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

RESPONSES TO THE
REQUEST FOR ADDITIONAL INFORMATION
DATED MAY 20, 2011

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUR OR DENOTED BY BRACKETS

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May 20, 2011

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, dated March 8, 2010.

Attached is our response to the subject Request for Additional Information.

Please feel free to contact us if you have any questions. Thank you.

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

Executed on 5/20/11

Stephen G. Frantz
Director, Reed Research Reactor.

Attachments:
- Attachment 1, Reed Research Reactor RAI Response
- Attachment A, Analysis of the Neutronic Behavior of the Reed Research Reactor
- Attachment B, Analysis of the Thermal Hydraulic Behavior of the Reed Research Reactor
- Attachment C, Foushee Letter dated March 1, 1966, Storage of TRIGA Fuel Elements
- SAR Chapter 14, Technical Specifications

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NRR

Attachment 1 to Reed Research Reactor RAI Response – May 2011

This document has three attachments to it.

- Attachment A, *Analysis of the Neutronic Behavior of the Reed Research Reactor*
- Attachment B, *Analysis of the Thermal Hydraulic Behavior of the Reed Research Reactor*
- Attachment C, Foushee Letter dated March 1, 1966, *Storage of TRIGA Fuel Elements*

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.

   The electrical system is shared with the rest of Reed College. The HVAC system for the reactor is separate from the rest of the campus ventilation system.

   The loss of electrical power to the facility results in the deenergization of all the systems at the reactor. There is no backup electrical supply system. Although much of the instrumentation has UPS backup supplies, they are not taken credit for in the analysis. The reactor will shutdown due to the control rod magnets deengergizing and the control rods falling into the core. The HVAC system, instrumentation, and alarms will all denenergize. The HVAC system fans will turn off and the dampers will fail as is. Thus the ventilation system will not go into isolation if the facility looses power. The accident analysis analyzes this condition as a leakage scenario.

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.

The facility and its components were constructed to comply with the building codes of the City of Portland, Multnomah County, and the State of Oregon in 1968. All modifications have been made in accordance with the applicable codes.

The facility was installed in accordance with designs provided by General Atomics and the architectural designs by Farnham and Peck, registered architects in the State of Oregon.

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.*

    During a flood, it is presumed that electrical contacts would be shorted out such that power would not be supplied to any of the RRR safety systems. However, because electro-magnates hold the control rods in position and without power the electro-magnates would not function resulting in the control rods inserting in the core, loss of electrical power is not an issue. It is also assumed that the RRR would not be operated if it were flooded.

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.

The RRR uses fuel discussed in NUREG-1282, *Safety Evaluation Report on High-Uranium Content Low-enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors*. In Table 1 of that document the RRR fuel is the first type, "Original." A detailed study of these fuels can be found in a paper by M. T. Simnad, "The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel," Nuclear Engineering and Design, Vol. 64, pp 403-422.

The difference in the length of the fuel meat will not have a safety significance other than a lower value for reactivity. The safety significance of the fuel is the inherent large negative temperature coefficient of reactivity.

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.

Temperature distribution within a fuel element during a rod movement is dependent on both time and spatial position within the element. This question can, however, be answered using a point kinetics approach with some conservative assumptions. A step insertion is discussed in RAI #55 and this discussion builds on that discussion and should be read first.

For a ramp insertion, the event will likely not be truly adiabatic, but most of the power production and energy release will occur very rapidly towards the end of the transient, so the adiabatic assumption is still reasonable as well as conservative.

The treatment and solution of the point reactor kinetics equations involved use of a code developed by OSU which was benchmarked against GA codes and the RELAP point kinetics module. This treatment is summarized in the to-be-published paper entitled <u>A Comparison of Pulsing Characteristics of the Oregon State University TRIGA® Reactor with FLIP and LEU Fuel</u> (Marcum, et al). The relevant section is included below. OSU specific parameters were of course changed to Reed parameters in order to analyze the behavior of the RRR under transient conditions.

PRKE MODEL

The Oregon State point reactor kinetics model was developed using a similar methodology to that seen above in the RELAP5-3D point reactor kinetics model. It is a simple single core average calculation identifying temporal average core reactivity, power, and fuel temperature. The primary difference in the Oregon State point reactor kinetics model when compared to the RELAP5-3D model is that it has no secondary calculation which isolates the hot rod characteristics. When inserting a given reactivity into the Oregon State model, it is distributed through all 88 fuel elements, similarly to that in RELAP5-3D. The derivation for the Oregon State point reactor kinetics model is presented in the following section.

We begin with the point reactor kinetics equations

$$\frac{\partial P(t)}{\partial t} = \left(\frac{\rho(t)-\beta}{\Lambda(t)}\right)P(t) + \sum_{i=1}^{6}\lambda_i C_i(t), \tag{1}$$

$$\frac{\partial C_i}{\partial t} = -\lambda_i C_i(t) + \frac{\beta_i}{\Lambda}P(t), \qquad i = 1...6. \tag{2}$$

Reactivity is related to the prompt fuel temperature coefficient of reactivity (α_F) by

$$\int_{\rho_o}^{\rho} d\rho = \int_{T_o}^{T} \alpha_F(T')dT' \tag{3}$$

The fuel temperature coefficient is a temperature dependence property of the fuel. Because of this extremely short time scale in which the majority of energy during a pulse is deposited in the fuel, it is assumed that the core fuel is adiabatic during this process. The temperature of the fuel is related to the power of the core through an energy balance equation:

$$C_p^* \frac{dT(t)}{dt} = P(t) \tag{4}$$

where C_p^* is representative of the specific heat for a prescribed mass or $C_p m$, most commonly referred to as the volumetric heat capacity , where m [kg] is the mass of an object and C_p is the specific heat [J/kg-K] of that object.

We employ an "integration factor" approach to discretize this stiff system of ordinary differential equations. In this approach, the mean neutron lifetime is taken as

$$\Lambda(t)=\frac{\ell}{k(t)}, \tag{5}$$

and the multiplication factor (k) is of the form

$$k(t)=\left(1-\rho(t)\right)^{-1} \tag{6}$$

Linearization

A reformulation of equation (1) may be performed given the relationships presented (5) and (6);

$$\frac{\partial P(t)}{\partial t}=\xi P(t)+\psi \tag{7}$$

where

$$\xi=\frac{\rho(t)-\beta}{\ell\left(1-\rho(t)\right)} \tag{8}$$

and

$$\psi=\sum_{i=1}^{6}\lambda_i C_i(t) \tag{9}$$

Multiplying through (7) with the integration factor yields

$$\frac{\partial P(t)}{\partial t}e^{-\xi t}=\xi P(t)e^{-\xi t}+\psi e^{-\xi t}, \tag{10}$$

or

$$\frac{\partial}{\partial t}\left(P(t)e^{-\xi t}\right)=\psi e^{-\xi t} \tag{11}$$

Carrying out the integration of (11) from time t_n to time t_{n+1},

$$\int_{t_n}^{t_{n+1}}\frac{\partial}{\partial t}\left(P(t)e^{-\xi t}\right)\partial t=\int_{t_n}^{t_{n+1}}\psi e^{-\xi t}\partial t \tag{12}$$

produces the following linearized equation for power

$$P_{n+1}=P_n e^{-\xi\Delta t}+\frac{\psi_n}{\xi}\left(e^{-\xi t_{n+1}}-1\right) \tag{13}$$

where Δt represents the time difference between $n+1$ and n. Equation (13) is the first of nine linearized equation which couple to produce the temporal solution. Considering (2), after multiplying through by its appropriate multiplication factor;

$$\frac{\partial C_i}{\partial t}e^{\lambda_i t} = -\lambda_i C_i(t)e^{\lambda_i t} + \frac{\beta_i}{\ell(1-\rho(t))}P(t)e^{\lambda_i t} \quad , \quad (14)$$

or

$$\frac{\partial}{\partial t}\left(C_i e^{\lambda_i t}\right) = \frac{\beta_i e^{\lambda_i t}}{\ell(1-\rho(t))}P(t) \quad (15)$$

Carrying out the integration of (15) from time t_n to time t_{n+1} yields

$$C_{i,n+1} = C_{i,n}e^{-\lambda_i \Delta t} + \frac{\beta_i}{\ell(1-\rho)_n}\int_{t_n}^{t_{n+1}} P(t')e^{-\lambda_i t'}\partial t' \quad . \quad (16)$$

The trapezoidal scheme was employed in order to solve for $P(t')$, where

$$P(t') = \frac{1}{2}(P_{n+1} + P_n) \quad (17)$$

Utilizing (17) in (16) yields

$$C_{i,n+1} = C_{i,n}e^{-\lambda_i \Delta t} + \frac{\beta_i(P_{n+1}+P_n)(1-e^{-\lambda_i \Delta t})}{2\lambda_i \ell(1-\rho)_n} \quad . \quad (18)$$

Equation (18) represents six of the nine equations that couple together to produce the temporal solution, where i represents one of six delayed neutron groups.

The reactivity feed is incorporated into the system of equations; this is done by integrating (3) from time t_n to time t_{n+1}, producing

$$\rho_{n+1} = \rho_n + \alpha_F\left(T_{n+1}^2 - T_{n+1}\cdot T_n\right) \quad (19)$$

Lastly, the temperature change in the fuel is accounted for through the linearized integration of (4). However the volumetric heat capacity of TRIGA® fuel is characteristically a function of temperature taking on the form

$$C_p^*(T) = C_{p0}^* + C_{p1}^* T \quad (20)$$

Integrating (20) over temperature yields

$$C_p^*(T) = C_{p0}^* T + \frac{C_{p1}^*}{2}T^2 \quad . \quad (21)$$

Performing the integration of (21) from time t_n to time t_{n+1}, produces

$$0 = \left(C_{p0}^* T + \frac{C_{p1}^*}{2}T^2\right)_{n+1} - \left(C_{p0}^* T + \frac{C_{p1}^*}{2}T^2\right)_n - \frac{\Delta t(P_{n+1}+P_n)}{2} \quad . \quad (22)$$

Solving for temperature at t_{n+1} generates

$$T_{n+1} = \frac{-C^{\bullet}_{p0,n+1} + \left(\left(C^{\bullet}_{p0,n+1}\right)^2 + 2C^{\bullet}_{p1,n+1} \cdot \left(\left(C^{\bullet}_{p0}T + \frac{C^{\bullet}_{p1}}{2}T^2 \right)_n + \frac{\Delta t\left(P_{n+1} + P_n\right)}{2} \right) \right)^{\frac{1}{2}}}{C^{\bullet}_{p1,n+1}} \quad (23)$$

Equation (23) is the last of the nine coupled equations to be linearized.

An explicit method was chosen while marching through time for the PRKE model solving, in order, (13), (18), (19), and (23).

Section 6 of Attachment A presents control rod worth and shutdown margin. Section 7 of Attachment B shows that the steady state MDNBR at 250 kW is 6.33 in the hottest channel. Analysis based on point reactor kinetics indicates that the maximum power level following a continuous rod withdrawal accident could rise as high as 1060 KW (26.8 KW in the hot rod), but the associated average temperature rise of the fuel is no more than 13.4 degrees due to the rapid nature of the event. This analysis is based on the following conservative assumptions using data from 2010 rod calibrations: 1) the maximum worth rod (Safety, \$3.37) is continuously withdrawn, 2) the withdrawal rate is that of the fastest moving rod (regulating, 36.1 sec withdrawal time), 3) the reactor scrams at a high power setpoint of 285 KW, 4) rod in motion commences 0.5 sec following the initiation of the scram signal, 5) only the two lowest worth rods are inserted during the scram (Reg + Shim = \$4.61) 6) rod insertion time due to the scram is the maximum allowable tech spec limit of 1.0 sec., and 7) reactivity addition due to rod runout terminates at the same time that inmotion of the other two rods commences, i.e., the withdrawal of the runout rod ceases when the SCRAM signal initiates insertion of the other two rods.

If the event is assumed to be sufficiently fast that heat transfer conditions are essentially adiabatic, then the ratio of peak temperature increase to average temperature increase will be the same as the steady state ratio of peak to average power. This ratio is shown in the thermal/hydraulic analysis to have a value of 2.952. If average fuel temperature is initially at the maximum pool temperature of 50°C, then the peak fuel temperature experienced during the rod runout accident will be no higher than 50+13.4*2.952 = 89.5°C. Since power peaks about 10 sec after accident initiation, conditions will likely not be truly adiabatic. Heat transfer from the hot element will result in a maximum peak temperature lower than 89.5°C. Given the rapid nature of the transient and the low temperature increases, DNB is not a concern during the most severe potential rod withdrawal accident scenario.

Typical Control Rod Worths were recently measured as:

Safety Rod	\$3.37
Shim Rod	\$3.27
Regulating Rod	<u>\$1.34</u>
	\$7.98

The Core Excess when critical at 5 W is typically:

Safety Rod	\$0.65
Shim Rod	\$0.65
Regulating Rod	<u>\$0.35</u>
	\$1.65

With a typical core excess and the most reactive control rod stuck out, the reactor will be subcritical by \$7.98 - \$1.65 - \$3.37 = \$2.96. With the maximum allowable core excess the reactor would be shutdown by \$7.98 - \$3.00 - \$3.37 = \$1.61 which is still greater than the \$0.50 minimum.

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.*

The water in the reactor pool is a moderator and reflector in addition to being a coolant. The naturally circulating water surrounds the core structure and flows up through the core area and fills the space between the fuel elements.

Sections 8 and 9 of Attachment A address the moderator void and temperature coefficients, respectively. A loss of moderator would preclude operating the reactor because of the large negative void coefficient.

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*

The neutron source is 1.64 Ci AmBe and was installed in 1968. Am-241 has a half-life of 432 days and emits an alpha particle. The alpha particle hits the Be-9 nucleus and produces neutrons through the (α,n) mechanism shown below:

$\alpha + {}^9Be \dashrightarrow a + {}^4He + {}^4He + n$

The energy of the neutrons is 1.5-11.5 MeV with an average neutron energy of 4.4 MeV. The source is clad with stainless steel and is designed to withstand the chemical, thermal, and radiation environment in a TRIGA reactor. The source provides a sufficient neutron count rate to satisfy the source-rod interlock on the Logarithmic Channel Nuclear Instrument.

The neutron source will be replaced if it indicates any leakage or if it does not supply sufficient neutrons to satisfy the source-rod interlock requirements.

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.*

    The reactor is standard TRIGA Mark I reactor that has been analyzed many times. There is nothing unique about the RRR reactor. The initial loading in 1968 showed that the core structure can support the weight of all its components in the empty condition. 40 years of operational experience has shown that the core structure can support the weight of all its components in flooded conditions.

    The structure is sound enough to last for the duration of the fuel. Routine in-service inspection and restrictions on the pool water quality minimize the corrosion of the components.

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.*

    There are no tank penetrations below the normal water level. There is no installed leak detection system other than monitoring pool level and tracking the amount of water added to compensate for evaporation. The pool level monitor is a capacitive level detector which displays in the control room. It alarms if level decreases by more than 10 cm below normal, and the alarm is visible outside the facility by periodic security patrols. Pool level and makeup water are checked weekly and the amount of water added is trended over time. Makeup averages less than 130 liters per week, which is consistent with the evaporation rate of our pool which has a surface area of 11.6 m^2 and an average temperature of 20°C. The only measureable changes in the makeup rate have been during long runs at high temperature, when evaporation increases. It is estimated that we would be able to detect a doubling in the makeup rate, which would equate to a leak of approximately 130 liters per week, or ≈ 0.77 l/h. The manual make up water system can supply well over 1000 l/h. If water level decreases too quickly or make up water use increases significantly, the tank will be examined for leaks. The measured pool activity after an extended run is approximately 1E-06 μCi/ml, primarily Na-24. The 10 CFR 20 Effluent Concentration Limit for Na-24 is 5E-05 μCi/ml, so any leakage would be within that limit. At the above measurable leak rate, that would be at most 7.7E-04 μCi/h. The leakage would go into the ground around the reactor pool. The nearest body of water is the Crystal Springs Creek, located at the bottom of the ravine approximately 100 meters the north of the reactor.

Forty years of experience has shown that the materials are compatible with the water and the environment. Corrosion control is maintained by limits on primary water conductivity and pH. The conductivity is measured weekly and the limit is set at 5 μSiemens/cm which is in agreement with NUREG 1537 Appendix 14 ¶3.3(9), Water Chemistry. The pH will be measured quarterly to ensure it is between 5 and 7.5, which is in agreement with NUREG 1537.

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.

The biological shield consists of the reactor pool water. The reactor is at the bottom of a tank that is 25 feet (7.6 m) deep. Normally at least 20 feet of water cover the core, providing the biological shield. [redacted] There are no accessible rooms or facilities below the floor level of the reactor. The biological shield is sufficient to keep doses within the requirements of 10 CFR 20.

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 +$3.00 of 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 1 inch 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.

Section 6 of Attachment A specifically addresses core excess, control rod worth, and shutdown margin. The smallest calculated shutdown margin of the current core configuration is $2.54 ± 0.04 with a calculated core excess of $1.65 ± 0.11. The shutdown margin is clearly met. The shutdown margin would still be about $1.00 with a core excess of $3.00. The $3.00 core excess limit more than adequate protects against low shutdown margin which is precisely its intent.

To evaluate the neutronic difference between the aluminum and the stainless steel clad fuel, an analysis was performed consistent with the methodology described in Attachment A where the highest power fuel element (i.e., grid position B-5) was replaced with an aluminum clad fuel element. The power generated in that fuel element went from 7.24 ± 0.014 kW to 7.25 ± 0.014 kW. While there appears to be an effect, this represents less than a percent difference and would likely be difficult to observe in a core excess measurement.

The core excess limit allows operation without the need to add or remove fuel elements. If operating with typical critical control rod worths of $0.65 (Safety), $0.65 (Shim), and $0.30 (Regulating), the calculated core excess is $1.65. Activities such as moving away from the reference state or adding negative worth experiments will make core excess more negative and shutdown margin less positive. The only activity which 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.

The reactivity limit of $1.00 for movable experiments is designed to prevent an inadvertent prompt critical condition from occurring and maintain a value below the shutdown margin. Movable experiments are by their very nature experiments in a position where it is possible for a sample to be inserted or removed from the core while critical. The reactivity worth limit for all experiments is designed to prevent an inadvertent prompt critical condition. This limit applies to movable, unsecured, and secured experiments. A maximum reactivity insertion of $2.00 for all experiments in agreement with Reg Guide 1537 Appendix 14 Section 3.8.1.

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.

A question arose whether the calculated value for the void coefficient of reactivity is too large, especially when compared with calculated values from other similar facilities. Section 8 of Attachment A states that the average void coefficient over the range from 0% void to 100% void is -$0.83 per percent void. This is very similar to the -$0.86 value calculated for the OSTR and compares reasonably well with the measured OSTR value of -$0.51 per percent void. As discussed below, it is highly improbable that the RRR could be operated with more than a few percent moderator void present in the core. The reactivity coefficient for the RRR within such an operating envelope is on the order of -$0.12 per percent void.

Figure 1 Core k-effective vs. % moderator void

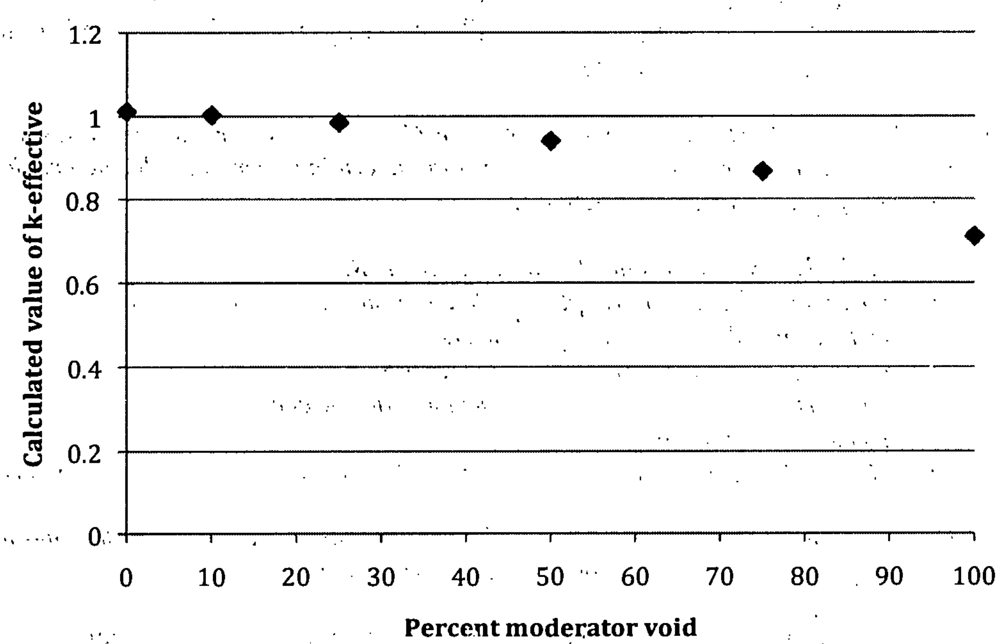

Figure 1 shows k_{eff} vs. percent of moderator void. Removal of 100% of water from the core results in the introduction of $54 negative reactivity. The graph indicates that removing any amount of water from the core will always result in the introduction of negative reactivity. The slope of the curve in terms of Δk per percent or $ per percent increases with increasing percent void.

It is understood that in order to calculate a void coefficient one varies density while holding temperature constant. To calculate the void coefficient, we first calculated k_{eff} of the reactor with the in-core water moderator at nominal temperature and normal corresponding density. We then reduced in-core water density to 90% of the original value and re-calculated k_{eff} at the same temperature. This process was repeated with in-core water densities of 75%, 50%, 25% and 0% of the original densities. The resulting eigenvalues for each core state are tabulated below and plotted in Figure 1.

Percent of original H_20 Density	Calculated k_{eff} eigenvalue	Reactivity (pcm / $ using $\beta = 0.0075$)
100	1.01209 +/- 0.00015	1194 / 1.54
90	1.00282 +/- 0.00015	281 / 0.37
75	0.98436 +/- 0.00015	-1588 / -2.12
50	0.93982 +/- 0.00015	-6403 / -8.54
25	0.86609 +/- 0.00014	-15461 / -20.61
0	0.71175 +/- 0.00014	-40498 / -54.00

The void coefficient is simply the slope of each line segment. As stated previously, it is highly unlikely that the reactor could ever operate with more than 10% void present. In the region between 0% and 10% void, the void coefficient is -91.3 pcm per percent void or -0.12 dollars per percent void. With increasing void percentage, the line segments become steeper, so the average void coefficient over the entire range of possible voiding (0% to 100%) is of greater magnitude and always negative.

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.

Attachment A addresses all the items requested in this section except for "changes in reactivity coefficients that result from changes to core configurations." The core configuration modeled is the core configuration in operation when the SAR was submitted. Changes in the core configuration are performed under the process described in 10 CFR 50.59.

An experiment was performed in March 2011 to measure the values for reactivity due to equilibrium and peak (post shutdown) xenon. Due to the limited core excess at the time, the reactor was unable to maintain full power for the 50-hour run. The xenon reduced the reactor power, with all rods fully out, to approximately 70% (175 kW), inserting $0.84 of negative reactivity. Extrapolating from this data using the equilibrium xenon equation

$$Xe_{\infty} = \frac{(\gamma_I + \gamma_{Xe})\Sigma_f \phi}{\lambda_{Xe} + \sigma_a^{Xe} \phi}$$

the equilibrium xenon worth at 250 kW is estimated to be $1.06. The measured power defect at 250 kW is approximately $1.33. Thus in order to maintain full power during a long run, a core excess of at least $1.06 + $1.33 = $2.39 will be necessary. To account for experiment worth, etc., a Core Excess limit of $3.00 seems prudent.

It has been noted that Figure 5 of Attachment A indicates that the calculated IRW value of the reg rod is $1.06 and the measured value is $1.34. Furthermore, some of the calculated reactivity values are non-physical, that is *withdrawing* the rod seems to introduce *negative* reactivity over certain increments. In response, during initial modeling attempts, it was found that the control rods in the MCNP model had a calculated worth that was significantly higher than the recent measured values. It was reasoned that not only had the fuel experienced some depletion, but the control rods had as well. In order to account for this, control rod boron content of the MCNP model was reduced as discussed in section 4 of Attachment A. All three control rods were depleted in exactly the same manner such that the calculated value of total rod worth was reduced to essentially the same value as measured total rod worth. Note, however, that the Reg rod is located in the E-ring while the other two rods are located in the C-ring. The reg rod is thus depleted significantly less than the other two rods. By depleting all three control rods in an identical manner, the end result was that the reg rod was significantly over-depleted and the shim and safe rods were slightly under-depleted. The goal of the depletion simulation was to achieve good agreement between measured and calculated total rod worth. A more accurate approach would have been to deplete each control rod in such a manner that the calculated and measured worth of each rod would be equal. Although this is a flaw in the model, safety calculations that involve control rod accidents are based on the *most* reactive rod sticking or ejecting. Values used in these safety calculations were based on measured rod worth, not calculated rod worth.

Regarding the non-physical behavior of calculated rod worth it is helpful to consider a concrete example. Consider the withdrawal from 74% insertion to 67% insertion. This 7% withdrawal corresponds to a 1.05 inch rod movement which results in an insertion of $0.10 reactivity (measured). To duplicate this manipulation with MCNP calculations, k_{eff} of the core is calculated at the initial rod position and then recalculated with the reg rod 1.05 inches higher. The k_{eff} value at the initial state should be close to 1.000. The reactivity value at the second state should be $0.10 higher, corresponding to $k_{eff} \approx 1.00075$. These two values of k_{eff} are extremely close. Since reactivity is dependent on the difference between two very close values, the uncertainty in calculated reactivity quantities tends to be much larger than the uncertainty in calculated k_{eff} values. Uncertainty can be reduced by running more particles. The rod worth calculations for each rod took more than a week of computer time. To cut the uncertainty (in the value of k_{eff}) in half would require an additional month of computer time. This is not considered feasible.

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.

Section 6 of Attachment A specifically addresses core neutron physics parameters. Based on the large negative values of the various temperature and void coefficients, the RRR can be operated safely and has been shown to be within the limits specified in the Technical Specifications for core excess and shutdown margin.

The eigenvalues used to generate the alpha-T values are tabulated below. The ZAIDs and corresponding ENDF information is also given. To perform these calculations, only the temperature (i.e. cross section library) for U235 and U238 were varied. There are no modern (post-1969) high temperature libraries for ZrH.

Temperature (°C)	keff	ZAID extension	ENDF Library
293.6	1.01209 +/- 0.00015	.66c	.66a(B-VI.6)
400	1.00547 +/- 0.00014	.12c	62mt(B-VI.2)
600	0.98819 +/- 0.00013	.14c	62mt(B-VI.2)
800	0.97619 +/- 0.00014	.15c	62mt(B-VI.2)
1200	0.92878 +/- 0.00013	.17c	62mt(B-VI.2)

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. The2007 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.*

Section 7 of Attachment B shows that the steady state MDNBR at 250 kW is 6.33 in the hottest channel using a pool temperature of 50°C which is well above the Technical Specification limit of 40°C set for the pool water temperature based on demineralizer resin. Experimentation has shown that even with the primary system off the core inlet temperature is never more than 2°C warmer than the pool temperature instrument used to comply with the technical specification temperature limit, so 50°C is very conservative.

The uncertainty in this analysis is largely driven by the uncertainty in the Bernath correlation coefficient, because the other variables in the calculation are well understood and are likely small in comparison. Given the conservative nature of the analysis, it is likely that the MDNBR number is lower than the lowest values for MDNBR vs. hot channel steady state power when uncertainty is included using a more applicable correlation coefficient.

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.*

The cooling system was designed to remove 250 kW of heat, thus is adequate for both normal operation and shutdown cooling. The heat exchanger size was doubled in 2009 to allow for a power upgrade to 500 kW if necessary. 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.

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.*

Corrosion control is maintained by limits on primary water conductivity. The conductivity is measured weekly and the limit is set at 5 μSiemens/cm which is in agreement with NUREG 1537 Appendix 14 ¶3.3(9), Water Chemistry. Additionally, the pH will be measured quarterly to ensure it is between 5 and 7.5, which is in agreement with NUREG 1537.

The fuel elements are inspected at least once every ten years for evidence of corrosion, wear, or damage. Gross failure or obvious visual deterioration of the fuel is sufficient to warrant declaration of the fuel as damaged. Visually inspecting fuel elements biennially will identify any developing fuel integrity issues throughout the core. The method of determining non-conforming fuel at the RRR has been exclusively visual inspection. Experience at many TRIGA reactors over many years has shown this to be adequate. Since the RRR is not a pulsing reactor, measurement of bow and elongation is not required under NUREG 1537.

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.*

    The secondary cooling system is designed to remove at least 250 kW of heat so as to sustain continuous operation at full power. Experience has shown that the pool temperature can be maintained below 40°C even during long operations in the summer. The limiting factor for pool temperature is damage to the primary resin.

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.*

    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 with a total activity less than 1 mCi, which is consistent with the ALARA program.

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 response to item 12.

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.*

The production of ^{16}N and resulting personnel exposures are discussed in Section 11.1.1.1.8. Additionally, the exposure rate measured at the surface of the primary water is approximately 2.5 mrem h^{-1}. It is very difficult to determine the exact contribution ^{16}N makes to this exposure rate but with a 7.2 s half life, the contribution of ^{16}N to occupational dose in the reactor bay is negligible and the contribution to dose to the general public is zero. In addition, the primary cooling system returns all of the primary coolant 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 7.2 seconds, more than enough time to decay before reaching the surface.

The results of surveys at the RRR is below. It should be noted that the above radiation levels were measured at the center and drop off rapidly, to a few μR, at the edge of the tank.

Table 1: RRR Radiation Levels at 230 kW in mrem/hour

	Water Surface	At Bridge Level	1 m above Pool
Diffuser ON	2.5	0.3	0.2
Diffuser OFF	10	3	1

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.*

    Attachment C demonstrates that the plane array one element thick used at the RRR will have a k_{eff} less than 0.80 under all conditions of moderation. The RRR fuel racks match those described in Attachment C in all pertinent aspects.

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 10.1.

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.*

This response replaces that in the SAR Chapter 11. The only significant source of ^{41}Ar that contributes to occupational radiation exposure is that which is generated in, and released from, the reactor tank, the rotating rack, the rabbit, and the vertical beam when it is used. The highest average ^{41}Ar concentration discharged while at full power in the last five years was 3.0E-7 μCi ml^{-1}. Assuming uniform mixing in the reactor bay, the concentration of ^{41}Ar in the reactor bay would also be 3.0E-7 μCi ml^{-1}. This is well below the 10 CFR 20 listed Derived Air Concentration (DAC) for ^{41}Ar of 3.0E-6 μCi ml^{-1}. Note that this value does not depend on how many hours the reactor is operated per year since it is based on the highest average concentration while at full power.

The Reed reactor discharges ^{41}Ar through an exhaust stack that is 3.6 meters above ground level. Atmospheric dilution will reduce the ^{41}Ar concentration before the exhaust plume returns to ground level. Based upon an annual average ^{41}Ar concentration of 3.0E-7 μCi ml^{-1} and measured flow rate of 2.26E9 ml hr^{-1}, the emission rate of ^{41}Ar in the stack effluent is approximately 0.188 μCi s^{-1}.

Based on this emission rate, the ground level concentration of ^{41}Ar (χ) as a function of distance can be calculated from the Gaussian plume model as follows:

$$\chi = \frac{Q}{\pi \sigma_y \sigma_z \mu} e^{\left[-\frac{1}{2}\left(\frac{y^2}{\sigma_y^2}+\frac{h_e^2}{\sigma_z^2}\right)\right]}$$

where:

Q = emission rate (μCi s^{-1});
σ_y = horizontal standard deviation of plume contaminant (m);
σ_z = vertical standard deviation of plume contaminant (m);
y = crosswind distance (0 m - centerline);
h_e = stack height (m); and
μ = mean wind speed (m s^{-1}).

Values for σ_y and σ_z can be determined from charts illustrating σ_y and σ_z vs. distance (Slade, D.H., Meteorology and Atomic Energy, TID-24190, 1968). Using the value of 1E-8 uCi ml^{-1} for ^{41}Ar found in 10 CFR 20 App B, Table 2, Col. 1, the TEDE as a function of distance received by a member of the general public may be estimated. However, the stack height is less than 2.5 times the building height. To correct for this, Equation 3 from Reg Guide 1.145 can be used to calculate X as follows:

$$\chi = \frac{Q}{\pi M \sigma_y \sigma_z \mu}$$

Where:

M=4 (Figure 3, Reg Guide 1.145)

The results of calculating the annual TEDE to the general public from routine releases of ^{41}Ar into the unrestricted area are given in Table 2. This calculation assumed a highly stable stability class and a low wind speed. It should be noted that in order to receive the doses shown in Table 2, an individual would be required to continuously occupy the specified location for a full year while the reactor operated continuously for a year with a constant atmospheric stability class. That being said, all calculated doses are well within all applicable limits in 10 CFR 20.

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 SAR Figure 2.3.

For comparison purposes, determination of radiation dose to the general public from airborne effluents may also be carried out using several computer codes recognized by regulatory authorities. One such method involves the use of COMPLY (V1.5D). Application of this code to the projected ^{41}Ar releases from the RRR using the data for atmospheric stability condition B predicts a maximum annual TEDE to the general public of 0.4 mrem. The method we used results in more conservative values than the COMPLY code.

Table 2 ^{41}Ar Concentrations and Annual Doses in the Unrestricted Area from ^{41}Ar Released During Routine Reactor Operations at Various Atmospheric Stability Classes

Atmospheric Stability Condition	μ ($m\ s^{-1}$)	σ_y (m)	σ_z (m)	χ ($\mu Ci\ m^{-3}$)	Distance (m)	TEDE (mrem)
F	1	4	2	1.63E-3	100	8.1
F	1	7	4	5.35E-4	200	2.7
F	1	11	6	2.48E-4	300	1.2
F	1	15	7	1.43E-4	400	0.7
F	1	18	9	9.80E-5	500	0.5
F	1	22	10	7.17E-5	600	0.4
F	1	24	11	5.68E-5	700	0.3

The direct exposure from ^{41}Ar in the reactor bay to a person in the adjacent Psychology Building was calculated using Microshield 7.02. The person was represented by a point 9.1 m from the building and 1 m in the air. There were two 19.24 cm thick concrete walls between the point and the reactor bay. The entire bay was assumed to be filled with ^{41}Ar at a concentration of 3E-7 $\mu Ci\ cm^{-3}$. The exposure rate was calculated to be 3.17E-5 $mR\ h^{-1}$.

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 and 11.1.6.

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 11.1.6.

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. All personnel leaving the reactor bay are surveyed for contamination. In the event of a personnel contamination, personnel decontamination processes would be employed in accordance with RRR Standard

Operating Procedures. This involves the use of a mild soap and lukewarm water.

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 replaces the analysis in SAR Chapter 13. For the RRR, the MHA has been defined as the cladding rupture of one highly irradiated fuel element with no radioactive decay followed by the instantaneous release of the noble gas and halogen fission products outside the cladding and into the air. The failed fuel element was assumed to have been operated at the highest core power for a continuous period of one year at 250 kW. This results in all of the halogens and noble gases (except Kr-85) reaching their saturated activities.

    This is the most severe accident for a TRIGA® reactor and was analyzed to determine the limiting or bounding potential radiation doses to the reactor staff and to the general public in the unrestricted area. A less severe, but more credible accident, involving this same single element having a cladding failure in water will also be analyzed.

During the lifetime of the RRR, used fuel within the core may be moved to new positions or removed from the reactor. Fuel elements are moved only during periods when the reactor is in a shutdown condition. Also, the RRR is seldom operated continuously at 250 kW for a period longer than 8-10 hours, let alone a period of one year. Nevertheless, this MHA has been analyzed for the RRR.

Three scenarios have been chosen for analysis:

- Scenario A:

  In this scenario, the entire north wall of the reactor room instantly vanishes. No credible cause for this occurrence can be imagined. The noble gas and halogen fission products that have been released to the reactor room air are assumed to mix instantly and uniformly with the room air. This reactor room air then moves out through the missing wall at the mean wind speed (1 m s^{-1}). This is assumed to be a ground level release. It takes 8.8 seconds for the entire volume of the reactor room air to be evacuated. Thus, individuals outside the reactor room will be exposed to a radioactive cloud for a period of 8.8 seconds;

- Scenario B:

  This scenario again assumes that the noble gas and halogen fission products instantly and uniformly mix with the reactor room air. The fission products that have been released to the reactor room air are then exhausted out the stack at the stack ventilation rate. However, this is assumed to be a ground level release. The time to evacuate the entire volume of the reactor room is 478 seconds, and this is, therefore, the exposure time for individuals outside the reactor room; and

- Scenario C:

  This scenario also assumes that the noble gas and halogen fission products instantly and uniformly mix with the reactor room air. The reactor room air then leaks from the room at ground level at the leak rate of 1.54E-3 m^3 s^{-1}. The leakage from the room is through the walls brought about by a pressure differential between the room and outside. This pressure differential was assumed to arise through the unlikely combination of a drop in atmospheric pressure of 1.5" Hg and an increase in room temperature of 40°C in 12 hours. Using the ideal gas law, $P_1V_1/T_1= P_2V_2/T_2$, where temperature is in K, pressure is in inches of Hg, and volume is in cubic centimeters. The initial conditions of T_1=295.2 K (22.2°C) and P_1=29.4" Hg in the RX bay of volume 339.8 m^3, the increase in temperature of 40°C would result in the pressure increasing to 33.4" Hg. The atmospheric drop of 1.5" Hg would mean an increase in room pressure to 34.9" Hg and assuming an outside temperature of 8.7°C, the resulting bay volume would be 273.3 m^3. The leak rate is determined by the difference in the initial and final bay volumes which is 66.5 m^3 divided by 12 hrs. or 43200 s to equal 1.54E-3 m^3 s^{-1}.

  In this case, it would take 1.95E+5 s for the entire volume of the reactor room air to be evacuated, and this is the exposure time for individuals outside the reactor room. This is also assumed to be a ground level release.

All halogens and noble gases (except Kr-85) are assumed to be at their saturation activities. The highest power element fails and releases the noble gases and halogens from the cladding gap to the air or water. This highest power element has an estimate power of 7.24 kW. The fission product inventory of halogen and noble gases are given in Table 3 for this element. The inventory assumes a saturated activity is present and is based upon the fission yield for each isotope.

Considerable effort has been expended to measure and define the fission product release fractions for TRIGA® fuels. Data on this aspect of fuel performance are reported in Refs. 1 through 7. Using this data, GA developed a conservative correlation for fission product release to be

$$e = 1.5x10^{-5} + 3.6x10^{3} \exp\left\{\frac{-1.34x10^{4}}{T + 273}\right\}$$

At an average fuel temperature of 264°C, this release fraction is 1.51E-5. This fuel temperature (264°C) is the calculated maximum expected fuel temperature.

Once the fission products are released to the gap, this activity is released when the cladding fails. If the release is in air (MHA), then this activity is released directly into the reactor room air. If the release occurs in the pool water, then the fission products must migrate through the water before being released to the reactor room air. Once released into the reactor room air, a further reduction of the halogen activity is expected to occur due to plateout in the building.

Thus, the fraction (w) of the fission product inventory released from a single fuel element which reaches the reactor room air and, subsequently, the atmosphere in the unrestricted environment is:

$$w = e\ f\ g\ h;$$

where:

e = the fraction released from the fuel to the fuel-cladding gap;
f = the fraction released from the fuel-cladding gap to the reactor room air (if no water is present), or to the pool water (if water is present);
g = the fraction released from the pool water to the reactor room air; and
h = the fraction released from the reactor room air to the outside unrestricted environment, due to plateout in the reactor room.

Table 3 Saturated Activities for Highest Power Density Fuel Element

[redacted]

For the accident where the cladding failure occurs in air, it is very conservatively assumed that 25% of the halogens released to the cladding gap are eventually available for release from the reactor room to the outside environment. This value is based on historical usage and recommendations (Refs. 1 through 9), where Reference 1 recommends a 50% release of the halogens from the gap to the air. References 2 and 3 apply a natural reduction factor of 50% due to plateout in the reactor building. Combining the 50% release from the gap with the 50% plateout results in the 25% total release. However, this value appears to be quite conservative, since References 6 and 7 quote a 1.7% release from the gap rather than 50%.

For the accident in air, 100% of the noble gases are assumed to be available for release to the unrestricted environment.

For the accident in water, it is conservatively assumed that most of the halogens released from the cladding gap remain in the water and are removed by the demineralizer. A small fraction, 5%, of the halogens is assumed to escape from the water to the reactor room air. Combining this with the 50% release from the gap to the water, the result is that 2.5% of the halogens in the gap are released to the reactor room. Again, 50% of these plateout in the reactor room before release to the outside environment. For the noble gases in water, 100% are assumed to be available for release to the unrestricted environment.

The experience at TMI-2, along with recent experiments, indicate that the 50% halogen release fraction is much too large. Possibly as little at 0.06% of the iodine reaching the cladding gap may be released into the reactor room due in part to a large amount of the elemental iodine reacting with cesium to form CsI, a compound much less volatile and more water soluble that elemental iodine (Ref. 7). The values for these various release fractions are given in Tables 4 and 5.

The more stable the atmospheric class, the higher the concentration. Therefore, it was assumed that the most stable atmospheric class (Pasquill F) prevailed for all scenarios. Also, the lower the wind speed, the higher the concentration. Thus, it was assumed that a low wind speed of 1 m s^{-1} existed for all scenarios.

Table 4 Release Fraction Components

Fission product	f No pool water	f With pool water	g No pool water	g With pool water	h
Noble gas	1.0	1.0	N/A	1.0	1.0
Halogens	0.5	0.5	N/A	0.05	0.5

Table 5 Total Release Fraction

Fission product	w No pool water	w With pool water
Noble gas	1.51 E-5	1.51 E-5
Halogens	3.76 E-6	1.88 E-7

Based on this emission rate, the ground level concentration as a function of distance can be calculated from the Gaussian plume model as follows:

$$\chi_{max} = \frac{Q}{\pi \sigma_y \sigma_z \mu} e^{\left[-\frac{1}{2}\left(\frac{y^2}{\sigma_y^2} + \frac{h_e^2}{\sigma_z^2}\right)\right]}$$

where:

Q = emission rate (µCi s^{-1});
σ_y = horizontal standard deviation of plume contaminant (m);
σ_z = vertical standard deviation of plume contaminant (m);
y = crosswind distance (0 m - centerline);
h_e = stack height (m); and
µ = mean wind speed (m s^{-1}).

Values for σ_y and σ_z can be determined from charts illustrating σ_y and σ_z vs. distance (Slade, D.H., Meteorology and Atomic Energy, TID-24190, 1968). The values for the dispersion coefficients and χ/Q are given in Table 6.

Furthermore, it was assumed that all of the fission products were released to the unrestricted area by a single reactor room air change, which would maximize the dose rate to persons exposed to the plume during the accident.

Table 6 Atmospheric Dispersion Coefficients and χ/Q Values for Pasquill F and Mean Wind Speed of 1 m sec^{-1}

Distance (m)	σ_y (m)	σ_z (m)	Ground Level Release χ/Q (s m^{-3})	Ventilation Release χ/Q (s m^{-3})
100	4	2	3.46 E-2	3.16 E-7
200	7	4	1.14 E-2	2.45 E-4
300	11	6	5.26 E-3	6.92 E-4
400	15	7	3.03 E-3	8.66 E-4
500	18	9	2.08 E-3	8.90 E-4
600	22	10	1.52 E-3	7.72 E-4
700	24	11	1.21 E-3	7.26 E-4

Additional parameters used in this accident were:

- reactor room ventilation exhaust rate: 0.628 m^3 s^{-1};
- reactor room leak rate: 1.54 E-3 m^3 s^{-1};
- reactor room volume: 300 m^3;
- area of north face of reactor building: 34.1 m^2;
- receptor breathing rate: 3.3 E-4 m^3 s^{-1}: (NRC "light work" rate); and
- dose conversion factors:
  - internal: based on DOE/EH-0071 (Ref. 12);
  - external: based on DOE/EH-0070 (Ref. 13).

The committed dose equivalent (CDE) to the thyroid and the committed effective dose equivalent (CEDE) for members of the general public at a given distance downwind from the facility for all isotopes of concern may each be calculated by:

$$(\text{CDE or CEDE})_D = \sum_i \left[\frac{\left(\frac{X}{Q}\right)_D \text{BR DCF}_{\text{int},i} A_i \lambda_v \left[e^{-\lambda_i t_i} - e^{-\lambda_i t_2} \right]}{\lambda_i} \right],$$

where:

$(\chi/Q)_D$ = atmospheric dispersion factor at a given distance D (s m^{-3});
BR = breathing rate (m^3 s^{-1});
DCF$_{int,i}$ = internal dose conversion factor for isotope i (mrem μCi^{-1}) [Ref. 12];
A$_i$ = initial activity of isotope i released into the reactor room (μCi);

R_v = ventilation or leakrate of air from the reactor bay ($m^3\ s^{-1}$);
V = reactor room volume (m^3);
λ_v = ventilation constant = R_v/V (s^{-1});
λ_i = decay constant for isotope i (s^{-1});
t_1 = time when plume first arrives at the receptor point (s); and
t_2 = time when plume has passed the receptor point (s).

The deep dose equivalent (DDE) to members of the general public at a given distance downwind from the facility for both the thyroid and the whole body may each be calculated by:

$$\left(DDE_{Thyroid} or DDE_{WB}\right)_D = \sum_i \left[\frac{\left(\frac{\chi}{Q}\right)_D DCF_{ext,i} A_i \lambda_v \left[e^{-\lambda_i t_1} - e^{-\lambda_i t_2}\right]}{\lambda_i} \right],$$

where:
$DCF_{ext,i}$ = external dose rate conversion factor for isotope i (mrem m^3 μCi^{-1} s^{-1}) [Ref. 13].

For calculating dose to occupational workers in the reactor room, stay times of 2 and 5 minutes were used. Experience indicates that the reactor room can easily be evacuated in 2 minutes. The value of 5 minutes is thought to be a reasonable longer period of time assuming a worker is performing some task (i.e., determining if a false alarm has occurred). The CDE and CEDE for personnel in the reactor room for a given stay-time may each be calculated by:

$$\left(CDE or CEDE\right)_{ST} = \sum_i \left[\frac{DCF_{int,i} A_i BR\left[1 - e^{\lambda_{eff} t_{ST}}\right]}{\lambda_{eff} V} \right]$$

where:
$\lambda_{eff} = \lambda_i + \lambda_v$; and
t_{ST} = stay-time of personnel (s).

The DDE to personnel in the reactor room for a given stay-time for both the thyroid and the whole body may be calculated by:

$$\left(DDE_{Thyroid} or DDE_{WB}\right)_{ST} = \sum_i \left[\frac{DCF_{ext,i} A\left[1 - e^{-\lambda_{eff} t_{ST}}\right]}{\lambda_{eff} V} \right]$$

The results of these calculations for all three scenarios are shown in Tables 7 through 10. As seen from the tables, Scenario A gives the highest doses to the general public at any distance, as might be expected since the activity was released in a very short time leaving little time for radioactive decay. Scenario B gives the lowest doses at any given distance since the release occurs through the stack at a higher elevation. In all cases, doses for the general public and occupational workers were all well below the annual dose limits specified by 10 CFR 20.

Table 7 Occupational Radiation Doses in the Reactor Room Following a Single Element Failure in Air

Scenario	Reactor Room Occupancy (minutes)	$CDE_{Thyroid} + DDE_{Thyroid}$ (mrem)	TEDE (mrem)
A	2	10.7	0.5
A	5	10.7	0.5
B	2	129.1	6.0
B	5	270.8	12.3
C	2	146.1	6.8
C	5	364.3	16.5

Table 8 Radiation Doses to Members of the General Public Following a Single Element Failure in Air

Distance (m)	Scenario A $CDE_{Thyroid}$ + $DDE_{Thyroid}$ (mrem)	Scenario A TEDE (mrem)	Scenario B $CDE_{Thyroid}$ + $DDE_{Thyroid}$ (mrem)	Scenario B TEDE (mrem)	Scenario C $CDE_{Thyroid}$ + $DDE_{Thyroid}$ (mrem)	Scenario C TEDE (mrem)
100	12.6	0.6	12.6	0.5	9.4	0.3
200	4.1	0.2	4.1	0.2	3.1	0.1
300	1.9	0.1	1.9	0.1	1.4	< 0.1
400	1.1	< 0.1	1.1	< 0.1	0.8	< 0.1
500	0.8	< 0.1	0.8	< 0.1	0.6	< 0.1
600	0.6	< 0.1	0.6	< 0.1	0.4	< 0.1
700	0.4	< 0.1	0.4	< 0.1	0.3	< 0.1

Table 9 Occupational Radiation Doses in the Reactor Room Following a Single Element Failure in Water

Scenario	Reactor Room Occupancy (minutes)	$CDE_{Thyroid} + DDE_{Thyroid}$ (mrem)	TEDE (mrem)
A	2	0.7	0.2
A	5	0.7	0.2
B	2	8.0	1.8
B	5	16.6	3.4
C	2	9.1	2.0
C	5	22.2	4.6

Table 10 Radiation Doses to Members of the General Public Following a Single Element Failure in Water

Distance (m)	Scenario A $CDE_{Thyroid}$ + $DDE_{Thyroid}$ (mrem)	Scenario A TEDE (mrem)	Scenario B $CDE_{Thyroid}$ + $DDE_{Thyroid}$ (mrem)	Scenario B TEDE (mrem)	Scenario C $CDE_{Thyroid}$ + $DDE_{Thyroid}$ (mrem)	Scenario C TEDE (mrem)
100	0.8	0.2	0.4	< 0.1	0.5	< 0.1
200	0.2	< 0.1	0.1	< 0.1	0.2	< 0.1
300	0.1	< 0.1	0.1	< 0.1	0.1	< 0.1
400	0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1
500	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1
600	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1
700	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1

The direct exposure from the isotopes given as the sourced term for the MHA (without primary water) uniformly distributed in the reactor bay to a person in the adjacent Psychology Building was calculated using Microshield 7.02. The person was represented by a point 9.1 m from the building and 1 m in the air. There were two 19.24 cm thick concrete walls between the point and the reactor bay. The exposure rate was calculated to be 8.49E-3 mR h^{-1}.

References:

1	"The Calculations of Distance Factors for Power and Test Reactor Sites," J.J.DiMunno et al, TID-14844, U.S. Atomic Energy Commission, March 1962.
2	Regulatory Guide 3.33, "Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Fuel Reprocessing Plant," U.S. Nuclear Regulatory Commission, April 1977.
3	Regulatory Guide 3.34, "Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Uranium Fuel Fabrication Plant," U.S. Nuclear Regulatory Commission, July 1979.
4	Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, June 1974 (Also see Regulatory Guide 1.3 on BWRs).
5	"A Guide to Radiological Accident Considerations for Siting and Design of DOE Nonreactor Nuclear Facilities," J.C. Elder et at, LA-10294-MS, Los Alamos National Laboratory, January 1986.
6	Nuclear Power Reactor Safety, E.E. Lewis, John Wiley and Sons, 1977, p. 521.
7	Nuclear Engineering, Theory and Technology of Commercial Nuclear Power, R.A. Knief, Hemisphere Publishing, 1992, pp. 353, 431.
8	"Fuel Elements for Pulsed TRIGA Research Reactors," M.T. Simnad et al, Nuc. Tech. Vol. 28, January 1976
9	"The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel," M.T. Simnad, General Atomic Report E-117-833, February 1980.
10	Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, August 1979.
11	"Calculated Atmospheric Radioactivity from the OSU TRIGA Research Reactor Using the Gaussian Plume Diffusion Model," Bright M.K. Wong, Oregon State University Department of Nuclear Engineering Report 7903, August 1979.

12	"Internal Dose Conversion Factors for Calculation of Dose to the Public," DOE/EH-0071, U.S. Department of Energy, Washington, D.C., 1988
13	"External Dose Rate Conversion Factors for Calculation of Dose to the Public," DOE/EH-0070, U.S. Department of Energy, Washington, D.C., 1988

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.*

Loss of Coolant Accident

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

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

The RRR original SAR from 1968 discussed this issue in-depth for a maximum power of 250 kW and aluminum-clad fuel. The calculations demonstrated that the maximum fuel temperature reached is 150°C under very conservative estimations, and dose rates from the core are summarized in Table 11.

Table 11: Radiation Dose Rates After Extended 250 kW Operation and Loss of All Shielding Water

Time from Shutdown	Direct Radiation (R/hr)	Scattered Radiation (R/hr)
10 seconds	2.5E3	0.650
1 day	3.0E2	0.075
1 week	1.3E2	0.035
1 month	3.5E1	0.010

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

Similar calculations of dose rates for the direct and scattered radiation in the

Torrey Pines Mark F were performed after a loss of coolant accident following a full year of operation at 1.5 MW. The ceiling of the reactor bay is considered to be a thick concrete ceiling 9 ft above the top of the reactor pool. The results are summarized in Table 12.

Table 12: Radiation Dose Rates After Extended 1.5 MW Operation and Loss of All Shielding Water

Time from Shutdown	Direct Radiation (R/hr)	Scattered Radiation (R/hr)
10 sec	1.5E4	3.90
1 day	1.8E3	0.45
1 week	8.1E2	0.21
1 month	2.1E2	0.06

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

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.*

Experiment reactivity worth is a negative value and failure in the experiment introduces positive reactivity. Reactivity worths are calculated using the most conservative assumptions regarding the worth of experiments, that is removal of the experiment results in a positive reactivity event, and adding the experiment also results in a positive reactivity event.

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.*

See updated SAR Chapter 14.

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.*

    Attachment B shows that at the Safety Limit value of 500 kW the maximum fuel centerline temperature is ≈ 410°C, which is acceptable by NUREG-1537, Appendix 14, Section 2.1. It further shows that at the Limiting Safety System Setting of 275 kW the maximum fuel centerline temperature is ≈ 300°C. NUREG-1537, Appendix 14, Section 2.2 says that the LSSS may be 10% to 20% above the licensed power. We have chosen 10% (275 kW) as the more conservative setting. At the licensed power of 250 kW the maximum fuel centerline temperature is ≈ 280°C.

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.*

    See updated SAR Chapter 14.

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 6OO0C 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.

See updated SAR Chapter 14.

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 at 110% 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. Experience and analysis have indicated that for the range of transients anticipated for a TRIGA reactor, a one second scram time is adequate to assure the safety of the reactor.

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.

Attachment B calculations show that the 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). 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.

To limit the dose rate in the reactor bay so that visitor may access the room the dose rate should be less than 2 mrem per hour. The pool activity is limited to 1 μCi/ml to reach this goal. Consider a slab of uniformly distributed radioactive material characterized by a linear absorption coefficient μ. If the slab is defined to be of cylindrical shape with a radius R and thickness t, the dose rate at point a distance h above the centerline of the slab is

$$D = \pi G (C_v/\mu)(1-e^{-\mu t}) \ln((R^2 + h^2)/h^2))$$

where

- D = dose rate (mR/hr) at a distance h from the surface
- G = gamma constant (mR-m2/hr-mCi)
- C_v = activity concentration in the slab (mCi/ml)
- t = thickness of slab (m)
- R = Radius of slab (m)

Ref: *Contemporary Health Physics: Problems and Solutions*, Bevelacqua, John Wiley and Sons, 1999, page 463

Assuming Na-24 for the slab with a radius of 1.52 m and thickness of 7.62 m to approximate the RRR Pool, dose rate one meter above the pool will be:

$$X_0(h) = \pi (1.84 \text{ mR/hr/m})((0.001 \text{ mCi/ml})/(0.06))(1-e^{-(0.06)(762)}) \ln(((1.52)^2 + 1^2)/1^2)))$$

$$X_0(h) = 1.15 \text{ mR/hr}.$$

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.

Temperature distribution within a fuel element during a rapid transient (i.e., a pulse) is dependent on both time and spatial position within the element. This question can, however, be answered using a point kinetics approach with some conservative assumptions.

Pulse transients are normally rapid enough that adiabatic conditions can be assumed. If adiabatic conditions are assumed, temperature distribution will mimic power distribution. The hottest part of the fuel element will be located at the outer radius of the fuel element, near the axial centerline. Following the rapid pulse transient, energy will be redistributed within the fuel element. Analysis of OSTR fuel which has very similar thermal behavior to the RRR fuel was done using the RELAP thermal/hydraulic code. This analysis showed that the temperature profile within a fuel element will be reversed (i.e., no longer hottest on the outer edge) within no more than 20 seconds following pulse peak power. The peak temperature following the reversal was also much lower than the peak temperature observed immediately after peak pulse power. It can therefore be conservatively assumed that peak temperature experienced by the fuel will be less than

$$T_0+\Delta T_{max}*PF$$

where T_0 is the initial fuel temperature, ΔT_{max} is the average temperature rise in a 'point reactor' under adiabatic conditions 20 seconds after the pulse peak and PF is the effective peaking factor. T_0 is assumed to be the maximum pool temperature of 50°C. PF is found from the power distribution analysis discussed in the thermal hydraulics report. ΔT_{max} is found by modeling the core as a point reactor and solving the point reactor kinetics equations using a semi implicit numerical solution method. The value of α_T, the reactivity temperature coefficient is taken from Attachment A:

$$\alpha_T = -5.18E\text{-}5 - T*9.18E\text{-}8\ \Delta k/k \text{ per degree C}$$

The heat capacity for an average fuel element is found by taking the volumetric heat capacity from the Simnad report for 8.5 weight percent LEU fuel and multiplying it by the average fuel volume of a RRR fuel element:

$C_p = 768.3+T*1.494$ Joule per degree C

There are 65 fuel elements in the modeled core. The average fueled volume is 358.3 cm^3. The remainder of the information needed to solve the point reactor kinetics equations is determined by U-235 delayed neutron fractions and decay constants. Mean neutron generation time for the RRR is not well known, but the temperature rise following a $1.00 reactivity insertion was found to depend only very weakly on mean neutron generation time. Changing mean neutron generation time from 30 microseconds to 300 microseconds produced a 3°C change in predicted temperature rise. Using this approach and the stated assumptions, a $1.00 reactivity insertion will produce a maximum fuel temperature no greater than 50 + 144 * 2.952 = 475.1°C. This upper bound could likely be significantly reduced using a RELAP based analysis.

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

Typical Control Rod Worths were recently measured as:

Safety Rod	$3.37
Shim Rod	$3.27
Regulating Rod	$1.34
	$7.98

The Core Excess when critical at 5 W is typically:

Safety Rod	$0.65
Shim Rod	$0.65
Regulating Rod	$0.35
	$1.65

With a typical core excess and the most reactive control rod stuck out, the reactor will be subcritical by $7.98 - $1.65 - $3.37 = $2.96. With the maximum allowable core excess the reactor would be shutdown by $7.98 - $3.00 - $3.37 = $1.61 which is still greater than the $0.50 minimum.

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. Routine in-service inspections look for evidence of deterioration or corrosion such there is a reasonable expectation that the components will be able to perform their functions safely for another 20 years.

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.

ANALYSIS OF THE NEUTRONIC BEHAVIOR

OF THE

REED RESEARCH REACTOR

Submitted By:

Radiation Center
Oregon State University
Corvallis, Oregon

March 8, 2011

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1. Introduction

This report contains the results of investigation into the neutronic behavior of the Reed Research Reactor (RRR). The objectives of this study were to: 1) create a model of the RRR to study the neutronic characteristics, and 2) demonstrate acceptable reactor performance and safety margins for the RRR core under normal conditions.

The design and analyses in this report provide comparisons of reactor parameters and safety margins for the currently configured RRR core. Neutronic behavior of the RRR core is analyzed under normal conditions. The current RRR core contains a loading of 64 fuel elements, three control rods, and various experimental facilities. The RRR core is shown to have acceptable reactivity coefficients. With proper fuel inventory management, the RRR core also maintains acceptable shutdown margin and excess reactivity characteristics.

2. Summary and Conclusions of Principal Safety Considerations

The conclusion of this investigation is that the MCNP model does an acceptable job of predicting behavior of the RRR core. As such the results suggest that the RRR core can be safely operated within the parameters set forth in the technical specifications. Discussion and specifics of the analysis are located in the following sections.

3. Reactor Fuel

The fuel utilized in the RRR is TRIGA® fuel manufactured by General Atomics. The use of high-uranium content, low-enriched uranium/zirconium hydride fuels in TRIGA® reactors has been previously addressed in NUREG-1282.[1] This document reviews the characteristics such as size, shape, material composition, dissociation pressure, hydrogen migration, hydrogen retention, density, thermal conductivity, volumetric specific heat, chemical reactivity, irradiation effects, prompt-temperature coefficient of reactivity and fission product retention. The conclusion of NUREG-1282 is that TRIGA® fuel, including that utilized in the RRR, is acceptable for use in reactors designed for such fuel.

The design of standard stainless steel clad fuel utilized in the RRR is shown in Figure 1. Stainless steel clad elements used at RRR all have fuel alloy length of 38.1 cm. The characteristics of standard fuel elements are shown in Table 1. Aluminum clad fuel elements with three different fuel alloy lengths are also used in the RRR core. Outer dimensions of aluminum and stainless steel clad elements are the same, but dimensions of interior structures differ. Fuel meats in the aluminum clad elements also lack the innermost zirconium pins present in the stainless clad elements. Characteristics of these aluminum clad fuel elements are shown in Table 2.

Table 1 Characteristics of Stainless Steel Clad Fuel Elements

Fuel Type	SS Clad
Uranium content [mass %]	[redacted]
BOL U-235 enrichment [mass % U]	19.75
Fuel alloy inner diameter [mm]	6.35
Fuel alloy outer diameter [mm]	36.449
Fuel alloy length [mm]	[redacted]
Cladding material	Type 304 SS
Cladding thickness [mm]	0.508
Cladding outer diameter [mm]	[redacted]

Table 2 Characteristics of Aluminum Clad Fuel Elements

Fuel Type	Full Height Al	½ Height Al	¾ Height Al
Uranium content [mass %]	[redacted]	[redacted]	[redacted]
BOL U-235 enrichment	19.89	20.2	20.1
Fuel alloy inner diameter [mm]	---	---	---
Fuel alloy outer diameter [mm]	[redacted]	[redacted]	[redacted]
Fuel alloy length [mm]	355.6	177 8	266.7
Cladding material	Al	Al	Al
Cladding thickness [mm]	0.8255	0.8255	0.8255
Cladding outer diameter [mm]	[redacted]	[redacted]	[redacted]

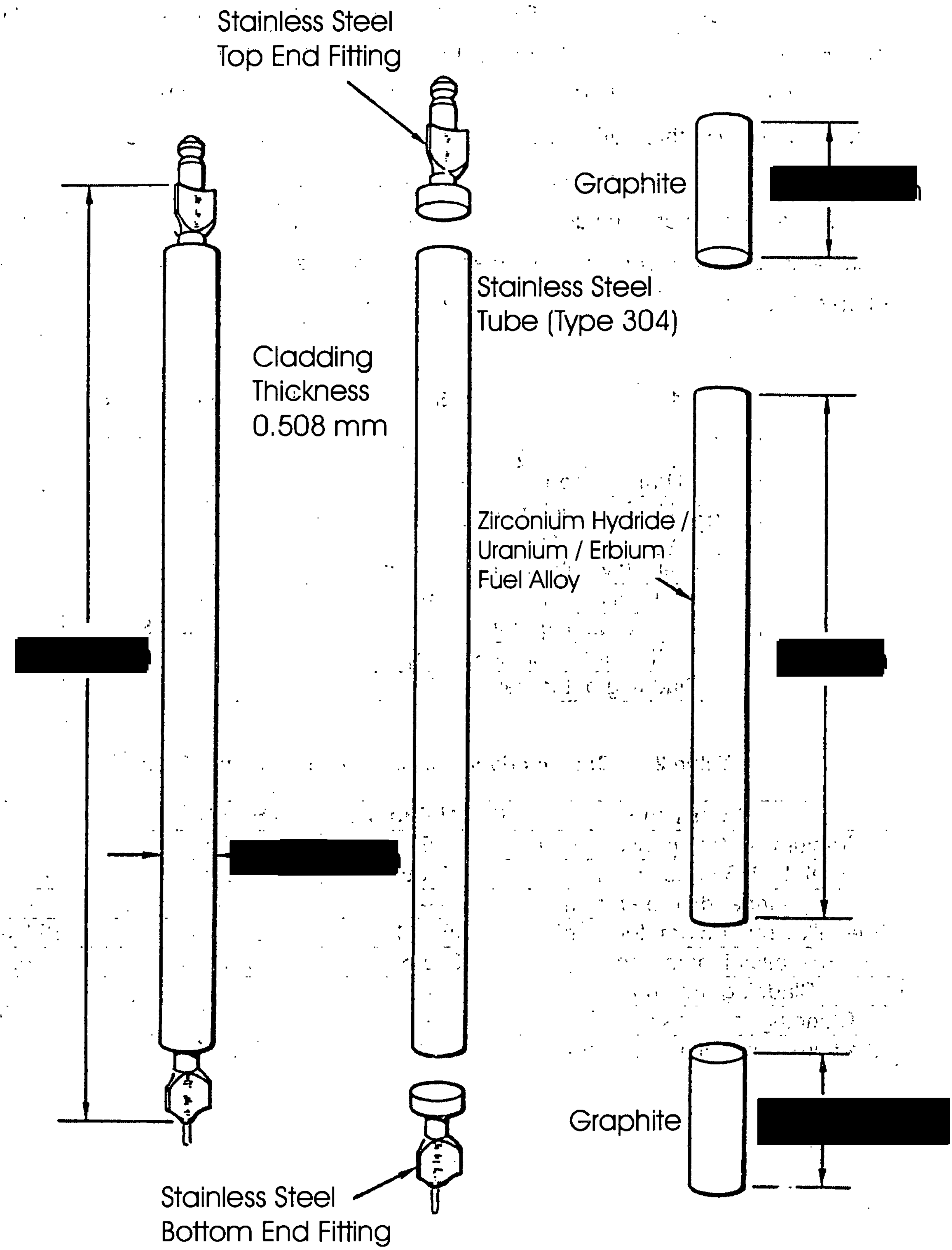

Figure 1 **TRIGA® Stainless Steel Clad Fuel Element Design Utilized in the RRR Core**

4. Reactor Core

The RRR core is a 6 ring (A through F) circular array composed of stainless steel clad and aluminum clad TRIGA fuel. The core also contains several non-fueled locations. The original core installed in 1968 comprised solely aluminum-clad fuel elements. There have been 46 distinct core configurations since inception. The current configuration established in 2009 includes 10 stainless steel clad elements installed in the B-ring and parts of the C-ring. These stainless clad elements were loaded between 1973 and 2008. The total number and type of elements used is shown in Table 3. The current core arrangement is shown in Figure 2. This arrangement was used to model and analyze the RRR core.

Table 3 Core Components for the current RRR core

Type of Elements	Number of Elements
Stainless steel clad fuel elements	10
Aluminum clad 14" fuel elements	52
Aluminum clad 3/4 fuel elements	1
Aluminum clad 1/2 fuel elements	1
Aluminum clad reflector elements	21

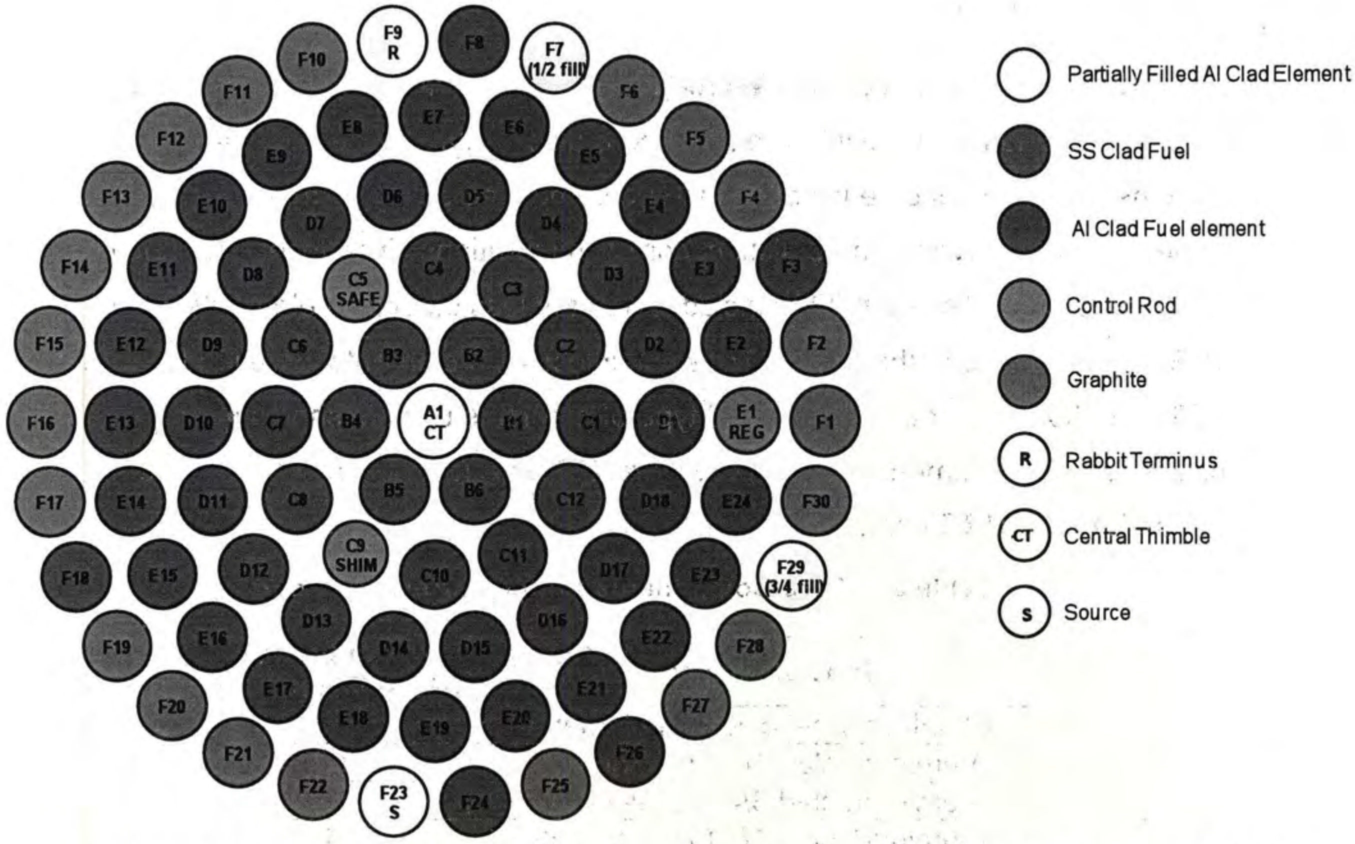

Figure 2 Schematic Illustration of the RRR Upper Grid Plate Showing the Typical Arrangement of Core Components

Detailed neutronic analyses of the RRR core were undertaken using MCNP5.[2] MCNP5 is a general purpose Monte Carlo transport code which permits detailed neutronic calculations of complex 3-dimensional systems. It is well suited to explicitly handle the material and geometric heterogeneities present in the RRR core. In the model developed to describe the RRR, facility drawings provided by the manufacturer at the time of construction of the facility were used to define the geometry of the core and surrounding structures. The geometry of the stainless steel clad fuel elements and control rods were based upon the manufacturing drawings for the assemblies, TOS210D210 Rev. R, TOS210J220 Rev. T, and TOS250D225 Rev. A, respectively. Geometry of the aluminum clad fuel elements were based upon the manufacturing drawings TOS210D130 Rev. K and TOS210C172 Rev. A. Representative Cross-sectional views of the MCNP5 model are shown in Figure 3 and Figure 4.

Figure 3 Horizontal Cross-section of the MCNP5 Model used to Perform Neutronic Analyses of the RRR Core (taken at the core mid-plane, showing model central region only)

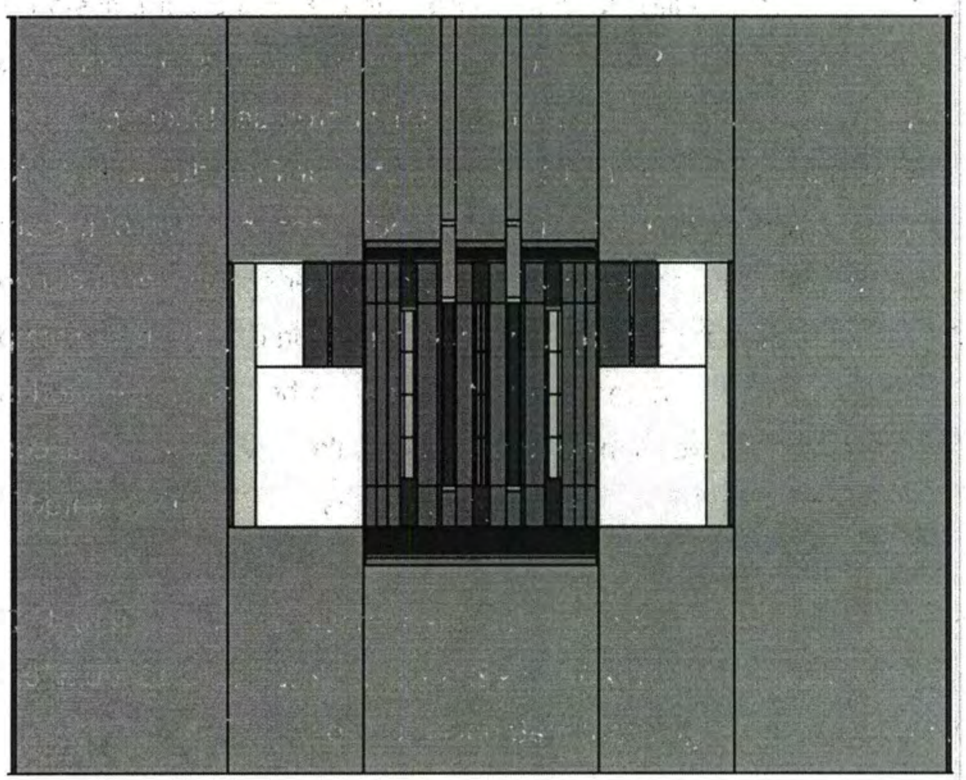

Figure 4 Vertical Cross-section of the MCNP5 Model used to Perform Neutronic Analyses of the RRR Core (different scale and larger extent than Figure 3)

Detailed start-up testing data was not available for the RRR. If the data was available, a series of MCNP5 analyses based upon various core configurations could be run to determine the bias of the model. This bias represents such things as differences in material properties that are difficult to determine or unknown (i.e., lack of manufacturer mass spectroscopy data on the exact composition of individual fuel meats and trace elements contained therein) or applicability of cross section data sets used to model the reactor (i.e., interpolation between temperatures). As a result, the validation of the model was based upon the ability of the code to accurately predict both core excess and control rod reactivity worth as compared with measurements made on the reactor in January 2010.

Because the entire core has only seen approximately 61.1 MW-days of operation, a detailed depletion analysis was not performed, however depletion was taken into account in two ways. First, U-235 was removed from the Beginning of Life (BOL) amount present in each fuel element in accordance with the total energy produced by each fuel element over core life. No other materials (i.e., plutonium, fission products) were added to fuel elements. Second, control rod absorber number density and absorber radius were reduced in an effort to mimic boron depletion. Boron distribution in the control rods at this point in core life would be non-uniform due to self shielding, with the highest degree of depletion occurring in the outermost portions of the absorber. Boron number density was reduced by 5% and absorber radius was reduced from 1.53 cm. to 1.30 cm. (i.e., 15% reduction) to account for boron depletion in a simplified manner. These reductions in number density and absorber radius resulted in reasonable agreement between measured and calculated values of total control rod worth (TRW). Measured TRW for the analyzed core was \$7.92. Calculated TRW for the analyzed core was \$8.11.

Fuel element meats were modeled as a homogeneous mixture of U-235, U-238, natural zirconium and hydrogen (ZAID's 92235.66c, 92238.66c, 40000.66c, 1001.66c). Compositions of all other significant materials are shown in Table 4.

Table 4 Physical Densities and Mass Fractions for Selected Core Components in the MCNP5 Model of the RRR

Material	Physical Density [g/cm³]	Nuclide	Mass Fraction
Type 304 SS (Fuel Clad)	7.857	C-12 Natural Cr Natural Ni Natural Fe	7.993E-04 1.900E-01 1.000E-01 7.092E-01
Graphite Reflector (Fuel)	1.560*	C-12	1.0000
Graphite Reflector Elements	1.560*	C-12	1.0000
Graphite Reflector (Core)	1.698	C-12	1.0000
Zr Fuel Pin (SS clad elements)	6.398	Natural Zr	1.0000
Stainless Steel + Water Mix	3.056	C-12 Natural Cr Natural Ni Natural Fe H-1 O-16	6.165E-04 1.465E-01 7.713E-02 5.470E-01 2.560E-02 2.032E-01
Pure Aluminum	2.700	Al-27	1.0000
Aluminum 6061-T6	2.700	Al-27 Natural Cr Natural Cu Natural Mg Natural Si B-10	0.9793 1.900E-03 2.800E-03 1.000E-02 6.000E-03 1.475E-06
Water	1.000	H-1 O-16	0.1119 0.8881
Air	1.29E-03	N-14 O-16	0.7671 0.2329

* Smeared density, accounting for graphite to clad gap. True physical density is 1.75 g/cm³.

5. Effective Delayed Neutron Fraction

The effective delayed neutron fraction for the RRR core was calculated with MCNP5 by utilizing the expression

$$\beta_{eff} = 1 - \frac{k_p}{k_{p+d}},$$

where k_p is the system eigenvalue assuming fission neutrons are born with the energy spectrum for prompt neutrons, and k_{p+d} is the system eigenvalue assuming fission neutrons are born with the appropriately weighted energy spectra for both prompt and delayed neutrons. The traditional effective delayed neutron fraction is 0.0075 and the calculation has verified this number. The calculation produced 0.00778 ± 0.00020 which is within 1.5 standard deviations of the value in use. This is also in reasonable agreement with values predicted in other LEU TRIGA cores (i.e., Oregon State University β_{eff} = 0.0076, Washington State University β_{eff} = 0.0075). The value β_{eff} = 0.0075 is used to express all dollar values of reactivities in this report.

6. Core Excess, Control Rod Worth and Shutdown Margin

The calculated excess reactivity of the RRR was $1.65 ± $0.11 assuming cold, clean conditions with no experiments in the core. This compares very favorably with the measured value of $1.66 from rod calibrations performed in January 2010. The reactivity worths of the individual control rods were calculated using the rod positions from rod calibrations performed by Reed College in January 2010. The results of the MCNP5 reactivity calculations are compared to the measured values in Figure 5 through 7. Uncertainty in the MCNP5 calculations is determined by 1σ values of successive k_{eff} quantities. Uncertainty of reactivity measurements using the rod pull method has been estimated as +/- 5% per rod pull[3]. The integral rod worths are summarized in Table 5. The calculated values are in reasonable agreement with the measured values, with TRW differing by 2.4%.

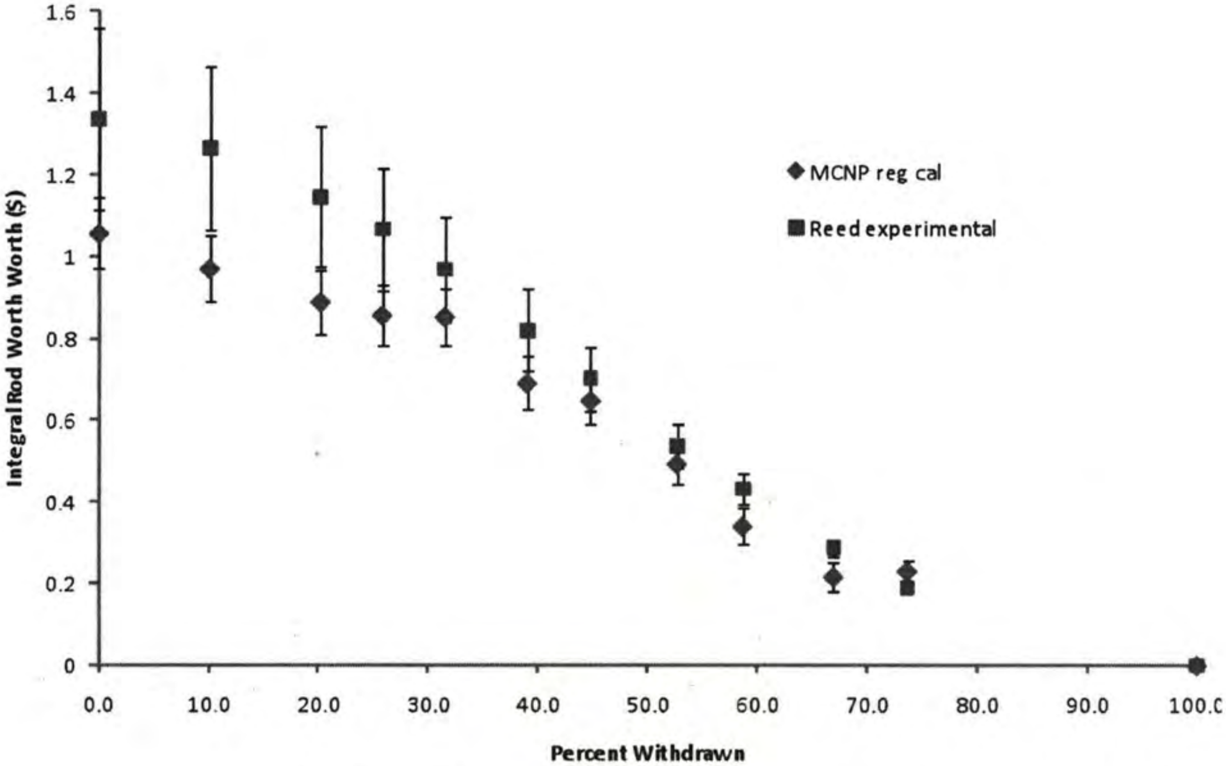

Figure 5 Reg Rod Worth for RRR Core

In accordance with Reed College operating procedures, the Reg rod is calibrated first, and is withdrawn sequentially from full insertion (0% withdrawn) to full withdrawal (100% withdrawn). After the reg rod is calibrated, the reactor is configured with the reg rod and shim rod fully withdrawn and the safe rod inserted to maintain criticality. This results in two control rods fully withdrawn and the third near mid height. The safe and shim rods are then calibrated by pulling the safe rod, measuring the period and then inserting the shim rod to re-establish criticality at the initial starting power. The result of this method is that the shim and safe rods curves are based on measurements taken on only the upper portions of the rods. The remainder of the rod worth curves is established by fitting a third order polynomial to the collected data. Results are also customarily plotted to show rod worth as $0.00 when the rod is 100% withdrawn. The remainder of the rod worth curves indicates the amount of reactivity inserted by inserting a control rod to the indicated position.

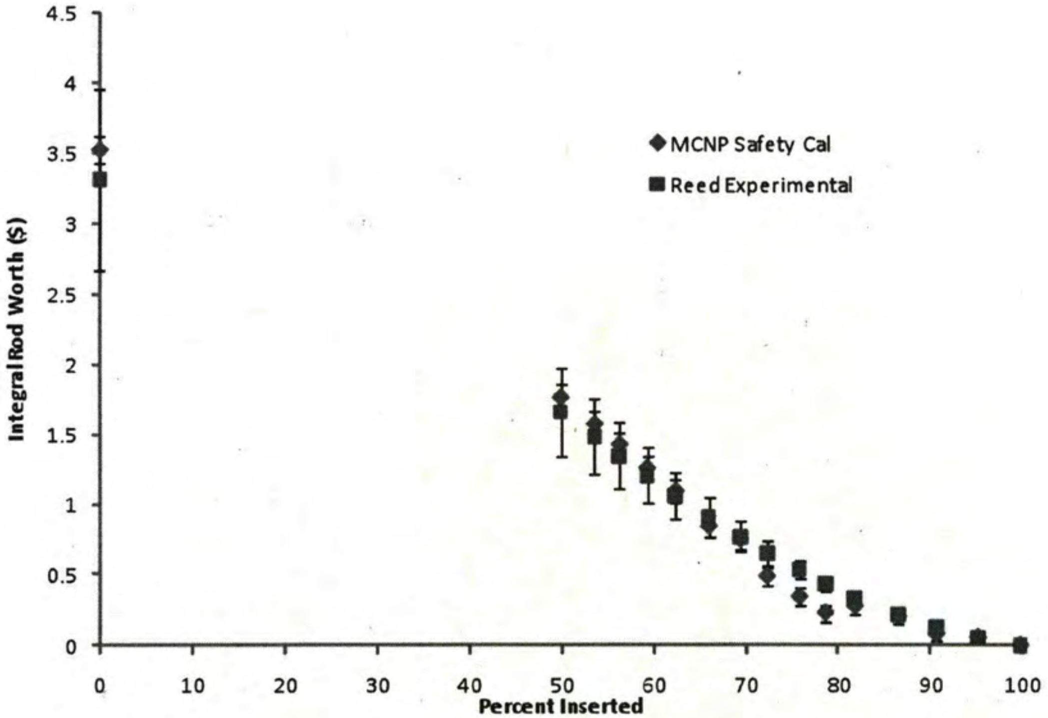

Figure 6 Safety Rod Worth for RRR Core

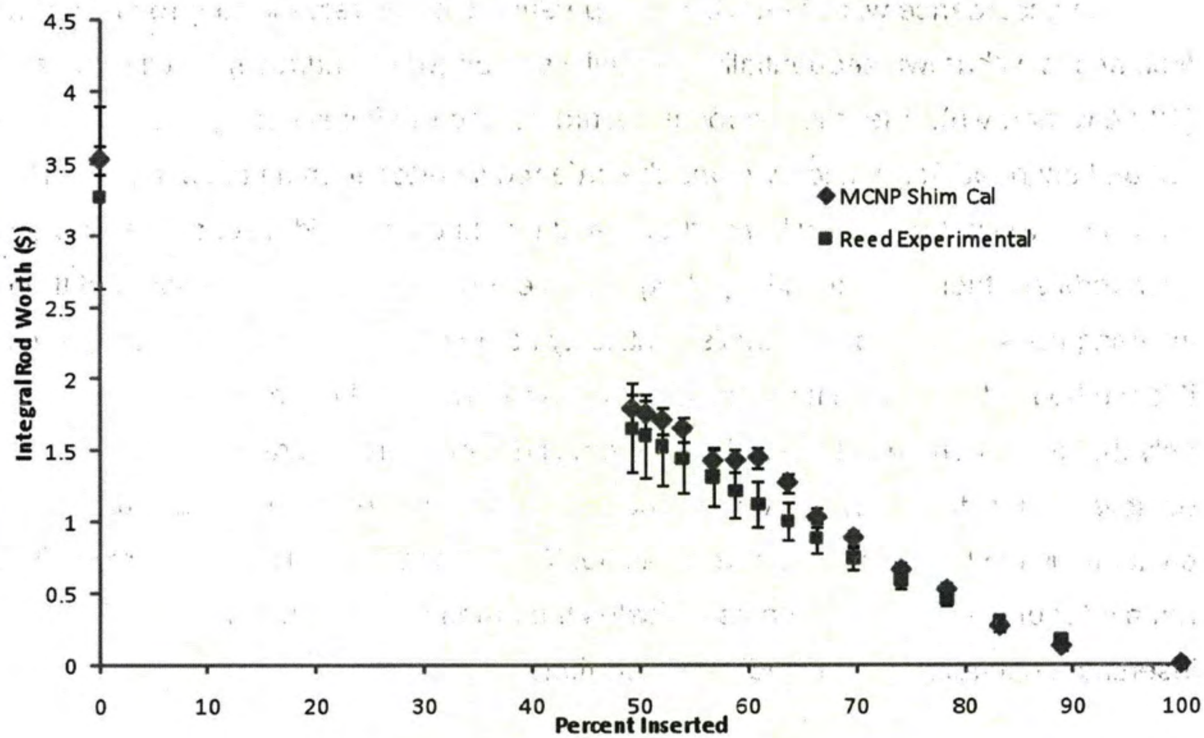

Figure 7 Shim Rod Worth for RRR Core

Table 5 Summary of Measured and Calculated Integral Control Rod Worth for the RRR

Control Rod	Measured Rod Worth [$]	MCNP5 Calculated Rod Worth [$]
Shim Rod	3.27 +/- 0.63	3.53 +/- 0.10
Safety Rod	3.31 +/- 0.64	3.52 +/- 0.10
Regulating Rod	1.34 +/- 0.22	1.06 +/- 0.09
Sum of all Rods	7.92 +/- 0.93	8.11 +/- 0.17

Shutdown margins for the RRR core were calculated using the MCNP5 model by fully inserting two rods with the remaining rod fully withdrawn. The results of the shutdown margin calculation are presented in Table 6. The calculated shutdown margin met the required Technical Specification shutdown margin (at the time of the start-up testing) of 0.4%Δk/k, or $0.53 shutdown with any rod fully withdrawn from the core. Shutdown margin values calculated directly using MCNP5 should be close to values inferred from individual rod worth data (measured or calculated), but it is not expected that they should be identical due to rod shadowing effects.

Table 6 Summary of Shutdown Margin Calculations for the RRR Core

Control Rod Fully Withdrawn	Shutdown Margin Calculated by MCNP5 model
Shim Rod	-2.54 +/- $0.04
Safety Rod	-2.70 +/- $0.04
Regulating Rod	-4.56 +/- $0.04

7. Fuel Prompt-temperature Coefficient

The prompt-temperature coefficient associated with the RRR fuel, α_F, was calculated by varying the fuel meat temperature while leaving other core parameters fixed. The MCNP5 model was used to simulate the reactor with all rods out at 300, 400, 600, 800, and 1200 K. The prompt-temperature coefficient for the fuel was calculated at the mid-point of the four temperature intervals, and the results were fit to a linear expression. The results are shown in Figure 8 and tabulated in **Table 7**. The prompt-temperature coefficient is observed to be negative for all evaluated temperature ranges with increasing magnitude as temperature increases. This behavior and range of magnitude compares favorably with results given in Simnad *et. al.*[4]

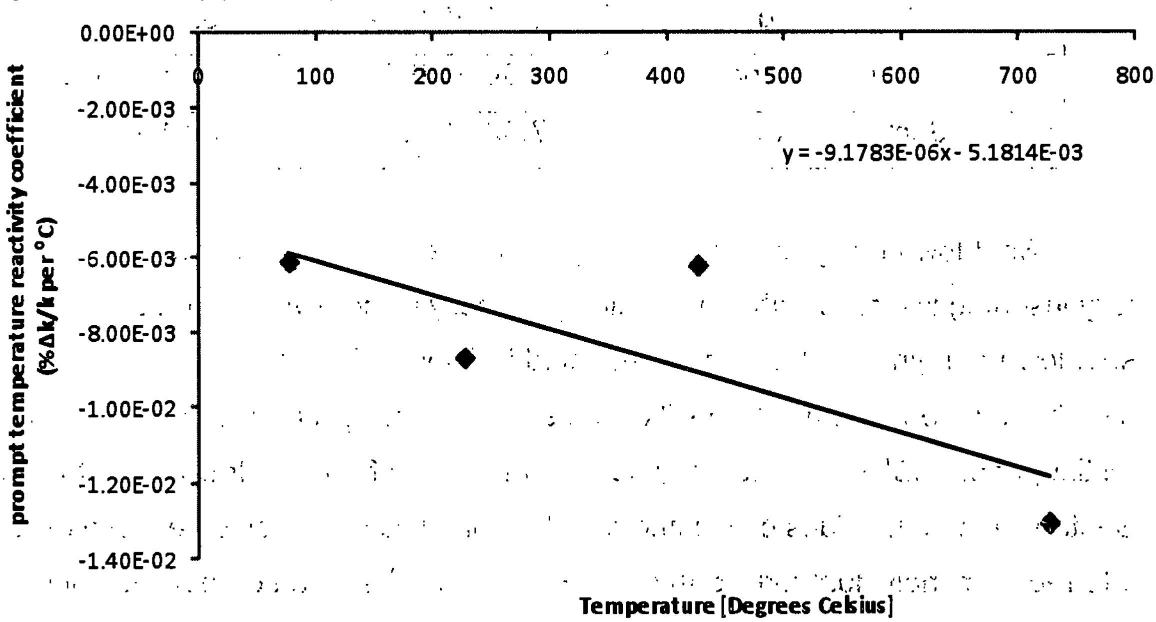

Figure 8 Prompt-Temperature Coefficient, α_F, as a Function of Temperature

Table 7 Prompt Temperature Coefficient

Temperature [°C]	Prompt Temperature Coefficient [%Δk/k/°C]
77	-6.11E-03
227	-8.70E-03
427	-6.22E-03
727	-1.31E-02

8. Moderator Void Coefficient

The moderator void coefficient of reactivity was also determined using the MCNP5 model. The voiding of the core was introduced by uniformly reducing the density of the liquid moderator in the entire core. The calculation was performed for several different voiding percentages (i.e., 100% to 75%, 100% to 50%, 75% to 50%, etc.). The void coefficient was negative for every interval and the average value was found to be -$0.83/% void.

9. Moderator Temperature Coefficient

The moderator temperature coefficient of reactivity, α_m, was determined by varying the moderator temperature within the MCNP5 model RRR core from 20°C to 60°C. Within this temperature range, the calculated moderator temperature coefficient of reactivity was -0.57¢/°C.

10. Core Power Distribution

The fuel element power distribution throughout the core is illustrated in Figure 9. For nominal operating power of 250 KW, the average fuel element power is 3.91 KW and the maximum element power is 7.24 KW (gridplate position B5).

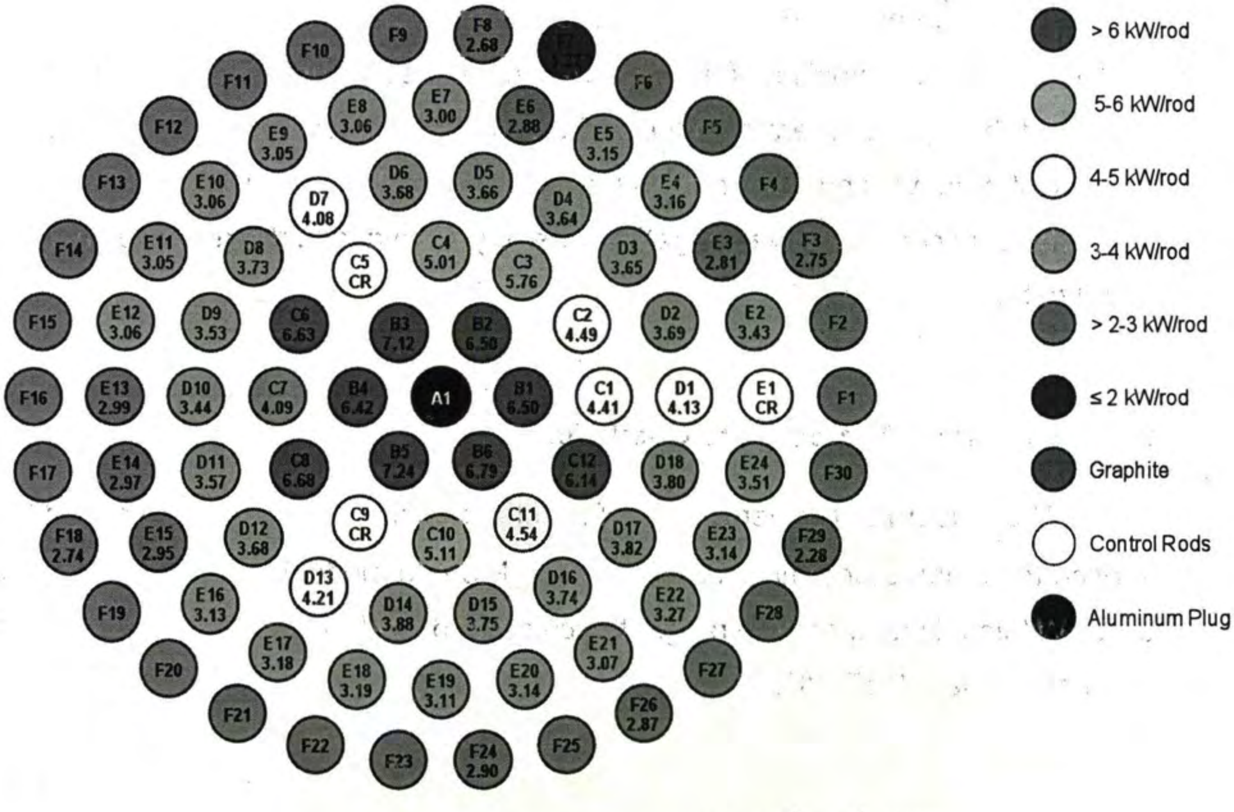

Figure 9 RRR Core Power Distribution

11. Summary

MCNP5 was used to calculate fundamental and operational parameters for the Reed Research Reactor. Values of fundamental parameters agree well with theoretical values. Values of operational parameters agree well with measured values. The results of this study indicate that the RRR can be operated safely within the Technical Specification bounding envelope.

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ANALYSIS OF THE THERMAL HYDRAULIC BEHAIVOR

OF THE

REED RESEARCH REACTOR

Submitted By:

Radiation Center
Oregon State University
Corvallis, Oregon

April 4, 2011

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1 Introduction

This report contains the results of investigation into the thermal hydraulic behavior of the Reed Research Reactor (RRR). The objectives of this study were to: 1) create a model of the RRR to study the thermal hydraulic characteristics, and 2) demonstrate acceptable reactor performance and safety margins for the RRR core under normal conditions.

2 Summary and Conclusions of Principal Safety Considerations

The conclusion of this investigation is that the thermal hydraulic model does an acceptable job of predicting behavior of the RRR core. As such, the results suggest that the RRR core can be safely operated within the parameters set forth in the technical specifications. Discussion and specific details of the analysis are located in the following sections.

3 Thermal Hydraulic Analysis

The TRIGA® system operating with cooling provided by natural convection water flow around the fuel elements was analysed. The predicted steady state thermal-hydraulic performance of the RRR was determined for the reactor operating at 250 kW with a water inlet temperature of 48.9°C. Operational data from the Oregon State University TRIGA® Reactor were used for benchmark comparisons. The maximum power fuel rod and maximum power heated subchannel were analysed for the RRR under steady-state conditions. The RELAP5-3D computer code (Ref. 1) was used to determine the Departure from Nucleate Boiling Ratio (DNBR) using the Bernath correlation critical heat flux tables. The power in the hottest rod at which critical heat flux is predicted to occur was also calculated, as well as the maximum fuel temperature in the hottest rod.

4 RELAP5-3D Model

The analysis was performed using a single flow channel divided into axial and radial segments (nodal distribution is described below). The RELAP5-3D model seen in Figure 1 consists of a Coolant Source, Cold Leg, Horizontal Connector, Hot Channel, and Coolant Sink. This model is representative of a single RRR core subchannel, assumed to be the hot channel.

The Coolant Source is modeled as a time dependant volume in RELAP5-3D allowing for an inlet pressure and temperature boundary condition to be imposed on the system during the analysis. The Cold Leg is incorporated into the RELAP5-3D model in order to create a pressure differential between the cold coolant entering the subchannel and the heated coolant passing through the subchannel. This pressure differential drives the natural circulation flow. The Horizontal Connector serves no physical purpose in the RRR, but is rather a nonphysical connector between the cold leg and hot channel to allow for communication between Volumes 101 and 103 during the computational process. The Hot Channel (Volume 103) is the volume which contains the fuel rod of a single RRR subchannel. It is assumed in the RELAP5-3D model that the hot channel has the most conservative thermal hydraulic parameters found in the RRR core and that it is located in the B ring.

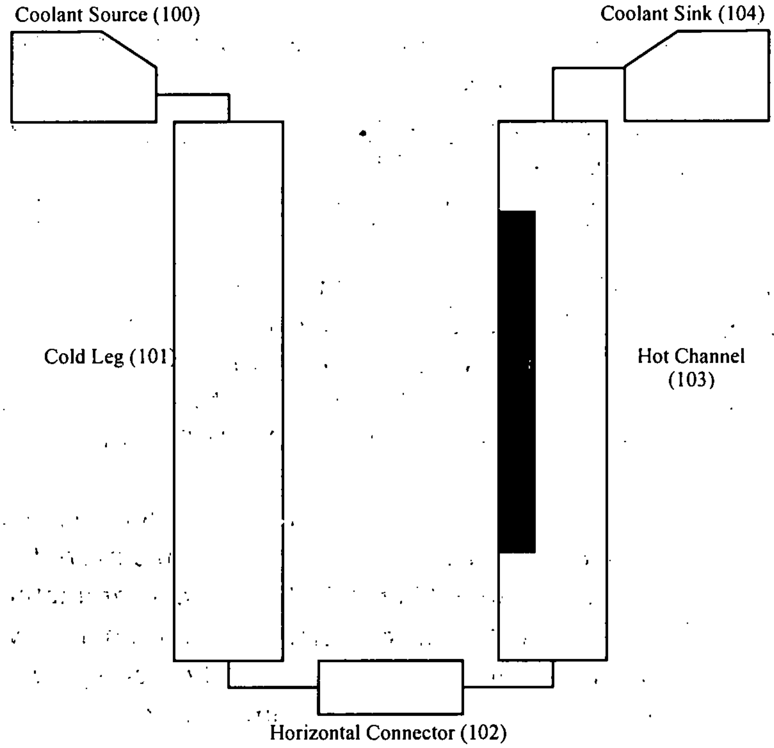

Figure 1 Single Channel RELAP5-3D Model Schematic

To simplify the RELAP5-3D model, it was assumed that there is no cross flow between adjacent channels so one channel would suffice. This assumption is conservative since higher values of temperature and lower margins to DNB are predicted when cross flow between adjacent channels is ignored. Furthermore, other work has shown that the single channel model provides critical heat flux results within ~1.0 % of those produced from two and eight channel models and that the single channel model produced the most conservative results relative to the two and eight channel models. (Ref. 4 and 8).

The reactor geometry and hydraulic data for the RELAP5-3D input are given in Table 1. The coefficients presented in Table 1 come from a study conducted by General Atomics which developed a methodology for calculating each effective subchannel form loss rather than local form losses within the core (Ref. 2). Flow channels in the RRR are triangular, square or irregular, depending on core location. The analysis assumes a triangular rod lattice configuration because the hottest flow channel is shown to occur adjacent to the A and B rings, and in this location, the lattice is triangular.

Table 1 RELAP5-3D Input for Reactor and Core Geometry and Heat Transfer

Hydraulic Data	
Inlet pressure loss coefficient	2.26
Exit pressure loss coefficient	0.63
Absolute pressure at the top of the core [Pa]	1.49E5

A constant pressure of 1.01E5 Pa (14.7 psia) is assumed to exist at the top of the reactor pool. The water column height above the top of the core was modelled as 4.88 m (16 feet), so this equivalent water column pressure boundary condition is used in the RELAP5-3D model. RELAP5-3D requires that input pressure conditions be entered as absolute pressure, therefore the input RELAP5-3D pressure used in the model at the top of the core is 1.49E5 Pa (21.639 psia).

The RELAP5-3D thermal hydraulic analysis was performed on the maximum powered channel. The analysis was conducted assuming (conservatively) that the maximum powered channel was also the most restrictive flow channel location found in the RRR. The flow parameters for the most restrictive flow channel are given in Table 2. The geometry of the maximum powered channel is shown in Figure 2. It is conservatively assumed that all fuel rods bordering the maximum powered subchannel are operating at the same power as the maximum powered rod.

Table 2 Hydraulic Flow Parameters for the Hot Channel

Parameter	Value
Flow area [m^2]	3.88E-04
Fuel Element Pitch [m]	0.04054
Wetted perimeter [m]	0.117
Hydraulic diameter [m]	1.301E-02
Heated diameter [m]	3.724E-02
Fuel element heated length [m]	0.381
Fuel element surface area [m^2]	4.469E-02
Fuel element surface roughness [m]	2.134E-06

The B Ring in the RRR contains the smallest pitch from fuel rod centerline to centerline, and also contains the smallest subchannel flow area. It is for this reason that the subchannel flow area for the RELAP5-3D model is calculated with reference to the B Ring subchannel flow area, depicted in Figure 2.

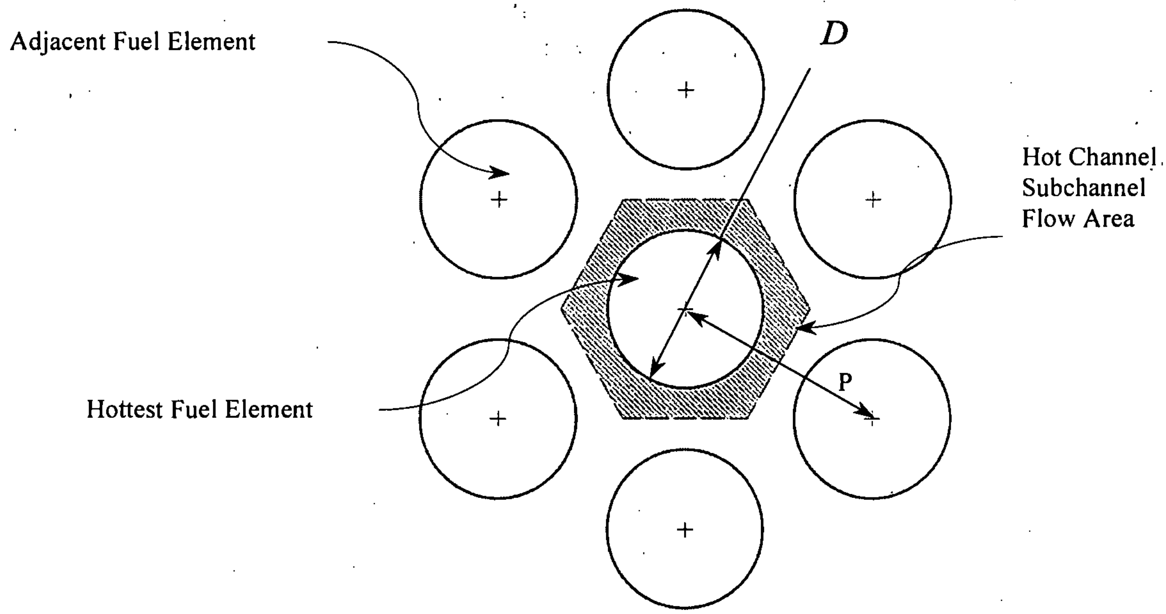

Figure 2 Hexagonal Array Axial Average unit subchannel dimensions

The pitch for the B Ring subchannel is 0.04054 m (1.596 in) (Ref. 4.51). The fuel rod outer diameter for the RRR core fuel rods is defined as 0.037 m (1.47 in). Equation (1) defines the subchannel flow area for a hexagonal array (Ref. 4).

$$A_{f\,i} = \frac{\sqrt{3}}{4}P^2 - \frac{\pi D^2_{\text{outer clad}}}{8} \tag{1}$$

where *P* represents fuel rod pitch and *D* represents the fuel rod outer diameter. From Equation (1) the subchannel flow area is calculated to be 1.74E-4 m^2 (0.2544 in^2).

The wetted perimeter for the subchannel only encompasses one half of an entire fuel rod in the figure above. Therefore the total flow area for the subchannel input into the RELAP5-3D model is 3.88E-4 m^2 (0.5088 in^2).

The axial length of the fueled portion of the fuel rod is 0.381 m (15.0 in) while the diameter of the outer cladding is 0.037 m (1.47 in). The total surface area of the fueled portion of the fuel rod is therefore 4.469E-2 m^2 (69.27 in^2). The wetted perimeter is defined as $P_{\text{wetted}} = \pi D_{\text{outer clad}}$. This equation produces a value of 0.117 m (4.618 in) for the wetted perimeter of a fuel rod.

The hydraulic diameter is calculated per Equation (2). With reference to the previously calculated wetted perimeter and subchannel flow area, the hydraulic diameter is calculated to be 1.327E-2 m (0.441 in).

$$D_h = \frac{4A_{fl}}{P_{wetted}} \qquad (2)$$

Figure 3 shows a comparison between a physical RRR fuel element and the RELAP5-3D discretized subchannel volume. Nodes 01 and 24 represent the lower and upper grid plates. The lower grid plate is 0.0191 m (0.75 in) thick. The upper gridplate is 0.0159 m (0.625 in) thick. The bottom surface of the upper grid plate is 0.6731 m (26.5 in) above the top surface of the lower grid plate. The fuel element axial nodal dimensions are given in Table 3.

[redacted]

Figure 3 Comparison of RRR Fuel Rod Axial Characteristics and RELAP5-3D Hot Channel

Node 02 extends from the bottom of the fuelled portion of the fuel rod to the top of the lower gridplate. The equation used to calculate the length of Node 02 (lower unheated node) is:

$$L_{02} = \frac{\left(0.6731 - L_{fuel} - L_{Upper\ graphite} - L_{Lower\ graphite}\right)}{2} + L_{Lower,graphite} \qquad (3)$$

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

The fuel nodal lengths must be discretized, and this can be done by use of Equation (4) for Nodes 03 through 22:

$$L_{03\rightarrow 22} = \frac{L_{fuel}}{n_{fuel}} \quad (4)$$

$L_{03\rightarrow 22}$ refers to the nodal length for Nodes 03 through 22, n_{fuel} is the number of nodes defined in the fuel region (i.e. 20 nodes).

Equation (5) is used to calculate the nodal length for Node 23 (upper unheated node).

$$L_{23} = \frac{\left(0.6731 - L_{fuel} - L_{Upper\ graphite} - L_{Lower\ graphite}\right)}{2} + L_{Upper,\ graphite} \quad (5)$$

Table 3 Hot Channel Axial Nodal Lengths

Core Volume Axial Nodal Lengths		
Nodal Description	Node Number	Nodal Length [m] (in)
Upper Grid Plate	24	0.01905 (0.75000)
Upper Graphite	23	0.14567 (5.73504)
Fuel	22	0.01905 (0.75000)
	21	0.01905 (0.75000)
	20	0.01905 (0.75000)
	19	0.01905 (0.75000)
	18	0.01905 (0.75000)
	17	0.01905 (0.75000)
	16	0.01905 (0.75000)
	15	0.01905 (0.75000)
	14	0.01905 (0.75000)
	13	0.01905 (0.75000)
	12	0.01905 (0.75000)
	11	0.01905 (0.75000)
	10	0.01905 (0.75000)
	09	0.01905 (0.75000)
	08	0.01905 (0.75000)
	07	0.01905 (0.75000)
	06	0.01905 (0.75000)
	05	0.01905 (0.75000)
	04	0.01905 (0.75000)
	03	0.01905 (0.75000)
Lower Graphite	02	0.14643 (5.76504)
Lower Grid Plate	01	0.01905 (0.75000)

A cross sectional view of a fuel element is shown in **Figure** 3. The radial nodal distribution is shown in Figure 4. The fuel portion of the fuel pin consists of an annular U/ZrH casting. The fuel slugs are hydrided and then forced into stainless steel tubes. The central void which aids the hydriding process is backfilled with a zirconium plug. A nominal gap exists between the fuel slug and the stainless steel clad. This gap is initially filled with air, but as burnup of the fuel progresses, hydrogen and fission gasses migrate into the gap. Note that the aluminium clad fuel elements do not have a central zirconium pin. Since the inner portions of the core contain stainless clad elements, only the stainless configuration was analysed. Fuel temperature gradient tends to be essentially flat in the center of a TRIGA fuel element. Aluminum clad elements also contain less fuel than stainless clad elements and thus generate less power. These two factors make it unlikely that the maximum temperature observed in an aluminum clad element would be significantly higher than that found in a stainless clad element.

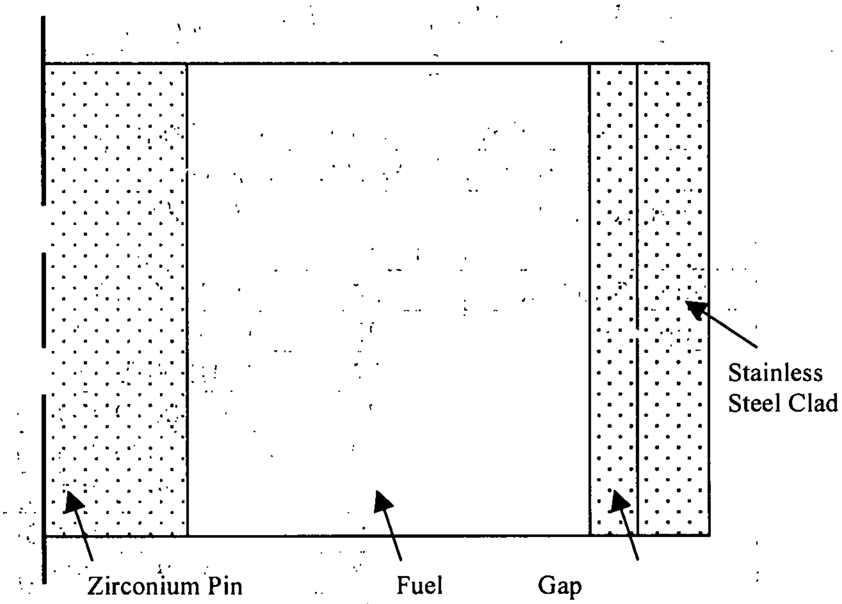

Figure 4 Cross Sectional View of Stainless Steel Clad Fuel Element

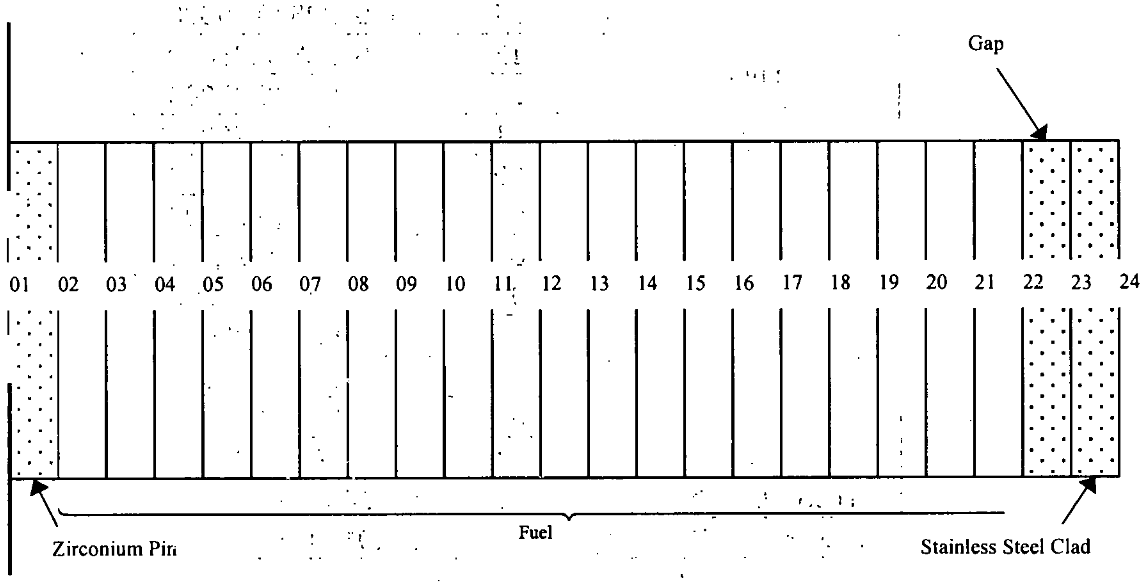

Figure 5 Radial Nodal Distribution in a Stainless Steel Clad Fuel Element

The mesh points within the fuel region used in the RELAP5-3D model correspond to one node for the central zirconium pin, twenty nodes of equal radial thickness for the fuel meat, one node for the fuel to clad gap and one node for the clad. The radial location of each node is identified in Table 4. The outer gap coordinate (Node 23) corresponds to a fuel-to-clad gap width of 2.54E-4 m (0.1 mils). RELAP5-3D input requires that radial mesh points be defined in order to specify all material properties and to calculate temperature gradients within the heat structure. All CHF correlations considered during the analysis use the heat flux of the outer cladding at a given nodal location. Because of this the outer cladding serves as both the wetted diameter and the heated diameter for the Bernath correlation ($D_H = D_{outer\ clad}$ = 3.724E-02 meters).

Power peaking factors for each core configuration were analyzed using MCNP5. The highest power rod for each configuration was determined by calculating the total power produced in each fuel element present in the configuration. After the highest power rod had been determined, further analyses were performed to find the detailed axial and radial power shapes associated with that rod. The axial and radial power shapes were determined for twenty equally spaced nodes in both the axial and radial directions. The MCNP5 results were used to calculate three peaking factors:

- Hot Channel Fuel Peak Factor = (maximum fuel rod power)/(core average fuel rod power)
- Hot channel Fuel Axial Peak Factor = (maximum axial power in the hot rod)/(average axial power in the hot rod)
- Hot Channel Fuel Radial Peak Factor = (maximum radial power in the hot rod)/(average radial power in the hot rod)

Table 4. Radial Fuel Element Nodal Locations (from fuel center)

Heat Structure Radial Node Lengths		
Nodal Description	Node Number	Node Coordinate [m] (in)
Inner Zirconium Pin	01	0.00000 (0.00000)
	02	0.00318 (0.12500)
Fuel	03	0.00355 (0.13976)
	04	0.00430 (0.16929)
	05	0.00506 (0.19921)
	06	0.00581 (0.22874)
	07	0.00656 (0.25827)
	08	0.00731 (0.28779)
	09	0.00807 (0.31772)
	10	0.00882 (0.34724)
	11	0.00957 (0.37677)
	12	0.01032 (0.40630)
	13	0.01108 (0.43622)
	14	0.01183 (0.46575)
	15	0.01258 (0.49527)

	16	0.01333 (0.52480)
	17	0.01409 (0.55472)
	18	0.01484 (0.58425)
	19	0.01559 (0.61378)
	20	0.01634 (0.64331)
	21	0.01710 (0.67323)
	22	0.01785 (0.70275)
Outer Gap	23	0.01785-0.01786 (0.70285-0.70305)
Outer Stainless Steel Clad	24	0.01873 (0.73750)

Average fuel rod power in the core, average axial power in the hot rod and average radial power in the hot rod must be properly calculated in order to obtain correct peak factors. The average fuel rod power in the core is calculated by taking the numerical average with each rod weighted equally. The fission rate in the hot rod is then calculated as discussed above and expressed in cylindrical (r,z) coordinates. The average axial power in the hot rod is calculated by taking the numerical average of the power density within each of the twenty axial segments.

The effective peak factor for each configuration is the product of these three individual peaking factors. The results of the MCNP5 analyses listing the location of the highest power rod along and its associated peaking factors are shown in Table 5 for the RRR core. Note that these peaking factors are calculated with control rods removed. This is conservative since the presence of control rods in the central regions of a core will result in flatter power distributions and lower hot channel peak factors.
It was assumed that all rods in the core have approximately the same axial heat distribution shape, and thus the maximum powered rod would produce the maximum local heat flux. Although minor variations in axial power shape occur throughout the core, we made the conservative assumption that the hot channel is bordered on all sides by a fuel rod having the same characteristics as the hot rod. In reality, the hot channel will likely be bordered by the hot rod, and two other rods of lower power. Thus even in the unlikely event that the maximum heat flux does not occur in the hot rod, the conditions in the hot channel are still expected to bound conditions at all other points in the core.

Using the Bernath (Ref. 3) correlation, CHF is defined in units of pound- centigrade per hr-ft^2 per the following equations.

$$CHF = h_{BO}\left(T_{w_{BO}} - T_b\right) \quad (6)$$

$$h_{BO} = 10890\left(\frac{D_h}{D_h + D_H}\right) + \Delta v \quad (7)$$

$$\Delta = \begin{cases} \dfrac{48}{D_h^{0.6}} & \text{if } D_h \le 0.1 \text{ ft} \\ \dfrac{10}{D_h} + 90 & \text{if } D_h \ge 0.1 \text{ ft} \end{cases} \tag{8}$$

$$T_{w_{BO}} = 57 \ln(P_{abs}) - 54\left(\frac{P_{abs}}{P_{abs} + 15}\right) - \frac{v}{4} \tag{9}$$

Where:

h_{BO}	Limiting film coefficient [p.c.u./hr-ft^2-°C]
T_b	Fluid bulk temperature [°C]
$T_{w_{BO}}$	Wall temperature at CHF [°C]
v	Fluid velocity [ft/sec]
Δ	"slope"
P_{abs}	Absolute pressure [psi]
D_H	Heated diameter [ft]
D_h	Hydraulic diameter [ft]

The Bernath correlation was used in this analysis because, (1) it is traditionally used in research reactor SARs, and (2) the correlation produces the most limiting CHFR values over all other correlations [Ref. 4, 5, and 7].

5 Fuel Meat to Cladding Gap Size and Content

The content of the gap gases was chosen to be the default setting for RELAP which assumes a mixture of He, Kr, and Xe at molar fractions of 0.1066, 0.134 and 0.7594, respectively. Although the backfill gas at the beginning-of-life for TRIGA® fuel is air, the content at later times in core life is unknown. Because of the difference in thermal conductivity between the gas mixture and air is different, the default RELAP mixture will produce higher fuel temperatures and is therefore conservative. Although the precise gap thickness is unknown, previous work suggests that the 0.1 mil gap value used is likely appropriate (Ref. 4 and 7). All results are based on a 0.1 mil fuel to clad gap.

6 Power Distribution

The RRR core power distribution, as well as the intra-fuel relative power distributions (radial and axial distribution in the hot rod) are provided in Figure through 8 (Ref. 8). Power distribution diagrams are used to derive Hot Channel Peak Factors. Axial power profiles are used to derive Hot Channel Fuel Axial Peak Factors. Radial power profiles are used to calculate Hot Channel Fuel Radial Peak Factors. The hot channel peak factor, axial power distribution and radial power distribution are used as RELAP5-3D inputs. A summary of all peak factors is given in Table 5.

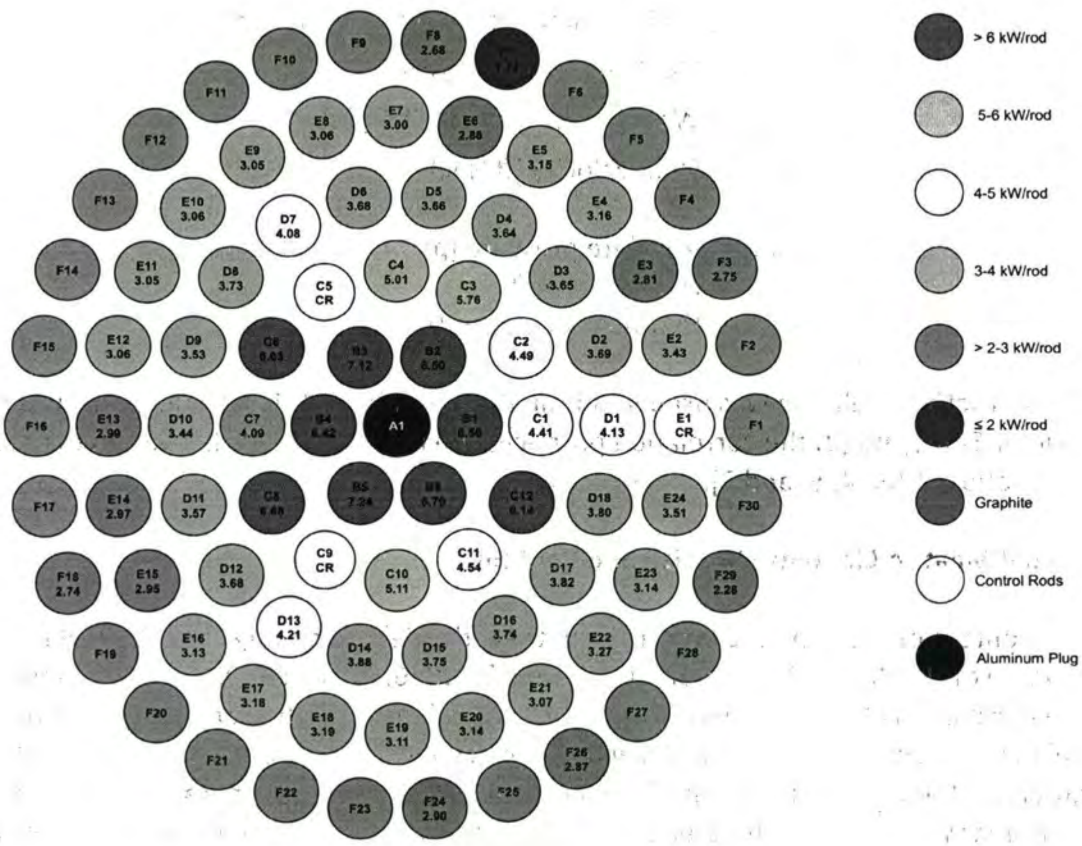

Figure 6 Core Power Distribution

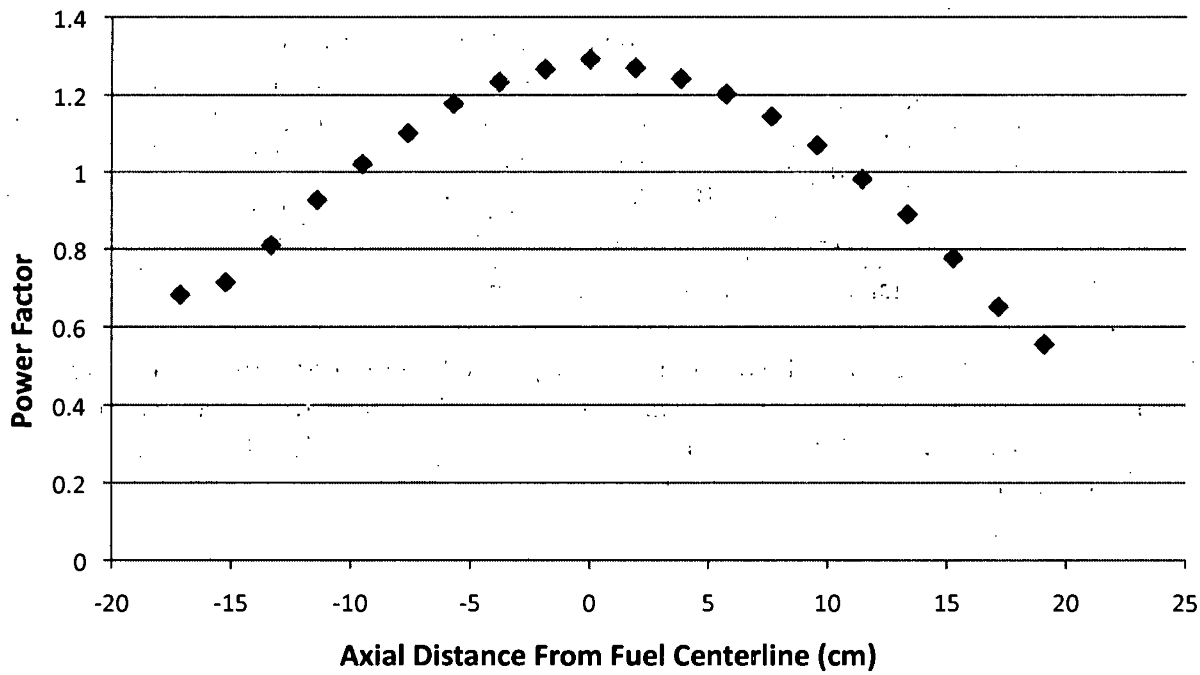

Figure 7 Axial Power Profile vs. Distance from Fuel Centerline

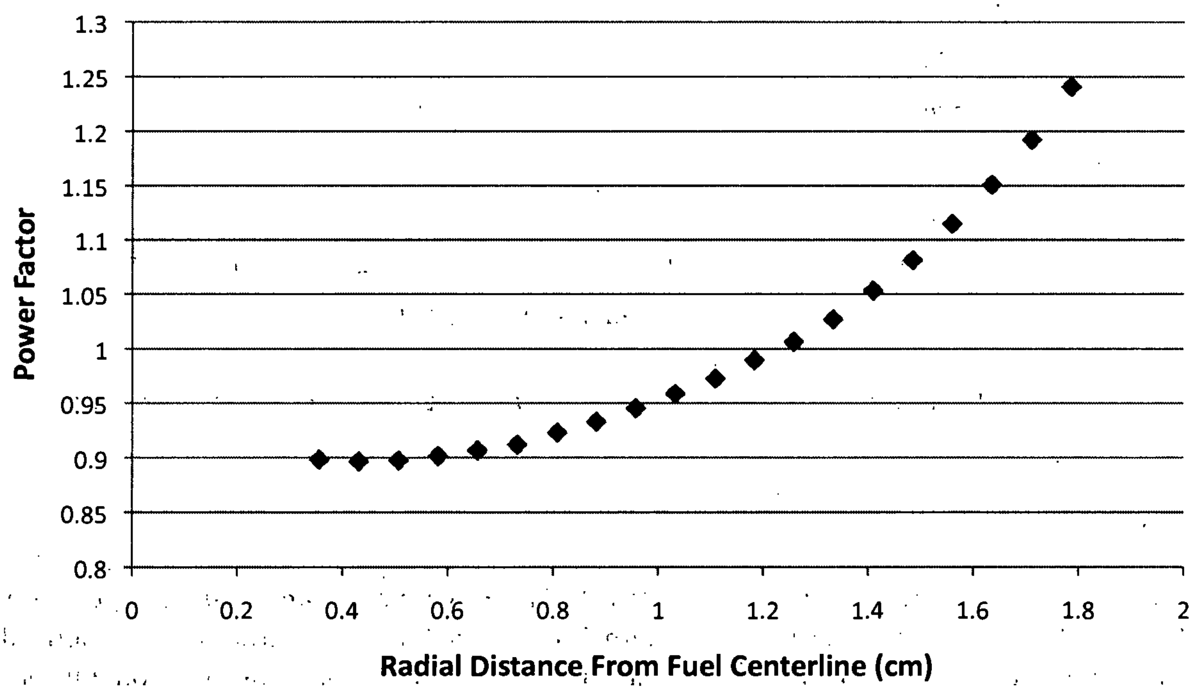

Figure 8 Radial Power Profile vs. Distance from Fuel Centerline

Table 5 Hot Channel Fuel Power Summary

Hot Rod Location	Hot Channel Fuel Thermal Power [kW]*	Hot Channel Fuel Peak Factor [P_{max}/P_{avg}]	Hot Channel Fuel Axial Peak Factor [P_{max}/P_{avg}]	Hot Channel Fuel Radial Peak Factor [P_{max}/P_{avg}]	Effective Peak Factor
B5	7.24	1.844	1.291	1.240	2.952

* Hot rod thermal power corresponds to core power of 250 kW.

An important consideration is the maximum steady state fuel temperature as a function of fuel element power. An analysis was performed with RELAP5-3D where the power was increased and the resulting maximum fuel temperature was determined. The results, shown in Figure , demonstrate the expected linear relationship.

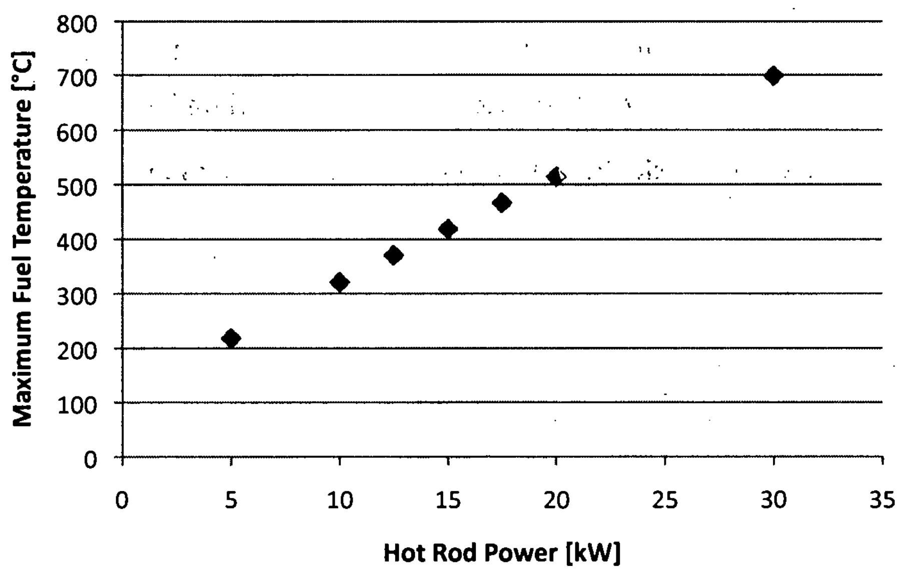

Figure 9 Maximum Fuel Temperature as a Function of Fuel Element Power

7 Predicted DNBR

The RRR core configuration modeled has a MDNBR of 6.33 with a maximum fuel temperature of 264°C at 250 kW steady state using the Bernath Correlation. Using the Bernath correlations, the RRR core therefore is operating at power well below that required for departure from nucleate boiling. The value of MDNBR is consistent with values reported for other TRIGA research reactors (Ref. 4 and 6).

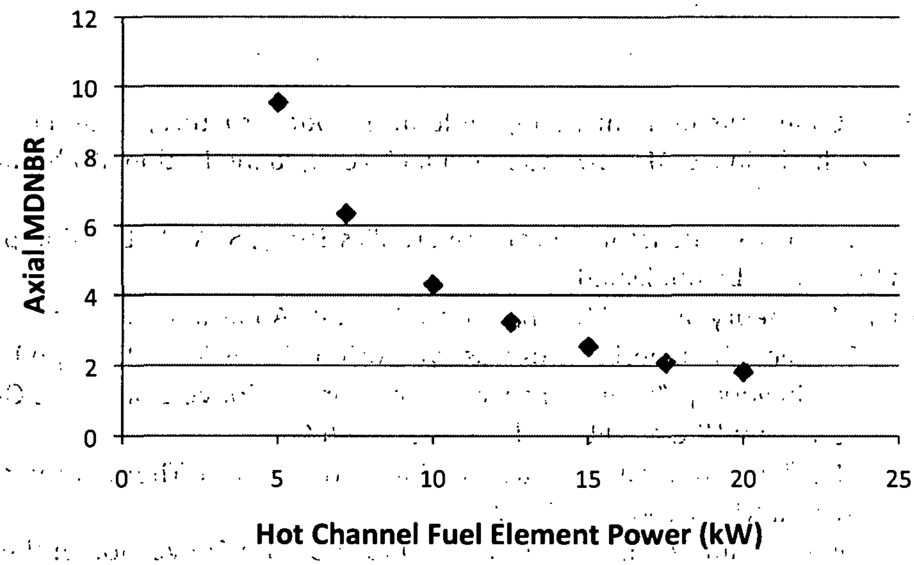

Figure 10 Hot Channel MDNBR

8 Summary

RELAP 5-3D was used to calculate fundamental thermal hydraulic parameters for the RRR. Values of these fundamental parameters agree well with values calculated and measured for other similar research reactors (Ref. 6 and 7). This analysis looked at the maximum license power of the RRR for steady state operations of 250 kW. Figure and Figure clearly show that the RRR can be operated safely within the Technical Specification bounding envelope.

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file TRIGA

To: Distribution

Date: March 1, 1966

From: Fabian C. Foushee FCF

Subject: Storage of TRIGA Fuel Elements

Introduction:

In GA-5402, "Criticality Safeguards Guide", general limits on the various dimensions, and concentrations of fissile material that must be imposed on any unit of such material (see p. 18, Table 1) are set forth. A further, very general limitation on the storage of well moderated U-235 is given (on p. 22) as an average of 300 gms of U-235 per square foot of aspect area. This latter limitation has been cited as evidence that the TRIGA fuel storage racks located in the reactor pool are indeed safe. In these racks, 10 TRIGA fuel elements are arranged in line with 2 inches separating the axial centerline of adjacent elements. Assuming, at the very most, [redacted] of U-235 per element, the concentration in the array is 197 gms of U-235/ft^2. This storage system, then, is certainly safe as the concentration for this finite array is only 2/3 of the recommended maximum safe value for an array that extends to infinity in two dimensions.

Recently, it has been suggested that it would be more appropriate to calculate the multiplication of a system containing fissile material as this would be a more meaningful measure of the safety of the system. Consequently, there have been completed some GAZE calculations of the TRIGA in-pool fuel storage racks which do, indeed, show a considerable margin of safety.

In the GAZE calculations to be described a set of six group (3 fast and 3 thermal) cross-sections derived for an 8.0 wt.-% U-ZrH (20% enriched U) system was used. These cross-sections were calculated for an homogeneous system with volume fractions of the various component materials as in TRIGA. The results of the calculations are consequently conservative as lumping the fuel (as in a fuel element) will lower k_{eff} because of self-shielding. The last statement must be qualified by noting that lumping the fuel has an advantageous effect in that the effective U-238 resonance integral is reduced. To examine the effect of this parameter on the multiplication of the system, two GAZE calculations were performed that were identical except one used U-238 cross-sections appropriate to an infinite dilution. From these calculations one could infer that using an infinite dilution U-238 resonance integral decreases k_{eff} by about 1.5%.

TRIGA Spent Fuel Storage

TRIGA fuel elements are stored in the reactor pool in racks fastened to the side of the pool. Each storage rack accommodates ten elements, the axial centerlines of which are two inches apart and lie in a single plane. The homogeneous model employed to describe the storage rack was an infinite plane 1.47 inches thick (the diameter of a fuel element) with 42.3% of the volume occupied by water (corresponding to the two inch separation distance). The nuclear densities of the constituents are given in Table I.

Table I

Composition of TRIGA Fuel Storage Rack System

Constituent	Nuclear Density x 10^{-24}
H-in $ZrH_{1.65}$	3.088×10^{-2} nuclei/cm^3
H-in H_2O	2.614×10^{-2}
Oxy	1.307×10^{-2}
Zr	1.939×10^{-2}
U-235	1.278×10^{-4}
U-238	5.111×10^{-4}
Stainless Steel	2.64×10^{-3}

Two calculations were performed, one for a single rack and one for two racks back-to-back. The two rack calculation assumed that the fuel elements touched (i.e., the mid-plane to mid-plane separation distance between adjoining racks was only 1.47 inches). Actually, the fuel storage racks, as presently constructed, cannot come closer together than about 2.5" center-to-center. Thus, there would be about 1 inch of water between racks providing significant de-coupling. The results of these calculations are shown in Table II.

Table II

	k_{eff}
Plane array one element thick	0.5096
Plane array two elements thick	0.7227

It should be noted that these calculations were made for 8 wt.-% uranium. Increasing the uranium content to 8.5 wt.-%, as in the Torrey Pines TRIGA Mark III, would increase k_{eff} by 2 to 3%, or to .53 and .74 for the one and two element arrays of Table II. These heavier elements contain something less than [redacted] gms of U-235 each. In the one element storage array there results a concentration of about [redacted] gms of U-235 per square foot of aspect; in the two element thick storage array, about [redacted] gms per square foot. Therefore one can conclude that whereas the one element thick storage array provides an areal concentration below the maximum prescribed (i.e., [redacted] gms - U-235/ft^2), the two element thick array results in a k_{eff} that is less than 0.8, a value found acceptable to the AEC Division of Licenses and Regulations, in the TRIGA Mark III Technical Specifications.

Chapter 14

Technical Specifications

Reed Research Reactor Safety Analysis Report

05/19/11

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Technical Specifications
Table of Contents

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**: A qualitative 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 which is used to establish neutron flux changes and to compensate for routine reactivity changes. A control rod may be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged. Types of control rods shall include:

a. **Regulating Rod** (Reg Rod): The regulating rod is a control rod having an electric motor drive and scram capabilities. Its position may be varied manually or by the servo-controller.

b. **Shim/Safety Rod**: A shim/safety rod is a control rod having an electric motor drive and scram capabilities. Its position is varied manually.

1.8 **Core Lattice Position**: A particular hole in the top grid plate of the core. It is specified by a letter indicating the specific ring in the grid plate and a number indicating a particular position within that ring.

1.9 **Excess Reactivity**: That amount of reactivity that would exist if all control rods were moved instantaneously to the maximum reactive condition from the point where the reactor is exactly critical ($k_{eff} = 1$) at reference core conditions.

1.10 **Experiment**: Any operation, hardware, or target (excluding devices such as detectors or foils) that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within an irradiation facility. Hardware rigidly secured to a core, shield, 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. **Secured Experiment**: Any experiment or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible malfunctions.

b. **Unsecured Experiment**: Any experiment or component of an experiment that does not meet the definition of a secured experiment.

c. **Movable Experiment**: A movable experiment is one where it is intended that the entire experiment maybe moved in or near the core or into and out of the core while the reactor is operating.

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

1.12 **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.

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

1.14 **Operable**: A system or component is operable when it is capable of performing its intended function.

1.15 **Operating**: A system or component is operating when it is performing its intended function.

1.16 **Operational Core**: A fuel element core which operates within the licensed power level and satisfies all the requirements of the Technical Specifications.

1.17 **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.

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

1.19 **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.

1.20 **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.

1.21 **Reactor Shutdown**: The reactor is shutdown if it is subcritical by at least $1.00 in the reference core condition with the reactivity worth of all installed experiments included and the following conditions exist:

a. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless the control rod drives are physically decoupled from the control rods;

b. No experiments are moved or serviced that have, on movement, a reactivity worth exceeding $1.00.

1.22 **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). Secured experiments can change the reference core conditions.

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

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

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

1.26 **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.

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

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

1.29 **Surveillance Intervals**: Allowable surveillance intervals shall not exceed the following.

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. Quarterly - interval not to exceed 16 weeks.
e. Monthly - interval not to exceed 6 weeks.
f. Weekly - interval not to exceed 10 days.

1.30 **True Value**: The actual value of a parameter.

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 500 kW.

Basis. The limiting parameter for the fuel is its temperature. Since Reed does not have any instrumented fuel elements, the safety limit is set on thermal power to limit fuel temperature. The *Analysis of the Thermal-Hydraulic Behavior of the Reed Research Reactor* (*RRR T-H Analysis)* performed in 2010, shows that at 500 kW the maximum fuel centerline temperature is ≈ 410°C. Since the temperature limit for TRIGA® fuel is in excess of 1000°C, this safety limit is acceptable. *RRR T-H Analysis* also shows that the Departure from Nucleate Boiling Ratio (DNBR) for the reactor at 500 kW is ≈ 2.5, which is above the minimum acceptable value of 2.0.

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. *RRR T-H Analysis*, shows that at 275 kW the maximum fuel centerline temperature is ≈ 300°C and the DNBR is ≈ 4.5 which is acceptable as discussed in 2.1. NUREG-1537, Appendix 14, Section 2.2 indicates that the LSSS may be 10% to 20% above the licensed power. 10% (275 kW) was chosen as the more conservative setting.

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. *RRR T-H Analysis*, shows that at 250 kW the maximum fuel centerline temperature is ≈ 264°C and the DNBR is ≈ 6.33 which is acceptable as discussed in 2.1.

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 shutdown 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 $0.50 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 shutdown 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.

The typical rod worths are $3.37 (Safety), $3.27 (Shim), and $1.34 (Regulating) with a total worth of $7.98. At critical conditions at 5 W the typical core excess is $1.65 with $0.65 (Safety), $0.65 (Shim), and $0.35 (Regulating). With a typical core excess and the most reactive control rod stuck out, the reactor will be subcritical by $7.98 - $1.65 - $3.37 = $1.96. With the maximum allowable core excess the reactor would be shutdown by $7.98 - $3.00 - $3.37 = $1.61 which is still greater than the $0.50 minimum.

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 shutdown 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 $3.00.

Basis. This core excess limit allows operation without the need to add or remove fuel elements to account for normal reactivity changes due to fission product poisons, experiments, power defect, fuel burn up, etc. Activities such as moving away from the reference state or adding negative worth experiments will make core excess more negative and shutdown margin less positive. The only activity which 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.

Power defect at full power adds approximately $1.33 of negative reactivity, and equilibrium xenon at full power adds approximately $1.06 of negative reactivity. A core excess of $3.00 leaves approximately $0.61 for the worth of experiments during extended operations.

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 not 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.
d. The interlocks in Table 3 of Section 3.2.3 of these TS are not operable.

Basis. This specification assures that the reactor shall be promptly shutdown 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. *RRR T-H Analysis* shows that the limit on reactivity addition rate is safe during normal operation and transients. The interlocks which prevent the simultaneous withdrawal of more than one control rod are addressed in the basis of Section 3.2.3.

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 all of the reactor power measuring channels in Table 1 are operable.

Table 1 – Power Measuring Channels

Safety Channel
Percent Power Channel
Linear Channel
Logarithmic Channel

Basis. Reactor power displayed at the control console gives continuous information on this parameter that has a specified safety limit.

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 2 and interlocks described in Table 3 are operable.

Table 2 - Minimum Reactor Safety Channels

Safety Channel	Function	Minimum Number
Power Level Scram	Scram at 275 kW or less	2
Loss of High Voltage Scram	Scram at less than 90% of nominal.	2
Console Manual Scram	Scram	1

Table 3 - 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

Basis. Power Level Scram: *RRR T-H Analysis*, shows that at 275 kW the maximum fuel centerline temperature is ≈ 300°C and the DNBR is ≈ 4.5 which is acceptable as discussed in Section 2.1 of these TS. The set points for both the Linear Channel and Percent Power Channel are normally set to 100% of 250 kW, which is the licensed power. Either channel will scram when its detector high voltage is less than 90% of nominal voltage since the channel is unreliable without proper high voltage.

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 manually removing multiple control rods simultaneously so that reactivity insertions from control rod manipulation are done in a controlled manner.

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 chemistry will 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 40°C;
c. The conductivity of the pool water shall be less than 5.0 microSiemens/cm.
d. The pH of the pool water shall be between 5.0 and 7.5.
e. The activity of the pool water shall be less than the limits in 10 CFR 20 Appendix B Table 2, Column 3.
f. If pool level decreases more than 10 cm below the normal pool level the cause shall be investigated.

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. The temperature limit also ensures the core inlet temperature is acceptable for the accident analysis. Experience at many research reactor facilities has shown that maintaining the conductivity and pH within the specified limit provides acceptable control of corrosion (NUREG-1537 Appendix 14, Section 3.3.(9)). Pool activity is limited to ensure dose rates are below 2 mrem/hour. Pool level is limited to a decrease of no more than 10 cm below normal to allow early detection of pool leakage.

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.

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. In the isolation mode it is much lower.

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 Continuous Air Radiation Monitor are operating.

Exception: When a single required radiation monitoring channel becomes inoperable, operations may continue only if portable 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. Calculations show that for both routine operations and accident scenarios predicted occupational and general public doses are below the applicable annual limits specified in 10 CFR 20.

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 1.0 mrem. 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 $1.00;
b. The sum of the absolute values of the reactivity worths of all experiments shall be less than $2.00.

Basis. The reactivity limit of $1.00 for movable experiments is designed to prevent an inadvertent prompt critical condition from occurring from an analyzed condition and maintain a value below the shutdown margin. Movable experiments are by their very nature experiments in a position where it is possible for a sample to be inserted or removed from the core while critical. The reactivity worth limit for all experiments is designed to prevent an inadvertent prompt critical condition. This limit applies to movable, unsecured, and secured experiments. A maximum reactivity insertion of $2.00 is acceptable because reactivity additions of $3.00 were analyzed in the SAR and shown to be safe.

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 from 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 Section 3 specification will establish the minimum performance level and a companion Section 4 surveillance specification will prescribe the frequency and scope of surveillance to demonstrate such performance.

Specifications.

a. Surveillance requirements may be deferred during reactor shutdown (except section 4.3); 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 Operations Committee. A system shall not be considered operable until after it is successfully tested.

Basis. This specification relates to surveillances of 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 prior to each day's operation or prior to each operation extending more than one day, 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 in 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 prior to each day's operation or prior to each operation extending more than one day.
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.
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 prior to each day's operation or prior to each operation extending more than one day.
b. A channel check of the water temperature monitor shall be performed prior to each day's operation or prior to each operation extending more than one day.
c. A channel test of the water level monitor and alarm shall be performed quarterly.
d. A channel test of the water temperature monitor and alarm shall be performed quarterly.
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 weekly.
h. The water pH shall be measured quarterly.
i. The activity of the pool water shall be measured quarterly.
j. The volume of water added to the pool shall be recorded and checked weekly.

Basis. Experience has shown that the frequencies of checks on systems that monitor reactor primary water level, temperature, pH 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. The ventilation will be verified to be operating in normal or isolation mode prior to each day's operation or prior to each operation extending more than one day
b. A channel test of the reactor bay confinement ventilation system's ability to be in isolation (exhaust through a HEPA filter and maintain a negative pressure in the reactor bay with respect to the control room) shall be performed quarterly.

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 prior to each day's operation or prior to each operation extending more than one day.
b. A channel test shall be performed quarterly.
c. A channel calibration shall be performed annually.
d. The average Ar-41 effluent concentration calculation 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 of these TS by the Reactor Operations Committee in accord with Section 6.5 of these TS 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 Operations 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 site boundary is that boundary extending 250 feet in every direction from the center of the reactor.
b. The restricted area is that area inside reactor facility. The unrestricted area is that area outside the reactor facility.
c. The reactor bay shall have a free air volume of 300,000 liters.
d. 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.
e. Controls to place the ventilation system in the isolation mode shall be available in the control room.

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). Control of the ventilation system is available from the control room, which will be habitable even during the MHA.

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 bulk pool water temperature alarm shall be provided to indicate high bulk water temperature if the temperature exceeds 40°C.
d. A pool level alarm visible outside the reactor facility shall be provided to indicate if the level decreases more than 10 cm below normal.

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).

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.

The pool level alarm is to allow timely detection of pool leaks. Visibility outside the facility allows it to be monitored by periodic security patrols.

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 stainless steel clad 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, or central thimble.
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 ZrH_x): between. 0.9 and 1.65;
c. Cladding: stainless steel, 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.

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). See Foushee's memo on *Storage of TRIGA Fuel Elements* dated March 1, 1966.

6 ADMINISTRATIVE CONTROLS

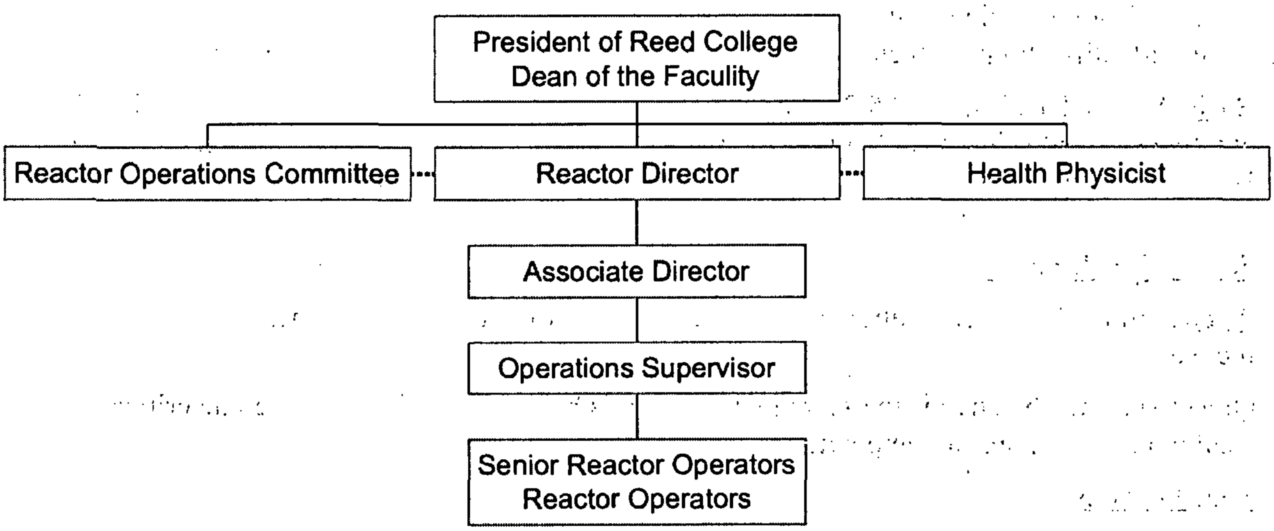
6.1 Organization

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. The Health Physicist shall communicate with the Reactor Director regarding health physics issues.

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. A list of reactor facility personnel by name and telephone number shall be readily available in the control room for use by the operator. The list shall include:
   1. Reactor Director;
   2. Reactor Associate Director;
   3. Operations Supervisor;
   4. Reactor Health Physicist;
   5. At least one other person who is a licensed SRO.

c. Events which require the presence of an SRO in the facility shall include:
   1. Initial criticality and approach to power of the day;
   2. All fuel or control rod relocations in the reactor core;
   3. Maintenance on any reactor safety system;
   4. Recovery from unplanned reactor scram or significant power reduction;
   5. 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 Operations Committee (ROC) shall have primary responsibility for review and audit of the safety aspects of reactor facility operations. Minutes, findings, or reports of the ROC shall be presented to the President and the Director within ninety days of completion.

6.2.1 ROC Composition and Qualifications

The ROC 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.

6.2.2 ROC Rules

The operation of the ROC shall be in accordance with written procedures including provisions for:

a. Meeting frequency (at least twice per year);
b. Voting rules;
c. Quorums (not fewer than half of the voting members);
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 ROC Review Function

The responsibilities of the ROC, 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 of these TS;
g. Review new experiments under Section 6.5 of these TS;
h. Review audit reports.

6.2.4 ROC Audit Function

The ROC, 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 ROC. 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 ROC 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 ROC 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 shutdown 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, the Chair of the ROC, and the President of Reed College;
c. A report, and any applicable followup report, shall be prepared and reviewed by the ROC. 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 ROC 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;
b. The number of unplanned shutdowns, including reasons for them;
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 the reactor is immediately shutdown;
   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 Operations 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 reactor operators and senior reactor operators shall be retained at all times the operator is employed or licensed at the facility.

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.