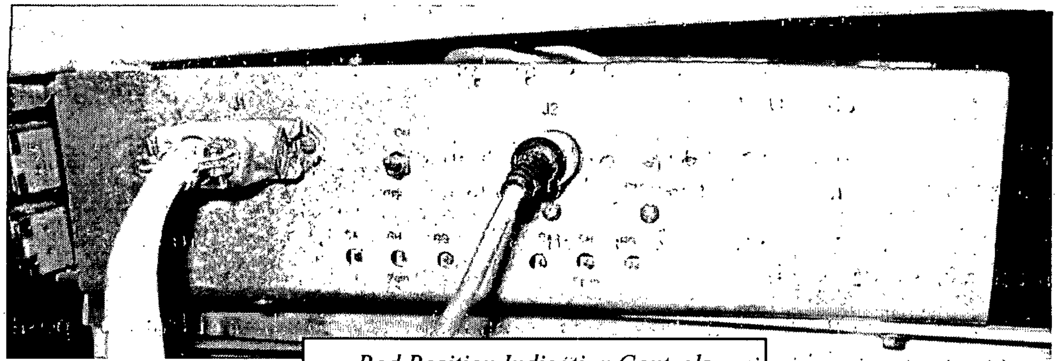
34.7.1.9 After review, the Operations Supervisor will store the Control Rod Calibration Form in the Other Checklists binder in the Control Room.

34.7.2 Adjusting the Rod Position Indication

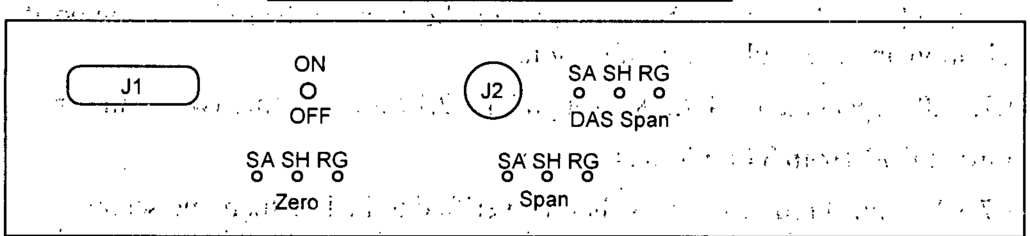
34.7.2.1 This procedure describes how to calibrate the control rod position indications on the console and Multitrend. This must be done before performing a control rod reactivity worth measurement per §34.7.4, and may be done at other times at the discretion of the Operations Supervisor.

34.7.2.2 The limit switches on the bridge control the rod motion. They must be correct before adjusting the position indication. If the limit switches need adjustment, see §34.6.1.

34.7.2.3 The position indications are adjusted with a small slot screw driver on the potentiometers in back of the rod position indication inside the console. Clockwise increases the value, counterclockwise decreases it. There are three adjustments for each rod: zero, span, and DAS Span (Multitrend). The rods are abbreviated SA (safety), SH (shim), and RG (reg).



Rod Position Indication Controls.



34.7.2.4 If this is part of a control rod calibration, record the information on SOP 34A. Otherwise record it in the Main Logbook and Maintenance Logbook.

34.7.2.5 Leave two control rods on the bottom at all times during this adjustment.

34.7.2.6 Drive the control rods all way down (the DOWN light on).

34.7.2.7 Record the "as found" bottom position indication for each rod from the position indication and the Multitrend indication.

34.7.2.8 Drive ONE control rod all way up (the UP light on).

34.7.2.9 Record the "as found" top position indication for the rod from the rod position indication and the Multitrend indication.

34.7.2.10 Adjust the span position indication for the rod to indicate between 99.5% and 100.5% (exactly 100.0% is best).

34.7.2.11 Adjust the DAS Span (Multitrend position indication) for the rod to indicate between 99.5% and 100.5% (exactly 100.0% is best). Note that it may take several minutes for the Multitrend display to stabilize.

34.7.2.12 Drive the rod to the bottom.

34.7.2.13 Adjust the bottom position indication for the rod to indicate between 0.0% and positive 0.5% (exactly 0.0% is best).

- Note that there is no negative sign, so you need to turn the screw back and forth to determine if you are looking at positive or negative numbers.
- Do not leave the value below 0.0% since the indication is not linear below 0.0%.
- There is no zero adjustment for the Multitrend.

CALIBRATION

34.7.2.14 Drive the control rod all way up (the UP light on).

34.7.2.15 Adjust the span position indication for the rod to indicate between 99.5% and 100.5% (exactly 100.0% is best).

34.7.2.16 Adjust the DAS Span (Multitrend position indication) for the rod to indicate between 99.5% and 100.5% (exactly 100.0% is best).

34.7.2.17 Repeat steps 34.7.2.12 through 34.7.2.16 until the top, bottom, and Multitrend are acceptable.

34.7.2.18 Record the "as left" bottom position indication for the rod from the position indication and the Multitrend indication.

34.7.2.19 Record the "as left" top position indication for the rod from the position indication and the Multitrend indication.

34.7.2.20 Repeat steps 34.7.2.8 through 34.7.2.19 for the other two control rods.

34.7.3 Control Rod Drop Time Measurement

34.7.3.1 Control rod drop times shall be verified to be less than one second semiannually and following removal of any control rod for maintenance or inspection (Tech Spec F.8 and F.9). Drop times may be measured at any time at the discretion of the Director.

34.7.3.2 Manual Control Rod Drop Times

34.7.3.2.1 This section shall be used if the Rod Drop Timer is considered non-operational or it is decided to manually measure the control rod drop times. If the electronic timer is to be used, go to §34.7.3.3.

34.7.3.2.2 The operator remains at the console as for normal operations with the two timers positioned in a way that they can easily view the control rod indicator lights. The operator will raise a single rod to its maximum height. The other two rods shall remain fully inserted.

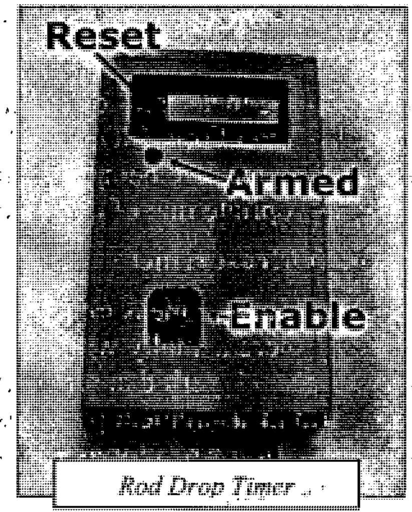
34.7.3.2.3 The operator manually scrams the rod. The timers start their stopwatches when the yellow "on" light (Magnet Power) is extinguished and stop them when the blue "cont" light goes off. The yellow light indicates that the rod has been released and the blue light indicates when the rod-down switch has been closed (rod fully down in core).

34.7.3.2.4 Repeat for each control rod. Average the two times for each rod to get the final value for the drop time.

34.7.3.2.5 Go to §34.7.3.13.

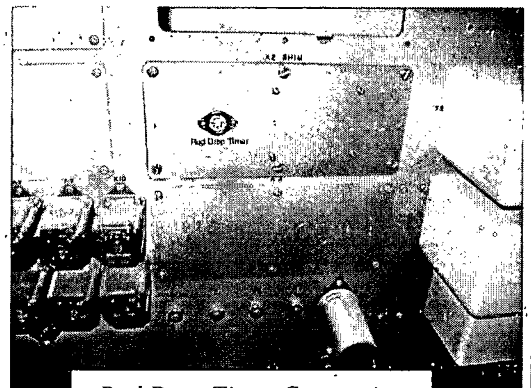
CALIBRATION

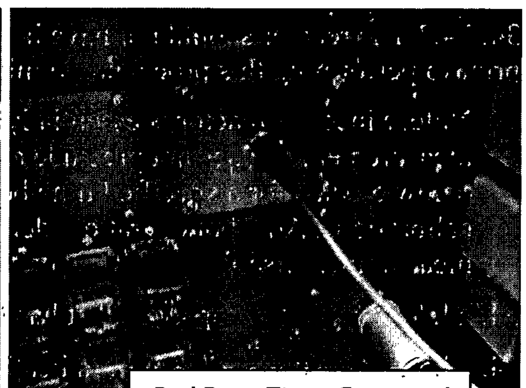
34.7.3.3 Get the Rod Drop Timer (pictured below) from its drawer in the Control Room under the AMS-4s. The Rod Drop Timer power supply is a 6V 200 mA AC adapter; it should be with the timer. Plug in the power supply transformer to the wall outlet and to the small plug at the base of the timer.



Rod Drop Timer

34.7.3.4 Connect the Rod Drop Timer to the connection inside the console near CB-1.


Rod Drop Timer Connection


Rod Drop Timer Connected

34.7.3.5 You must press the black reset button on the rod drop timer before the next measurement to clear the display

34.7.3.6 Raise the rod to be tested until the red UP light on the rod is lit.

34.7.3.7 Press the red enable button on the Rod Drop Timer. Actually, it can be pressed as soon as the rod is off the bottom. The green armed light on the rod drop timer will turn on.

34.7.3.8 Manually scram the reactor. This will not work by pressing the individual rod drop button. The SCRAM light must come on.

34.7.3.9. The timer will show, in milliseconds, how long between the scram signal and a rod down limit switch being actuated.

CALIBRATION

CALIBRATION

34.7.3.10 Reset the scram.

34.7.3.11 Repeat §34.7.3.5 to §34.7.3.10 for each additional rod to be tested.

34.7.3.12 Disconnect the rod drop timer when done.

34.7.3.13. If the drop time for any rod is greater than 1 second, notify the Operations Supervisor and Director.

34.7.3.14 If the drop times are being measured for a control rod calibration, record the values on the Control Rod Calibration Form (SOP 34A); otherwise, record the drop times in the Main Logbook. If the times are being measured for the semiannual checklist, initial and date the appropriate line of the checklist.

34.7.4 Control Rod Reactivity Worth Measurement

34.7.4.1 This procedure describes how to measure the differential reactivity worth of the control rods. This is done by incrementally withdrawing a control rod and measuring the resulting reactor period; period and reactivity are related by the In-Hour equation. Control rod worth measurement provides the means to determine the core-excess and shutdown margin of the reactor as well as the reactivity change induced by changes in control rod heights.

34.7.4.2 Record all information on SOP 34A. Preliminary steps for filling out the form are described in §34.7.1.

34.7.4.3 The reactor should not have been run at powers above 5 watts for 48 hours prior to performing this procedure to minimize the xenon effects.

Note: Since the reactor is essentially at 5 watts during this control rod reactivity measurement, it does not preclude activities that require the reactor to have been below 5 watts for a specified time before the activity, e.g., fuel movement, power calibrations, Aquabot operation, demineralizer changes, and control rod reactivity measurement itself.

34.7.4.4 Ensure a Startup Checklist has been completed.

34.7.4.5 Ensure the rod position indication is correct per §34.7.2.

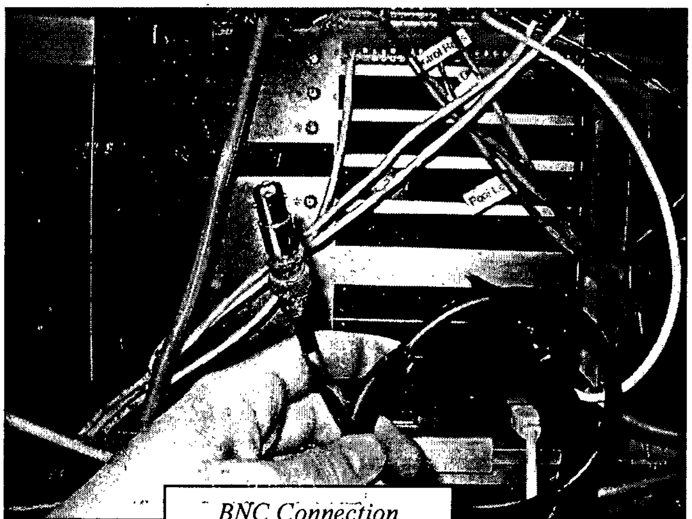
34.7.4.6 To avoid moving the wrong control rod at the wrong time, it is useful to cover the rod buttons that should not be used with a plastic ruler or other item.

CALIBRATION

CALIBRATION

34.7.4.7 Period Timer Calibration

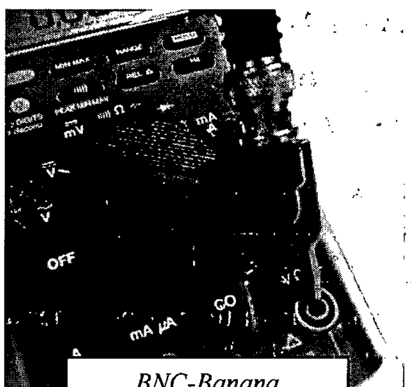
34.7.4.7.1 At the back of the Multitrend, there is a BNC connection coming off of the cable that carries information from the Linear Channel to the Multitrend. Attach a BNC cable to this connection (long enough to reach the front of the console where you will be working).



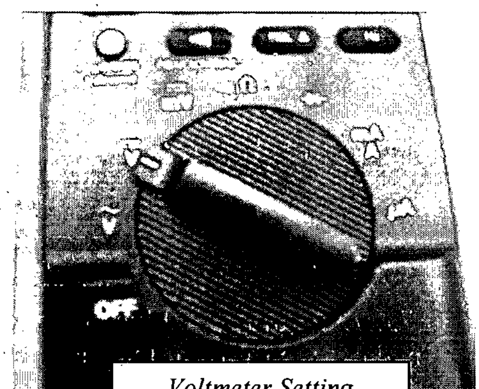
BNC Connection

34.7.4.7.2 Attach the other end of the cable to a digital voltmeter using a BNC-banana plug adaptor (normally in a small drawer on the bottom shelf of Classroom Cabinet E).

34.7.4.7.3 Set the voltmeter for direct current volts ($\overline{\mathbf{V}}$). The voltage is less than 10 VDC.



BNC-Banana



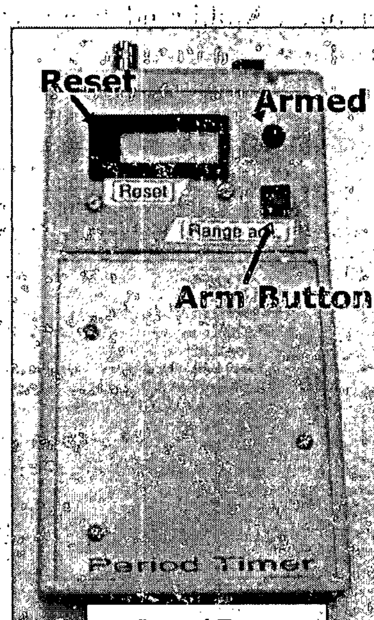
Voltmeter Setting

34.7.4.7.4 Place the linear channel in "Current" mode and adjust it to exactly 30% of scale on the linear channel module. Record the voltage on SOP 34A.

34.7.4.7.5 Adjust the linear channel in the "Current" mode to exactly 81.5% of scale on the linear channel module. Record the voltage on SOP 34A.

34.7.4.7.6 Remove the banana plug-BNC adaptor from the digital voltmeter and hook up the normal meter probes.

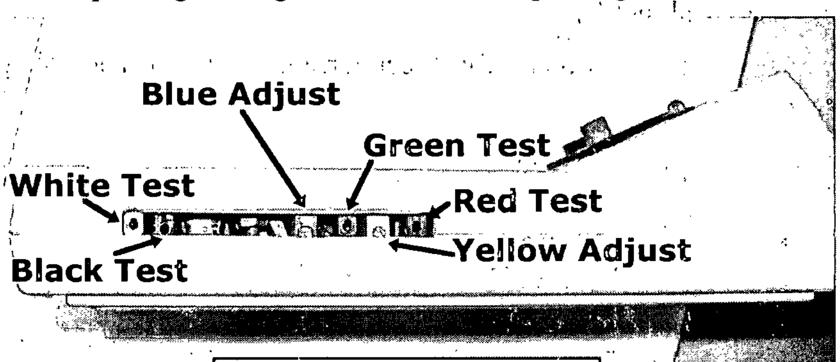
34.7.4.7.7 Get the period timer from its drawer in the Control Room under the AMS-4s.



Period Timer

34.7.4.7.8 Use the digital voltmeter probes to measure the voltage between the green test point and the black test point on side of the period timer.

- The red probe goes to green and the black probe goes to black.



Period Timer Adjustments

34.7.4.7.9. Adjust the blue potentiometer on side of the period timer as necessary to make it the same voltage as measured above at 30%.

34.7.4.7.10 Use the digital voltmeter probes to measure the voltage between the red test point and the black test point on side of the period timer.

- The red probe goes to red and the black probe goes to black.

34.7.4.7.11 Adjust the yellow potentiometer on side of the period timer as necessary to make it the same voltage as measured above at 81.5%.

34.7.4.7.12 Connect the BNC cable to the top of the period timer, so that the period timer receives information from the Linear Channel.

34.7.4.8 Reg Rod Worth Measurement

34.7.4.8.1 Per SOP 1, establish a critical condition at 5 watts with the Reg Rod on the bottom.

- The Safety and Shim Rods should be withdrawn to approximately the same height.

34.7.4.8.2 Allow power level to stabilize at approximately 5 watts (18% to 22% on the 25 watt scale) for at least two minutes with minimal rod motion.

- It is much more important that the power be stable than that it be at exactly 5 watts. This allows delayed neutrons to reach equilibrium.

34.7.4.8.3 With the reactor stable for at least two minutes with minimal rod motion, withdraw the Reg Rod to approximately the new target height.

- The reactivity change should take place in one continuous pull. When you release the UP button do not adjust the height. Leave it wherever it was when you stopped.

34.7.4.8.4 Allow the power level to increase one full range on the linear channel so that the "prompt jump" is gone.

34.7.4.8.5 When the linear channel auto-ranges to the 250-watt scale, arm the period timer with its red button; the green armed LED will come on.

34.7.4.8.6 When the linear channel crosses 30%, the timer will start.

34.7.4.8.7 When linear channel crosses 81.5%, the timer will stop and indicate the period in milliseconds.

34.7.4.8.8 After power exceeds 82% of 250-watts and the timer stops, return the power level to 5 watts by lowering the Safety and Shim Rods.

- Try to keep the Safety and Shim rod banked at about the same height.
- Do not reposition the Reg Rod during this step!

34.7.4.8.9 Record the period on SOP 34A.

34.7.4.8.10 Record the rod positions after stabilizing at 5 watts on SOP 34A.

34.7.4.8.11 Reset the timer with the black button on the left of the display.

34.7.4.8.12 Remember to complete a Status Stamp every hour.

34.7.4.8.13 Repeat §34.7.4.8.3 to §34.7.4.8.12 until the Reg Rod UP light is lit.

34.7.4.9 Safety and Shim Rod Worth Measurement

34.7.4.9.1 If the data are taken carefully and processed correctly, it is possible to measure the Safety and Shim rods simultaneously since they have similar worths. This is done by starting with the Shim fully out, the Reg fully out, and the Safety rod as far in as possible while critical. Data for the Safety rod is taken as normal. Data for the Shim rod is taken as it is inserted to restore critical conditions at 5 watts after each pull. The Reg rod is not moved at all during the process.

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34.7.4.9.2 If it is desired to measure the worth of the Safety and Shim rods individually, follow this procedure for the Safety Rod and then repeat it with the Shim and Safety rods exchanged; you will need an additional copy of the relevant page of SOP 34A to record the extra data.

34.7.4.9.3 Per SOP 1, establish a critical condition at 5 watts with the Shim Reg and Reg Rod all the way out and the Safety Rod as far in as possible while remaining critical.

- It is not possible to be critical with the Safety or Shim on the bottom.

34.7.4.9.4 Allow power level to stabilize at approximately 5 watts (18% to 22% on the 25 watt scale) for at least 2 minutes with minimal rod motion.

- It is much more important that the power be stable than that it be at exactly 5 watts. This allows delayed neutrons to reach equilibrium.

34.7.4.9.5 With the reactor stable for at least two minutes with minimal rod motion, withdraw the Safety Rod to approximately the new target height.

- The reactivity change should take place in one continuous pull. When you release the UP button do not adjust the height. Leave it wherever it was when you stopped.

34.7.4.9.6 Allow the power level to increase one full range on the linear channel so that the "prompt jump" is gone.

34.7.4.9.7 When the linear channel auto-ranges to the 250-watt scale, arm the period timer with its red button; the green armed LED will come on.

34.7.4.9.8 When the linear channel crosses 30%, the timer will start.

34.7.4.9.9 When linear channel crosses 81.5%, the timer will stop and indicate the period in milliseconds.

34.7.4.9.10 After power exceeds 82% of 250-watts and the timer stops, return the power level to 5 watts by lowering Shim Rod.

- Do not reposition the Safety Rod or Reg Rod during this step!

34.7.4.9.11 Record the period on SOP 34A.

34.7.4.9.12 Record the rod positions after stabilizing at 5 watts on SOP 34A.

34.7.4.9.13 Reset the timer with the black button on the left of the display.

34.7.4.9.14 Remember to complete a Status Stamp every hour.

34.7.4.9.15 Repeat §34.7.4.9.5 to §34.7.4.9.14 until the Safety Rod UP light is lit.

34.7.4.9.16 Return the reactor to 5 watts to record the critical rod height on the Shim Rod. This is essential for the Shim Rod worth measurement

34.7.4.10 Use a stopwatch to measure the time it takes each rod to travel from bottom to top, and from top to bottom and record the times on SOP 34A.

- This is measured using the motor UP and motor DOWN lights.

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34.7.4.11 Record the most recent xenon-free 5 W and 230 kW banked rod heights. You may use a previous operation from the Main Logbook or you may operate the reactor at 5 W and 230 kW to get *very* recent values. Record the date and time of each operation.

34.7.4.12 There is an Excel spreadsheet on the Control Room computer and on the server called "Rod Cal." Enter the data on the tab labeled "Enter Data Here" and follow the instructions there. The new rod worth measurements are individual worksheets, and the "Print This" worksheet will calculate maximum reactivity addition rate, core excess, shutdown margin, and minimum shutdown margin.

34.7.4.13 Print the "Print This" tabs of the worksheet.

34.7.4.14 Record the core excess, shutdown margin, shutdown margin with the most reactive rod all the way out, and the maximum reactivity insertion rate on SOP 34A.

34.7.4.15 Tape the new rod worth charts in the back of the Main Logbook. Old rod worth charts removed from the Main Logbook should be filed in the Counting Room.

34.7.4.16 Finish the Control Rod Calibration Form (SOP 34A) per §34.7.1.

34.7.5 Control Rod Inspection

34.7.5.1 Each of the three control rods shall be visually inspected at least once every two years, usually during the January annual maintenance. (Tech Spec F.2)

34.7.5.2 Staffing is required per SOP 35 for fuel inspection since it is necessary to move fuel first.

34.7.5.3 The reactor is not shutdown while control rods are being removed or inserted into the reactor, so a licensed operator must be at the console and a second person must be in the facility. Since this is maintenance on a control mechanism, the SRO must be present in the facility.

34.7.5.4 Inspect one control rod at a time and replace it into the core before going on to the next one.

34.7.5.5 Wear gloves and a lab coat when handling the control rods. Monitor anything that comes out of the pool for radioactivity.

34.7.5.6 Only aluminum clad control rods are in the core, but if stainless-steel clad control rods are being inspected, there may be a significant radiation field present. In that case, the reactor shall be shutdown 48 hours before control rod removal and a Radiation Work Permit (SOP 53) shall be used.

34.7.5.7 Complete the Control Rod Inspection Checklist (Appendix B) while doing this procedure.

34.7.5.8 Perform a Startup Checklist per SOP 20. Do not perform a core excess.

34.7.5.9 Using SOP 35, move any two elements from the B-ring **or** three elements of the C-ring to the fuel storage racks.

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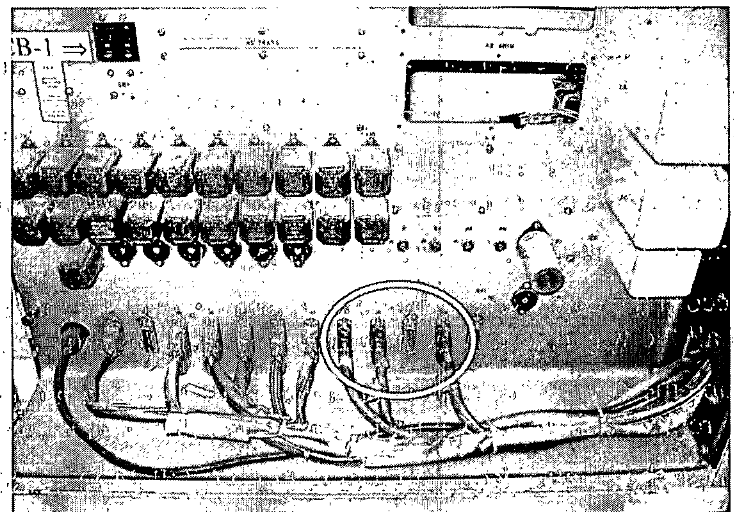
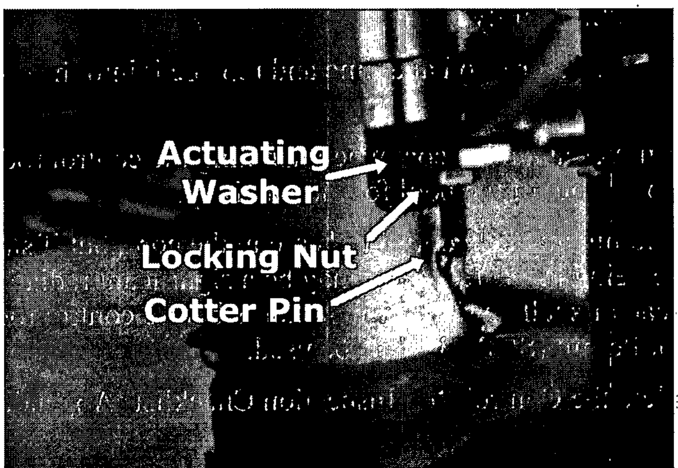
34.7.5.10 Unplug all three console-to-rod-drive motor plugs inside the console: J-7 (Reg Rod), J-8 (Shim Rod), and J-10 (Safety Rod). The rod UP and DOWN lights will come on when this is done.



Rod Plugs J-7, J-8, J-10

34.7.5.11 Remove the rod-down actuator cotter pin, nut, and actuating washer (located a few inches above the water) from the pull rod. You will have to lean over the pool to do this; be careful not to drop them. The cotter pin is really a paperclip that has to be unbent before removing it from the small hole at the bottom of the pull rod. The nut and actuating washer unscrew as normal. It may be necessary to tighten the actuating washer (the top one) a little bit in order to get at the nut. You should be able to do it all with your fingers.



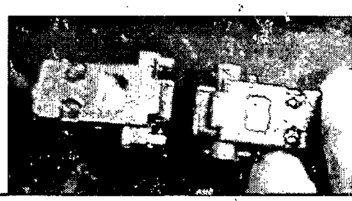
Pull Rod Bottom

CALIBRATION

34.7.5.12 Unplug the position indication for the rod being removed. It is at the potentiometer at the side of the motor.

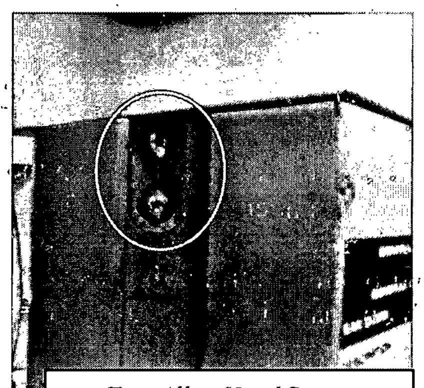
Rod Position Connection

34.7.5.13 For the regulating rod only, unplug the motor power supply next to the motor. The pool level alarm may actuate when you do this.



Reg Rod Power Connection

34.7.5.14 Remove the small Allen head 3/32-inch screws from the upper edge of each base mount. The are locations for six screws, but normally only five are used. Do not removed the large bolts that hold the motor housing to the bridge.



Two Allen Head Screws

34.7.5.15 Lift the rod motor off and place it on the floor.

34.7.5.16 An operator must be at the console for the next steps.

34.7.5.17 Lift the guide barrel up through the base mount, thereby lifting the connecting rod. This is done by reaching inside the barrel from the top and lifting up. As the rod is raised, be sure that someone watches the UPPER part of the rod as it

CALIBRATION

approaches the region of the ceiling lights. Have someone watch the LOWER part of the rod to ensure it doesn't hit other control rods or the central thimble.

34.7.5.18 The operator shall log the control rod removed from the core in black and underline in red ink in the Main Logbook.

34.7.5.19 Lift the rod high enough to permit removal of the first set of three 7/16-inch bolts and nuts; this will allow the top of the rod extension to be removed. Ensure someone is holding the lower part of the control rod when the three sets of bolts and nuts are removed, otherwise the control rod will fall onto the core.

Unbolting Rod Extension

Unbolting Rod Extension

34.7.5.20 While holding onto the control rod, pass it down through the control rod opening on the bridge and hand it out under the bridge. This will allow removal of the control rod from the pool (the ceiling is too low to pull it straight out the opening in the bridge). Be very careful not to drop the control rod. This is probably best done by one person lying on the bridge with one hand in the control rod opening and one hand around the side of the bridge.

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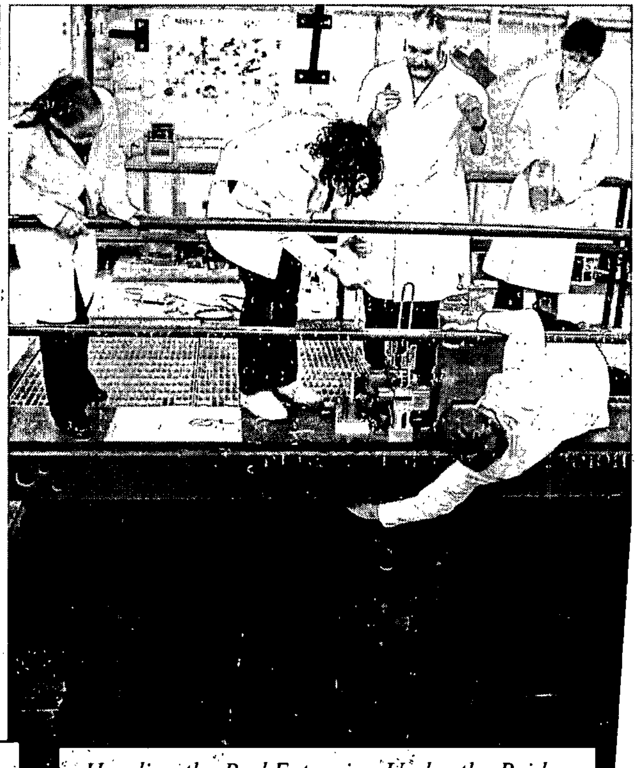
Rod Free From Core



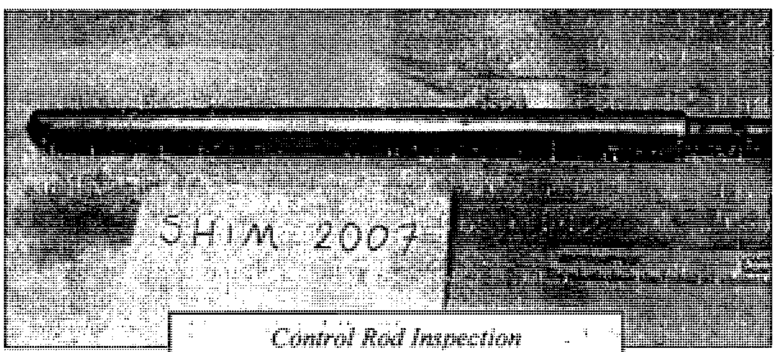
Handing the Rod Extension Under the Bridge

34.7.5.21 With an ion chamber monitoring the immediate area, remove the control rod and lay it on a piece of absorbent paper.

34.7.5.22 Inspect the control rod for scratches, wear, etc. Use a caliper to measure the rod diameter at the top, center, and bottom. Carefully measure the extent of any wear and compare it to previous measurements. Record observations on the Control Rod Inspection Form (Appendix C).

- A standard control rod is 20 inches long and 1.25 inches in diameter.

34.7.5.23 Take photographs of the rod with a ruler and a sign in the picture saying what rod it is and the year. Store the photographs with the Control Rod Inspection Form.



Control Rod Inspection

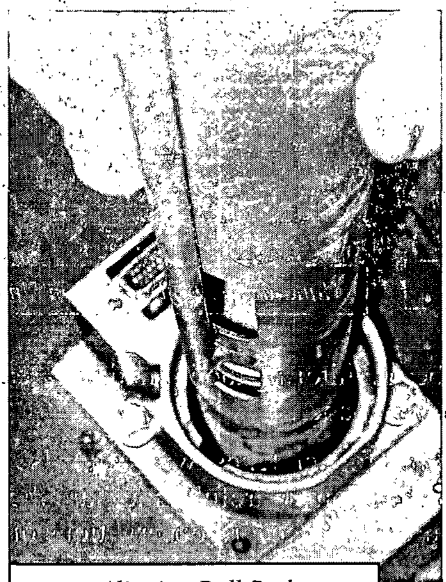
CALIBRATION

34.7.5.24 If indication of significant distortion or deterioration is found, the element shall be replaced (Tech Spec F.2).

34.7.5.25. Reinstall the rod by reversing the order of removal. First put the control rod back in the pool. While holding onto the control rod, hand it down under the bridge and pass it up through the control rod opening on the bridge. Be very careful not to drop the control rod or to hit anything with it.

34.7.5.26 Lift the rod through the opening high enough to permit reinstallation of the first set of three 7/16-inch bolts and nuts. Ensure someone is holding the lower part of the control rod and the upper part. If the bolts and nuts on the control rod drive shaft don't seem to fit, try rotating one of the shafts by 180°.

34.7.5.27 Lower the rod and barrel down into the rod housing. Align the pull rod with the hole in the motor housing when returning it.



Aligning Pull Rod

34.7.5.28 The operator shall log the control rod installed in the core in black and underline in red ink in the Main Logbook.

34.7.5.29 Reinstall the rod motor on the rod motor mount. Make sure the spring on the rod down limit switch is still there.

Install the small Allen head 3/32-inch screws in the upper edge of each base mount.

34.7.5.30 Reinstall the rod-down actuator cotter pin, nut, and actuating washer (located a few inches above the water) from the pull rod. You will have to lean over the pool to do this; be careful not to drop them.

34.7.5.31 Plug in the position indication for the rod.

34.7.5.32 For the regulating rod only, plug in the motor power supply.

34.7.5.33 Repeat §34.7.5.11 through §34.7.5.32 for the other two control rods.

34.7.5.34 Reconnect all three console to rod drive motor plugs inside the console: J-7, J-8, and J-10.

34.7.5.35 Adjust the micro-switches per §34.6.1.

34.7.5.36 Check each rod by raising it a short distance and then releasing it by a scram.

34.7.5.37 Check the work area for contamination.

34.7.5.38 Perform §34.7.3 to verify that the rod drop time of each control rod is acceptable.

34.7.5.39 Using SOP 35, Fuel Inspection, return the fuel elements to the core that were removed in §34.7.5.9.

34.7.5.40 Perform a control rod calibration per §34.7.1.

34.7.5.41 Complete the second part of the Control Rod Inspection Checklist (SOP 34B).

34.7.6 Calibrating Auto Demand

34.7.6.1 This procedure describes how to calibrate Auto Demand so that the setting on the demand potentiometer matches the corresponding reading on the Linear Channel. A startup checklist is not required for this procedure.

34.7.6.2 Turn on console power and note this action in the Main Logbook.

34.7.6.3 Connect a voltmeter to the test positions on the auto rod control circuit board. The negative probe goes to TP 10 (which is black); the positive probe goes to TP 2 (which is red).

34.7.6.4 Dial in 20% on the demand potentiometer.

34.7.6.5. Place the Linear Channel in Current mode and adjust it to 20% of any range.

34.7.6.6 Adjust R11 on the auto rod control circuit board to read 0 VDC.

34.7.6.7 Dial in 96% on the demand potentiometer.

34.7.6.8 Place the Linear Channel in Current mode and adjust it to 96% of any range.

34.7.6.9 Adjust R8 on the auto rod control circuit board to read 0 VDC.

34.7.6.10 Repeat steps 34.7.6.3 to 34.7.6.8 until both values are less than 0.1 VDC.

34.7.6.11 Disconnect the voltmeter.

34.7.6.12 If desired, turn off console power and note this action in the Main Logbook.

34.7.6.13 If the calibration was done for the Semiannual Checklist, record the date on the checklist. Otherwise, log the calibration in the Main Logbook and Maintenance Logbook.

CALIBRATION

TROUBLESHOOTING

34.8 TROUBLESHOOTING

34.8.1 Inoperable Control Rod

34.8.1.1 If the **source interlock light is lit**:

- Ensure the neutron source is in the core by visual observation in the Reactor Bay.
- Reset the log channel trips. To reset the log channel, place it in test at Hi Current, and press the reset button, then return it to operate. This is necessary because although the signal is high enough so that the source interlock will not trip, the normal reading is not high enough to reset the trip – a higher signal is necessary.

34.8.1.2 If the **reg rod won't move**:

- Ensure that Rod Control is not in Auto.

34.8.1.3 If the **yellow magnet power "ON" lights are not lit**:

- Ensure that console power is on.
- Ensure that the key is inserted and fully turned to the "operate" position.
- Ensure that all the scrams are reset, including the red ones on the Linear and Percent Power modules.
- See if the ON light bulb is burnt out by exchanging it with the button for a rod that does work. Instructions for removing console buttons are in §34.6.4. If the light bulb is burnt out, replace it per §34.6.4.

34.8.1.4 If a **rod moves up a few percent and stops**:

- If a rod moves up a few percent and stops when the "DOWN" light turns off, the limit switches are probably misaligned. This can be verified because the blue "CONT" light will also turn off when the white "DOWN" turns off.
- The problem is that the motor down limit switch cleared before the rod down limit switch cleared, so the rod circuitry tried to drive in the motor. When you release the "UP" button, the motor will drive to the bottom and the "DOWN" and "CONT" lights will both turn on.
- To correct this problem, you will have to adjust the limit switches per §34.6.1. Be careful since you will have to use metal tools and the circuit boards on the top of the rod motors are exposed and energize.

TROUBLESHOOTING

Control Rod Calibration Form

34B

Date: 1/14/10

Operator Names: Dan Lidral-Porter
Todd Garon
Stephen Von Kugelgen
Ellen McManus
Neal Reynolds

Preliminary

No operations above 5 watts for 48 hours: [✓]

Startup Checklist Complete: [✓]

Rod Position Indication is accurate with ±0.5% of full motion: [✓]

Rod drop times are less than 1 second for full motion: [✓] *

As Found	**DOWN** light lit		**UP** light lit	
	Rod Indication	Multitrend	Rod Indication	Multitrend
Safety Rod	0.2	-0.1	100.5	100.3
Shim Rod	0.0	0.4	100.2	99.3
Reg Rod	0.0	0.5	99.8	99.8

As Left	**DOWN** light lit (0.0-0.5%)		**UP** light lit (99.5-100.5%)	
	Rod Indication	Multitrend	Rod Indication	Multitrend
Safety Rod	0.0	-0.4	100.0	100.0
Shim Rod	0.0	0.6	100.0	100.0
Reg Rod	0.0	0.6	~~99.8~~ 100.1	100.0

	Withdrawal Time (s)	Insertion Time (s)	Drop time (ms)
Safety Rod	47.68	45.05	614
Shim Rod	77.33	74.63	761
Reg Rod	36.10	35.94	533 ms

Period Timer Calibrated at 30% and 81.5% of Linear Channel: [✓]

Voltage Reading with Linear at 30% ~~1.49~~ IF ~~VDC~~ 1.977 VDC
80% ~~3.75~~ ~~VDC~~ 3.780 VDC

Period Timer Connected to Linear Channel: [✓]

Attachment to RRC RAI Response 20100730

34B Control Rod Calibration Form 34B

Reg Rod Calibration

Initial data when stable at 5 Watts for at least 2 minutes.	Reg Rod (at bottom)	Safety Rod	Shim Rod
	0.00	88.4	88.5

Target Reg Rod After Pull	Actual Reg Rod After Pull	Period (ms)	Height when back down at 5 Watts and stable for ≥ 2 min.	
			Safety	Shim
25	26.3	32002	84.5	~~84.6~~ IF 84.9
34	33.0	84919	83.0	82.9
41	41.3	47802	80.4	80.4
48	47.3	78082	78.7	78.8
55	55.2	40034	76.4	76.2
61	61.0 ~~60.6~~ SVK	65245	74.5	74.7
67	68.5	47048	72.7	72.6
74	74.2	82783	71.4	71.1
81	79.8	112045	70.2	70.3
89	89.9	63483	68.8	68.8
UP Light	100.0	126215	67.8	68.2

Control Rod Calibration Form

34B

Safety and Shim Rod Calibration

If it is a different day than the Reg Rod Calibration

No operations above 5 watts for 48 hours: [____]

Startup Checklist Complete: [____]

Rod Position Indication is accurate with ±0.5% of full motion: [____]

Period Timer Calibrated at 30% and 81.5% of Linear Channel: [____]

Voltage Reading with Linear at 30% ______ VDC

80% ______ VDC

Period Timer Connected to Linear Channel: [____]

Initial data when stable at 5 Watts for at least 2 minutes.	Safety Rod	Shim Rod (at top)	Reg Rod (at top)
	49.9	100.0	100.0

Target Safety Rod After Pull	Actual Safety Rod After Pull	Period (ms)	Shim Height when back down at 5 Watts and stable for ≥ 2 min
~~50~~			
53	53.6	37233	88.9
56	56.3	53119	83.2
59	59.3	50344	78.3
62	62.4	50851	74.1
66	66.0	42581	69.7
69	69.4	51506	66.3
72	72.3	67373	63.6
75	75.8	64073	~~61.1~~ 60.9 SVK
78	78.6	84510	58.8
82	81.8	85406	56.7
86	~~86.9~~ 86.5 NR 86.5	58161	53.9
90	90.6	103361	52.0
95	95.1	123398	50.5
UP Light	99.7	165529	49.2

You must drive the Shim Rod in to stabilize at 5 watts after Safety Rod is at the Top!

Concluding Analysis

Data entered in spreadsheet and checked: [✓]

Rod Worths printed and taped in back of main logbook: [✓]

Core Excess: SVK ~~1.29% Δk/k~~ (< 2.25 %Δk/k) * 1.41% Δk/k

Shutdown Margin: SVK ~~5.16% Δk/k~~ (> 0.7 %Δk/k) * 5.28% Δk/k

Shutdown Margin with most reactive control rod stuck out: SVK ~~2.36% Δk/k~~ (> 0.4 %Δk/k) * 2.49% Δk/k

Maximum reactivity addition rate: 0.084% Δk/k·s (< 0.12 %Δk/k/s) *

Operator: [signature] 1/27/10
Date

Supervisor review: [signature] 1/27/10
Date

- Only change the yellow cells. Version **20100315**
- These sheets assume that you calibrated the Safety Rod and used the simultaneous "inserted" data to de
  If that is not the case and you calibrated the Shim Rod separately, You will have to enter that data in th
- Check the equations for Integral and Differential Rod Worth on each of the Rod's sheets and Banked Rod
- When you are done, review all the sheets to make sure they are correct and print the sheets labeled "Pri

	Reg Rod Height After pull		Period (ms)
14-Jan-10	**Target**	**Actual**	
	Start at bottom	0.0	
	25	26.3	32002
	34	33.0	84919
	41	41.3	47802
	48	47.3	78082
	55	55.2	40034
	61	61.0	65245
	67	68.5	47048
	74	74.2	82783
	81	79.8	112045
	89	89.9	63483
	UP Light	100.0	126215

	Safety Height after pull		Period (ms)	Shim Height after stable	Shim Height after pull	Period (ms)
14-Jan-10	**Target**	**Actual**				
Start at lowest critical		49.9		100.0	49.2	
	53	53.6	37233	88.9	50.5	165529
	56	56.3	53119	83.2	52.0	123398
	59	59.3	50344	78.3	53.9	103361
	62	62.4	50851	74.1	56.7	58161
	66	66.0	42581	69.7	58.8	85406
	69	69.4	51506	66.3	60.9	84510
	72	72.3	67373	63.6	63.6	64073
	75	75.8	64073	60.9	66.3	67373
	78	78.6	84510	58.8	69.7	51506
	82	81.8	85406	56.7	74.1	42581
	86	86.5	58161	53.9	78.3	50851
	90	90.6	103361	52.0	83.2	50344
	95	95.1	123398	50.5	88.9	53119
	UP Light	99.7	165529	49.2	100.0	37233

	Withdrawal Time (secs)		Critical Power	Banked Rod Height	
Safe	47.68		230 kW	94	After being shutdown for
Shim	77.30		5 W	72	After being shutdown for
Reg	36.10				

	Max Reactivity Addition Rate ($/%)	Max Reactivity Addition Rate ($/sec)	Max Reactivity Addition Rate (% Δk/k/s)	Less than 0.12% Δk/k/s?
Safety	0.04752	0.100	0.075	**OK**
Shim	0.04579	0.059	0.044	**OK**
Reg	0.01989	0.055	0.041	**OK**

	Height with Safety IN	Total Worth	Core Excess
Safety	49.9	$3.31	$1.66
Shim	100	$3.27	$0.00
Reg	100	$1.34	$0.00
		$7.92	$1.66

Shutdowm Margin	$6.26	4.69%
Minimum Shutdowm Margin	$2.94	2.21%

Core Excess Less Than 2.25%/Δk/k ($3)	**OK**
SDM Greater Than 0.7%/Δk/k ($0.93)	**OK**
Min SDM Greater Than 0.4%/Δk/k ($0.53)	**OK**

	Height with Shim IN	Total Worth	Core Excess
Safety	100	$3.31	$0.00
Shim	50.5	$3.27	$1.66
Reg	100	$1.34	$0.00
		$7.92	$1.66

Shutdowm Margin	$6.26	4.69%
Minimum Shutdowm Margin	$2.94	2.21%

Core Excess Less Than 2.25%/Δk/k ($3)	**OK**
SDM Greater Than 0.7%/Δk/k ($0.93)	**OK**
Min SDM Greater Than 0.4%/Δk/k ($0.53)	**OK**

Cut this out and tape it on the console faceplate:

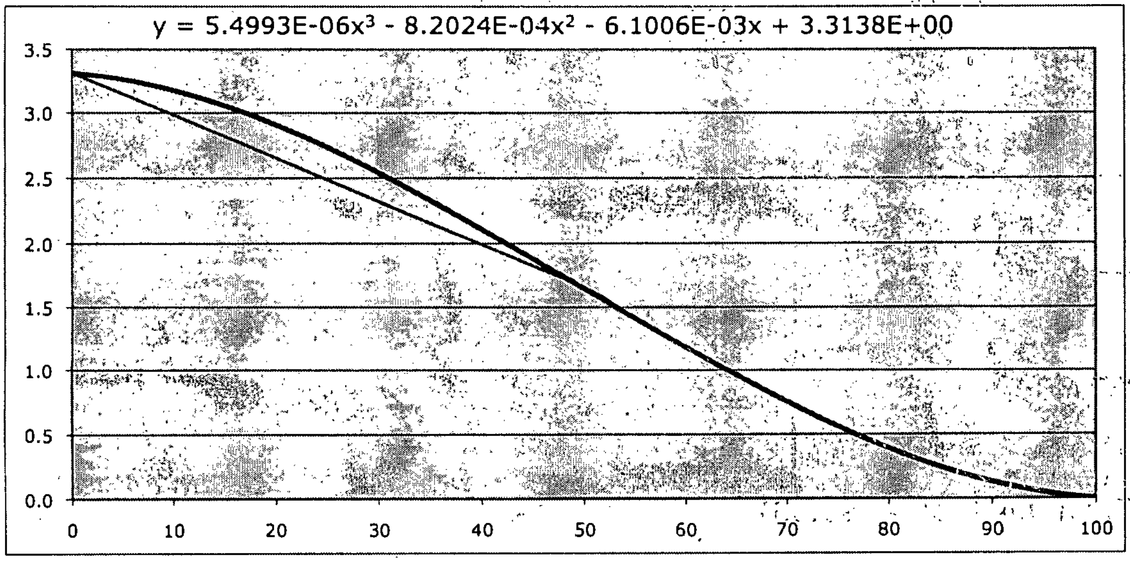
Jan-2010 Approximate Values

kW	% of Range	Rod Height	Core Excess
230	92	94	$0.16
200	80	90	$0.35
175	70	86	$0.52
150	60	84	$0.68
125	50	81	$0.84
100	40	79	$1.01
75	30	78	$1.17
50	20	76	$1.33
1	40	72	$1.65
0.005	20	72	$1.66

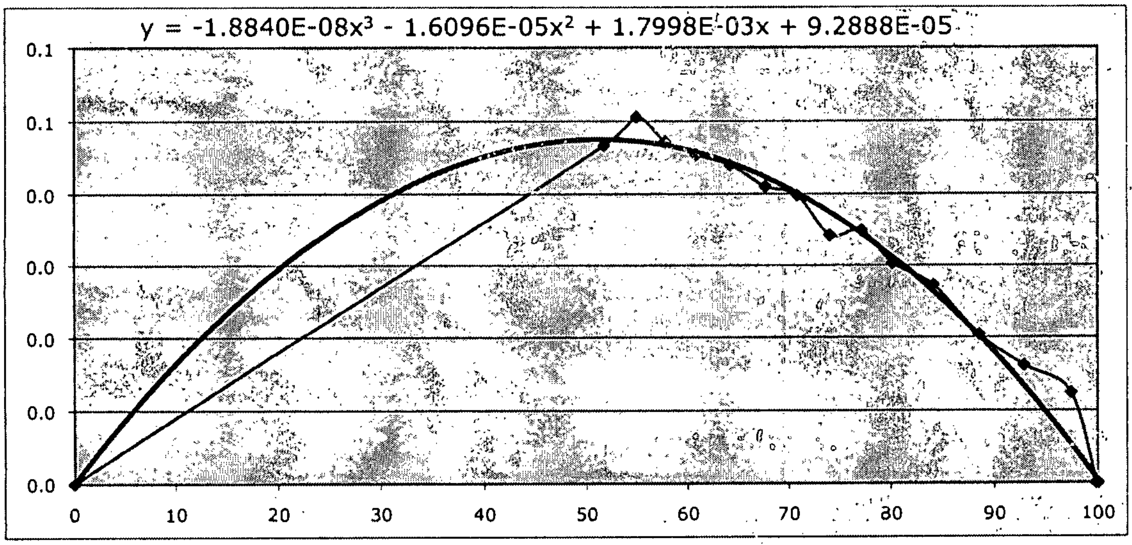
20100315 Verson

Safety **1/14/10**

	x^3	x^2	x	const	
Integral	5.4990E-06	-8.2020E-04	-6.1021E-03	3.3138E+00	**Ensure these numbers ma the equation on the graph**



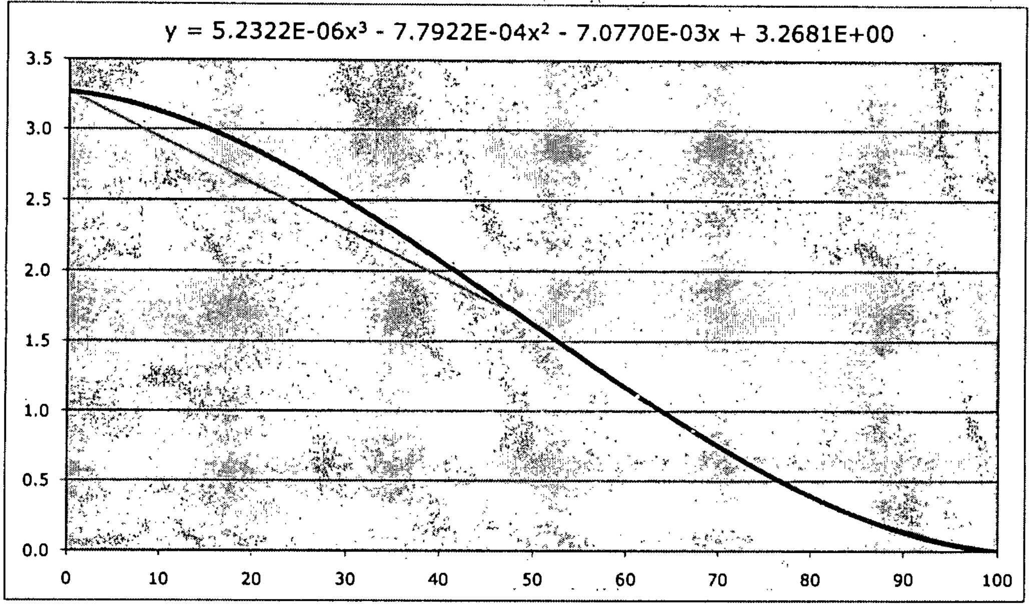
	x^3	x^2	x	const	
Differential	-1.8840E-08	-1.6096E-05	1.7998E-03	9.2888E-05	**Ensure these numbers ma the equation on the graph**



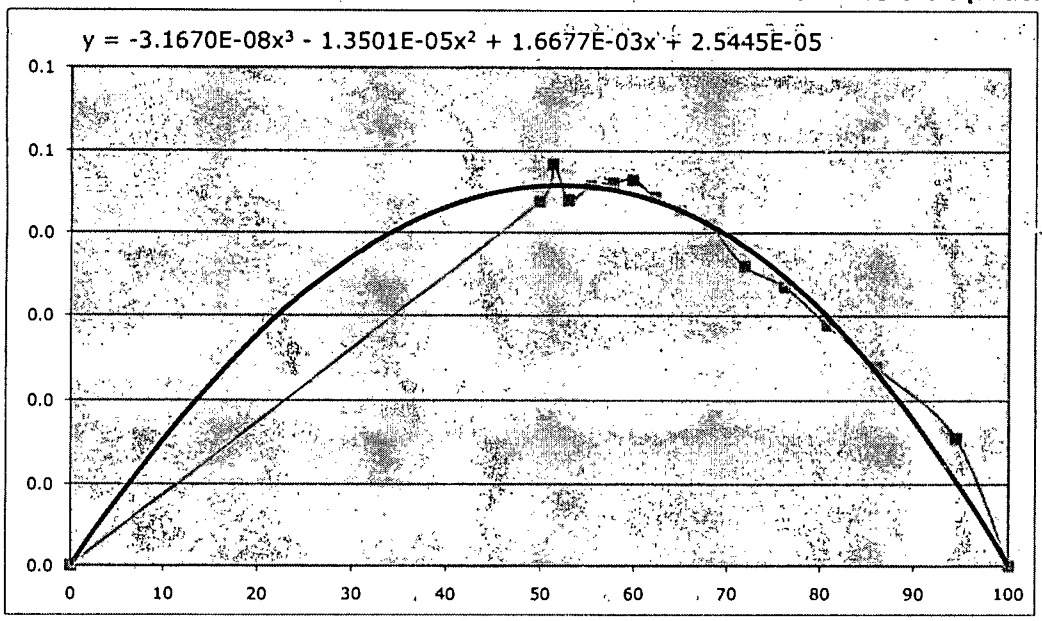
Safe Height after pull	Period (sec)	Reactivity (Δk/k)	Reactivity ($)	Integral ($)	Midpoint	Addition Rate $/%
0.0			0	3.31	0.0	0.0000
49.9	37.2	0.001294	0.1725	1.66	51.8	0.0466
53.6	53.1	0.001024	0.1365	1.49	55.0	0.0506
56.3	50.3	0.001062	0.1416	1.35	57.8	0.0472
59.3	50.9	0.001055	0.1406	1.21	60.9	0.0454
62.4	42.6	0.001187	0.1583	1.07	64.2	0.0440
66.0	51.5	0.001046	0.1394	0.91	67.7	0.0410
69.4	67.4	0.000866	0.1155	0.77	70.9	0.0398
72.3	64.1	0.000898	0.1197	0.66	74.1	0.0342
75.8	84.5	0.000732	0.0976	0.54	77.2	0.0349
78.6	85.4	0.000726	0.0968	0.44	80.2	0.0303
81.8	58.2	0.000961	0.1282	0.34	84.2	0.0273
86.5	103.4	0.000627	0.0836	0.21	88.6	0.0204
90.6	123.4	0.000544	0.0726	0.13	92.9	0.0161
95.1	165.5	0.000427	0.0569	0.06	97.4	0.0124
99.7			0.0000	0.00	99.9	0.0000
100.0				0.00	100.0	0.0000

Shim **1/14/10**

	x^3	*x^2*	*x*	*const*	
Integral	5.2323E-06	-7.7924E-04	-7.0764E-03	3.2681E+00	**Ensure these numbers match the equation on the graph be**

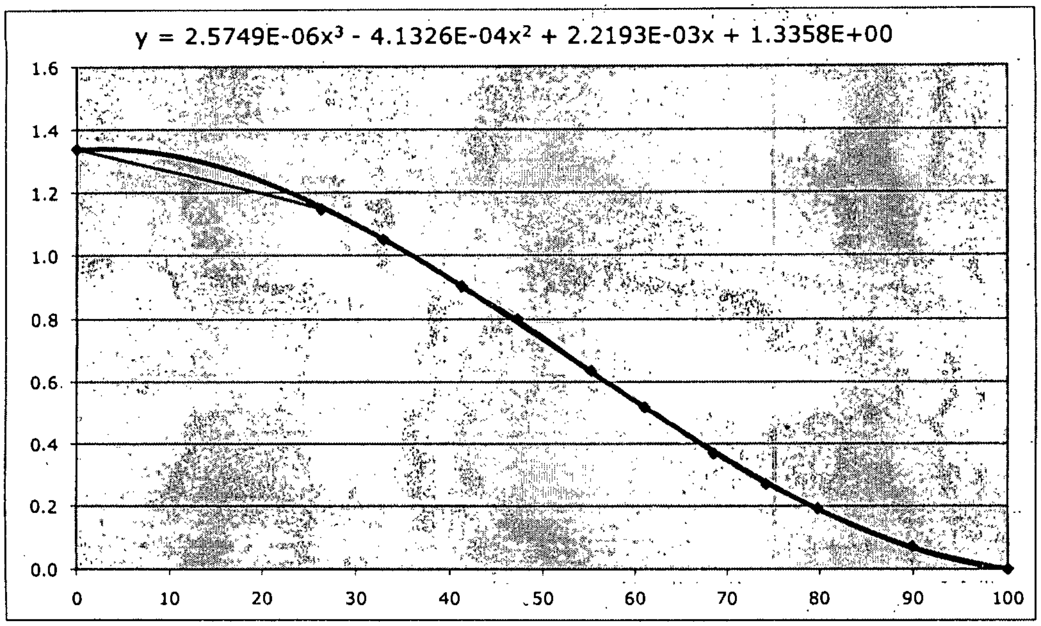


	x^3	*x^2*	*x*	*const*	
Differential	-3.1619E-08	-1.3508E-05	1.6679E-03	2.5385E-05	**Ensure these numbers match the equation on the graph be**

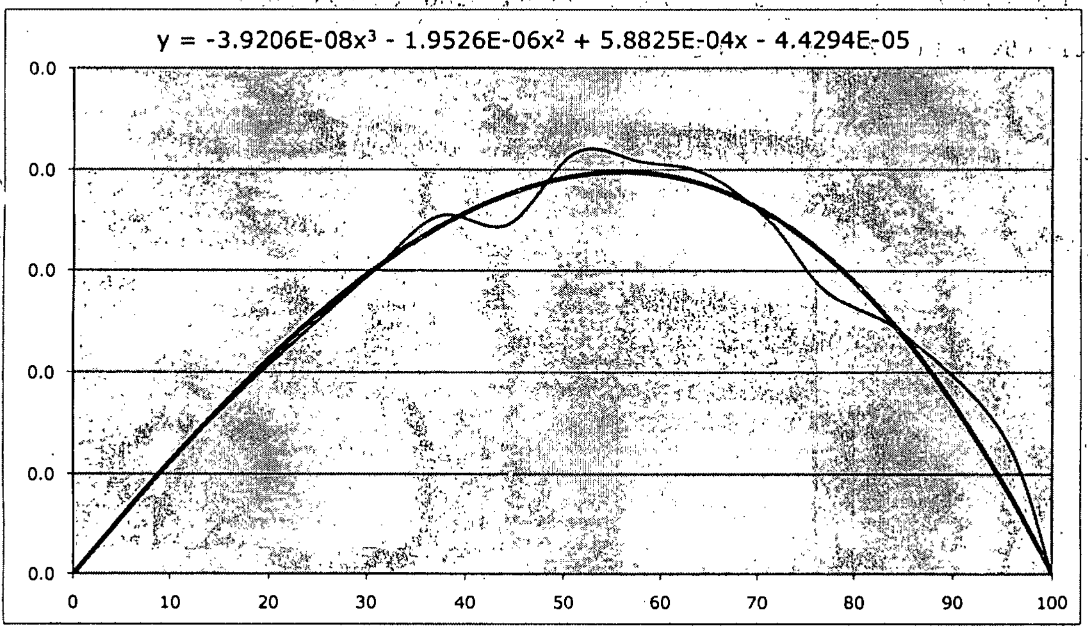


Shim Height before pull	Period (sec)	Reactivity (Δk/k)	Reactivity ($)	Integral ($)	Midpoint	Addition Rate $/%
0.0			0	3.27	0.0	0.0000
49.2	165.53	0.000427	0.0569	1.66	49.9	0.0438
50.5	123.40	0.000544	0.0726	1.60	51.3	0.0484
52.0	103.36	0.000627	0.0836	1.53	53.0	0.0440
53.9	58.16	0.000961	0.1282	1.45	55.3	0.0458
56.7	85.41	0.000726	0.0968	1.32	57.8	0.0461
58.8	84.51	0.000732	0.0976	1.22	59.9	0.0465
60.9	64.07	0.000898	0.1197	1.12	62.3	0.0443
63.6	67.37	0.000866	0.1155	1.00	65.0	0.0428
66.3	51.51	0.001046	0.1394	0.89	68.0	0.0410
69.7	42.58	0.001187	0.1583	0.75	71.9	0.0360
74.1	50.85	0.001055	0.1406	0.59	76.2	0.0335
78.3	50.34	0.001062	0.1416	0.45	80.8	0.0289
83.2	53.12	0.001024	0.1365	0.31	86.1	0.0240
88.9	37.23	0.001294	0.1725	0.17	94.5	0.0155
100.0				0.00	100.0	0.0000
100.0				0.00	100.0	0.0000

1/14/10	*x^3*	*x^2*	*x*	*const*	**Ensure these numbers match**
Integral	2.5749E-06	-4.1326E-04	2.2193E-03	1.3358E+00	**the equation on the graph be**



Reg	*x^3*	*x^2*	*x*	*const*	**Ensure these numbers match**
Differential	-3.9206E-08	-1.9526E-06	5.8825E-04	-4.4294E-05	**the equation on the graph be**



Reg Height after pull	**Period (sec)**	**Reactivity (Δk/k)**	**Reactivity ($)**	**Integral ($)**	**Midpoint**	**Addition Rate $/%**
0.0			0	1.34	0.0	0.0000
26.3	32.00	0.001421	0.1894	1.15	13.2	0.0072
33.0	84.92	0.000729	0.0973	1.05	29.7	0.0145
41.3	47.80	0.001100	0.1466	0.90	37.2	0.0177
47.3	78.08	0.000777	0.1036	0.80	44.3	0.0173
55.2	40.03	0.001235	0.1647	0.64	51.3	0.0209
61.0	65.25	0.000886	0.1182	0.52	58.1	0.0204
68.5	47.05	0.001112	0.1482	0.37	64.8	0.0198
74.2	82.78	0.000744	0.0992	0.27	71.4	0.0174
79.8	112.05	0.000588	0.0784	0.19	77.0	0.0140
89.9	63.48	0.000904	0.1205	0.07	84.9	0.0119
100.0	126.22	0.000534	0.0712	0.00	95.0	0.0071
100.0				0.00	100.0	0.0000
100.0				0.00	100.0	0.0000
100.0				0.00	100.0	0.0000

Reed Research Reactor

Safety Analysis Report

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 [redacted] 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 operation of the reactor at 250 kW thermal power. 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 250 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 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 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 250 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 250 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 250 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 250 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 250 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 [redacted] of 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 250 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 about [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 250 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.

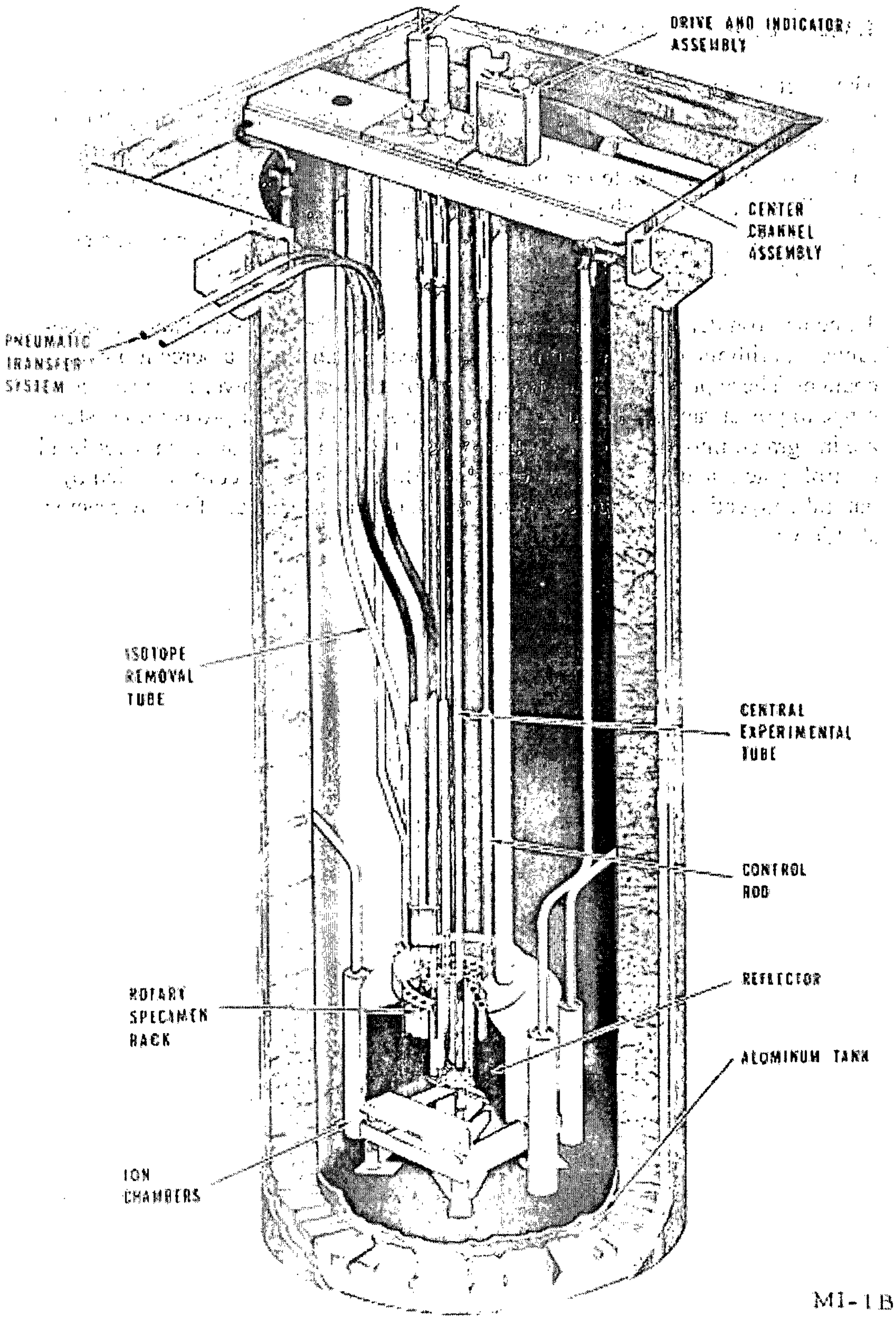

Figure 1.1 Cutaway View of Typical TRIGA® Reactor

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 is normally 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. The facility has some uninterruptible power supplies (UPS) to provide power for some instruments and [redacted] following loss of electricity, and the facility has emergency lighting. The facility has no other backup power capability, so if the campus looses electricity, the ventilation fans, cooling pumps, and other equipment would turn off. The dampers and valves fail as-is. The UPS will allow detection of airborne radiation, area radiation, reactor pool level, and reactor power.

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 250 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.

The RRR is a standard Mark I TRIGA® reactor [redacted] tank. Operating issues at other TRIGA® reactors are therefore applicable to the RRR, such as corrosion, leaks, component wear, etc. A unique characteristic of the RRR is that the reactor pool [redacted] The RRR does not have beam ports other than the vertical central thimble. The RRR does not use fuel element followers on its control rods.

Table 1.1 Major Design Parameters

Parameter	Value
Max steady-state thermal power	250 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

Chapter 2

Site Characteristics

Reed Research Reactor Safety Analysis Report

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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 a[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.

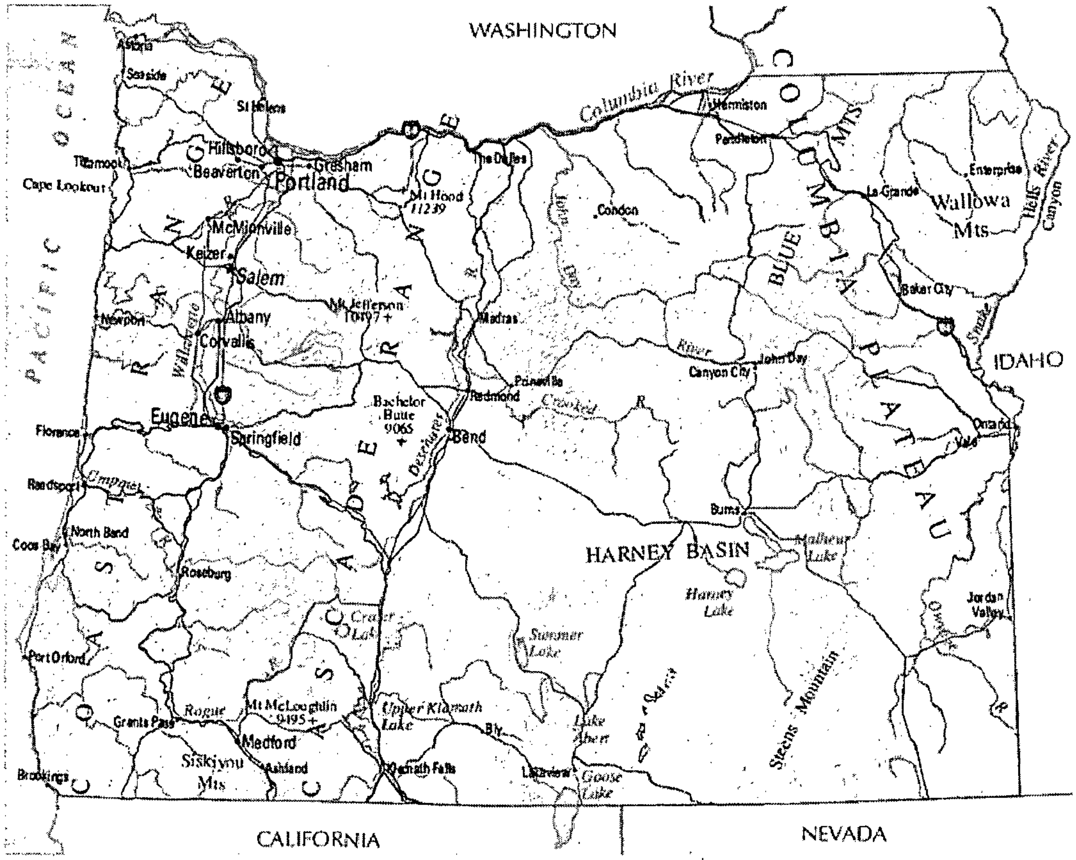


Figure 2.1 State Map of Oregon

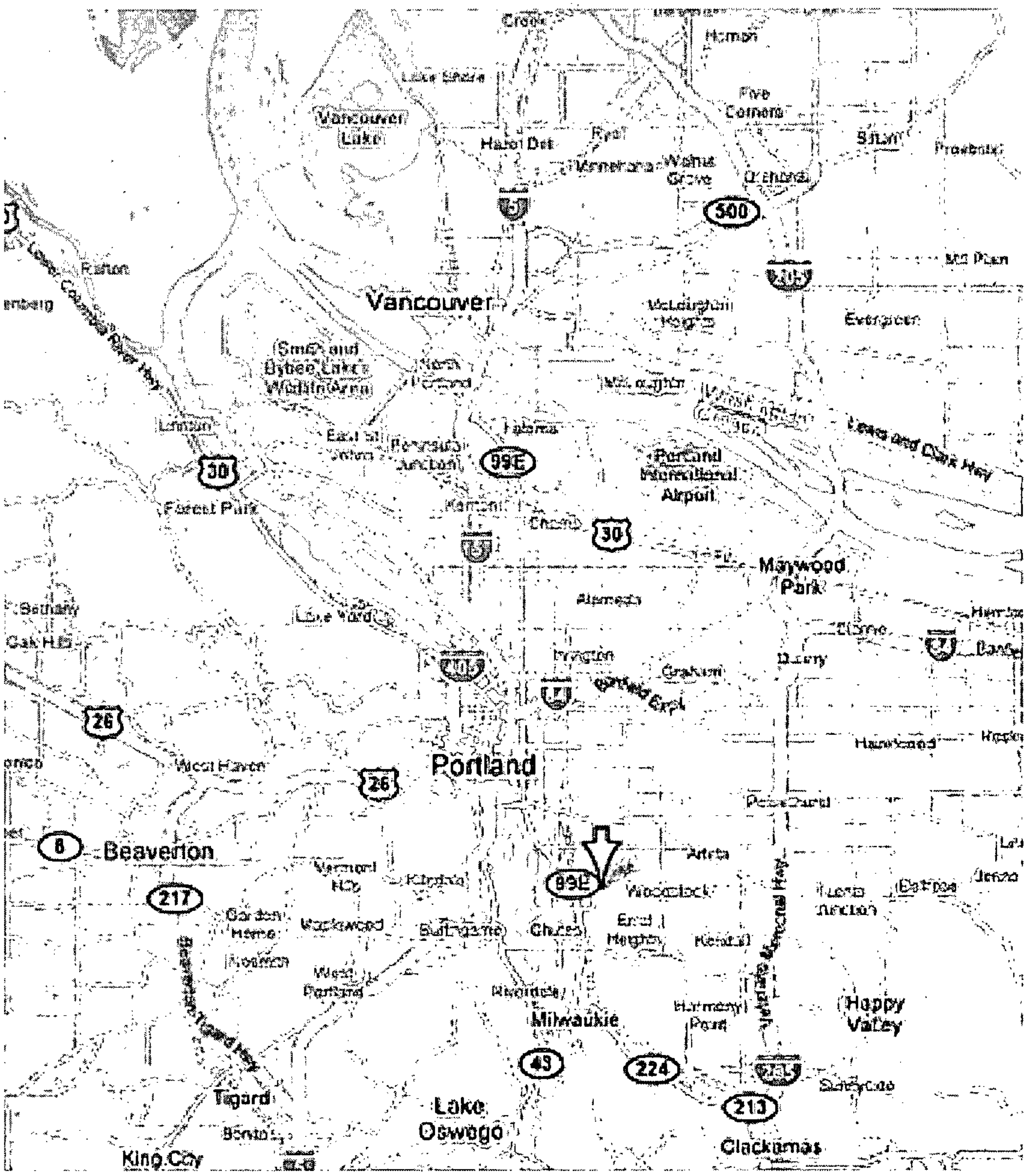
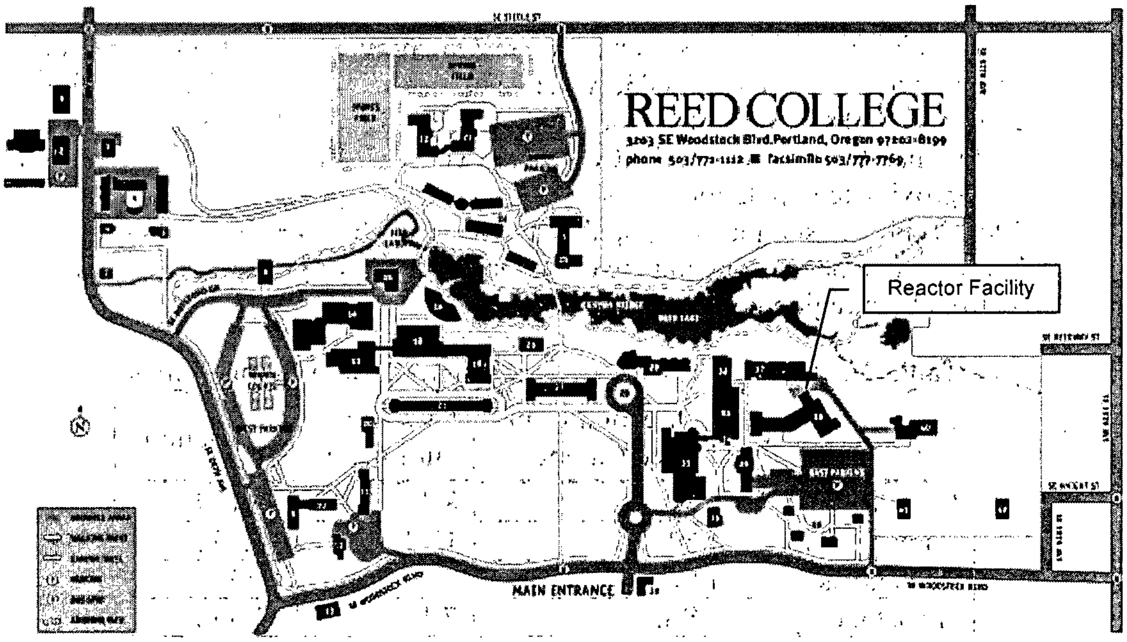


Figure 2.2 Portland and Surrounding Highways, Streams, Rivers and Bodies of Water. The facility location is indicated by the arrow



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.

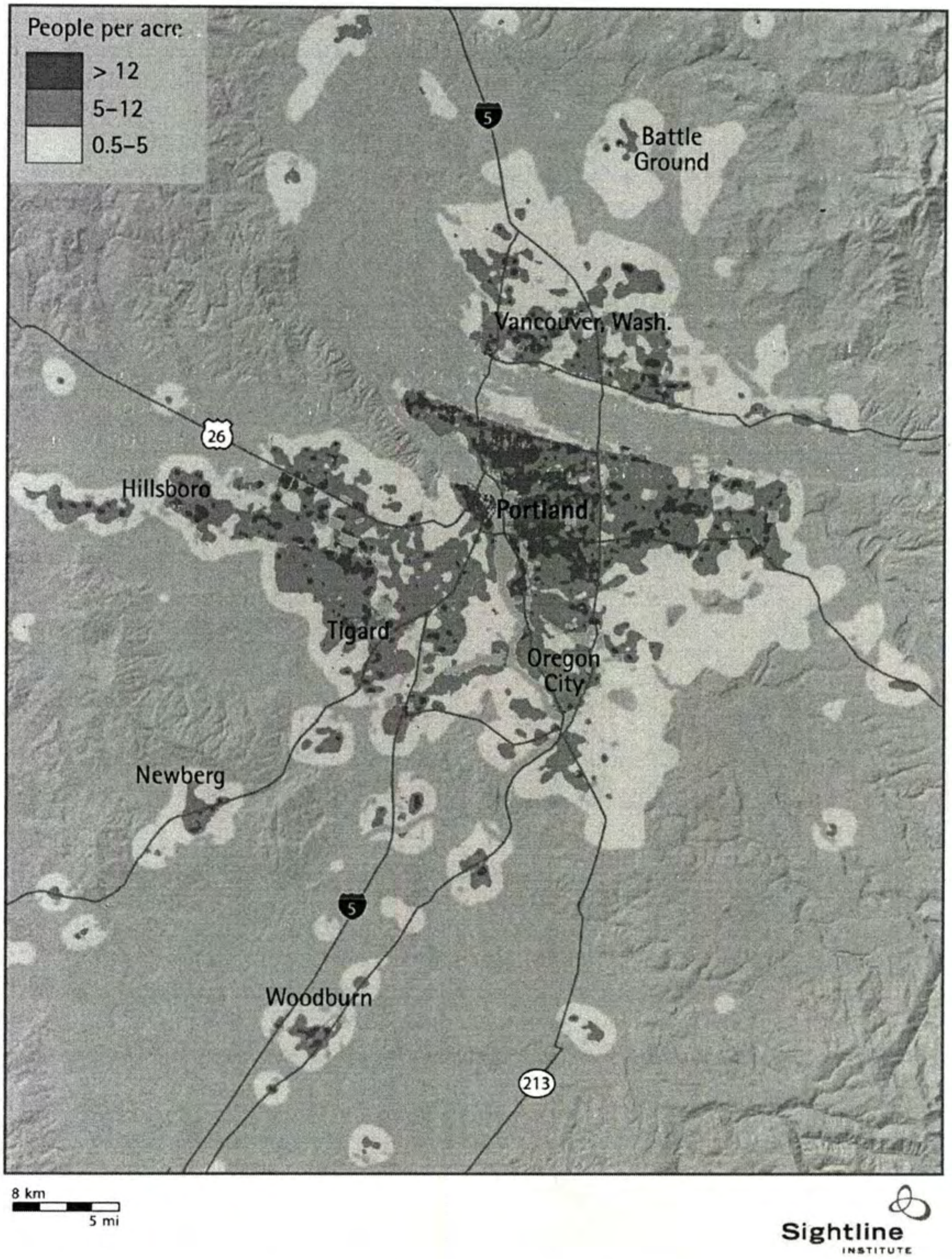


Figure 2.4 Population Density in the Portland Metropolitan Region

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. There are no nearby military facilities.

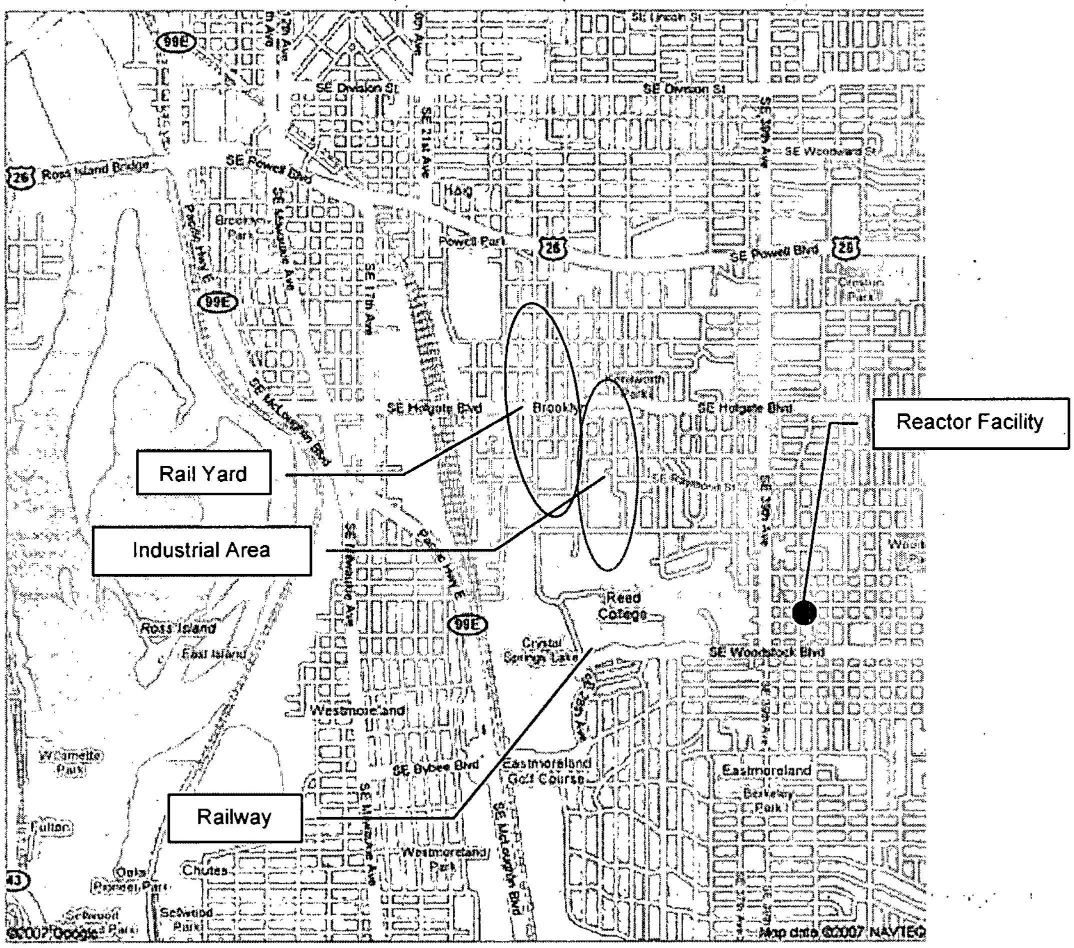


Figure 2.5 Railways Near the Reed Campus

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.

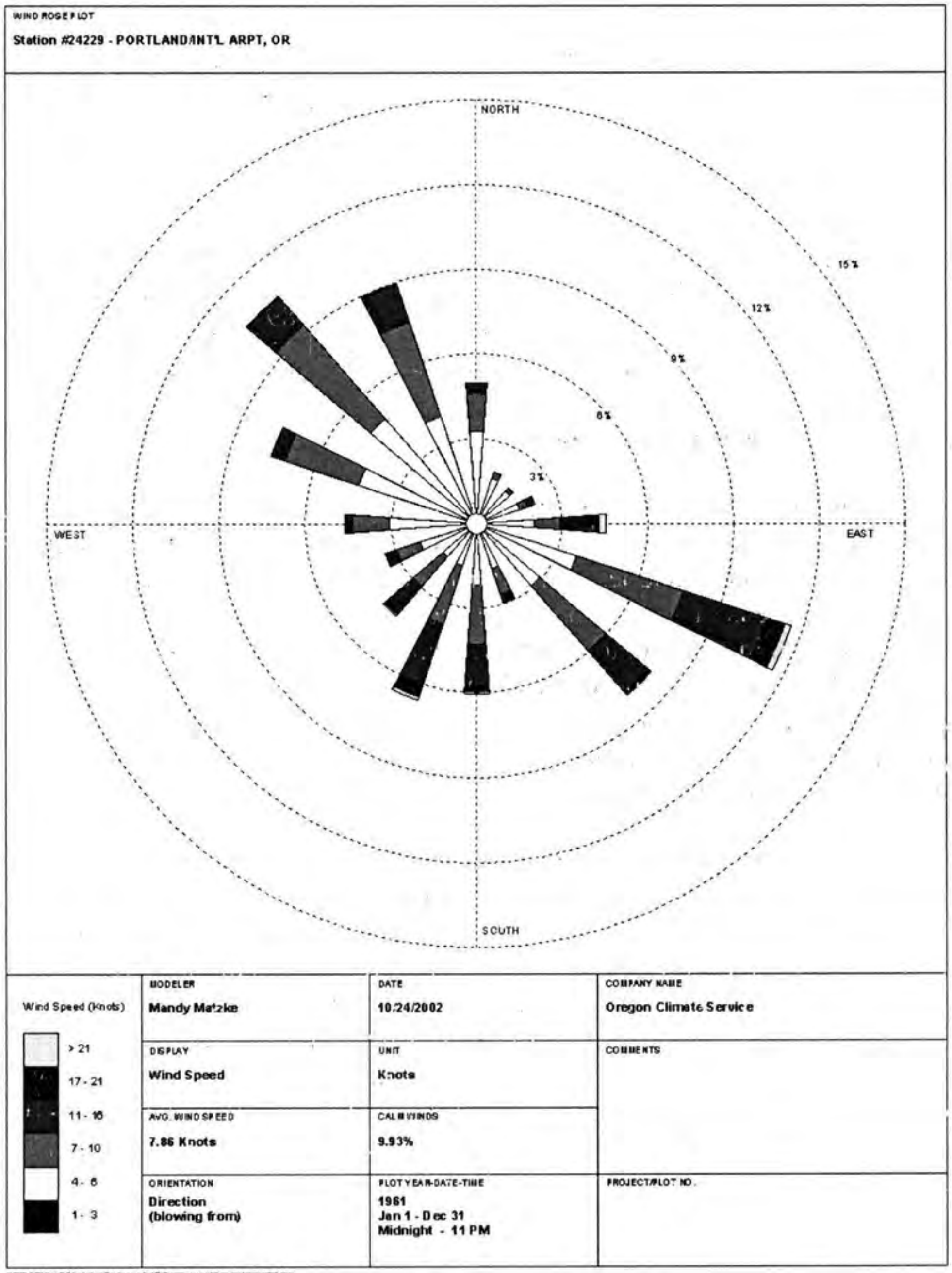


Figure 2.6 Portland Wind Rose (Annual Average)

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]

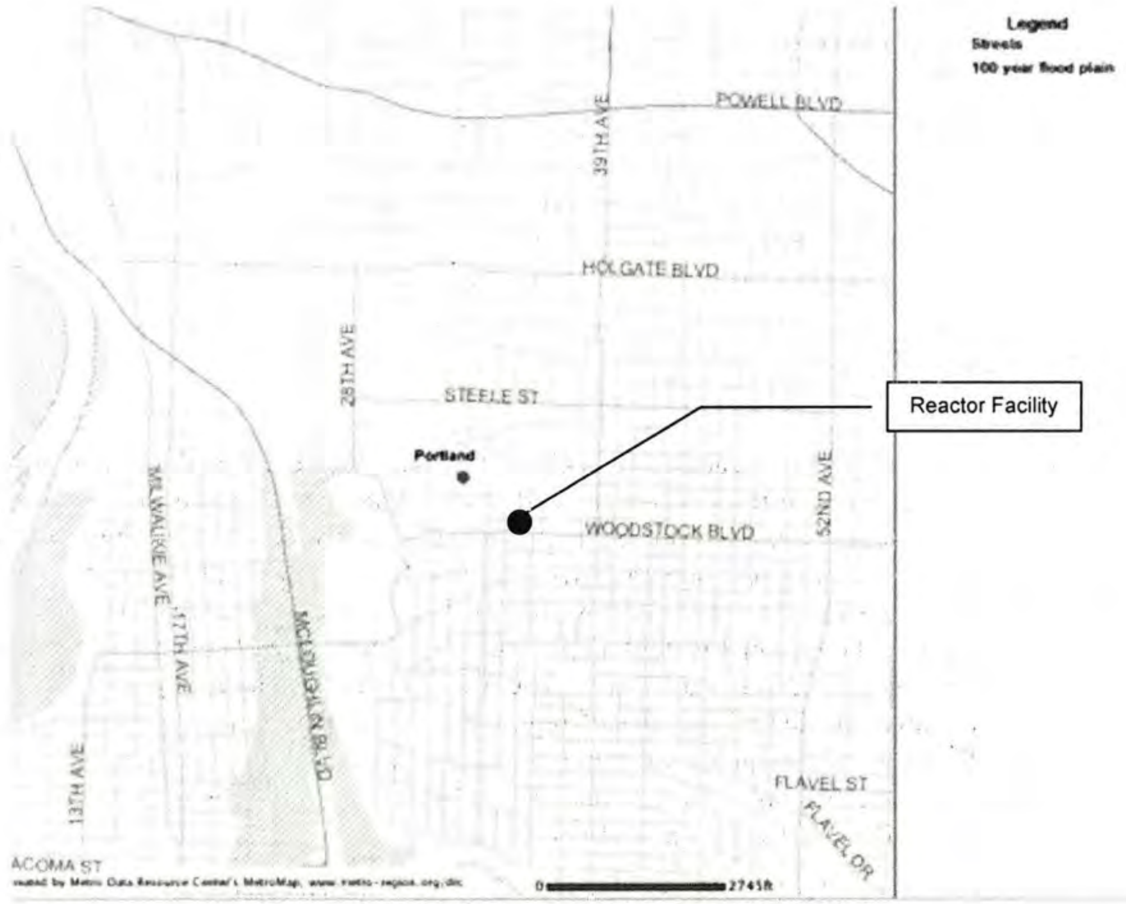


Figure 2.6 Reed Campus 100-year Flood Plain

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

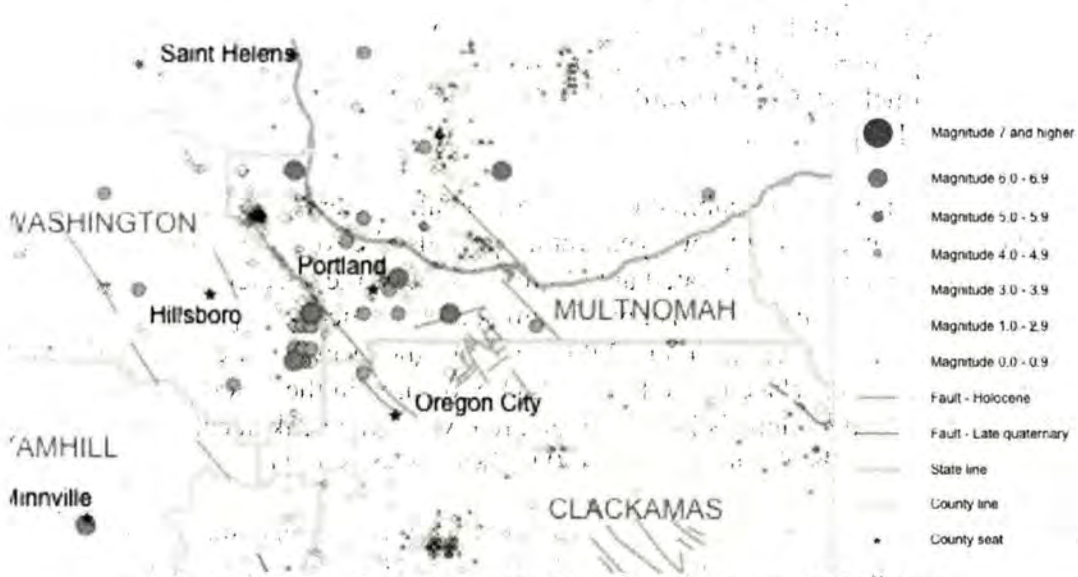


Figure 2.7 Earthquakes near Portland Oregon, 1841-2002

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]

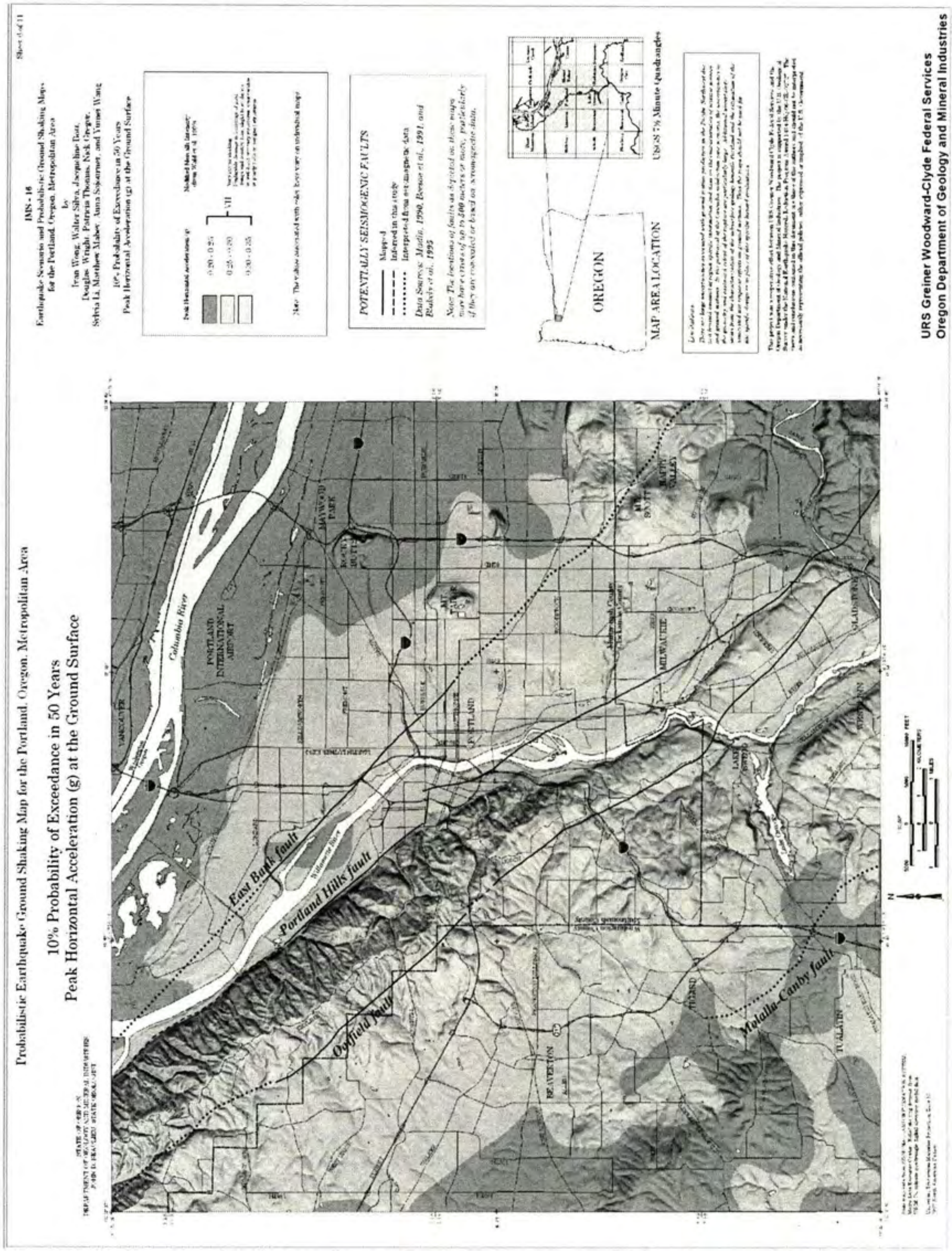


Figure 2.8: 500 year return Probabilistic Earthquake Ground Shaking Map for the Portland Area

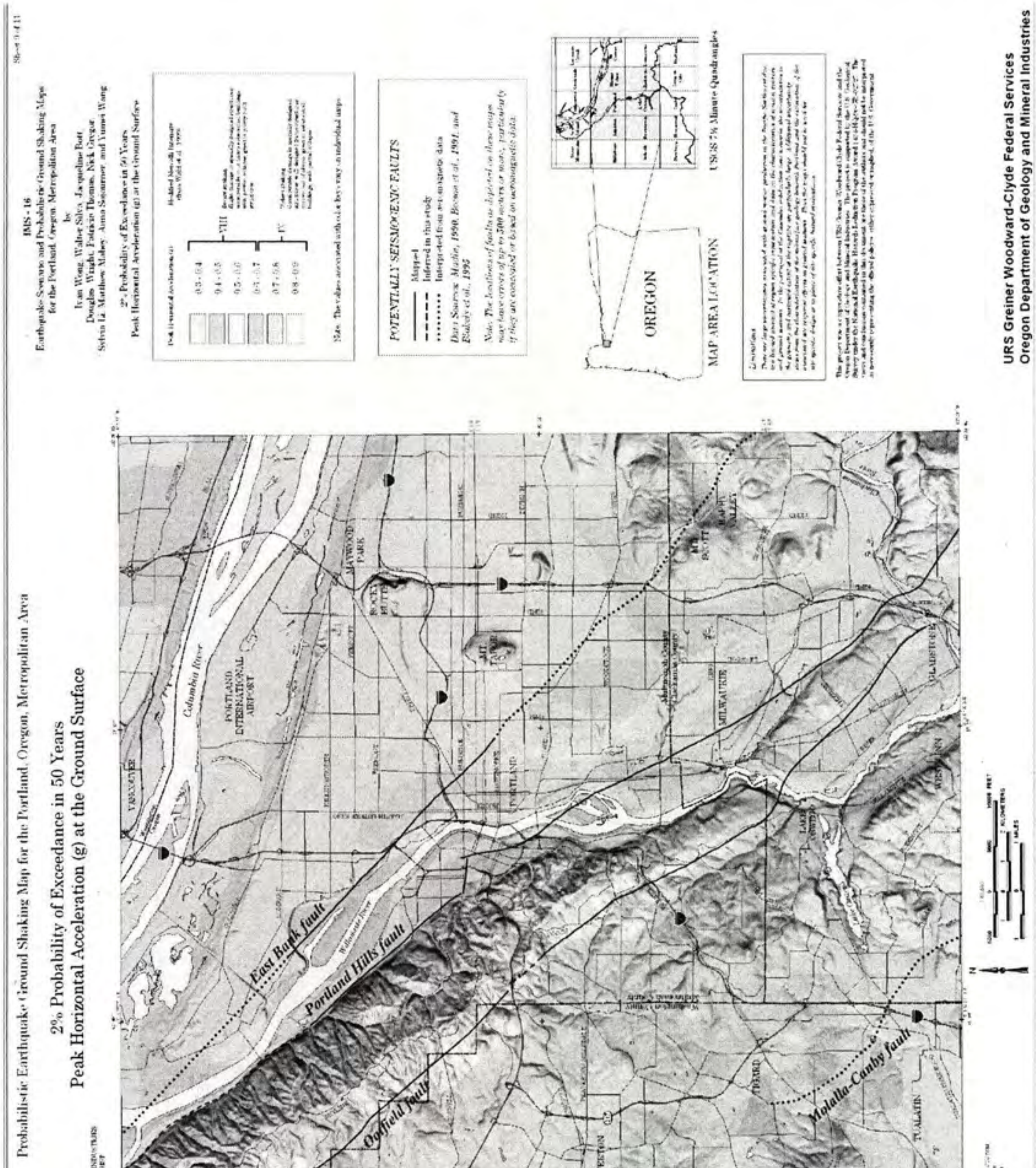


Figure 2.9: 2500 year Return Earthquake Scenario Ground Shaking Map for the Portland Area

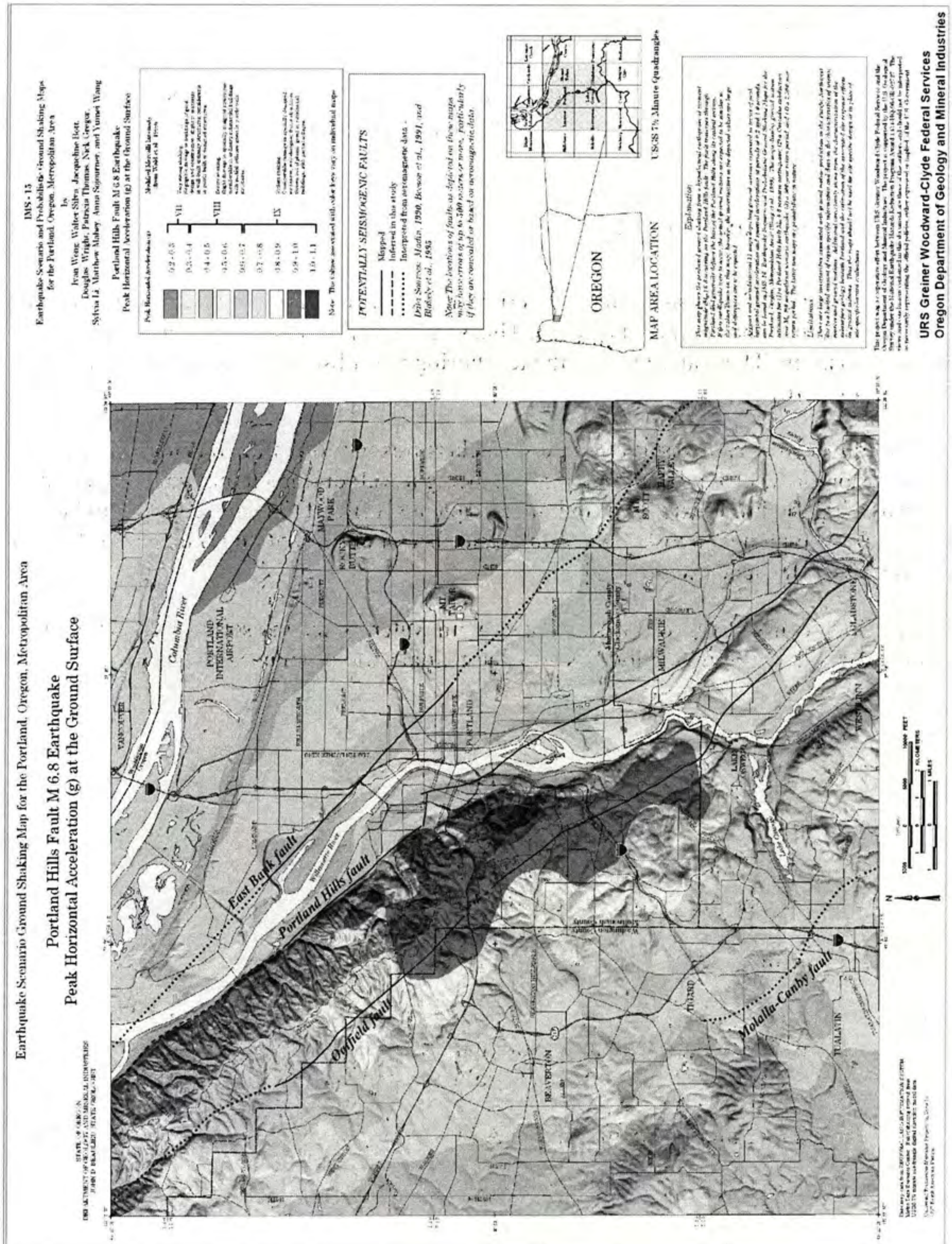


Figure 2.10: Earthquake Scenario Ground Shaking Map for the Portland Area

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

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Chapter 3

Design of Structures, Systems, and Components

Reed Research Reactor Safety Analysis Report

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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 [redacted] 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. All building modifications and equipment additions were in conformance with Oregon and Portland building codes in existence at the time, to meet fire, safety, seismic, and flood requirements. The contract architect for installation was Farnham & Peck.

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

[redacted] 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. The reactor building is designed for anticipated rain or snow loads. However, even if flooding occurred, core safety would not be an issue since the core is located in a water pool. Water may damage electrical components by causing electrical shorts and a loss of power. Loss of power to electrical systems is addressed in Chapter 13.

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. Automatic scrams are provided for high power indication on the core neutron detectors. A manual scram is available to the operator at any time. 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.

Interlocks are provided to limit reactivity insertion. One interlock will prevent control rod withdrawal unless there is a neutron-induced signal on one of the nuclear instruments. The other interlock prevent simultaneous manually withdrawal of more than one control rod.

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 250 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. In the event of a loss of electrical power, the electromagnets will deenergize and drop the control rods into the core, which is the safe condition. Details of the control rod scram system are addressed in Chapter 4, Reactor Description, and Chapter 7, Instrumentation and Control.

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 is designed to remove 250 kW of heat, thus is adequate for both normal operation and shutdown cooling. 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.

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 an alarm if water level is above or below normal.

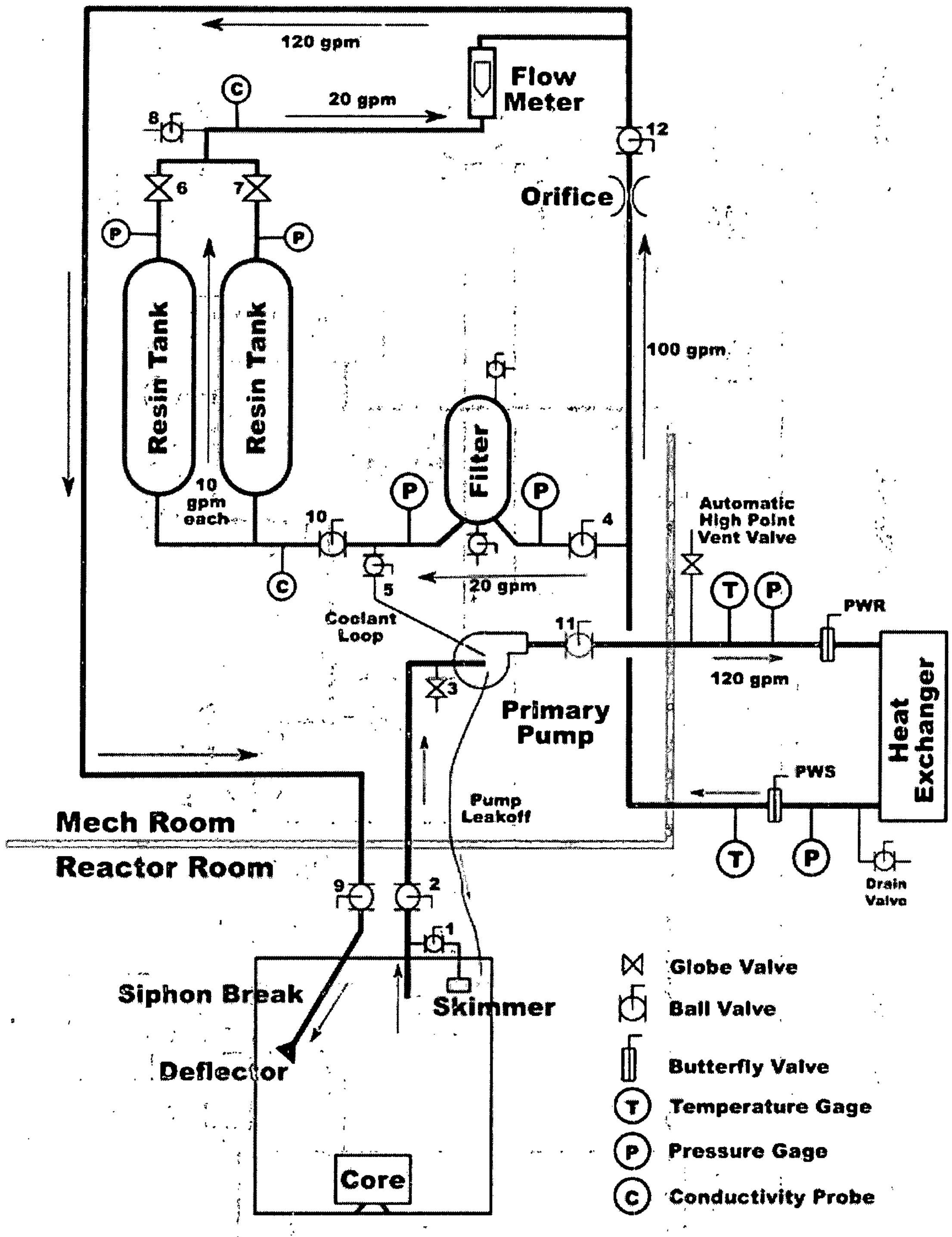

Figure 5.1 Primary Cooling System

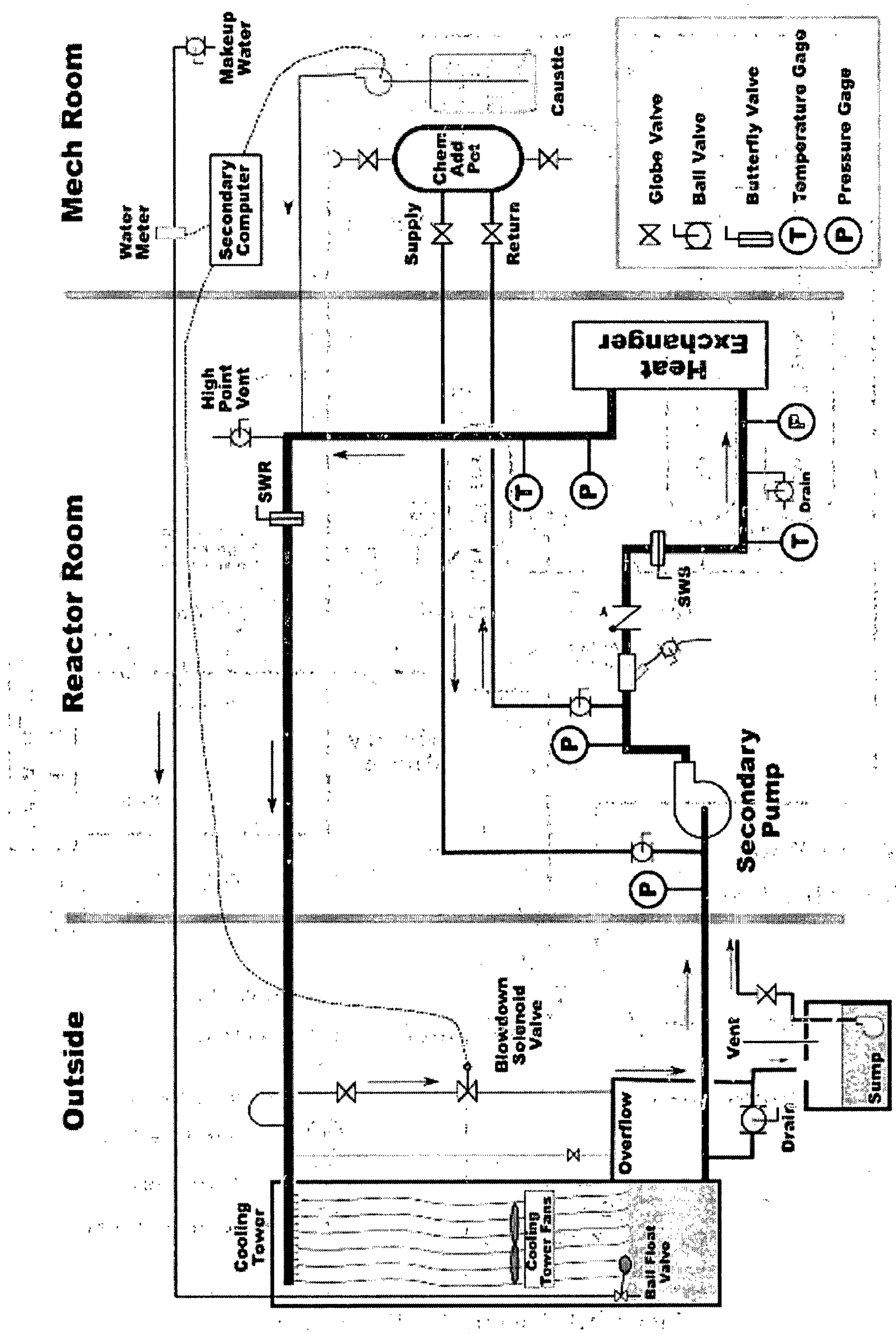

Figure 5.2 Secondary Cooling System

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 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 piping and a valve.

The skimmer is a plastic cylinder that contains a basket in its upper end. 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.

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 2009. It is a plate-type heat exchanger rated for 500 kW at 250°F (120°C) and 150 psi (1 MPa).

5.3 Secondary Cooling System

The secondary cooling system circulates water from the heat exchanger through the cooling tower. The secondary cooling system is designed to remove at least 250 kW of heat so as to sustain continuous operation at full power. 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 Primary Coolant Cleanup System

The primary coolant cleanup system maintains reactor pool water conductivity below 2 microSeimens/cm by means of a filter and a mixed bed demineralizer to limit corrosion of the reactor tank and the primary system. The mixed bed demineralizer tends to keep the water pH neutral, but this is not measured. This limit has been shown through 40 years of experience to adequately limit the corrosion.

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 and will be disposed of in accordance with Chapter 11, Radiation Protection and Waste Management. Historically the dose rate at 30 cm from the demineralizers after continuous at full power operation is less than 100 mrem per hour, which is consistent with the ALARA program. Historically the annual shipment or spent resin and filters average 110 cubic feet, which is consistent with the ALARA program.

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.

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.

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

5.5 Primary Coolant Makeup Water System

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

The amount of water added through the make up system is recorded weekly to detect increases in the make up rate which might indicate leaks from the primary system or the reactor tank.

5.6 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.

Historically the concentration of N-16 at the site boundary has averaged 1×10^{-10} µCi/ml, which would produce a dose to a member of the public at the site boundary of approximately 0.5 mrem per year, well in compliance with the regulations and the ALARA program.

5.7 Auxiliary Systems using Primary Coolant

There are no auxiliary system which use primary coolant.

5.8 References

1. DOE Fundamentals Handbook Chemistry, Volume 1 of 2, DOE-HDBK-1015/1-93, January 1993.
2. Metallurgy of TRIGA® Fuel Elements, W.P. Wallace and M.T. Simnad, General Atomics, January 4, 1961.
3. Corrosion Issues in the Long Term Storage of Aluminium-Clad Spent Nuclear Fuels, H.B. Peacock, R.L. Sindelar, McIntyre R. Louthan, CORROSION 96, March 24 - 29, 1996 , Denver, Co

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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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 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.

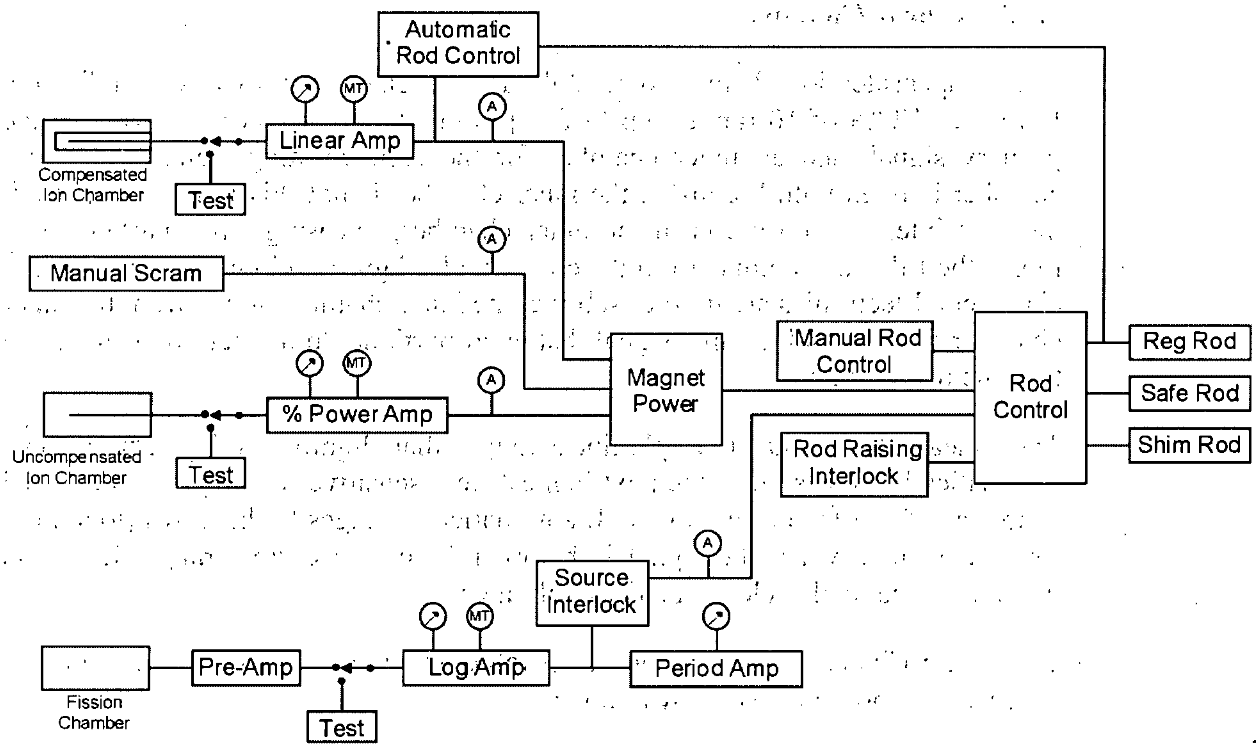

Figure 7.1 Instrumentation and Control

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. 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. 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.

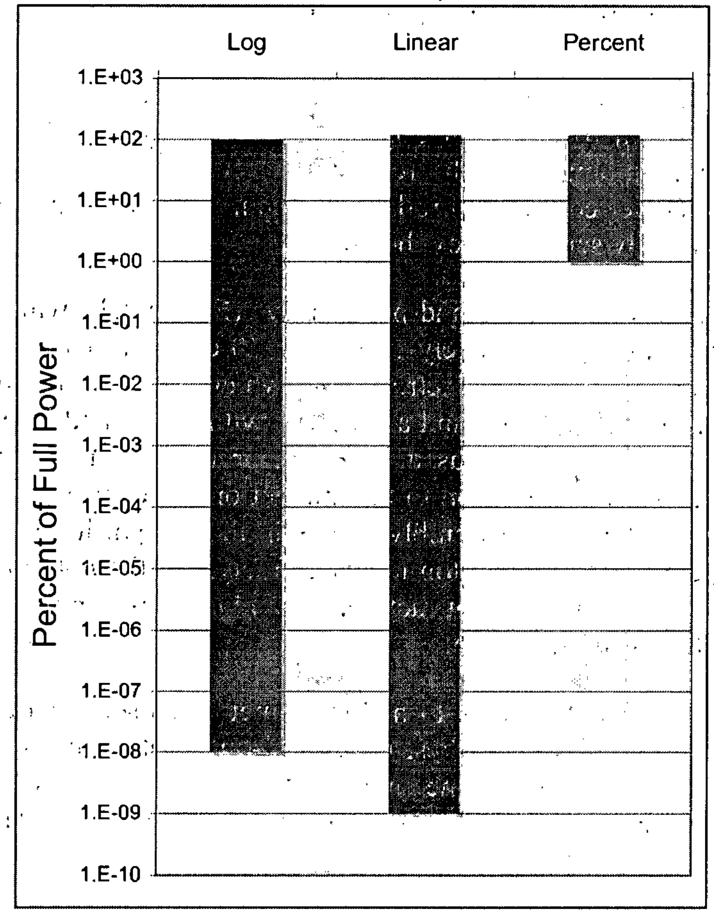

Figure 7.2 Relative Ranges of the Power Indications

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 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. 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. The limit switches are shown in Figure 7.4.

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.

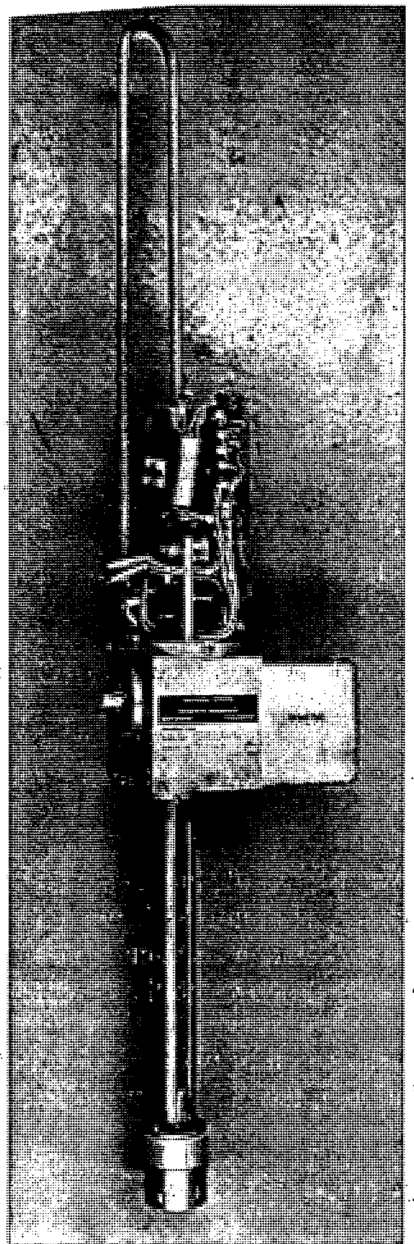

Figure 7.3 Control Rod Drive

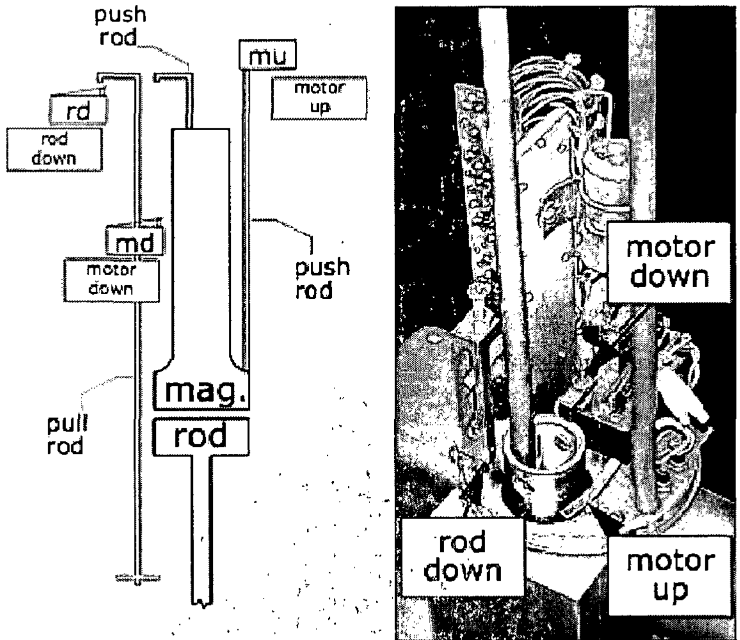

Figure 7.4 Control Rod Limit Switches

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 gravitational force. 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 neutron detectors are displayed in separate electronic modules on the console. The information is also supplied to a console data recorder.

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. There are no audible annunicators on the console. The annunciators only reset when the console key is moved to the reset position. There are Annunciators for the Manual Scram, Linear Channel Scram, and Percent Power Channel Scram. In addition there is an annunciator for the Source Interlock.

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 will alarm on high airborne activity.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 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. Routine maintenance and service of these systems is the responsibility of the Reed College Physical Plant.

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.

Upon a high radioactivity alarm on the reactor bay air monitor (typically 1 x 10^{-8} μCi/ml), the system switches 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 100 cfm (50 L/s). This maintains the reactor bay at a negative pressure which serves to restrict air leakage from the reactor bay. This serves to contain any possible airborne radioactivity. Once the system is in isolation mode, it will remain in isolation mode until it is manually reset. The system can be set to isolation mode by a button on the reactor console. Technical Specifications require periodic testing to ensure that the isolation system works properly.

Figure 9.1 Ventilation System in Normal Operation

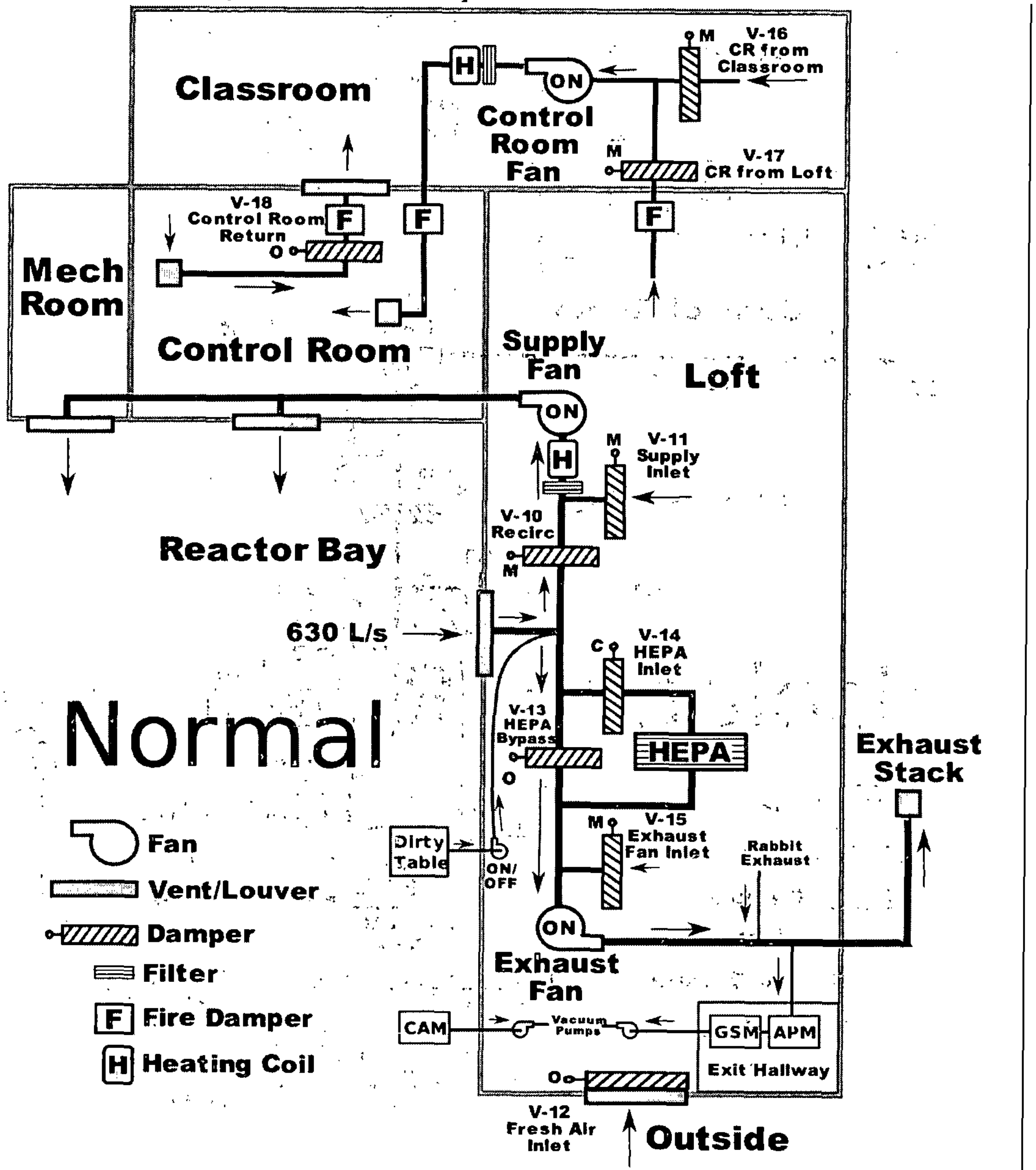
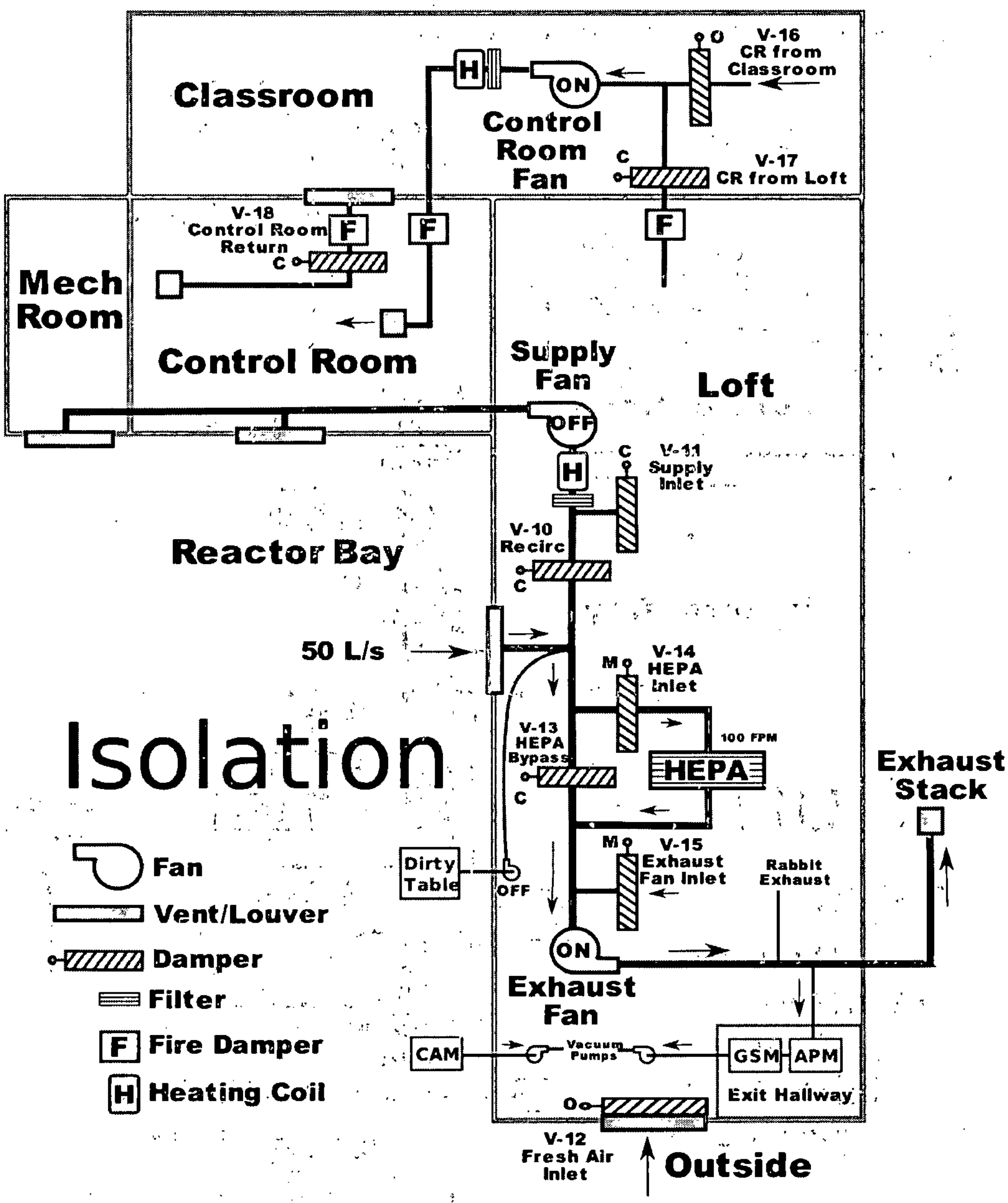


Figure 9.2 Ventilation System in Isolation



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. No neutron poisons are used.

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

Fuel inspection is accomplished with a small underwater video camera attached to a rigid pole. The fuel element is in a special rack designed to hold and measure the fuel 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 four pull-stations. There is a smoke detector in the control room, two in the reactor room, and one in the mechanical equipment room.

A number of fire extinguishers are readily available to reactor personnel. There are no fire hose stations in the facility.

Reed College Environmental Health & Safety contracts with an off-campus agency for fire extinguisher testing and maintenance on a regular basis.

The primary potential source of fires in the reactor bay, control room, mechanical equipment room, and ventilation loft is an electrical malfunction. 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, and cabinets. The doors separating the facility from the outside are fire resistant doors. The doors to the control room and the reactor bay are fire resistant. Inside the reactor facility, the door to the mechanical room is a fire resistant door. The fire loading (flammable materials) is very low in theses areas. The trash cans are metal. Solvents and flammables are kept in fireproof metal cabinets. No smoking is allowed in the entire facility. The walls and ceiling are structural concrete. The cable conduit from the control console to the reactor bay are steel pipe set in concrete.

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.

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 [redacted] uranium-235 in enrichments less than 20%. The license also allows possession of [redacted] americium-beryllium neutron source, and no more than [redacted] 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). It is inspected by a professional contractor every year. Its use is administratively controlled to prohibit movement of the crane over the reactor core when the reactor is operating, and to avoid movement of the crane over the pool, unless necessary for the activity. A procedure exists for the use of the crane and for certification of crane operators. It is normally stored in a wall mounted rack in the southeast corner of the reactor bay.

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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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 Reactor Review Committee (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. The experimental facilities are comparable in design, construction, utilization, and purpose to experimental facilities at other similar research reactors. The experimental facilities have been successfully and safely utilized during the period of the current facility license.

Accidents such as loss of coolant and reactivity insertion that experimental facilities could be subject to are discussed in Chapter 13. The design, construction, and utilization of the experimental facilities are such that these accidents are extremely unlikely. Chapter 11 discusses radiation hazards. Access to experimental facilities is controlled by the use of operating and radiation protection procedures. Use of appropriate radiation detection equipment, radiation protection practices (including the ALARA program), and established experiment review procedures provide reasonable assurance that doses from experimental facilities will meet the requirements of 10 CFR Part 20 for personnel and members of the general public.

The reactor safety systems are independent of the experimental facilities. The experimental facilities are maintained by in service inspections on a regular schedule. The reactivity worth of experiments and their materials are limited by TS 3.6.

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 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 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. 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)

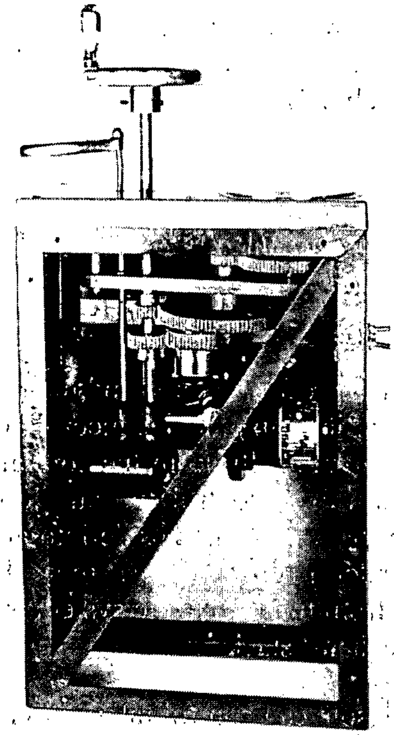

Figure 10.2 Rotary Specimen Rack Motor

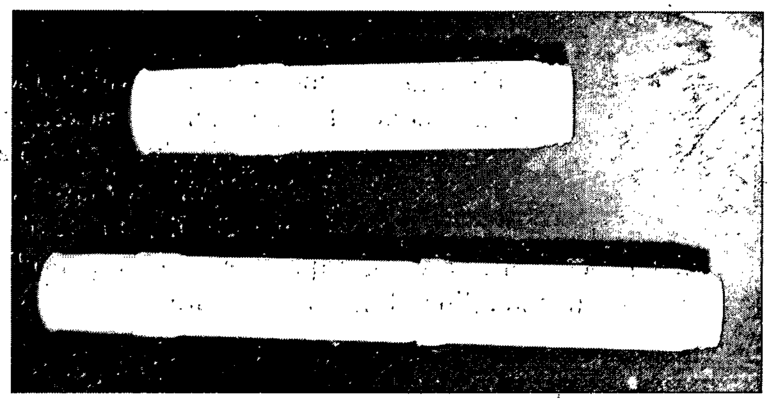

Figure 10.3 TRIGA® Irradiation Tubes

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 conveyed to a receiver/sender station via tubing at least 1.08 inches

(2.74 cm) ID. The in-tank and in-core portion of the pneumatic transfer system is illustrated in figure 10.6. This system, shown schematically in figure 10.4 and 10.5, 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.

The operator at the reactor console can control rabbit insertion by means of a permissive switch in the control room that will deenergize the four solenoids. This will prevent inserting any rabbits and will cause any rabbit in the core to move back to the receiving area.

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 10.6) 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 centerlines 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 10.7). 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 10.10) 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 an electric motor, which is maintained periodically. 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. The minimum filter efficiency is 99.97%. This filter is replaced when periodic visual inspection indicates a reduction in efficiency due to the buildup of impurities on the filter. Two rods with wing nuts hold the filter sandwiched between the chambers.

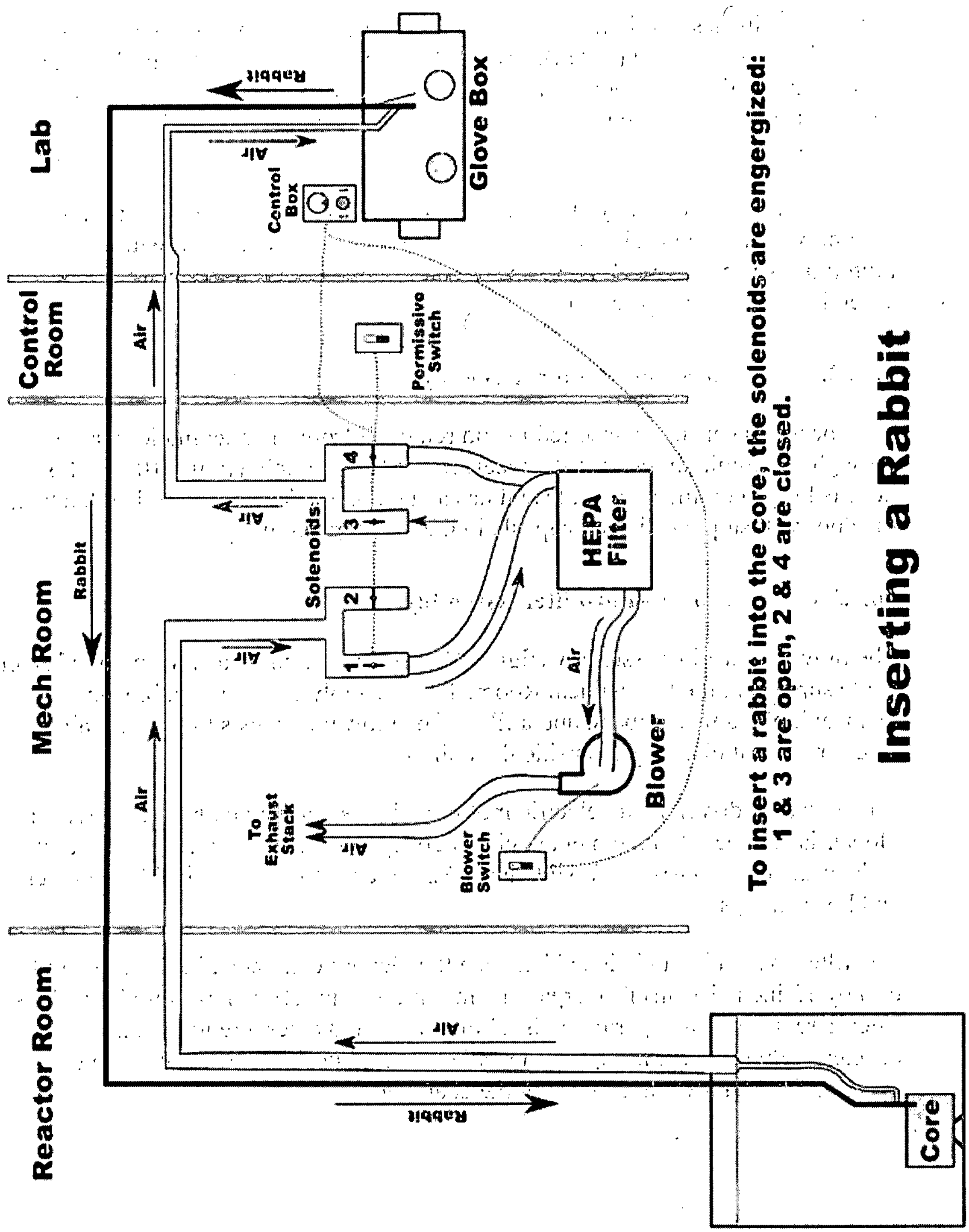

Figure 10.4 Diagram of the pneumatic transfer system inserting a sample

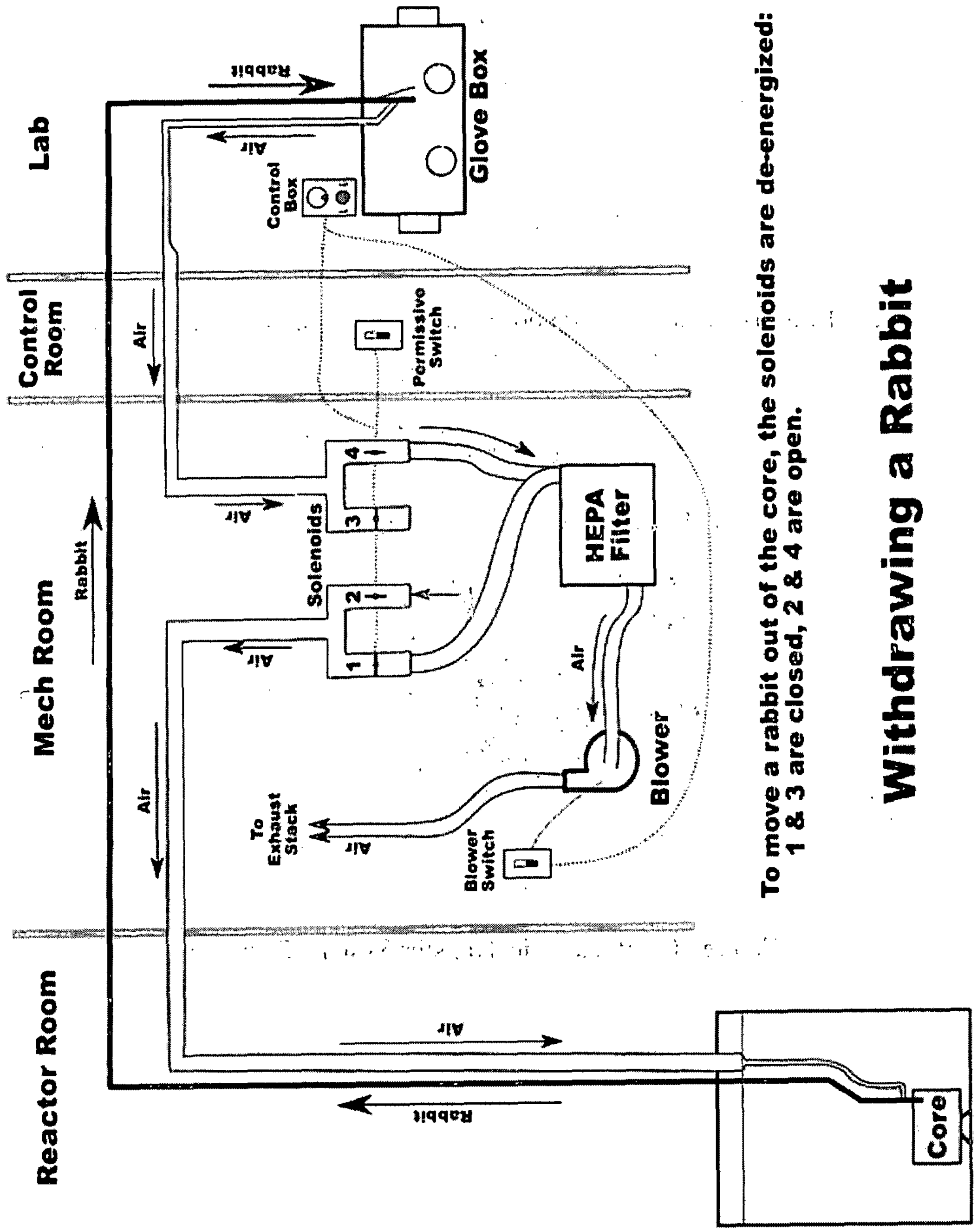

Figure 10.5 Diagram of the pneumatic transfer system removing a sample

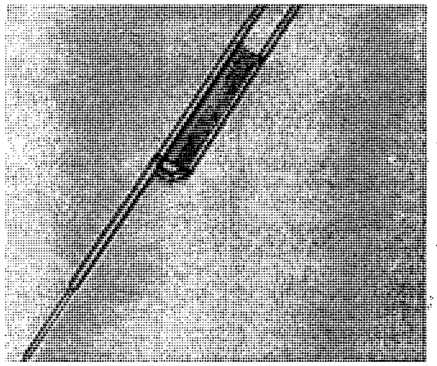

Figure 10.6 In-tank and in-core portions of the Pneumatic Transfer System

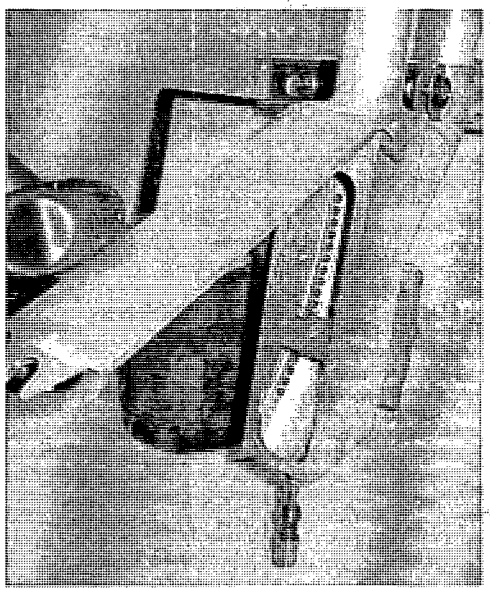

Figure 10.7 Pneumatic Transfer System Terminus

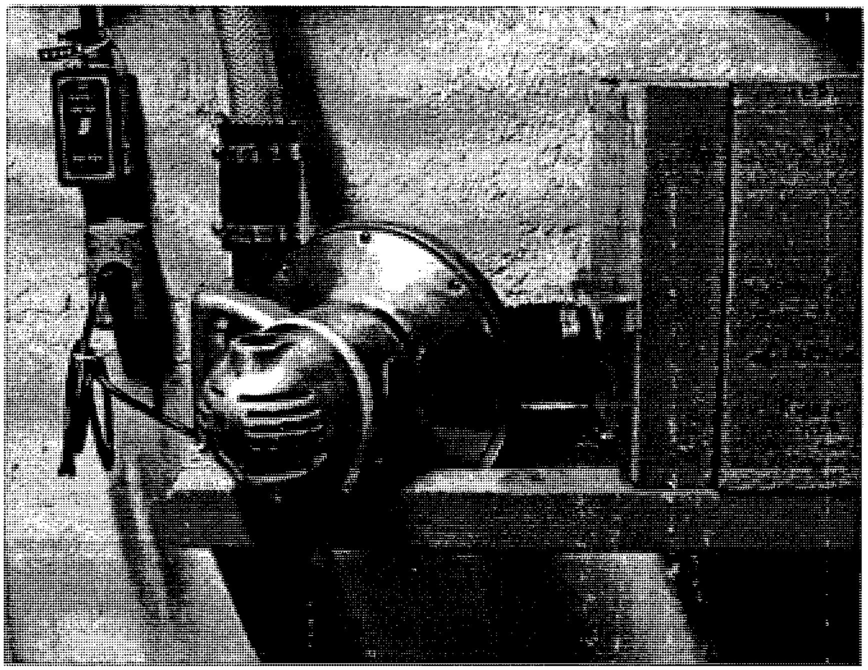

Figure 10.8 Pneumatic Transfer System Blower-and-Filter Assembly

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 [redacted] fuel element and an [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.

[redacted]

Figure 10.9 Source Holder / "Dummy Element"

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

The Reactor Review Committee (RRC) approves the types of experiments to be performed at the facility and the procedures for performing the experiement. If an experiment falls within the scope of the set of approved experiment, a request for operation is submitted to the Director. The Director verifies that operation is within the scope of approved 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 and referred to the Reactor Review Committee (RRC) for approval.

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 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:

- <u>Breach of fission product barriers</u> (which could occur through reactivity effects, thermal effects, mechanical forces, and/or chemical attack)
- <u>Interference with reactor control system functions</u> (which could occur through local flux perturbations or mechanical forces that can affect shielding or confinement)
- <u>Introduction or exacerbation of radiological hazards</u> (which could occur through irradiation of dispersible material, mechanical instability, inadequate shielding and/or inadequate controls for safe handling)
- <u>Interferences with other experiments or operations activities</u> (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)
- <u>Determination that the proposed activity is in compliance with Technical Specifications</u>

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.

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

For purposes of radiation dose calculations the actually measurements during operation are used. 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.

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 (630 L/s). The site boundary is 250 feet (76 m) from the center of the reactor.

During normal operations the primary airborne sources of radiation are argon-41 and nitrogen-16. Argon-41 results from irradiation of the air in experimental facilities and dissolved air in the reactor pool water. The primary means of argon-41 production is the rotating specimen rack. Other production sources include the pool water and the pneumatic irradiation system. Nitrogen-16 is produced when oxygen in the pool water is irradiated by the reactor core.

Nitrogen-16, argon-41, and radiation from the reactor core contribute to this dose rate. Nitrogen-16 has a very short half-life (about 7 seconds) and the reactor has a core diffuser system (see Chapter 5) that creates a water circulation pattern designed to suppress nitrogen-16 transport to the surface of the pool and reduce the reactor pool surface dose rate. The maximum measured dose rate at the top of the reactor tank at full power with the primary cooling system operating is 0.3 mrem/hr. The maximum measured dose rate at the top of the reactor tank at full power is with the primary cooling system off is 2.5 mrem/hr. Assuming that the 2.5 mrem/hr dose at the pool surface was from nitrogen-16 (a conservative assumption), the occupational airborne is within 10 CFR 20 limits. Because of the short half-life of nitrogen-16, exposure to the public is negligible.

The largest source of argon-41 production is the rotating specimen rack. The Gaseous Stack Monitor measures the noble gasses released from the facility through the exhaust stack. It is tracked in DAC-hours. The highest annual total release rate measured since the totalizing equipment was installed in 2002 was 760 DAC-hours when the facility was performing an unusually long operation for a project. Every other recorded year was well less than half that amount. Using the 10 CFR 20 Appendix B argon-41 DAC of 3E10-6 and dividing by the number of hours in a year (8760), the average concentration at he point of release during the year was 1.15E-7 μCi/ml. Using the dilution factor for travel to the site boundary of 880, the average concentration a the site boundary during the year was 3.0E-10 μCi/ml. Assuming that the 10 CFR 20 Appendix B argon-41 EC of 1.00E-8 is equivalent to 50 mrem/year, the dose to a member of the public during that year was 1.48 mrem, well within 10 CFR 20 limits.

This is below the 10 CFR Part 20, Appendix B, Table 2 limit of 50 mrem for inhalation.

TS 3.5.2 limits the concentration of argon-41 discharged into the environment as follows:

The annual average concentration of Ar-41 discharged into the unrestricted area shall not exceed 1.5E-6 μCi/ml at the point of discharge.

The concentration of 1.5E-6 μCi/ml in TS 3.5.2 bounds the concentration of the calculated maximum concentration of 1.15E-7 μCi/ml. The concentration in TS 3.5.2 represents a dose of about 8 mrem/yr under the assumptions discussed above. This dose meets the regulatory requirements for radiation dose limits for individual members of the public in 10 CFR 20.1301, "Dose Limits for Individual Members of the Public," and the constraint on air emissions of radioactive material in 10 CFR 20.1101(d).

Releases from accident conditions can occur if the fuel cladding is breached. Conservative calculations performed in Chapter 13 of the SAR demonstrate that, even if a fuel pin clad rupture were to occur in air and the radioactive material quickly released to the environment at ground level, the dose to population directly outside the RRR would be within 10 CFR Part 20 requirements

11.1.1.1.1 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 250 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

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

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

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

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

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

If a worker were exposed to the full course of the decay, cumulative concentration (μCi-h/cm³) in the reactor bay would be

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

Assuming an occupational exposure of 2000 hours per year, the value of 1.91E-4 μCi-h/cm³ above produces

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

This is well below the 2000 hour annual limit of 3E-6 μCi/cm³ specified in Appendix B of 10 CFR Part 20.

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 11.7 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 11.8. 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 11.7, 11.8, and 11.9. These sources are addressed in Section 11.2.

Table 11.8 Representative Special Nuclear Material Inventory

Source description	Radionuclide	Physical characteristics	Weight % U-235	App. mass (g) U-235	App. mass (g) Total U
[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 (ALARA) 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 Radiation Safety 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].

Table 11.9 Significant Byproduct Inventory

Source description	Radionuclide	Activity	Location
Reactor startup	[redacted]		
Instrument calibration			
Instrument calibration			
Instrument calibration			
Instrument calibration			

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. The RRC is described in detail in Chapter 21, Administration.

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.

The Director is responsible for implementation of the Radiation Protection Program and compliance with 10 CFR 20 at the reactor. The Environmental Health and Safety Officer is responsible for radiation safety for the campus. The Reactor Health Physicist oversees the implementation of the Radiation Protection Program at the reactor. The Reactor Health Physicist is independent of reactor operations management. The Reactor Director and the Reactor Health Physicist report to the Dean of the Faculty; they coordinate activities and program implementation the Environmental Health and Safety office. During an emergency the Emergency Coordinator is responsible for radiation safety as described in the Emergency Plan.

Procedures are in place for radiation protection during normal operations and reactor experiments. The procedures are reviewed by the RRC periodically. Records of the implementation of the radiation protection program are maintained for review and audit. A radiation work permit process addresses unusual activities and special experiments. Emergency procedures are in place for radiation protection during emergencies.

11.1.2.2 Training

Implementation of training for radiation protection is the responsibility of the Environmental Health and Safety office and the Reactor Director. 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 Surveying

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 Surveys

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

Continuously: Area monitors in reactor bay using energy compensated geiger counters. Air monitors in the reactor bay and the effluent path with proportional counters.

Daily prior to operation: Contamination surveys with a geiger counter or equivalent. Area dose rates with ion chambers, scintillation detectors, or energy compensated geiger counters.

Weekly: Solid waste is surveyed with a geiger counter or equivalent. Liquid waste is solidified prior to disposal.

Biweekly: Wipe test reactor bay, control room, and facility with a wipe test counter.

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 11.10. 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 11.10 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 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 11.11 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 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.

Samples that have been activated in the reactor are monitored for contamination and radioactivity upon removal from the reactor. Personnel protective clothing (gloves, lab coats, etc.) are worn while handling samples from the reactor. Samples are transferred from their irradiation vials into counting vials or are placed in plastic bags to limit the spread of contamination. All personnel are surveyed for contamination following handling irradiated samples.

Table 11.12 lists sample locations for routine monitoring of surface contamination control measures. On a biweekly basis, 100 cm^2 swipe tests with ethanol are analyzed in a wipe test counter 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.

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

Gamma-ray exposure-rate data, based on quarterly measurements over the most recent 7-year period is indicated in Table 11.13. Source terms are related to reactor power levels; therefore maximum radiation levels during operation at 250 kW should not exceed twice the maximum historical values in Table 11.13.

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.*

15) *Slade, D.H. (ed.), "Meteorology and Atomic Energy," Report TID-24190, U.S. Atomic Energy Commission, 1968).*

16) *Handbook of Health Physics and Radiological Health. Shleien, Slaback, Birky, editors. 3rd ed. Baltimore (MD): Williams and Wilkins, 1998.*

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
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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.

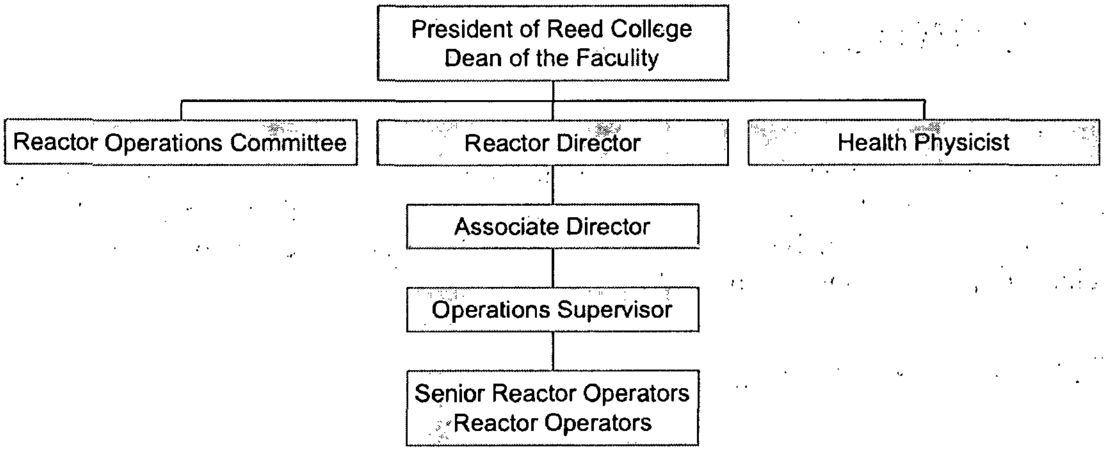

Figure 12.1 Organization and Management Structure for the Reed Reactor

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 15.4-1988, *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 have at least five voting members, at least two of which are knowledgeable in fields that relate to physics and nuclear safety. The Reactor Director and Associate Director shall be nonvoting members. The Dean of the Faculty, the Reactor Health Physicist, and the campus Radiation Safety Officer shall be voting members. The President shall appoint the ROC members except those who are members by virtue of their position described above.

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 approved 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 violations of Technical Specifications Surveillance requirements;
6) Review of abnormal performance of equipment and operating anomalies;
7) Review of unusual or abnormal occurrences and incidents which are reportable under 10 CFR Part 20 and 10 CFR Part 50; and
8) 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

Changes to the facility through replacement, modification, and changes are performed under the safety review process of 10 CFR 50.59 to ensure proper quality assurance. Procedures are in place for performing the reviews. The RRC oversees the process.

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 14

Technical Specifications

Reed Research Reactor Safety Analysis Report

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Technical Specifications
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Included in this document are the Technical Specifications (TS) and the "Bases" for the TS. These bases, which provide the technical support for the individual TS, are included for informational purposes only. They are not part of the TS and they do not constitute limitations or requirements to which the licensee must adhere.

1 DEFINITIONS

1.1 **Audit**: An examination of records, procedures, or other documents after implementation from which appropriate recommendations are made.

1.2 **Channel**: The combination of sensor, line, amplifier, and output devices that are connected for the purpose of measuring the value of a parameter.

1.3 **Channel Calibration**: An adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter that the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall include a Channel Test.

1.4 **Channel Check**: A qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

1.5 **Channel Test**: The introduction of a signal into the channel for verification that it is operable.

1.6 **Confinement**: An enclosure of the reactor bay that is designed to only allow the release of effluents between the enclosure and its external environment through controlled or defined pathways.

1.7 **Control Rod**: A device fabricated from neutron absorbing material that is used to establish neutron flux changes and to compensate for routine reactivity changes.

1.8 **Excess Reactivity**: That amount of reactivity that would exist if all control rods were moved to the maximum reactive condition from the point where the reactor is exactly critical (k_{eff}) at reference core conditions.

1.9 **Experiment**: Any operation, hardware, or target that is designed for non-routine investigation of reactor characteristics or that is intended for irradiation within an irradiation facility. Hardware rigidly secured to a core or tank structure so as to be a part of their design to carry out experiments is not normally considered an experiment. Specific experiments shall include:

a. **Fixed Experiment**: Any experiment or component of an experiment that is held in a consistent position relative to the reactor by mechanical means. This includes the rotating specimen rack, the central thimble, and fuel locations.
b. **Movable Experiment**: Any experiment that is not fixed. It is intended that the experiment may be moved in or near the core or into and out of the core while the reactor is operating. This included the pneumatic transfer system.

1.10 **Fuel Element**: A single TRIGA® fuel rod.

1.11 **Irradiation Facilities**: The central thimble, the rotating specimen rack, the pneumatic transfer system, sample holding dummy fuel elements, and any other in-pool irradiation facilities.

1.12 **Measured Value**: The value of a parameter as it appears on the output of a channel.

1.13 **Operable**: Capable of performing its intended function.

1.14 **Operating**: Performing its intended function.

1.15 **Reactor Facility**: The physical area defined by the Reactor Bay, the Mechanical Equipment Room, the Control Room, the Hallway, the Loft, the Classroom, the Radiochemistry Lab, the Counting Room, the Break Room, the Storeroom, the sump area, the stairway, and the Restroom.

1.16 **Reactor Operating**: The reactor is operating whenever it is not shutdown.

1.17 **Reactor Safety Systems**: Those systems, including their associated input channels, that are designed to initiate, automatically or manually, a reactor scram for the primary purpose of protecting the reactor.

1.18 **Reactor Secured**: The reactor is secured when:

a. Either there is insufficient fissile material in the reactor to attain criticality under optimum available conditions of moderation and reflection; or,

b. All of the following exist:

1. The three control rods are fully inserted.
2. The reactor is shutdown;
3. No experiments or irradiation facilities in the core are being moved or serviced that have, on movement or servicing, a reactivity worth exceeding one dollar;
4. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods.
5. The console key switch is in the "off" position and the key is removed from the console.

1.19 **Reactor Shutdown**: The reactor is shutdown when it is subcritical by at least one dollar both in the reference core condition and for all allowed ambient conditions, with the reactivity worth of all installed experiments and irradiation facilities included;

1.20 **Reference Core Condition**: The reactivity condition of the core when it is at ambient temperature and the reactivity worth of xenon is negligible (< $0.30). Fixed experiments can change the reference core conditions.

1.21 **Review**: An examination of records, procedures, or other documents prior to implementation from which appropriate recommendations are made.

1.22 **Safety Channel**: A measuring channel in a reactor safety system.

1.23 **Scram Time**: The elapsed time between reaching a limiting safety system set point and the instant that the slowest control rod reaches its fully-inserted position.

1.24 **Shall, Should, and May**: The word "shall" is used to denote a requirement; the word "should" is used to denote a recommendation; and the word "may" to denote permission, neither a requirement nor a recommendation.

1.25 **Shutdown Margin**: The amount of reactivity by which the reactor is subcritical, or would be subcritical if the control rods were inserted, except that the most reactive rod is in its most reactive position.

1.26 **Shutdown Margin Limit**: The minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems and will remain subcritical without further operator action, starting from any permissible operating condition with the most reactive rod in its most reactive position.

1.27 **Substantive Changes**: Changes in the original intent or safety significance of an action or event.

1.28 **Surveillance Intervals**: Allowable surveillance intervals shall not exceed the following. These intervals are to provide operational flexibility only and are not to be used to reduce frequency.

a. Biennial - interval not to exceed 130 weeks.
b. Annual - interval not to exceed 65 weeks.
c. Semi-annual - interval not to exceed 32 weeks.
d. Monthly - interval not to exceed 6 weeks.
e. Weekly - interval not to exceed 10 days.
f. Daily - prior to each day's operation or prior to each operation extending more than one day.

2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTING

2.1 Safety Limit-Reactor Power

Applicability. This specification applies to the reactor thermal power.

Objective. The objective is to define the maximum thermal power that can be permitted with confidence that no damage to the fuel element cladding shall result.

Specifications. The thermal power of the reactor shall not exceed **<to be completed>** kW.

Basis.

> **<This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted in November 2010.>**

2.2 Limiting Safety System Setting

Applicability, This specification applies to the scram settings that prevent the safety limit from being reached.

Objective. The objective is to prevent the safety limits from being reached.

Specifications'The limiting safety system setting shall be equal to or less than 275 kW as measured by a power measuring channel.

Basis.

> **<This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted in November 2010.>**

3 LIMITING CONDITIONS OF OPERATION

3.0 General

Limiting Conditions for Operation (LCO) are those administratively established constraints on equipment and operational characteristics that shall be adhered to during operation of the facility. The LCOs are the lowest functional capability or performance level required for safe operation of the facility.

3.1 Reactor Core Parameters

3.1.1 Operation

Applicability. This specification applies to the energy generated in the reactor during operation.

Objective. The objective is to assure that the thermal power safety limit shall not be exceeded during operation.

Specifications. The steady-state reactor power level shall not exceed 250 kW.

Basis.

> **<This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted in November 2010.>**

3.1.2 Shutdown Margin

Applicability. These specifications apply to the reactivity condition of the reactor and the reactivity worths of control rods and experiments during operation.

Objective. The objective is to assure that the reactor can be shut down at all times and to assure that the thermal power safety limit shall not be exceeded.

Specifications. The reactor shall not be operated unless the shutdown margin provided by control rods is greater than **<to be completed>** with:

a. Irradiation facilities and experiments in place and the total worth of all non-secured experiments in their most reactive state;
b. The most reactive control rod fully withdrawn;
c. The reactor in the reference core condition.

Basis. The value of the shutdown margin assures that the reactor can be shut down from any operating condition even if the most reactive control rod remains in the fully withdrawn position.

The shutdown margin calculation assumes a) irradiation facilities and experiments in place and the total worth of all non-secured experiments in their most reactive state, b) the most reactive control rod fully withdrawn and c) the reactor in the reference core condition. The only activity that could result in requiring fuel movement to meet shutdown margin and core excess limits would be the unusual activity of adding an experiment with large positive reactivity worth.

3.1.3 Core Excess Reactivity

Applicability. This specification applies to the reactivity condition of the reactor and the reactivity worths of control rods and experiments during operation.

Objective. The objective is to assure that the reactor can be shut down at all times and to assure that the thermal power safety limit shall not be exceeded.

Specifications. The maximum available excess reactivity based on the reference core condition shall not exceed **<to be completed>**.

Basis.

> **<This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted in November 2010.>**

3.1.4 Fuel Parameters

Applicability. This specification applies to all fuel elements.

Objective. The objective is to maintain integrity of the fuel element cladding.

Specifications. The reactor shall not operate with damaged fuel elements, except for the purpose of locating damaged fuel elements. A fuel element shall be considered damaged and must be removed from the core if:

a. A cladding defect exists as indicated by release of fission products; or
b. Visual inspection identifies bulges, gross pitting, or corrosion.

Basis. Gross failure or obvious visual deterioration of the fuel is sufficient to warrant declaration of the fuel as damaged.

3.2 Reactor Control And Safety Systems

3.2.1 Control Rods

Applicability. This specification applies to the function of the control rods.

Objective. The objective is to determine that the control rods are operable.

Specification. The reactor shall not be operated if any control rods are operable. Control rods shall not be considered operable if:

a. Damage is apparent to the rod or rod drive assemblies; or
b. The scram time exceeds 1 second.
c. The reactivity addition rate exceeds $0.16 per second.

Basis. This specification assures that the reactor shall be promptly shut down when a scram signal is initiated and that the reactivity addition rates are safe. Experience and analysis have indicated that for the range of transients anticipated for a TRIGA® reactor, the specified scram time is adequate to assure the safety of the reactor.

3.2.2 Reactor Power Measuring Channels

Applicability. This specification applies to the information that shall be available to the reactor operator during reactor operation.

Objective. The objective is to specify the minimum number of reactor power measuring channels that shall be available to the operator to assure safe operation of the reactor.

Specifications. The reactor shall not be operated unless at least two reactor power measuring channels are operable.

(1) Any single power measuring channel may be inoperable while the reactor is operating for the purpose of performing a channel check, test, or calibration.
(2) If any required reactor power measuring channel becomes inoperable while the reactor is operating for reasons other than that identified in Section 3.2.2(1) above, the channel shall be restored to operation within 5 minutes. If after 5 minutes the channel has not been returned to service the reactor shall be immediately shutdown.

Basis. Reactor power displayed at the control console gives continuous information on this parameter that has a specified safety limit. For note (1), taking these measuring channels off-line for short durations for the purpose of a check, test, or calibration is considered acceptable because in some cases, the reactor must be operating in order to perform the check, test, or calibration. Additionally there exist a power level indication operating while a single channel is off-line. For note (2), events that lead to these circumstances are self-revealing to the operator. Furthermore, recognition of appropriate action on the part of the operator as a result of an instrument failure would make this consistent with TS 6.7.2.

3.2.3 Reactor Safety Systems and Interlocks

Applicability. This specification applies to the reactor safety system channels and interlocks.

Objective. The objective is to specify the minimum number of reactor safety system channels and interlocks that shall be available to the operator to assure safe operation of the reactor.

Specifications. The reactor shall not be operated unless the minimum number of safety channels described in Table 1 and interlocks described in Table 2 are operable.

Table 1 - Minimum Reactor Safety Channels

Safety Channel	Function	Minimum Number
Power Level Scram	Scram at 275 kW or less	2
Console Manual Scram	Scram	1

Table 2 - Minimum Interlocks

Interlock	Function	Minimum Number
Low Power Level	Prevents control rod withdrawal with no neutron induced signal	1
Control Rod Drive Circuit	Prevents simultaneous manual withdrawal of two rods	1

(1) Any single safety channel or interlock may be inoperable while the reactor is operating for the purpose of performing a channel check, test, or calibration.

(2) If any required safety channel or interlock becomes inoperable while the reactor is operating for reasons other than that identified in Section 3.2.3(1) above, the channel shall be restored to operation within 5 minutes. If after 5 minutes the channel has not been returned to service the reactor shall be immediately shutdown.

Basis.

Power Level Scram: The set point for both the linear and percent power channels are normally set to 100% of 250 kW, which is the licensed power.

Manual Scram: The manual scram must be functional at all times the reactor is in operation. It has no specified value for a scram set point; it is manually initiated by the reactor operator.

Low Power Level Interlock: The rod withdrawal prohibit interlock prevents the operator from adding reactivity when there in no neutron induced signal on a low power channel. When this happens, the indication is insufficient to produce meaningful instrumentation response. If the operator were to insert reactivity under this condition, the period could quickly become very short and result in an inadvertent power excursion. A neutron source is added to the core to create sufficient instrument response that the operator can recognize and respond to changing conditions.

Control Rod Drive Circuit: The single rod withdrawal interlock prevents the operator from removing multiple control rods simultaneously so that reactivity insertions from control rod manipulation are done in a controlled manner.

<This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted in November 2010.>

For note (1), taking these safety channels off-line for short durations for the purpose of a check, test, or calibration is considered acceptable because in some cases, the reactor must be operating in order to perform the check test or calibration. Additionally there exist a power level indication operating while the single channel is off-line. For note (2), events that lead to these circumstances are self-revealing to the operator. Furthermore, recognition of appropriate action on the part of the operator as a result of an instrument failure would make this consistent with TS 6.7.2.

3.3 Reactor Primary Pool Water

Applicability. This specification applies to the primary water of the reactor pool.

Objective. The objective is to assure that there is an adequate amount of water in the reactor pool for fuel cooling and shielding purposes, that the bulk temperature of the reactor pool water remains sufficiently low to guarantee demineralizer resin integrity, and that pool conductivity remains low enough to limit corrosion.

Specifications. The reactor primary water shall exhibit the following parameters:

a. The pool water level shall be greater than 5 meters above the upper core plate;
b. The bulk pool water temperature shall be less than 50°C;
c. The conductivity of the pool water shall be less than 2.0 microSiemens/cm.

Basis. The minimum height of 5 meters of water above the upper core plate guarantees that there is sufficient water for effective cooling of the fuel and that the radiation levels at the top of the reactor are within acceptable levels. The bulk water temperature limit is necessary, according to the resin manufacturer, to ensure that the resin does not break down. Experience at many research reactor facilities has shown that maintaining the conductivity within the specified limit provides acceptable control of corrosion (NUREG-1537).

3.4 Ventilation System

Applicability. This specification applies to the operation of the reactor bay ventilation system.

Objective. The objective is to assure that the ventilation system shall be in operation to mitigate the consequences of possible releases of radioactive materials resulting from reactor operation or when moving irradiated fuel.

Specifications. The reactor shall not be operated nor irradiated fuel moved unless the facility ventilation system is operating in the normal mode or isolation mode, except for periods of time not to exceed two hours to permit repair, maintenance, or testing of the ventilation system.

Basis. During normal operation of the ventilation system, the annual average ground concentration of Ar-41 in unrestricted areas is well below the applicable effluent concentration limit in 10 CFR 20.

<This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted in November 2010.>

3.5 Radiation Monitoring Systems and Effluents

3.5.1 Radiation Monitoring Systems

Applicability. This specification applies to the radiation monitoring information that must be available to the reactor operator during reactor operation.

Objective. The objective is to specify the minimum radiation monitoring channels that shall be available to the operator to assure safe operation of the reactor.

Specifications. The reactor shall not be operated unless one Area Radiation Monitor and one Air Radiation Monitor are operating.

Exception: When a single required radiation monitoring channel becomes inoperable, operations may continue only if other instruments may be substituted for the normally installed monitor within one hour of discovery for periods not to exceed one month.

Basis. The radiation monitors provide information to operating personnel regarding routine releases of radioactivity and any impending or existing danger from radiation. Their operation will provide sufficient time to evacuate the facility or take the necessary steps to prevent the spread of radioactivity to the surroundings.

3.5.2 Effluents

Applicability. This specification applies to the release rate of Ar-41.

Objective. The objective is to ensure that the concentration of the Ar-41 in the unrestricted areas is below the applicable effluent concentration value in 10 CFR 20.

Specifications. The annual average concentration of Ar-41 discharged into the unrestricted area shall not exceed 1.5×10^{-6} μCi/ml at the point of discharge.

Basis. If Ar-41 is continuously discharged at 1.5×10^{-6} μCi/ml, measurements and calculations show that Ar-41 released to the unrestricted areas under the worst-case weather conditions would result in an annual TEDE of 8 mrem (SAR 11.1.1.1). This is less than the applicable limit of 10 mrem (Regulatory Guide 4.20).

3.6 Limitations on Experiments

3.6.1 Reactivity Limits

Applicability. This specification applies to experiments installed in the reactor and its irradiation facilities.

Objective. The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications. The reactor shall not be operated unless the following conditions governing experiments exist:

a. The absolute value of the reactivity worth of any single moveable experiment shall be less than **<to be completed>**;
b. The sum of the absolute values of the reactivity worths of all experiments shall be less than **<to be completed>**.

Basis.

<This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted in November 2010.>

3.6.2 Materials

Applicability. This specification applies to experiments installed in the reactor and its irradiation facilities.

Objective. The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications. The reactor shall not be operated unless the following conditions governing experiments exist:

a. Explosive materials, such as gunpowder or nitroglycerin, in quantities greater than 25 mg shall not be irradiated in the reactor or irradiation facilities. Explosive materials in quantities less than 25 mg may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than half of the design pressure of the container;
b. Experiments containing corrosive materials shall be doubly encapsulated. If the encapsulation of material that could damage the reactor fails, it shall be removed for the reactor and a physical inspection of potentially damaged components shall be performed.

Basis. This specification is intended to prevent damage to reactor components resulting from failure of an experiment involving explosive or corrosive materials. Operation of the reactor with the reactor fuel or structure potential damaged is prohibited to avoid potential release of fission products.

3.6.3 Failures and Malfunctions

Applicability. This specification applies to experiments installed in the reactor and its irradiation facilities.

Objective. The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications. Where the possibility exists that the failure of an experiment under nominal operating conditions of the experiment or reactor, credible accident conditions in the reactor, or possible accident conditions in the experiment could release radioactive gases or aerosols to the reactor bay or the unrestricted area, the quantity and type of material in the experiment shall be limited such that the airborne radioactivity in the reactor bay or the unrestricted area will not result in exceeding the applicable dose limits in 10 CFR 20, assuming that 100% of the gases or aerosols escape from the experiment;

Basis. This specification is intended to meet the purpose of 10 CFR 20 by reducing the likelihood that released airborne radioactivity to the reactor bay or unrestricted area surrounding the RRR will result in exceeding the total dose limits to an individual as specified in 10 CFR 20.

4 SURVEILLANCE REQUIREMENTS

4.0 General

Applicability. This specification applies to the surveillance requirements of any system related to reactor safety.

Objective. The objective is to verify the proper operation of any system related to reactor safety. Typically, a specific system from a Section 3 specification will establish the minimum performance level, and a companion Section 4 surveillance specification requirement will prescribe the frequency and scope of surveillance to demonstrate such performance.

Specifications.

a. Surveillance requirements may be deferred during reactor shutdown (except Sections 4.3.a, 4.3.b, and 4.g); however, they shall be completed prior to reactor operation unless reactor operation is required for performance of the surveillance. Such surveillance shall be performed as soon as practicable after reactor operation. Scheduled surveillance that cannot be performed with the reactor operating may be deferred until a planned reactor shutdown.
b. Any additions, modifications, or maintenance to the ventilation system, the core and its associated support structure, the pool, the pool coolant system, the rod drive mechanism radiation monitors, or the reactor safety systems shall be made and tested in accordance with the specifications to which the systems were originally designed and fabricated or to specifications reviewed by the Reactor Review Committee. A system shall not be considered operable until after it is successfully tested.

Basis. This specification relates to surveillances of reactor systems reactor systems that could directly affect the safety of the reactor, to ensure that they are operable. It also relates to surveillances of reactor systems that could affect changes in reactor systems that could directly affect the safety of the reactor. As long as changes or replacements to these systems continue to meet the original design specifications it can be assumed that they meet the presently accepted operating criteria.

4.1 Reactor Core Parameters

Applicability. This specification applies to the surveillance requirements for reactor core parameters.

Objective. The objective is to verify that the reactor does not exceed the authorized limits for power, shutdown margin, core excess reactivity, specifications for fuel element condition, and verification of the total reactivity worth of each control rod.

Specifications.

a. The shutdown margin shall be determined daily, or following any significant change (>$0.25) from a reference core.
b. The core excess reactivity shall be determined annually or following any significant change (>$0.25) from a reference core.

c. Twenty percent of the fuel elements comprising the core shall be inspected visually for damage or deterioration biennially such that the entire core is inspected over a ten year period.

Basis. Experience has shown that the identified frequencies will ensure performance and operability for each of these systems or components. The value of a significant change in reactivity (>$0.25) is measurable and will ensure adequate coverage of the shutdown margin after taking into account the accumulation of poisons. For inspection, looking at fuel elements biennially will identify any developing fuel integrity issues throughout the core. Furthermore, the method of determining non-conforming fuel at the RRR has been exclusively visual inspection.

4.2 Reactor Control and Safety Systems

Applicability. This specification applies to the surveillance requirements of reactor control and safety systems.

Objective. The objective is to verify performance and operability of those systems and components that are directly related to reactor safety.

Specifications.

a. A channel test of each item in Tables 1 and 2 in Section 3.2.3 shall be performed daily.
b. A channel calibration shall be made of the each reactor power level monitoring channel by the calorimetric method annually.
c. The scram time shall be measured annually.
d. The total reactivity worth and reactivity addition rate of each control rod shall be measured annually or following any significant change (>$0.25) from a reference core, not including experiments.
e. The control rods and drives shall be visually inspected for damage or deterioration biennially.

Basis. Experience has shown that the identified frequencies will ensure performance and operability for each of these systems or components.

4.3 Reactor Primary Pool Water

Applicability. This specification applies to the surveillance requirements for the reactor pool water.

Objective. The objective is to assure that the reactor pool water level, the water temperature, and the conductivity monitoring systems are operating, and to verify appropriate alarm settings.

Specifications.

a. A channel check of the water level monitor shall be performed daily.
b. A channel check of the water temperature monitor shall be performed daily.
c. A channel test of the water level monitor shall be performed monthly.
d. A channel test of the water temperature monitor shall be performed monthly.
e. A channel calibration of the water level monitor shall be performed annually.
f. A channel calibration of the water temperature monitor shall be performed annually.
g. The water conductivity shall be measured monthly.

Basis. Experience has shown that the frequencies of checks on systems that monitor reactor primary water level, temperature, and conductivity adequately keep the pool water at the proper level and maintain water quality at such a level to minimize corrosion and maintain safety.

4.4 Ventilation System

Applicability. This specification applies to the reactor bay confinement ventilation system.

Objective. The objective is to assure the proper operation of the ventilation system in controlling releases of radioactive material to the unrestricted area.

Specifications. A channel test of the reactor bay confinement ventilation system's ability to be in isolation shall be performed monthly.

Basis. Experience has demonstrated that tests of the ventilation system on the prescribed basis are sufficient to assure proper operation of the system and its control over releases of radioactive material.

4.5 Radiation Monitoring System

Applicability. This specification applies to the surveillance requirements for the area radiation monitoring equipment and the air monitoring systems.

Objective. The objective is to assure that the radiation monitoring equipment is operating properly.

Specifications. For each radiation monitoring system in Section 3.5.1:

a. A channel check shall be performed daily.
b. A channel test shall be performed monthly.
c. A channel calibration shall be performed annually.

Basis. Experience has shown that an annual calibration is adequate to correct for any variation in the system due to a change of operating characteristics over a long time span.

4.6 Experimental Limits

Applicability. This specification applies to the surveillance requirements for experiments installed in the reactor and its irradiation facilities.

Objective. The objective is to prevent the conduct of experiments that may damage the reactor or release excessive amounts of radioactive materials as a result of experiment failure.

Specifications.

a. The reactivity worth of an experiment shall be estimated or measured, as appropriate, before reactor operation with said experiment.
b. An experiment shall not be installed in the reactor or its irradiation facilities unless a safety analysis has been performed and reviewed for compliance with Section 3.6 by the Reactor Review Committee in accord with Section 6.5 and the procedures that are established for this purpose.

Basis. Experience has shown that experiments that are reviewed by the RRR staff and the Reactor Review Committee can be conducted without endangering the safety of the reactor or exceeding the limits in the TS.

5 DESIGN FEATURES

5.0 General

Major alterations to safety related components or equipment shall not be made prior to appropriate safety reviews.

5.1 Site and Facility Description

Applicability. This specification applies to the Reed College TRIGA® Reactor site location and specific facility design features.

Objective. The objective is to specify the location of specific facility design features.

Specifications.

a. The restricted area is that area inside reactor facility. The unrestricted area is that area outside the reactor facility.
b. The reactor bay shall have a free air volume of 300,000 liters.
c. The reactor bay shall be equipped with ventilation systems designed to exhaust air or other gases from the reactor bay and release them from a stack at a minimum of 3.6 meters from ground level.
d. Emergency controls for the ventilation systems shall be located in the reactor control room.
e. When in the isolation mode, the ventilation system shall exhaust through a HEPA filter and maintain a negative pressure in the reactor bay with respect to the control room.
f. The ventilation system shall enter the isolation mode upon a high activity alarm from an Air Radiation Monitor.
g. The radiation levels in the reactor bay shall be monitored by an Area Radiation Monitor.

Basis. The reactor facility and site description are strictly defined (SAR 2.0). Proper handling of airborne radioactive materials (in emergency situations) can be conducted from the reactor control room with a minimum of exposure to operating personnel (SAR 9.1).

5.2 Reactor Coolant System

Applicability. This specification applies to the pool containing the reactor and to the cooling of the core by the pool water.

Objective. The objective is to assure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding.

Specifications.

a. The reactor core shall be cooled by natural convective water flow.
b. The pool water inlet and outlet pipes shall be equipped with siphon breaks not less than 5 meters above the upper core plate.
c. A pool water level alarm shall be provided to indicate loss of coolant if the pool level drops 15 cm below normal level.
d. A bulk pool water temperature alarm shall be provided to indicate high bulk water temperature if the temperature exceeds 50°C.

Basis. This specification is based on thermal and hydraulic calculations that show that the TRIGA® core can operate in a safe manner at power levels up to 250 kW with natural convection flow of the coolant water.

In the event of accidental siphoning of pool water through inlet and outlet pipes the pool water level will drop to a level no less than 5 meters from the upper core plate either due to a siphon break or due to the pipe ending (SAR 5.2).

Loss-of-coolant alarm caused by a water level drop of no more than 15 cm provides a timely warning so that corrective action can be initiated. This alarm is located in the control room (SAR 5.2).

The bulk water temperature alarm provides warning so that corrective action can be initiated in a timely manner to protect the demineralizer resin. The alarm is located in the control room.

5.3 Reactor Core and Fuel

5.3.1 Reactor Core

Applicability. This specification applies to the configuration of fuel and in-core experiments.

Objective. The objective is to assure that provisions are made to restrict the arrangement of fuel elements and experiments so as to provide assurance that excessive power densities shall not be produced.

Specification.

a. The core assembly shall consist of TRIGA® fuel elements.
b. The fuel shall be arranged in a close-packed configuration except for single element positions occupied by in-core experiments, irradiation facilities, graphite dummies, control rods, startup sources, central thimble, or may be empty.
c. The reflector, excluding experiments and irradiation facilities, shall be water and graphite.

Basis.

Only TRIGA® fuel is anticipated to ever be used. In-core water-filled experiment positions have been demonstrated to be safe in the TRIGA® Mark I reactor. The largest values of flux peaking will be experienced in hydrogenous in-core irradiation positions. Various non-hydrogenous experiments positioned in element positions have been demonstrated to be safe in TRIGA® fuel element cores up to 500 kW operation. The core will be assembled in the reactor grid plate that is located in a pool of light water. Water in combination with graphite reflectors can be used for neutron economy and the enhancement of irradiation facility radiation requirements.

5.3.2 Control Rods

Applicability. This specification applies to the control rods used in the reactor core.

Objective. The objective is to assure that the control rods are of such a design as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications. The shim, safety, and regulating control rods shall have scram capability and contain boron compounds as a poison, in aluminum or stainless steel cladding.

Basis. The poison requirements for the control rods are satisfied by using neutron absorbing boron compounds. These materials must be contained in a suitable clad material such as aluminum or stainless steel to ensure mechanical stability during movement and to isolate the poison from the pool water environment. Scram capabilities are provided for rapid insertion of the control rods that is the primary safety feature of the reactor.

5.3.3 Reactor Fuel

Applicability. This specification applies to the fuel elements used in the reactor core.

Objective. The objective is to assure 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 physical and nuclear characteristics.

Specifications.

The individual unirradiated TRIGA® fuel elements shall have the following characteristics:

a. Uranium content: nominal 8.5 weight percent enriched to less than 20% in U-235;
b. Hydrogen-to-zirconium atom ratio (in the ZrHx): between. 0.9 and 1.65;
c. Cladding: stainless steel or aluminum, nominal 0.020 inches thick;
e. Identification: each element shall have a unique identification number.

Basis. Material analysis of 8.5/20 fuel shows that the maximum weight percent of uranium in any fuel element is less than 8.5 percent, and the maximum enrichment of any fuel element is less than 20.0 percent.

> **<This will be completed as part of the analysis being performed by Oregon State and General Atomics and will be submitted in November 2010.>**

5.4 Fuel Storage

Applicability. This specification applies to the storage of reactor fuel at times when it is not in the reactor core.

Objective. The objective is to assure that fuel that is being stored shall not become critical and shall not reach an unsafe temperature.

Specifications.

a. All fuel elements shall be stored in a geometrical array where the k_{eff} is less than 0.8 for all conditions of moderation.
b. Irradiated fuel elements shall be stored in an array that will permit natural convection cooling by water.

Basis. The limits imposed are conservative and assure safe storage (NUREG-1537).

6 ADMINISTRATIVE CONTROLS

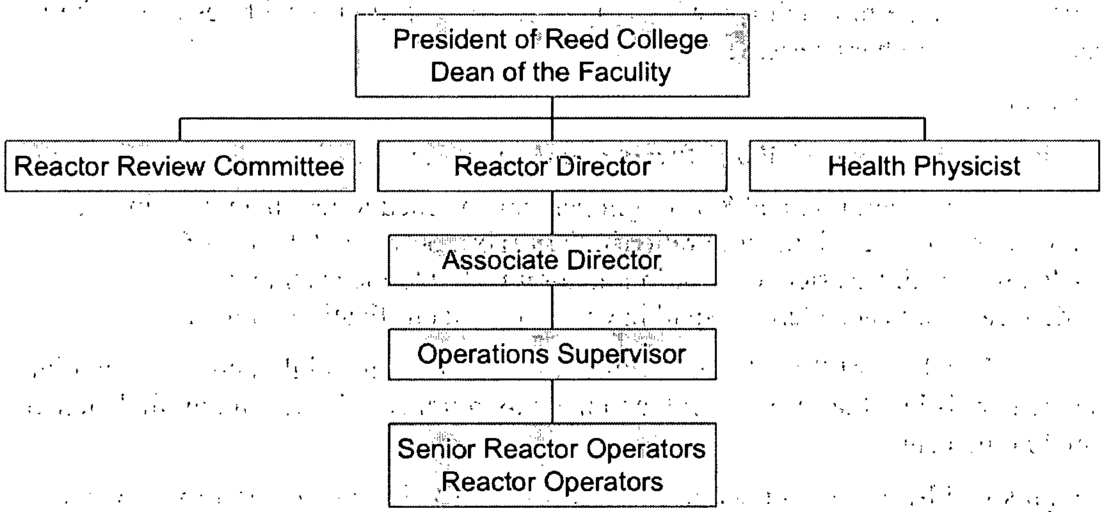
6.1 Organization

Individuals at the various management levels, in addition to being responsible for the policies and operation of the reactor facility, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license, TS, and federal regulations.

6.1.1 Structure

The reactor administration shall be as shown in Figure 1.

Figure 1 - Administrative Structure



6.1.2 Responsibility

The following specific organizational levels and responsibilities shall exist. Note that the Levels refer to ANSI/ANS-15.4-1988; R1999.

a. President (Level 1): The President of Reed College is responsible for the facility license and representing Reed College.
b. Director (Level 2): The Director reports to the President of Reed College via the Dean of the Faculty, and is accountable for ensuring that all regulatory requirements, including implementation, are in accordance with all requirements of the NRC and the Code of Federal Regulations.
c. Associate Director (Level 3): The Associate Director reports to the Director and is responsible for guidance, oversight, and technical support of reactor operations.
d. Health Physicist (Level 3): The Health Physicist reports to the President of Reed College via the Dean of the Faculty and is responsible for directing health physics activities including implementation of the radiation safety program.
e. Operations Supervisor (Level 3): The Operations Supervisor reports to the Associate Director and Director and is responsible for directing the activities of the reactor staff and for the day-to-day operation and maintenance of the reactor.
f. Reactor Operator and Senior Reactor Operator (Level 4): The Reactor Operators (RO) and Senior Reactor Operators (SRO) report to the Operations Supervisor, Associate

Director, and the Director, and are primarily involved in the manipulation of reactor controls, monitoring of instrumentation, and operation and maintenance of reactor related equipment.

g. During a vacancy in any position individuals may fill multiple positions if they meet the qualifications.

6.1.3 Staffing

a. The minimum staffing when the reactor is operating shall be:
   1. A reactor operator in the control room;
   2. A second person present in the reactor facility able to scram the reactor and summon help;
   3. If neither of these two individuals is an SRO, a designated SRO shall be readily available on call. Readily available on call means an individual who:
      a) Can be contacted quickly by the operator on duty;
      b) Is capable of getting to the reactor facility within 15 minutes.

b. Events requiring the presence of an SRO in the facility:
   1. Initial criticality of the day;
   2. All fuel or control rod relocations in the reactor core;
   2. Maintenance on any reactor safety system;
   3. Recovery from unplanned reactor scram;
   4. Relocation of any in-core experiment or irradiation facility with a reactivity worth greater than one dollar.

6.1.4 Selection and Training of Personnel

The selection, training, and requalification of personnel should be in accordance with ANSI/ANS 15.4-1988; R1999, "Standard for the Selection and Training of Personnel for Research Reactors."

6.2 Review And Audit

The Reactor Review Committee (RSC) shall have primary responsibility for review and audit of the safety aspects of reactor facility operations. Minutes, findings, or reports of the RSC shall be presented to the President and the Director within ninety days of completion.

6.2.1 RSC Composition and Qualifications

The RRC shall have at least five voting members, at least two of which are knowledgeable in fields that relate to physics and nuclear safety. The Reactor Director and Associate Director shall be nonvoting members. The Dean of the Faculty, the Reactor Health Physicist, and the campus Radiation Safety Officer shall be voting members. The President shall appoint the RSC members except those who are members by virtue of their position described above.

6.2.2 RSC Rules

The operations of the RSC shall be in accordance with written procedures including provisions for:

a. Meeting frequency;
b. Voting rules;
c. Quorums;
d. Method of submission and content of presentation to the committee;
e. Use of subcommittees;
f. Review, approval, and dissemination of minutes.

6.2.3 RSC Review Function

The responsibilities of the RSC, or designated subcommittee thereof, include, but are not limited to, the following:

a. Review changes made under 10 CFR 50.59;
b. Review new procedures and substantive changes to existing procedures;
c. Review proposed changes to the TS or license;
d. Review violations of TS, license, or violations of internal procedures or instructions having safety significance;
e. Review operating abnormalities having safety significance;
f. Review events from reports required in Section 6.7.2;
g. Review new experiments under Section 6.5;
h. Review audit reports.

6.2.4 RSC Audit Function

The RSC, or a subcommittee thereof, shall audit reactor operations at least annually. The annual audit shall include at least the following:

a. Facility operations for conformance to these TS and applicable license conditions;
b. The requalification program for the operating staff;
c. The results of action taken to correct deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operation that affect reactor safety;
d. The Emergency Plan and implementing procedures.

6.3 Radiation Safety

The Health Physicist shall be responsible for implementation of the radiation safety program. The requirements of the radiation safety program are established in 10 CFR 20. The program should use the guidelines of the ANSI/ANS 15.11 -1993; R2004, "Radiation Protection at Research Reactor Facilities."

6.4 Procedures

Written operating procedures shall be adequate to assure the safety of operation of the reactor, but shall not preclude the use of independent judgment and action if the situation requires. Operating procedures shall be in effect for the following:

a. Startup, operation, and shutdown of the reactor;
b. Fuel loading, unloading, and movement within the reactor;
c. Maintenance of major components of systems that could have an effect on reactor safety;
d. Surveillance checks, calibrations, and inspections required by the TS or those that have an effect on reactor safety;
e. Radiation protection;
f. Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity;
g. Shipping of radioactive materials;
h. Implementation of the Emergency Plan.

Substantive changes to the above procedures shall be made only after review by the RSC. Unsubstantive changes shall be reviewed prior to implementation by the Director or Associate Director.

Temporary deviations from the procedures may be made by the responsible SRO when the procedure contains errors or in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported by the next working day to the Director or Associate Director.

6.5 Experiments Review and Approval

The following apply to experiments:

a. Experiments shall be carried out in accordance with established and approved procedures;
b. All new experiments or class of experiments shall be reviewed by the RSC and approved in writing by the Director or Associate Director prior to initiation;
c. Substantive changes to previously approved experiments shall be made only after review by the RSC and approved in writing by the Director or Associate Director;
d. Minor changes that do not significantly alter the experiment may be approved by the Operations Supervisor, Associate Director, or Director.

6.6 Required Actions

6.6.1 Actions to Be Taken in Case of Safety Limit Violation

In the event a safety limit (reactor power) is exceeded:

a. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC;
b. An immediate notification of the occurrence shall be made to the Director, and the Chair of the RSC;

c. A report, and any applicable followup report, shall be prepared and reviewed by the RSC. The report shall describe the following:
   1. Applicable circumstances leading to the violation including, when known, the cause and contributing factors;
   2. Effects of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public;
   3. Corrective action to be taken to prevent recurrence.

6.6.2 Actions to Be Taken in the Event of an Occurrence of the Type Identified in Section 6.7.2 Other than a Safety Limit Violation

For all events that are required by regulations or TS to be reported to the NRC within 24 hours under Section 6.7.2, except a safety limit violation, the following actions shall be taken:

a. The reactor shall be secured and the Director or Associate Director notified;
b. Operations shall not resume unless authorized by the Director or Associate Director;
c. The RSC shall review the occurrence at or before their next scheduled meeting;
d. A report shall be submitted to the NRC in accordance with Section 6.7.2.

6.7 Reports

6.7.1 Annual Operating Report

An annual report shall be created and submitted by the Director to the NRC by November 1 of each year consisting of:

a. A brief summary of operating experience including the energy produced by the reactor and the hours the reactor was critical;
b. The number of unplanned shutdowns, including reasons therefore;
c. A tabulation of major preventative and corrective maintenance operations having safety significance;
d. A brief description, including a summary of the safety evaluations, of changes in the facility or in procedures and of tests and experiments carried out pursuant to 10 CFR 50.59;
e. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25 percent of the concentration allowed or recommended, a statement to this effect is sufficient;
f. A summarized result of environmental surveys performed outside the facility;
g. A summary of exposures received by facility personnel and visitors where such exposures are greater than 25 percent of that allowed.

6.7.2 Special Reports

In addition to the requirements of applicable regulations, and in no way substituting therefore, the Director shall report to the NRC as follows:

a. A report not later than the following working day by telephone and confirmed in writing by facsimile to the NRC Operations Center, to be followed by a written report that describes the circumstances of the event within 14 days to the NRC Document Control Desk of any of the following:

   1. Violation of the safety limit;
   2. Release of radioactivity from the site above allowed limits;
   3. Operation with actual safety system settings from required systems less conservative than the limiting safety system setting;
   4. Operation in violation of limiting conditions for operation unless prompt remedial action is taken as permitted in Section 3;
   5. A reactor safety system component malfunction that renders or could render the reactor safety system incapable of performing its intended safety function. If the malfunction or condition is caused by maintenance, then no report is required;
   6. An unanticipated or uncontrolled change in reactivity greater than one dollar. Reactor trips resulting from a known cause are excluded;
   7. Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks) where applicable; or
   8. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

b. A report within 30 days in writing to the NRC Document Control Desk of:

   1. Permanent changes in the facility organization involving Level 1-2 personnel;
   2. Significant changes in the transient or accident analyses as described in the Safety Analysis Report.

6.8 Records

6.8.1 Records to be Retained for a Period of at Least Five Years or for the Life of the Component Involved if Less than Five Years

a. Normal reactor operation;
b. Principal maintenance activities;
c. Reportable occurrences;
d. Surveillance activities required by the TS;
e. Reactor facility radiation and contamination surveys;
f. Experiments performed with the reactor;
g. Fuel inventories, receipts, and shipments;
h. Approved changes to the operating procedures;
i. Reactor Review Committee meetings and audit reports.

6.8.2 Records to be Retained for at Least One Requalification Cycle

Records of retraining and requalification of licensed operators shall be retained until the operator's license is terminated.

6.8.3 Records to be Retained for the Lifetime of the Reactor Facility

a. Gaseous and liquid radioactive effluents released to the environs;
b. Offsite environmental monitoring surveys;
c. Radiation exposures for all personnel monitored;
d. Drawings of the reactor facility;
e. Reviews and reports pertaining to a violation of the safety limit, the limiting safety system setting, or a limiting condition of operation.

Chapter 16

Other License Conditions

Reed Research Reactor Safety Analysis Report

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16 OTHER LICENSE CONSIDERATIONS

16.1 Prior Use of Reactor Components

16.1.1 General

All safety related equipment is evaluated prior to use at the Reed Research Reactor (RRR) as required by 10 CFR 50.59. This includes equipment, components, and fuel received from other research reactors.

All safety related equipment is inspected periodically as required to ensure its continued safety, suitability, and effectiveness, including component aging or corrosion.

16.1.2 Fuel Elements

The fuel used in the RRR is the same TRIGA® fuel installed by General Atomics in 1968. Over the course of 40 years of operation, only approximately [redacted] have been consumed. [redacted] have had pinhole leaks in their cladding and have were removed from service. [redacted] was dropped during an inspection and its bottom pin was bent, so it was removed from service.

[redacted] unused stainless steel clad elements were received from the Berkeley TRIGA® reactor when it shutdown. They were never used before installation at RRR and were evaluate prior to installation in the reactor core.

16.1.3 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.

16.2 Medical Uses

The RRR is not used for medical purposes, other than possible research with non-human subjects.