PENNSTATE

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September 1, 2009

US Nuclear Regulatory Commission
ATTN: Mr. William Kennedy, Project Manager
Office of Nuclear Reactor Regulation
Mail Stop O12-G13
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

Reference: Pennsylvania State University Breazeale Nuclear Reactor
Docket No. 50-005, License No. R-2
USNRC Request for Additional Information (RAI) dated July 28, 2009

Subject: Response to RAI dated July 28, 2009

Dear Mr. Kennedy:

The attachment to this letter answers the questions presented in the RAI dated September 1, 2009. The response and attachments do not contain any sensitive information.

If there are any questions regarding the information submitted, please contact Mr. Mark Trump, Associate Director for Operations at the RSEC. I declare under penalty of perjury that the foregoing is true and correct.

Executed on September 1, 2009.

Sincerely yours,

Kenan Ünlü
Director, Radiation Science and Engineering Center
Professor, Department of Mechanical and Nuclear Engineering

cc: E.J. Pell (w/o)
A.A. Atchley (w/o)
M.A. Trump

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PSBR response to NRC RAI

Dtd. 07/28/09

1. **Provide clarification as to whether the peak-to-measured pulse fuel temperature ratio of 1.6 accounts for the power peaking effects caused by vacant core positions (i.e., water holes).**

Recent analytical modeling work performed in support of the use of LEU fuel in TRIGA reactors has raised questions concerning TRIGA fuel temperature performance during pulsing. These issues were originally satisfied by extensive testing done during the TRIGA development and deployment stage in the 1960's.[1] In the 1970s and 80s, PSU and others conducted tests on 8.5 and 12 wt% elements that builds on the earlier work and formed the basis for the limiting reactor pulse ($3.50) in the PSBR SAR.[2,3] Review of the development work for the SAR shows the tests were conducted with new and used instrumented 12 wt% elements in multiple locations (including next to a water hole) with the element rotated to indicate maximum temperatures. The limiting case for the PSBR SAR analysis involved new 12 wt% fuel elements. 8.5 wt% and used (~8000 MWD/MTU) 12wt% elements were shown to have peak temperatures that do not approach or challenge the fuel safety limit based on lower "in-pin" peaking factors of ~1.1 instead of 1.6. for the 12 wt% elements. Therefore PSU believes the effects of the waterhole peaking are adequately represented in this work and the analysis to ensure fuel safety.

However, the discussion and documentation in the available references is insufficient to provide the level of assurance needed to meet today's expectations for analytical verification without the involvement of the vendor and development of additional PSBR specific neutronic and thermo-hydraulic models. Development of such analysis is not warranted considering the remaining number of new 12 wt% elements and fuel management methodology used at PSBR. A restriction on the use of new 12 wt% elements will eliminate the question raised by the aforementioned recent water hole peaking analysis. Future fuel procurement at PSBR will be 20 and 30 wt% standard LEU TRIGA elements that have been well analyzed.[4] Use of these standard fuels at PSBR will require additional site-specific analysis (in accordance with NUREG-1282) and a license amendment.

Therefore PSU requests a change to the facility technical specifications that restricts the use of new (less than 8000 MWD/MTU) 12 wt% fuel elements adjacent to a water hole (open fuel element position in the contiguous core array not filled with fuel, control rod, graphite, or experiment that displaces water).

Additionally, provide a discussion of requirements in the technical specifications (TS) that ensure the peak-to-measured pulse fuel temperature ratio of 1.6 is conservative for any allowed core configuration.

The 1.6:1 ratio of maximum to measured fuel temperature (which is only applicable during a pulse) was calculated in a 1974 paper by then Director- Breazeale Nuclear Reactor Dr. Samuel Levine (et al.).[2] It was the more conservative of two analyses that expanded the work of Goodwin on 8 wt.% TRIGA fuel.[3] The analysis that the authors considered more likely, set the ratio of temperatures at 1.45:1. The use of the higher ratio institutionalized a conservative bias of 1 – (1.60/1.45) = 10% to the pulse temperatures.

PSBR response to NRC RAI

Dtd. 07/28/09

In order to maintain the assumptions in the SAR analysis, the following technical specifications are in place at Penn State.

1.) **TS 2.2: Limited Safety System Setting**. This specification limits the maximum operating temperature of the reactor to 650°C so that there is ample margin to the safety limit. The basis to this specification states that "*Thus, 500°C of safety margin exists before the 1150°C safety limit is reached. This safety margin provides adequate compensation for variations in the temperature profile of depleted and differently loaded fuel" (i.e. 8.5wt% vs. 12wt% fuel elements.)*.
2.) **TS 3.1.5: Core Configuration Limitation.** (subsections a, b, c, and e of technical specifications included in this RAI response) These specifications limit the type and configuration of fuel (subsection a) used in the Penn State reactor, the amount of power that the fuel can produce (subsection b), the normalized power and pulsing (subsection c), and the arrangement of the fuel (subsection e waterholes). These actions help to protect the assumptions in the SAR, including the fuel temperature ratio of 1.6.
3.) **TS 5.1: Reactor Fuel.** This specification limits the type of fuel used in the Penn State reactor to 8.5 wt% and 12 wt% U-ZrH stainless steel clad TRIGA fuel-moderator elements which were analyzed in the SAR. The use of other types of TRIGA fuel is not yet analyzed and therefore prohibited.
4.) **TS 5.2: Reactor Core.** This specification limits the type of reflectors used in the Penn State reactor in order to limit the effects of flux perturbation caused by unanalyzed materials. Subsection 5.2.b limits the reflector to "*water, D_2O, or graphite or any combination of the three moderator materials.*"

The above specifications ensure that the only fuel used in the Penn State reactor is as analyzed in the SAR with a maximum peaking factor of 1.6:1.

PSBR response to NRC RAI

Dtd. 07/28/09

2. **The table at the top of page 22 of the supplement to your application dated October 31, 2008, shows that the maximum deviation between the measured and calculated pulse fuel temperatures was 13.2 percent for element I-13 located in core position G-8. A discussion of the deviation applied to steady-state fuel temperatures follows the table.**

   Provide a similar discussion of the deviation between the measured and calculated pulse fuel temperatures as it applies to maximum fuel temperature during a pulse.

There are four analyzed cases in the Penn State SAR.

1. New 12 wt % fuel element at steady state power of 1MW (fuel element I-14)
2. Used 12 wt % fuel element at steady state power of 1MW (fuel element I-13)
3. New 12 wt % fuel element during pulsing (fuel element I-14)
4. Used 12 wt % fuel element during pulsing (fuel element I-13)

The used 12 wt % fuel element (I-13) had a burnup of 2.2 MWd or 8000 MWD/MTU for a 275 gram loading of uranium in a nominal 12 wt % TRIGA fuel element.

Tables 2-1 to 2-4 below show data for cases 1-4 from the SAR and the previous submission question 13.2. Position G-8 is in the B-Ring of the core adjacent to the central water hole. Position G-10 is in the C-Ring and surrounded by fuel.

Table 2-1: Deviation for a new 12 wt % fuel element at steady state power of 1MW (I-14)

Fuel Element	Core Position	NP (SS)	SS 1MW Meas.	SS 1MW Calc.	% Deviation
I-14	G-8	2.01	466	467	-0.2%
I-14	G-10	1.84	450	439	2.4%

Table 2-2: Deviation for a Used 12 wt % fuel element at steady state power of 1MW (I-13)

Fuel Element	Core Position	NP (SS)	SS 1MW Meas.	SS 1MW Calc.	% Deviation
I-13	G-8	1.56	411	411	-0.12%
I-13	G-10	1.39	381	382	-0.24%

Table 2-3: Deviation for a new 12 wt % fuel element during Pulsing (I-14)

Fuel Element	Core Position	NP (pulse)	\$2.75 Pulse Meas.	\$2.75 Pulse Calc.	% Deviation
I-14	G-8	2.07	517	518	-0.1%
I-14	G-10	1.8	466	453	2.8%

Table 2-4: Deviation for a used 12 wt % fuel element during Pulsing (I-13)

Fuel Element	Core Position	NP (pulse)	\$2.75 Pulse Meas.	\$2.75 Pulse Calc.	% Deviation
I-13	G-8	1.62	511	443	13.2%
I-13	G-10	1.54	453	422	6.7%

An important distinction in this discussion is that the deviation tabulated here is the difference between the calculated value of the temperature at the thermocouple location and the measured value at the thermocouple location. The actual measurement uncertainty is quite small (on the order of 1-2 degrees Celsius) and is bore out on the steady state temperature predictions to be a based on daily comparisons between a cold fuel element and the pool water temperature.

With that in mind, the SAR explains the relatively large deviation for the partially burned element I-13 was due to:

1.) Excess burnup of U-235 at the periphery of the fuel element that is not accounted for in the $A_o+B_or^2$ assumed shape of the power distribution and,

2.) The gap temperature difference is probably different in I-13 and I-14 (as would be expected for any two instrumented fuel elements)

A partially burned fuel element has a much lower peak-to-measured fuel temperature ratio during a pulse as compared to a new 12 wt % fuel element (1.1 as compared to 1.6).

Using a conservative assumption that the observed calculated/measured difference is a multiplicative error the maximum temperature for a $3.50 pulse in NP 2.2 fuel can be calculated. For the cases of pulsing the two types of elements, the calculations below are the same as the calculation in the SAR (page XIII-25) and would proceed as follows:

Case 3 (I-14): $t_{poj} = 1.177x10^4 * NP_j * \partial k_p + 95.8 * NP_j + T_o$ (SAR eqn 34)

$$t_{poj} = 1.177x10^4 * 2.2 * (\$3.50 * 0.007 - 0.007) + 95.8 * 2.2 + 20$$

t_{poj} =684°C * Uncertainty of 2.8% = 703°C

703°C * pin radial peaking factor of 1.6 = 1125°C

Case 4 (I-13): $t_{poj} = 1.475x10^4 * NP_j * \partial k_p + 80 * NP_j + T_o$ (SAR eqn 37)

$$t_{poj} = 1.475x10^4 * 2.2 * (\$3.50 * 0.007 - 0.007) + 80 * 2.2 + 20$$

t_{poj} =764°C * Uncertainty of 13.2% = 865°C

865°C * pin radial peaking factor of 1.1 = 951°C

Both cases are below the safety limit with the new (I-14) being limiting. Since I-13 is not a limiting case the SAR provides not further discussion of the deviations.

Explain any uncertainties in the calculation and measurement of the maximum pulse fuel temperature. Explain how the uncertainties are treated to ensure that the maximum pulse fuel temperature will not exceed the safety limit for any core configuration allowed by the TS.

There is no method to measure the maximum pulse fuel temperature. All methods are computer models (such as the General Atomics work cited below) or extrapolation of experimental data to the to the point of maximum temperature.

The PSBR SAR was based on extrapolation to 12 wt% of analytically derived flux and heat production profiles that were experimentally verified using sophisticated techniques for 8 wt% as described by Goodwin (SAR Chapter 13 Ref. 15). Goodwin's work showed that volumetric heat generation from a pulse was related to the thermal neutron flux profile in an element at the start of the pulse. Fourteen high speed thermocouple junctions were placed in an 8 wt% TRIGA fuel element that was then subjected to multiple pulse events to validate the analytical work and develop a temperature (hence heat generation and flux) profile for the element. This enabled Goodwin to predict the element's performance and peak temperatures with a good degree of accuracy up to the "adiabatic limit" of the fuel.

The work done by Levine for the PSU 12 wt% elements used the basic theoretical premises of Goodwin – that the temperature profile during a pulse can be predicted based on the volumetric heat generation. Both General Atomics (GA) and Levine (et al.)[2] concluded that the installed thermocouples in an instrumented fuel element perform adequately to be used to develop a temperature relationship between measured values and peak pulse temperatures. Levine used measured temperature values to develop a predictive model.

Penn State experimental data (including any temperature measurement uncertainties) for a new 12 wt% element (I-14) agreed closely with the calculated predictions. Therefore the sum of the errors and uncertainties regarding flux profile and heat generation were determined to be relatively small differences from the actual values and provided appropriate accuracy for the measurement and prediction of pulsing performance. The SAR does not give additional consideration to the uncertainties on the partially depleted 12 wt% fuel element (I-13) since that case was not limiting due to the burnup and reduced peaking.

Larger uncertainties were encountered with the total heat generation in a 12 wt% element versus an 8 wt%, the flux depression in the 12 wt%, and the resultant heat generation profile. These uncertainties yielded two different values for the peak to measured temperature ratios: 1.45:1 (which Levine considered more likely [2]) and 1.6:1. The SAR was written with the more conservative 1.6:1 ratio to assure compliance with the safety limit for a maximum allowed pulse of the maximum power element.

In 1975, GA published information on TRIGA performance that included temperature predictions and profiles for 12 wt% elements at the Sandia ACPR (E-117-393 Fuel Elements for Pulsed TRIGA Reactor dated 1975). Figure 2-1 shows the calculated temperature distribution of the highest power new 12 wt% TRIGA fuel element during a $4.80 pulse calculated for the Sandia ACPR (20,000 MW pulse peak power). The highest ratio of the maximum temperature anywhere in the fuel to the temperature at the thermocouple location is 1.57:1 and occurs immediately with the pulse. The PSBR SAR assumption of 1.6:1 bounds the GA analysis. Note the maximum predicted temperature anywhere in the hot element is ~1000 °C.

One of the principal reasons the PSBR SAR is so conservative is the SAR uses the <u>peak measured</u> fuel temperature at the thermocouple location to calculate the maximum peak temperature. In earlier GA work, it was concluded that the time response was sufficiently fast (SAR Chapter 13 Reference 25 – GA-6216 1965) to measure the peak temperature at the thermocouple following pulsing. Levine's work concluded the same. However, unlike 8 wt% fuel where there is little change in the thermocouple location temperature for seconds after the pulse the temperature in a 12 wt% at the thermocouple rises significantly for ~10 seconds after the pulse as conduction equalizes the element temperature (see Figure 14 GA-6216 which is on page 24 of 40 in the October 31, 2008 RAI response and Figure 2-1 below). Since the SAR applies the 1.6:1 ratio to the <u>peak measured</u> value (not temperature at the thermocouple during the maximum temperature profile), the predicted maximum fuel temperatures in the SAR are conservative.

From Figure 2-1, the ratio of the maximum fuel temperature (at ~.1 second) to peak measured (at ~10 seconds) is ~1.3:1, not 1.6:1. Applying the 1.6 ratio to a peak measured temperature in 12 wt% fuel inserts conservancy of 1.6/1.3 = 23%.

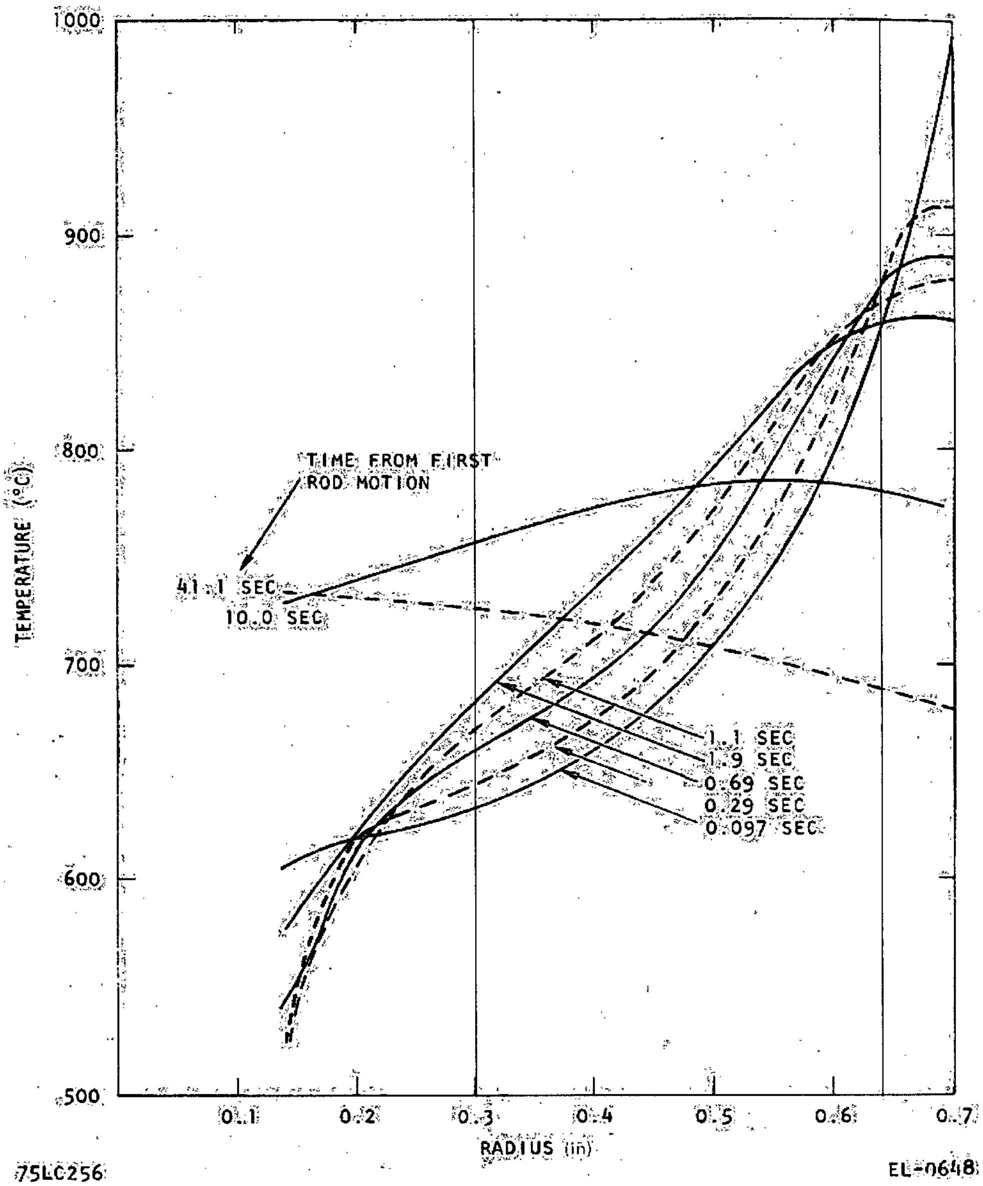


Figure 2-1: Standard Pulse Radial Temperature Distribution in a Fuel Rod (General Atomics E-117-393)

3. **Your application describes qualified locations within the reactor pool for operation of the reactor. Page 7 of the supplement to your application dated October 31, 2008, states:**

> ***"Qualification of a new operating location is governed by operating procedures. New locations are analyzed for reactivity coupling effects in addition to stresses and radiation effects."***

Provide clarification as to whether the qualification of a new operating location requires an evaluation, consistent with the criteria of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.59(c), to determine that the new operating location does not require a license amendment.

The qualification of a new operating position per the Standard Operating Procedure (SOP) does not require a re-evaluation of criteria of 10CRF50(c). The design criterion of the original modification that increased the degrees of movement (reactor bridge modification) for the reactor included that there are no unreviewed reactor safety questions as a result of the motion or new position.

A simple movement of the reactor and subsequent operation at a "new" position does not qualify as a change as defined by 50.59(a)(1). Movement of the reactor and operation of the reactor in the pool are described by the SAR. *Qualification* of a new position is **not** a *modification* or *addition to* or *removal from* the *facility or procedures that affects a design function.* It is not a change to a *method of performing or controlling that function,* nor a *change to the evaluation that demonstrates that intended functions will be accomplished.* Qualification of a new operating position is the verification that the new operating position **did not** result in any unexpected radiation levels, reactivity couplings, or affect any design functions.

Since the installation of the bridge modification, two new operating positions have been qualified (and one removed). Both are associated with new irradiation fixtures (D_2O thermal column and Fast Neutron Irradiator). SOP-5 *Experiment Evaluation and Authorization* requires an evaluation of 10CFR50.59 for new experiments. AP-12 *Change* would be used to approve any new irradiation fixture and requires screening for 10CFR50.59 applicability.

4. TS 5.1.b requires that the hydrogen-to-zirconium atom ratio shall be a nominal 1.65 hydrogen atoms to 1.0 zirconium atoms. Section 4.2.1 of the safety analysis report states that this ratio is 1.7 to 1.0 for 8.5 weight percent fuel elements.

a. Clarify this apparent discrepancy.

The optimal ratio of H/Zr has been refined by new research results since the inception of the TRIGA reactor design. TS 5.1.b states that the nominal ratio shall be 1.65. Nominal is interpreted to be a both target value for fuel fabrication and a value representative of previously procured fuel in use. In 1980 General Atomics revised their specifications for TRIGA fuel H/Zr ratio to a nominal value of 1.6:1 with a maximum of 1.65:1 for both 8.5 and 12.0 wt%[6] (page 4-1). The original loading of 8.5 wt% fuel at Penn State had a nominal value of 1.68 with a maximum of 1.70[7]. The fuel-follower control rods had ratios of 1.72, 1.75 and 1.76, but an explanation for this difference can no longer be located. The documented range of H/Zr in PSU fuel is 1.57 to 1.76.

TRIGA fuel was originally constructed with aluminum cladding, 8 wt% uranium and hydrogen to zirconium ratio of 1:1 on an atomic basis[1]. The change to stainless steel cladding and the higher hydrogen ratio made the fuel more reliable and robust. Once the hydride concentration gets above 1.5, the fuel is no longer subject to thermal phase separation or the thermal cycling effects that cause hydrogen migration in the earlier fuels[6] (page 2-14/2-16). The nominal H/Zr ratio was stated as 1.6:1[6] (page 1-1). Figure 4-1 shows the phase diagram for H/Zr. For any ratio $\geq$ 1.65, the fuel stays in the delta or epsilon phase for all temperatures below the safety limit (1150°)[6] (Fig 2.4). The presence of uranium in the matrix drops the transition temperatures slightly[6] (page2-5).

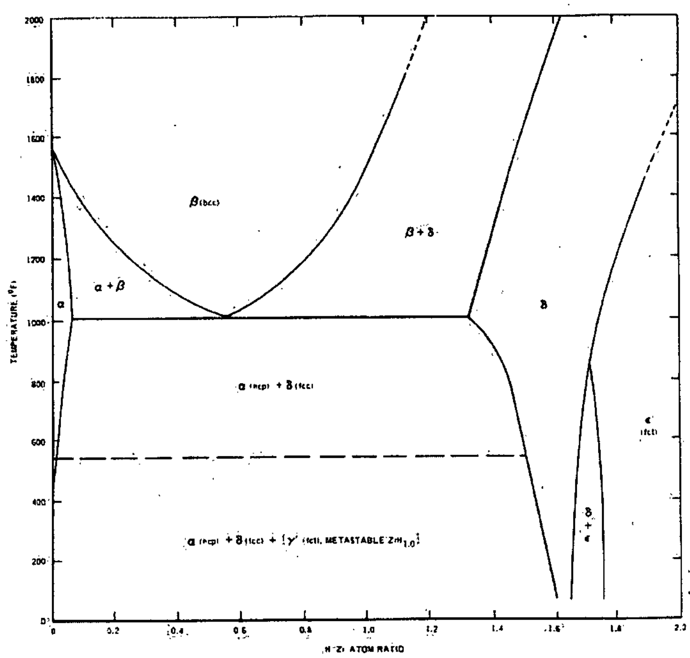


Fig. 2-4. Zirconium hydride phase diagram, showing boundary determination (from Ref. 19)

Figure 4-1: Phases of Zirconium-Hydride as a Function of Temperature

The safety limit for TRIGA fuel with a stainless steel cladding and a H/Zr ratio of 1.65:1 is listed as 1150°C under all conditions when the cladding temperature is ≤ 500°C[6] (page 4-1).

Figure 4-2 shows the change in internal pressure for a given H/Zr ratio versus temperature[6] (Fig 2.8). This shows that the internal fuel pressure for a given temperature will be slightly higher for the higher H-Zr ratios in the earlier fuel. Since this is only 8.5 wt% fuel for Penn State, it is not the limiting case.

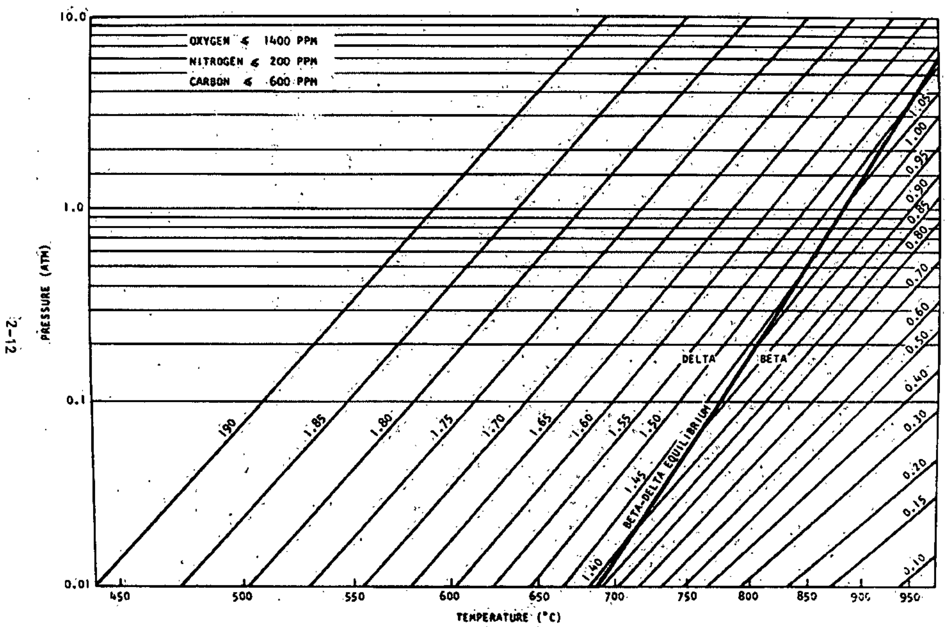


Fig. 2-8. Dissociation pressure isochores of zirconium hydride (expressed as H/Zr atom ratios) (from Ref. 23)

Figure 4-2: Hydrogen Disassociation Pressure from TRIGA Fuel

b) Additionally, provide a discussion of any controls in place that ensure TS 5.1.b is satisfied for new fuel received at the facility (e.g., review of vendor quality assurance documentation).

Historically there were no PSU specific administrative controls in place to ensure the Department of Energy (DOE) contract vendors (fuel providers) provided fuel that met the specifications required by the DOE purchase contract and TS 5.1.b. Receipt validation was done on an ad hoc basis. A change to SOP-3 *Core Loading and Fuel Handling* has been initiated to verify and document compliance with TS 5.1.b prior to loading any new elements into the core.

5. In accordance with 10 CFR 50.36(a)(1), provide bases for proposed technical specifications 5.1 through 5.6. Your response should include the proposed technical specifications in their entirety.

Attached is a copy of the proposed technical specifications which includes the following changes from Amendment 37:

1. Reformatted the entire document – to remove multiple font sizes and margins shift used to accommodate previous amendments;
2. Removed all amendment references and change bars – to support reissue with license renewal;
3. Added a header with title and footer to each page with "page # of #" format – convenience and clarity;
4. Added a table of contents – not part of PSBR Technical Specifications, provided for convenience only;
5. Updated multiple references to sections of the Safety Analysis Report in technical speculations to reflect the current (new SAR) chapter numbers and if necessary chapter titles – to support reissue with new SAR and license renewal.
6. TS 1.1.34.g changed "containment" to "confinement" – Correct typographic error, the facility does not have a containment;
7. Split TS 3.1.5.c into two specifications 3.1.5.c and 3.1.5.d, changed new TS 3.1.5.d "Reactivity Accident in the Safety Analysis Report" to "Insertion of Excess Reactivity", provided guidance on when the calculation in TS 3.1.5.d is required, and modified basis to support the revision – for clarity of requirement and match wording used the SAR;
8. Added TS 3.1.5.e and associated basis – new restriction on core configuration resulting from review of peaking issues in the vicinity of internal water filled core positions. Refer to question 1 above;
9. Deleted reference to SAR in TS 3.3.2 – information removed from new SAR for security reasons.
10. Added to TS 4.0 permission to operate the reactor to accomplish surveillance requirements that exceed the specified performance interval – current Technical Specifications provide no mechanism to accomplish overdue requirements that require operation of the reactor. This addition corrects that oversight and limits operations to those required to complete the surveillance(s);
11. Added basis to TS Section 5 – to support compliance with 10CFR50.36(a)(1).

PSBR response to NRC RAI
Dtd. 07/28/09

References:

1. Goodwin, W., "The Measurement of Radial Power Distributions in a TRIGA Fuel Element during Reactor Excursions," Ph.D. Thesis, University of Illinois, 1969.

2. Levine, S., G. Geisler and R. Totenbier, "Temperature Behavior of 12 WT.% U TRIGA Fuel", TOC-5, TRIGA Owners' Conference III, February 1974.

3. Simnad, M., "Fuel Elements for Pulsed TRIGA Research Reactors E-117-393" General Atomics 1975.

4. U.S. Nuclear Regulatory Commission, "NUREG-1282 Safety Evaluation Report on High-Uranium Content, Low-Enriched Zirconium Hydride Fuels for TRIGA Reactors", Government Printing Office, Washington D.C., 1987.

5. Penn State Safety Analysis Report Chapter XIII, 1997.

6. Simnad, M., "The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel E-117-833" General Atomics 1980.

7. PSBR Fuel Receipt Records 1965.

TECHNICAL SPECIFICATIONS: PENN STATE BREAZEALE REACTOR (PSBR)
FACILITY LICENSE NO. R-2

1.0 INTRODUCTION

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

1.1 Definitions

1.1.1 ALARA

The ALARA (As Low As Reasonably Achievable) program is a program for maintaining occupational exposures to radiation and release of radioactive effluents to the environs as low as reasonably achievable.

1.1.2 Automatic Control

Automatic control mode operation is when normal reactor operations, including start up, power level change, power regulation, and protective power reductions are performed by the reactor control system without, or with minimal, operator intervention.

1.1.3 Channel

A channel is the combination of sensor, line, amplifier, and output devices which are connected for the purpose of measuring the value of a parameter.

1.1.4 Channel Calibration

A channel calibration is an adjustment of the channel such that its output responds, with acceptable range, and accuracy, to known values of the parameter which the channel measures. Calibration SHALL encompass the entire channel, including equipment actuation, alarm, or trip, and SHALL be deemed to include a Channel Test.

1.1.5 Channel Check

A channel check is 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.

1.1.6 Channel Test

A channel test is the introduction of a signal into the channel to verify that it is operable.

1.1.7 Cold Critical

Cold critical is the condition of the reactor when it is critical with the fuel and bulk water temperatures both below 100°F (37.8°C).

1.1.8 Confinement

Confinement means an enclosure on the overall facility which controls the movement of air into it and out through a controlled path.

1.1.9 Excess Reactivity

Excess reactivity is that amount of reactivity that would exist if all control rods (safety, regulating, etc.) were moved to the maximum reactive condition from the point where the reactor is exactly critical (keff=1) in the reference core condition.

1.1.10 Experiment

Experiment SHALL mean (a) any apparatus, device, or material which is not a normal part of the core or experimental facilities, but which is inserted in these facilities or is in line with a beam of radiation originating from the reactor core; or (b) any operation designed to measure reactor parameters or characteristics.

1.1.11 Experimental Facility

Experimental facility SHALL mean beam port, including extension tube with shields, thermal column with shields, vertical tube, central thimble, in-core irradiation holder, pneumatic transfer system, and in-pool irradiation facility.

1.1.12 Instrumented Element

An instrumented element is a TRIGA fuel element in which sheathed chromel-alumel or equivalent thermocouples are embedded in the fuel.

1.1.13 Limiting Conditions for Operation

Limiting conditions for operation of the reactor are those constraints included in the Technical Specifications that are required for safe operation of the facility. These limiting conditions are applicable only when the reactor is operating unless otherwise specified.

1.1.14 Limiting Safety System Setting

A limiting safety system setting (LSSS) is a setting for an automatic protective device related to a variable having a significant safety function.

1.1.15 Manual Control

Manual control mode is operation of the reactor with the power level controlled by the operator adjusting the control rod positions.

1.1.16 Maximum Elemental Power Density

The maximum elemental power density (MEPD) is the power density of the element in the core producing more power than any other element in that loading. The power density of an element is the total power of the core divided by the number of fuel elements in the core multiplied by the normalized power of that element. This definition is only applicable for non-pulse operation.

1.1.17 Maximum Power Level

Maximum Power Level is the maximum measured value of reactor power for non-pulse operation.

1.1.18 Measured Value

The measured value is the value of a parameter as it appears on the output of a channel.

1.1.19 Movable Experiment

A movable experiment is one where it is intended that the entire experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

1.1.20 Normalized Power

The normalized power, NP, is the ratio of the power of a fuel element to the average power per fuel element.

1.1.21 Operable

Operable means a component or system is capable of performing its intended function.

1.1.22 Operating

Operating means a component or system is performing its intended function.

1.1.23 Pulse Mode

Pulse mode operation SHALL mean operation of the reactor allowing the operator to insert preselected reactivity by the ejection of the transient rod.

1.1.24 Reactivity Limits

The reactivity limits are those limits imposed on reactor core reactivity. Quantities are referenced to a reference core condition.

1.1.25 Reactivity Worth of an Experiment

The reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter experiment position or configuration.

1.1.26 Reactor Control System

The reactor control system is composed of control and operational interlocks, reactivity adjustment controls, flow and temperature controls, and display systems which permit the operator to operate the reactor reliably in its allowed modes.

1.1.27 Reactor Interlock

A reactor interlock is a device which prevents some action, associated with reactor operation, until certain reactor operation conditions are satisfied.

1.1.28 Reactor Operating

The reactor is operating whenever it is not secured or shutdown.

1.1.29 Reactor Secured

The reactor is secured when:

a. It contains insufficient fissile material or moderator present in the reactor, adjacent experiments, or control rods, to attain criticality under optimum available conditions of moderation, and reflection, or

b. A combination of the following:

   1) The minimum number of neutron absorbing control rods are fully inserted or other safety devices are in shutdown positions, as required by technical specifications, and

   2) The console key switch is in the off position and the key is removed from the lock, and

   3) 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, and

   4) No experiments in or near the reactor are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment or one dollar whichever is smaller.

1.1.30 Reactor Shutdown

The reactor is shutdown if it is subcritical by at least one dollar in the reference core condition and the reactivity worth of all experiments is included.

1.1.31 Reactor Safety System

Reactor safety systems are those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

1.1.32 Reference Core Condition

The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible ($<0.21\%\ \Delta k/k$ (~\$0.30)).

1.1.33 Research Reactor

A research reactor is defined as a device designed to support a self-sustaining neutron chain reaction for research, development, educational, training, or experimental purposes, and which may have provisions for the production of radioisotopes.

1.1.34 Reportable Occurrence

A reportable occurrence is any of the following which occurs during reactor operation:

a. Operation with the safety system setting less conservative than specified in Section 2.2, Limiting Safety System Setting.

b. Operation in violation of a limiting condition for operation.

c. Failure of a required reactor safety system component which could render the system incapable of performing its intended safety function.

d. Any unanticipated or uncontrolled change in reactivity greater than one dollar.

e. An observed inadequacy in the implementation of either administrative or procedural controls which could result in operation of the reactor outside the limiting conditions for operation.

f. Release of fission products from a fuel element.

g. Abnormal and significant degradation in reactor fuel, cladding, coolant boundary or confinement boundary that could result in exceeding 10 CFR Part 20 exposure criteria.

1.1.35 Rod-Transient

The transient rod is a control rod with SCRAM capabilities that is capable of providing rapid reactivity insertion for use in either pulse or square wave mode of operation.

1.1.36 Safety Limit

Safety limits are limits on important process variables which are found to be necessary to reasonably protect the integrity of certain physical barriers which guard against the uncontrolled release of radioactivity. The principal physical barrier is the fuel element cladding.

1.1.37 SCRAM Time

SCRAM time is the elapsed time between reaching a limiting safety system set point and a specified control rod movement.

1.1.38 Secured Experiment

A secured experiment is any experiment, experimental facility, 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 to 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.

1.1.39 Secured Experiment with Movable Parts

A secured experiment with movable parts is one that contains parts that are intended to be moved while the reactor is operating.

1.1.40 Shall, Should, and May

The word "shall" is used to denote a requirement; the word "should" to denote a recommendation; and the word "may" to denote permission, neither a requirement nor a recommendation.

1.1.41 Shim, Regulating, and Safety Rods

A shim, regulating, or safety rod is a control rod having an electric motor drive and SCRAM capabilities. It has a fueled follower section.

1.1.42 Shutdown Margin

Shutdown margin SHALL mean 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 although the most reactive rod is in its most reactive position, and that the reactor will remain subcritical without further operator action.

1.1.43 Square Wave Mode

Square wave (SW) mode operation SHALL mean operation of the reactor allowing the operator to insert preselected reactivity by the ejection of the transient rod, and which results in a maximum power within the license limit.

1.1.44 Steady State Power Level

Steady state power level is the nominal measured value of reactor power to which reactor power is being controlled whether by manual or automatic actions. Minor variations about this level may occur due to noise, normal signal variation, and reactivity adjustments. During manual, automatic, or square wave modes of operation, some initial, momentary overshoot may occur.

1.1.45 TRIGA Fuel Element

A TRIGA fuel element is a single TRIGA fuel rod of standard type, either 8.5 wt% U-ZrH in stainless steel cladding or 12 wt% U-ZrH in stainless steel cladding enriched to less than 20% uranium-235.

1.1.46 Watchdog Circuit

A watchdog circuit is a circuit consisting of a timer and a relay. The timer energizes the relay as long as it is reset prior to the expiration of the timing interval. If it is not reset within the timing interval, the relay will de-energize thereby causing a SCRAM.

2.0 SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTING

2.1 Safety Limit-Fuel Element Temperature

Applicability

The safety limit specification applies to the maximum temperature in the reactor fuel.

Objective

The objective is to define the maximum fuel element temperature that can be permitted with confidence that no damage to the fuel element and/or cladding will result.

Specification

The temperature in a water-cooled TRIGA fuel element SHALL NOT exceed 1150°C under any operating condition.

Basis

The important parameter for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification especially since it can be measured at a point within the fuel element and the relationship between the measured and actual temperature is well characterized analytically. A loss in the integrity of the fuel element cladding could arise from a build-up of excessive pressure between the fuel-moderator and the cladding if the maximum fuel temperature exceeds 1150°C. The pressure is caused by the presence of air, fission product gases, and hydrogen from the dissociation of the hydrogen and zirconium in the fuel-moderator. The magnitude of this pressure is determined by the fuel-moderator temperature, the ratio of hydrogen to zirconium in the alloy, and the rate change in the pressure.

The safety limit for the standard TRIGA fuel is based on data, including the large mass of experimental evidence obtained during high performance reactor tests on this fuel. These data indicate that the stress in the cladding due to the increase in the hydrogen pressure from the dissociation of zirconium hydride will remain below
the ultimate stress provided that the temperature of the fuel does not exceed 1150°C and the fuel cladding is below 500°C. See Safety Analysis Report, Ref. 13 and 30 in Section 13 and Simnad, M.T., F.C. Foushee, and G.B. West, "Fuel Elements for Pulsed Reactors," Nucl. Technology, Vol. 28, p. 31-56 (January 1976).

2.2 Limiting Safety System Setting (LSSS)

Applicability

The LSSS specification applies to the SCRAM setting which prevents the safety limit from being reached.

Objective

The objective is to prevent the safety limit (1150°C) from being reached.

Specification

The limiting safety system setting SHALL be a maximum of 650°C as measured with an instrumented fuel element if it is located in a core position representative of the maximum elemental power density (MEPD) in that loading. If it is not practical to locate the instrumented fuel in such a position, the LSSS SHALL be reduced. The reduction of the LSSS SHALL be by a ratio based on the calculated linear relationship between the normalized power at the monitored position as compared to normalized power at the core position representative of the MEPD in that loading.

Basis

The limiting safety system setting is a temperature which, if reached, SHALL cause a reactor SCRAM to be initiated preventing the safety limit from being exceeded. Experiments and analyses described in the Safety Analysis Report, Section 13 - Accident Analysis, show that the measured fuel temperature at steady state power has a simple linear relationship to the normalized power of a fuel element in the core. Maximum fuel temperature occurs when an instrumented element is in a core position of MEPD. The actual location of the instrumented element and the associated LSSS SHALL be chosen by calculation and/or experiment prior to going to maximum reactor operational power level. The measured fuel temperature during steady state operation is close to the maximum fuel temperature in that element. Thus, 500°C of safety margin exists before the 1150°C safety limit is reached. This safety margin provides adequate compensation for variations in the temperature profile of depleted and differently loaded fuel elements (i.e. 8.5 wt% vs. 12 wt% fuel elements). See Safety Analysis Report, Chapter 13.

If it is not practical to place an instrumented element in the position representative of MEPD the LSSS SHALL be reduced to maintain the 500°C safety margin between the 1150°C safety limit and the highest fuel temperature in the core if it was being measured. The reduction ratio SHALL be determined by calculation using the accepted techniques used in Safety Analysis Report, Chapter 13.

In the pulse mode of operation, the same LSSS SHALL apply. However, the temperature channel will have no effect on limiting the peak power or fuel temperature, generated, because of its relatively long time constant (seconds), compared with the width of the pulse (milliseconds).

3.0 LIMITING CONDITIONS FOR OPERATION

The limiting conditions for operation as set forth in this section are applicable only when the reactor is operating. They need not be met when the reactor is shutdown unless specified otherwise.

3.1 Reactor Core Parameters

3.1.1 Non-Pulse Mode Operation

Applicability

These specifications apply to the power generated during manual control mode, automatic control mode, and square wave mode operations.

Objective

The objective is to limit the source term and energy production to that used in the Safety Analysis Report.

Specifications

a. The reactor may be operated at steady state power levels of 1 Mw (thermal) or less.

b. The maximum power level SHALL be no greater than 1.1 Mw (thermal).

c. The steady state fuel temperature SHALL be a maximum of 650°C as measured with an instrumented fuel element if it is located in a core position representative of MEPD in that loading. If it is not practical to locate the instrumented fuel in such a position, the steady state fuel temperature SHALL be calculated by a ratio based on the calculated linear relationship between the normalized power at the monitored position as compared to normalized power at the core position representative of the MEPD in that loading. In this case, the measured steady state fuel temperature SHALL be limited such that the calculated steady state fuel temperature at the core position representative of the MEPD in that loading SHALL NOT exceed 650°C.

Basis

a. Thermal and hydraulic calculations and operational experience indicate that a compact TRIGA reactor core can be safely operated up to power levels of at least 1.15 Mw (thermal) with natural convective cooling.

b. Operation at 1.1 Mw (thermal) is within the bounds established by the SAR for steady state operations. See Chapter 13, Section C of the SAR.

c. Limiting the maximum steady state measured fuel temperature of any position to 650°C places an upper bound on the fission product release fraction to that used in the analysis of a Maximum Hypothetical Accident (MHA). See Safety Analysis Report, Chapter 13.

3.1.2 Reactivity Limitation

Applicability

This specification applies to the reactivity condition of the reactor and the reactivity worth of control rods, experiments, and experimental facilities. It applies to all modes of operation.

Objective

The objective is to ensure that the reactor is operated within the limits analyzed in the Safety Analysis Report and to ensure that the safety limit will not be exceeded.

Specification

a. The maximum excess reactivity above cold, clean, critical plus samarium poison of the core configuration with experiments and experimental facilities in place SHALL be 4.9% $\Delta k/k$ (~$7.00).

b. During initial measurements of maximum excess reactivity for a new core/experimental configuration this specification is suspended provided the reactor is operated at power levels no greater than 1 kw. If the power level exceeds 1 kw, power SHALL be reduced to less that 1 kw within one minute. This exemption does not apply for the annual confirmatory measurement of excess reactivity required by TS 4.1.2.

Basis

Limiting the excess reactivity of the core to 4.9% $\Delta k/k$ (~$7.00) prevents the fuel temperature in the core from exceeding 1150°C under any assumed accident condition as described in the Safety Analysis Report, Chapter 13. The exemption allows the initial physics measurement of maximum excess reactivity for a new core/experimental configuration to be measured without creating a reportable occurrence. Maintaining the power level less that 1 kw during this exemption assures there is no challenge to the safety limit on fuel temperature.

3.1.3 Shutdown Margin

Applicability

This specification applies to the reactivity condition of the reactor and the reactivity worth of control rods, experiments, and experimental facilities. It applies to all modes of operation.

Objective

The objective is to ensure that the reactor can be shut down at all times and to ensure that the safety limit will not be exceeded.

Specification

The reactor SHALL NOT be operated unless the shutdown margin provided by control rods is greater than 0.175% $\Delta k/k$ (~$0.25) with:

a. All movable experiments, experiments with movable parts and experimental facilities in their most reactive state, and

b. The highest reactivity worth control rod fully withdrawn.

Basis

A shutdown margin of 0.175% $\Delta k/k$ (~$0.25) ensures that the reactor can be made subcritical from any operating condition even if the highest worth control rod should remain in the fully withdrawn position. The shutdown margin requirement may be more restrictive than Specification 3.1.2.

3.1.4 Pulse Mode Operation

Applicability

These specifications apply to the energy generated in the reactor as a result of a pulse insertion of reactivity.

Objective

The objective is to ensure that the safety limit will not be exceeded during pulse mode operation.

Specifications

a. The stepped reactivity insertion for pulse operation SHALL NOT exceed 2.45% Δk/k (~$3.50) and the maximum worth of the poison section of the transient rod SHALL be limited to 2.45% Δk/k (~$3.50).

b. Pulses SHALL NOT be initiated from power levels above 1 kw.

Basis

a. Experiments and analyses described in the Safety Analysis Report, Chapter 13, show that the peak pulse temperatures can be predicted for new 12 wt% fuel placed in any core position. These experiments and analyses show that the maximum allowed pulse reactivity of 2.45% Δk/k (~$3.50), prevents the maximum fuel temperature from reaching the safety limit (1150°C) for any core configuration that meets the requirements of 3.1.5.

   The maximum worth of the pulse rod is limited to 2.45% Δk/k (~$3.50) to prevent exceeding the safety limit (1150°C) with an accidental ejection of the transient rod.

b. If a pulse is initiated from power levels below 1 kw, the maximum allowed full worth of the pulse rod can be used without exceeding the safety limit.

3.1.5 Core Configuration Limitation

Applicability

These specifications apply to all core configurations except as noted.

Objective

The objective is to ensure that the safety limit (1150°C) will not be exceeded due to power peaking effects in the various core configurations.

Specifications

a. The critical core SHALL be an assembly of either 8.5 wt% U-ZrH stainless steel clad or a mixture of 8.5 wt% and 12 wt% U-ZrH stainless steel clad TRIGA fuel-moderator elements placed in water with a 1.7 inch center line grid spacing.

b. The maximum calculated MEPD SHALL be less that 24.7 kw per fuel element for non-pulse operation.

c. The NP of any core loading with a maximum allowed pulse worth of 2.45% $\Delta k/k$ (~\$3.50) SHALL be limited to 2.2. IF the maximum allowed pulse worth is less than 2.45% $\Delta k/k$ (\$~3.50) for any given core loading (i.e. the pulse can be limited by the total worth of the transient rod, by the core excess, or administratively), THEN the maximum NP may be increased above 2.2 as long as the calculated maximum fuel temperature does not exceed the safety limit with that maximum allowed pulse worth and NP.

d. IF the maximum NP is increased above 2.2 as described in TS 3.1.5.c above, THEN the Insertion of Excess Reactivity analysis in the Safety Analysis Report SHALL be evaluated to ensure that the safety limit is not exceeded with the new conditions (See Safety Analysis Report, Chapter 13.1.2.).

e. The core SHALL NOT be configured such that a 12 wt% U-ZrH stainless steel clad TRIGA fuel-moderator element with a burnup less than a nominal 8000 MWD/Metric Ton of Uranium is located adjacent to a vacant (water-filled) internal core position during pulse mode operation.

Basis

a. The safety analysis is based on an assembly of either 8.5 wt% U-ZrH stainless steel clad or a mixture of 8.5 wt% and 12 wt% U-ZrH stainless steel clad TRIGA fuel-moderator elements placed in water with a 1.7 inch center line grid spacing.

b. Limiting the MEPD to 24.7 kw per element for non-pulse operation places an upper bound on the elemental heat production and the source term of the PSBR to that used in the analysis of a Loss Of Coolant Accident (LOCA) and Maximum Hypothetical Accident (MHA) respectively. See Safety Analysis Report, Chapter 13.

c. The maximum NP for a given core loading determines the peak pulse temperature with the maximum allowed pulse worth. If the maximum allowed pulse worth is reduced the maximum NP may be increased without exceeding the safety limit (1150°C). The amount of increase in the maximum NP allowed SHALL be calculated by an accepted method documented by an administratively approved procedure.

d. If the core loading deviates from the limits set in TS 3.1.5.c then revalidation of the Insertion of Excess Reactivity analysis in the Safety Analysis Report will ensure that the new loading does not inadvertently exceed the safety limit (See Safety Analysis Report, Chapter 13.1.2.).

e. Radial peaking effects in unirradiated 12 wt% U-ZrH stainless steel clad TRIGA fuel-moderator elements located adjacent to water-filled internal core position may cause a reduction in the safety margin during pulse mode operation with the maximum allowed pulse worth of 2.45% Δk/k (~$3.50) and the maximum allowed NP of 2.2. Locating an 8.5 wt% or moderately-irradiated (~8000 Megawatt Days per Metric Ton of Uranium) 12 wt% U-ZrH stainless steel clad TRIGA fuel-moderator element adjacent to vacant water-filled internal core positions provides additional safety margin. 12 wt% elements in the periphery of the core are not subject to this concern as the NP is too low to make these elements limiting.

3.1.6 <u>TRIGA Fuel Elements</u>

<u>Applicability</u>

These specifications apply to the mechanical condition of the fuel.

<u>Objective</u>

The objective is to ensure that the reactor is not operated with damaged fuel that might allow release of fission products.

<u>Specifications</u>

The reactor SHALL NOT be operated with damaged fuel except to detect and identify the fuel element for removal. A TRIGA fuel element SHALL be considered damaged and SHALL be removed from the core if:

a. In measuring the transverse bend, the bend exceeds the limit of 0.125 inch over the length of the cladding.

b. In measuring the elongation, its length exceeds its original length by 0.125 inch.

c. A clad defect exists as indicated by release of fission products.

Basis

a. The limit of transverse bend has been shown to result in no difficulty in disassembling the core. Analysis of the removal of heat from touching fuel elements shows that there will be no hot spots which cause damage to the fuel.

b. Experience with TRIGA reactors has shown that fuel element bending that could result in touching has occurred without deleterious effects. This is because (1) during steady state operation, the maximum fuel temperatures are at least 500°C below the safety limit (1150°C), and (2) during a pulse, the cladding temperatures remain well below their stress limit. The elongation limit has been specified to ensure that the cladding material will not be subjected to strains that could cause a loss of fuel integrity and to ensure adequate coolant flow

3.2 Reactor Control and Reactor Safety System

3.2.1 Reactor Control Rods

Applicability

This specification applies to the reactor control rods.

Objective

The objective is to ensure that sufficient control rods are operable to maintain the reactor subcritical.

Specification

There SHALL be a minimum of three operable control rods in the reactor core.

Basis

The shutdown margin and excess reactivity specifications require that the reactor can be made subcritical with the most reactive control rod withdrawn. This specification helps ensure it.

3.2.2 Manual Control and Automatic Control

Applicability

This specification applies to the maximum reactivity insertion rate associated with movement of a standard control rod out of the core.

Objective

The objective is to ensure that adequate control of the reactor can be maintained during manual and 1, 2, or 3 rod automatic control.

Specification

The rate of reactivity insertion associated with movement of either the regulating, shim, or safety control rod SHALL be NOT greater than 0.63% $\Delta k/k$ (~\$0.90) per second when averaged over full rod travel. If the automatic control uses a combination of more than one rod, the sum of the reactivity of those rods SHALL be not greater than 0.63% $\Delta k/k$ (~\$0.90) per second when averaged over full travel.

Basis

accident analysis (refer to Safety Analysis Report, Chapter 13) indicates that the safety limit (1150°C) will not be exceeded if the reactivity addition rate is less than 1.75% $\Delta k/k$ (~\$2.50) per second, when averaged over full travel. This specification of 0.63% $\Delta k/k$ (~\$0.90) per second, when averaged over full travel, is well within that analysis.

3.2.3 Reactor Control System

Applicability

This specification applies to the information which must be available to the reactor operator during reactor operation.

Objective

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

Specification

The reactor SHALL NOT be operated unless the measuring channels listed in Table 1 are operable. (Note that MN, AU, and SW are abbreviations for manual control mode, automatic control mode, and square wave mode, respectively).

Table 1
Measuring Channels

Measuring Channel	Min. No. Operable	Effective Mode MN, AU & SW	Effective Mode Pulse
Fuel Element Temperature	1	X	X
Wide Range Instrument			
Linear Power	1	X	
Log Power	1	X	
Reactor Period/SUR	1	X	
Power Range Instrument			
Linear Power	1	X	
Pulse Peak Power	1		X

Basis

Fuel temperature displayed at the control console gives continuous information on this parameter which has a specified safety limit. The power level monitors ensure that the reactor power level is adequately monitored for the manual control, automatic control, square wave, and pulsing modes of operation. The specifications on reactor power level and reactor period indications are included in this section to provide assurance that the reactor is operated at all times within the limits allowed by these Technical Specifications.

3.2.4 Reactor Safety System and Reactor Interlocks

Applicability

This specification applies to the reactor safety system channels, the reactor interlocks, and the watchdog circuit.

Objective

The objective is to specify the minimum number of reactor safety system channels and reactor interlocks that must be operable for safe operation.

Specification

The reactor SHALL NOT be operated unless all of the channels and interlocks described in Table 2a and Table 2b are operable.

Basis

a. A temperature SCRAM and two power level SCRAMs ensure the reactor is shutdown before the safety limit on the fuel element temperature is reached. The actual setting of the fuel temperature SCRAM depends on the LSSS for that core loading and the location of the instrumented fuel element (see Technical Specification section 2.2).

Table 2a
Minimum Reactor Safety System Channels

Channel	Number Operable	Function	Effective Mode MN, AU	Pulse	SW
Fuel Temperature	**1**	SCRAM ≤ 650°C*	**X**	**X**	**X**
High Power	**2**	SCRAM ≤ 110% of maximum reactor operational power not to exceed 1.1 Mw	**X**		**X**
Detector Power Supply	**1**	SCRAM on failure of supply voltage	**X**		**X**
SCRAM Bar on Console	**1**	Manual SCRAM	**X**	**X**	**X**
Preset Timer	**1**	Transient Rod SCRAM 15 seconds or less after pulse		**X**	
Watchdog Circuit	**1**	SCRAM on software or self-check failure	**X**	**X**	**X**

* The limit of 650°C SHALL be reduced as required by specification 2.2.

Table 2b
Minimum Reactor Interlocks

Channel	Number Operable	Function	Effective MN, AU	Mode Pulse	SW
Source Level	1	Prevent rod withdrawal without a neutron induced-signal on the log power channel	X		
Pulse Mode Inhibit	1	Prevent pulsing from levels above 1 kw		X	
Transient Rod	1	Prevent applications of air unless cylinder is fully inserted	X		
Shim, Safety, and Regulating Rod	1	Prevent movement of any rod except the transient rod		X	
Simultaneous Rod Withdrawal	1	Prevent simultaneous manual withdrawal of two rods	X		X

b. The maximum reactor operational power may be administratively limited to less than 1 Mw depending on section 3.1.5.b of this Technical Specification. The high power SCRAMs SHALL be set to no more than 110% of the administratively limited maximum reactor operational power if it is less than 1 Mw.

c. Operation of the reactor is prevented by SCRAM if there is a failure of the detector power supply for the reactor safety system channels.

d. The manual SCRAM allows the operator to shut down the reactor in any mode of operation if an unsafe or abnormal condition occurs.

e. The preset timer ensures that the transient rod will be inserted and the reactor will remain at low power after pulsing.

f. The watchdog circuit will SCRAM the reactor if the software or the self-checks fail (see Safety Analysis Report, Chapter 7).

g. The interlock to prevent startup of the reactor without a neutron induced signal ensures that sufficient neutrons are available for proper startup in all allowable modes of operation.

h. The interlock to prevent the initiation of a pulse above 1 kw is to ensure that fuel temperature is approximately pool temperature when a pulse is performed. This is to ensure that the safety limit is not reached.

i. The interlock to prevent application of air to the transient rod unless the cylinder is fully inserted is to prevent pulsing the reactor in the manual control or automatic control mode.

j. In the pulse mode, movement of any rod except the transient rod is prevented by an interlock. This interlock action prevents the addition of reactivity other than with the transient rod.

k. Simultaneous manual withdrawal of two rods is prevented to ensure the reactivity rate of insertion is not exceeded.

3.2.5 <u>Core Loading and Unloading Operation</u>

<u>Applicability</u>

This specification applies to the source level interlock.

<u>Objective</u>

The objective of this specification is to allow bypass of the source level interlock during operations with a subcritical core.

<u>Specification</u>

During core loading and unloading operations when the reactor is subcritical, the source level interlock may be momentarily defeated using a spring loaded switch in accordance with the fuel loading procedure.

<u>Basis</u>

During core loading and unloading, the reactor is subcritical. Thus, momentarily defeating the source level interlock is a safe operation. Should the core become inadvertently supercritical, the accidental insertion of reactivity will not allow fuel temperature to exceed the 1150°C safety limit because no single TRIGA fuel element is worth more than 1% $\Delta k/k$ (~$1.43) in the most reactive core position.

3.2.6 <u>SCRAM Time</u>

<u>Applicability</u>

This specification applies to the time required to fully insert any control rod to a full down position from a full up position.

<u>Objective</u>

The objective is to achieve rapid shutdown of the reactor to prevent fuel damage.

<u>Specification</u>

The time from SCRAM initiation to the full insertion of any control rod from a full up position SHALL be less than 1 second.

Basis

This specification ensures that the reactor will be promptly shut down when a SCRAM signal is initiated. Experience and analysis, Safety Analysis Report, Chapter 13, have indicated that for the range of transients anticipated for a TRIGA reactor, the specified SCRAM time is adequate to ensure the safety of the reactor. If the SCRAM signal is initiated at 1.1 Mw, while the control rod is being withdrawn, and the negative reactivity is not inserted until the end of the one second rod drop time, the maximum fuel temperature does not reach the safety limit.

3.3 Coolant System

3.3.1 Coolant Level Limits

Applicability

This specification applies to operation of the reactor with respect to a required depth of water above the top of the bottom grid plate.

Objective

The objective is to ensure that water is present to provide adequate personnel shielding and core cooling when the reactor is operated, and during a LOCA.

Specification

The reactor SHALL NOT be operated with less than 18 ft. of water above the top of the bottom grid plate.

Basis

When the water is more than approximately 18 ft. above the top of the bottom grid plate, the water provides sufficient shielding to protect personnel during operation at 1 Mw, and core cooling is achieved with natural circulation of the water through the core. Should the water level drop below approximately 18.25 ft. above the top of the bottom grid plate while operating at 1 Mw, a low pool level alarm (see Technical Specifications 3.3.2) will alert the operator who is required by administratively approved procedure to shutdown the reactor. Once this alarm occurs it will take longer than 1300 seconds before the core is completely uncovered because of a break in the 6" pipe connected to the bottom of the pool. Tests and calculations show that, during a LOCA, 680 seconds is sufficient decay time after shutdown (see Safety Analysis Report, Chapter 13) to prevent the fuel temperature from reaching 950°C. To prevent cladding rupture, the fuel and the cladding temperature must not exceed 950 °C (it is assumed that the fuel and the cladding are the same temperature during air cooling).

3.3.2 Detection of Leak or Loss of Coolant

Applicability

This specification applies to detecting a pool water loss.

Objective

The objective is to detect the loss of a significant amount of pool water.

Specification

A pool level alarm SHALL be activated and corrective action taken when the pool level drops 26 cm from a level where the pool is full.

Basis

The alarm occurs when the water level is approximately 18.25 ft. above the top of the bottom grid plate. The point at which the pool is full is approximately 19.1 ft. above the top of the bottom grid plate. The reactor staff SHALL take action to keep the core covered with water according to existing procedures. The alarm is also transmitted to the Police Services annunciator panel which is monitored 24 hrs. a day. The alarm provides a signal that occurs at all times. Thus, the alarm provides time to initiate corrective action before the radiation from the core poses a serious hazard.

3.3.3 Fission Product Activity

Applicability

This specification applies to the detection of fission product activity.

Objective

The objective is to ensure that fission products from a leaking fuel element are detected to provide opportunity to take protective action.

Specification

An air particulate monitor SHALL be operating in the reactor bay whenever the reactor is operating. An alarm on this unit SHALL activate a building evacuation alarm.

Basis

This unit will be sensitive to airborne radioactive particulate matter containing fission products and fission gases and will alert personnel in time to take protective action.

3.3.4 Pool Water Supply for Leak Protection

Applicability

This specification applies to pool water supplies for the reactor pool for leak protection.

Objective

The objective is to ensure that a supply of water is available to replenish reactor pool water in the event of pool water leakage.

Specification

A source of water of at least 100 GPM SHALL be available either from the University water supply or by diverting the heat exchanger secondary flow to the pool.

Basis

Provisions for both of these supplies are in place and will supply more than the specified flow rate. This flow rate will be more than sufficient to handle leak rates that have occurred in the past or any anticipated leak that might occur in the future.

3.3.5 Coolant Conductivity Limits

Applicability

This specification applies to the conductivity of the water in the pool.

Objectives

The objectives are:

a. To prevent activated contaminants from becoming a radiological hazard, and

b. To help preclude corrosion of fuel cladding and other primary system components.

Specification

The reactor SHALL NOT be operated if the conductivity of the bulk pool water is greater than 5 microsiemens/cm (5 micromhos/cm).

Basis

Experience indicates that 5 microsiemens/cm is an acceptable level of water contaminants in an aluminum/stainless steel system such as that at the PSBR. Based on experience, activation at this level does not pose a significant radiological hazard, and significant corrosion of the stainless steel fuel cladding will not occur when the conductivity is below 5 microsiemens/cm.

3.3.6 Coolant Temperature Limits

Applicability

This specification applies to the pool water temperature.

Objective

The objective is to maintain the pool water temperature at a level that will not cause damage to the demineralizer resins.

Specification

An alarm SHALL annunciate and corrective action SHALL be taken if during operation the bulk pool water temperature reaches 140°F (60°C).

Basis

This specification is primarily to preserve demineralizer resins. Information available indicates that temperature damage will be minimal up to this temperature.

3.4 Confinement

Applicability

This specification applies to reactor bay doors.

Objective

The objective is to ensure that no large air passages exist to the reactor bay during reactor operation.

Specifications

The reactor bay truck door SHALL be closed and the reactor bay personnel doors SHALL NOT be blocked open and left unattended if either of the following conditions are true.

a. The reactor is not secured, or

b. Irradiated fuel or a fueled experiment with significant fission product inventory is being moved outside containers, systems or storage areas.

Basis

This specification helps to ensure that the air pressure in the reactor bay is lower than the remainder of the building and the outside air pressure. Controlled air pressure is maintained by the air exhaust system and ensures controlled release of any airborne radioactivity.

3.5 Engineered Safety Features - Facility Exhaust System and Emergency Exhaust System

Applicability

This specification applies to the operation of the facility exhaust system and the emergency exhaust system.

Objective

The objective is to mitigate the consequences of the release of airborne radioactive materials resulting from reactor operation.

Specification

a. If the reactor is not secured, at least one facility exhaust fan SHALL be operating and, except for periods of time less than 48 hours during maintenance or repair, the emergency exhaust system SHALL be operable.

b. If irradiated fuel or a fueled experiment with significant fission product inventory is being moved outside containers, systems or storage areas, at least one facility exhaust fan SHALL be operating and the emergency exhaust system SHALL be operable.

Basis

During normal operation, the concentration of airborne radioactivity in unrestricted areas is below effluent release limits as described in the Safety Analysis Report, Chapter 13. In the event of a substantial release of airborne radioactivity, an air radiation monitor and/or an area radiation monitor will sound a building evacuation alarm which will automatically cause the facility exhaust system to close and the exhausted air to be passed through the emergency exhaust system filters before release. This reduces the radiation within the building. The filters will remove ≈ 90% all of the particulate fission products that escape to the atmosphere.

The emergency exhaust system activities only during an evacuation where upon all personnel are required to evacuate the building (section 3.6.2). If there is an evacuation while the emergency exhaust system is out of service for maintenance or repair, personnel evacuation is not prevented.

Personnel dose to the public will be equivalent or less whether or not the emergency exhaust system functions in the unlikely event an accident occurs during the maintenance or repair.

3.6 Radiation Monitoring System

3.6.1 Radiation Monitoring Information

Applicability

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

Objective

The objective is to ensure that sufficient radiation monitoring information is available to the operator to ensure personnel radiation safety during reactor operation.

Specification

The reactor SHALL NOT be operated unless the radiation monitoring channels listed in Table 3 are operating.

Table 3
Radiation Monitoring Channels

Radiation Monitoring Channels	Function	Number
Area Radiation Monitor	Monitor radiation levels in the reactor bay.	1
Continuous Air (Radiation) Monitor	Monitor radioactive particulates in the reactor bay air.	1
Beamhole Laboratory Monitor	Monitor radiation in the Beamhole Laboratory required only when the laboratory is in use.	1

Basis

a. The radiation monitors provide information to operating personnel of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and to take the necessary steps to control the spread of radioactivity to the surroundings.

b. The area radiation monitor in the Beamhole Laboratory provides information to the user and to the reactor operator when this laboratory is in use.

3.6.2 <u>Evacuation Alarm</u>

<u>Applicability</u>

This specification applies to the evacuation alarm.

<u>Objective</u>

The objective is to ensure that all personnel are alerted to evacuate the PSBR building when a potential radiation hazard exists within this building.

<u>Specification</u>

The reactor SHALL NOT be operated unless the evacuation alarm is operable and audible to personnel within the PSBR building when activated by the radiation monitoring channels in Table 3 or a manual switch.

<u>Basis</u>

The evacuation alarm produces a loud pulsating sound throughout the PSBR building when there is any impending or existing danger from radiation.. The sound notifies all personnel within the PSBR building to evacuate the building as prescribed by the PSBR emergency procedure.

3.6.3 <u>Argon-41 Discharge Limit</u>

<u>Applicability</u>

This specification applies to the concentration of Argon-41 that may be discharged from the PSBR.

<u>Objective</u>

The objective is to ensure that the health and safety of the public is not endangered by the discharge of Argon-41 from the PSBR.

<u>Specification</u>

All Argon-41 concentrations produced by the operation of the reactor SHALL be below the limits imposed by 10 CFR Part 20 when averaged over a year.

<u>Basis</u>

The maximum allowable concentration of Argon-41 in air in unrestricted areas as specified in Appendix B, Table 2 of 10 CFR Part 20 is 1.0×10^{-8} µCi/ml. Measurements of Argon-41 have been made in the reactor bay when the reactor operates at 1 Mw. These measurements show that the concentrations averaged over a year produce less than 1.0×10^{-8} µCi/ml in an unrestricted area (see Environmental Impact Appraisal, December 12, 1996).

3.6.4 As Low As Reasonably Achievable (ALARA)

Applicability

This specification applies to all reactor operations that could result in occupational exposures to radiation or the release of radioactive effluents to the environs.

Objective

The objective is to maintain all exposures to radiation and release of radioactive effluents to the environs ALARA.

Specification

An ALARA program SHALL be in effect.

Basis

Having an ALARA program will ensure that occupational exposures to radiation and the release of radioactive effluents to the environs will be ALARA. Having such a formal program will keep the staff cognizant of the importance to minimize radiation exposures and effluent releases.

3.7 Limitations of Experiments

Applicability

These specifications apply to experiments installed in the reactor and its experimental facilities.

Objective

The objective is to prevent damage to the reactor and to minimize 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 reactivity of a movable experiment and/or movable portions of a secured experiment plus the maximum allowed pulse reactivity SHALL be less than 2.45% _k/k (~$3.50). However, the reactivity of a movable experiment and/or movable portions of a secured experiment SHALL have a reactivity worth less than 1.4% Δk/k (~$2.00). During measurements made to determine specific worth, this specification is suspended provided the reactor is operated at power levels no greater than 1 kw. When a movable experiment is used, the maximum allowed pulse SHALL be reduced below the allowed pulse reactivity insertion of 2.45% Δk/k (~$3.50) to ensure that the sum is less 2.45% Δk/k (~$3.50).

b. A single secured experiment SHALL be limited to a maximum of 2.45% $\Delta k/k$ (~\$3.50). The sum of the reactivity worth of all experiments SHALL be less than 2.45% $\Delta k/k$ (~\$3.50). During measurements made to determine experimental worth, this specification is suspended provided the reactor is operated at power levels no greater that 1 kw.

c. When the keff of the core is less than 1 with all control rods at their upper limit and no experiments in or near the core, secured negative reactivity experiments may be added without limit.

d. An experiment may be irradiated or an experimental facility may be used in conjunction with the reactor provided its use does not constitute an unreviewed safety question. The failure mechanisms that SHALL be analyzed include, but are not limited to corrosion, overheating, impact from projectiles, chemical, and mechanical explosions.

   Explosive material SHALL NOT be stored or used in the facility without proper safeguards to prevent release of fission products or loss of reactor shutdown capability.

   If an experimental failure occurs which could lead to the release of fission products or the loss of reactor shutdown capability, physical inspection SHALL be performed to determine the consequences and the need for corrective action. The results of the inspection and any corrective action taken SHALL be reviewed by the Director or a designated alternate and determined to be satisfactory before operation of the reactor is resumed.

e. Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under (1) normal operating conditions of the experiment and reactor, (2) credible accident conditions in the reactor, or (3) possible accident conditions in the experiment, SHALL be limited in activity such that the airborne concentration of radioactivity averaged over a year SHALL NOT exceed the limit of Appendix B Table 2 of 10 CFR Part 20.

   When calculating activity limits, the following assumptions SHALL be used:

   1) If an experiment fails and releases radioactive gases or aerosols to the reactor bay or atmosphere, 100% of the gases or aerosols escape.

   2) If the effluent from an experimental facility exhausts through a holdup tank which closes automatically on high radiation level, at least 10% of the gaseous activity or aerosols produced will escape.

   3) If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of these vapors can escape.

   4) For materials whose boiling point is above 130°F and where vapors formed by boiling this material can escape only through an undisturbed column of water above the core, at least 10% of these vapors can escape.

f. Each fueled experiment SHALL be controlled such that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 curies. In addition, any fueled experiment which would generate an inventory of more than 5 millicuries (mCi) of I-131 through I-135 SHALL be reviewed to ensure that in the case of an accident, the total release of iodine will not exceed that postulated for the MHA (see Safety Analysis Report, Chapter 13).

Basis

a. This specification limits the sum of the reactivities of a maximum allowed pulse and a movable experiment to the specified maximum reactivity of the transient rod. This limits the effects of a pulse simultaneous with the failure of the movable experiment to the effects analyzed for a 2.45% $\Delta k/k$ (~\$3.50) pulse. In addition, the maximum power attainable with the ramp insertion of 1.4% $\Delta k/k$ (~\$2.00) is less than 500 kw starting from critical.

b. The maximum worth of all experiments is limited to 2.45% $\Delta k/k$ (~\$3.50) so that their inadvertent sudden removal from the cold critical reactor will not result in the reactor achieving a power level high enough to exceed the temperature safety limit (1150°C). The worth of a single secured experiment is limited to the allowed pulse reactivity insertion as an increased measure of safety. Should the 2.45% $\Delta k/k$, (~\$3.50) reactivity be inserted by a ramp increase, the maximum power attainable is less than 1 Mw.

c. Since the initial core is subcritical, adding and then inadvertently removing all negative reactivity experiments leaves the core in its initial subcritical condition.

d. The design basis accident is the MHA (See Safety Analysis Report, Chapter 13). A chemical explosion (such as detonated TNT) or a mechanical explosion (such as a steam explosion or a high pressure gas container explosion) may release enough energy to cause release of fission products or loss of reactor shutdown capability. A projectile with a large amount of kinetic energy could cause release of fission products or loss of reactor shutdown capability. Accelerated corrosion of the fuel cladding due to material released by a failed experiment could also lead to release of fission products.

   If an experiment failure occurs a special investigation is required to ensure that all effects from the failure are known before operation proceeds.

e. This specification is intended to reduce the likelihood that airborne activities in excess of the limits of Appendix B Table 2 of 10 CFR Part 20 will be released to the atmosphere outside the facility boundary.

f. The 5 mCi limitation on I-131 through I-135 ensures that in the event of failure of a fueled experiment, the exposure dose at the exclusion area boundary will be less than that postulated for the MHA (See Safety Analysis Report, Chapter 13) even if the iodine is released in the air.

4.0 SURVEILLANCE REQUIREMENTS

IF a Surveillance Requirement(s) is not accomplished in the specified interval that prohibits reactor operation; **THEN** the reactor SHALL NOT be operated until the Surveillance Requirement(s) is satisfied EXCEPT as required to accomplish the required Surveillance(s).

4.1 Reactor Parameters

4.1.1 Reactor Power Calibration

Applicability

This specification applies to the surveillance of the reactor power calibration.

Objective

The objective is to verify the performance and operability of the power measuring channel.

Specification

A thermal power channel calibration SHALL be made on the linear power level monitoring channel biennially, not to exceed 30 months.

Basis

The thermal power level channel calibration will ensure that the reactor is operated at the authorized power levels.

4.1.2 Reactor Excess Reactivity

Applicability

This specification applies to surveillance of core excess reactivity.

Objective

The objective is to ensure that the reactor excess reactivity does not exceed the Technical Specifications and the limit analyzed in Safety Analysis Report, Chapter 13.

Specification

The excess reactivity of the core SHALL be measured annually, not to exceed 15 months, and following core or control rod changes equal to or greater than 0.7% $\Delta k/k$ (~$1.00).

Basis

Excess reactivity measurements on this schedule ensure that no unexpected changes have occurred in the core and the core configuration does not exceed excess reactivity limits established in the Specification 3.1.2.

4.1.3 TRIGA Fuel Elements

Applicability

This specification applies to the surveillance requirements for the TRIGA fuel elements.

Objective

The objective is to verify the continuing integrity of the fuel element cladding.

Specification

Fuel elements and control rods with fuel followers SHALL be inspected visually for damage or deterioration and measured for length and bend in accordance with the following:

a. Before being placed in the core for the first time or before return to service.

b. Every two years, not to exceed 30 months, or at intervals not to exceed the sum of $3,500 in pulse reactivity, whichever comes first, for elements with a NP greater than 1 and for control rods with fueled followers.

c. Every four years, not to exceed 54 months, for elements with a NP of 1 or less.

d. Upon being removed from service. Those removed from service are then exempt from further inspection.

Basis

The frequency of inspection and measurement schedule is based on the parameters most likely to affect the fuel cladding of a pulsing reactor operated at moderate pulsing levels and utilizing fuel elements whose characteristics are well known.

4.2 Reactor Control and Safety System

4.2.1 Reactivity Worth

Applicability

This specification applies to the reactivity worth of the control rods.

Objective

The objective is to ensure that the control rods are capable of maintaining the reactor subcritical.

Specification

The reactivity worth of each control rod and the shutdown margin for the core loading in use SHALL be determined annually, not to exceed 15 months, or following core or control rod changes equal to or greater than 0.7% $\Delta k/k$ (~\$1.00).

Basis

The reactivity worth of the control rod is measured to ensure that the required shutdown margin is available and to provide an accurate means for determining the core excess reactivity, maximum reactivity, insertion rates, and the reactivity worth of experiments inserted in the core.

4.2.2 Reactivity Insertion Rate

Applicability

This specification applies to control rod movement speed.

Objective

The objective is to ensure that the reactivity addition rate specification is not violated and that the control rod drives are functioning.

Specification

The rod drive speed both up and down and the time from SCRAM initiation to the full insertion of any control rod from the full up position SHALL be measured annually, not to exceed 15 months, or when any significant work is done on the rod drive or the rod.

Basis

This specification ensures that the reactor will be promptly shut down when a SCRAM signal is initiated. Experience and analysis have indicated that for the range of transients anticipated for a TRIGA reactor, the specified SCRAM time is adequate to ensure the safety of the reactor. It also ensures that the maximum reactivity addition rate specification will not be exceeded.

4.2.3 Reactor Safety System

Applicability

The specifications apply to the surveillance requirements for measurements, channel tests, and channel checks of the reactor safety systems and watchdog circuit.

Objective

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

Specifications

a. A channel test of the SCRAM function of the wide range linear, power range linear, fuel temperature, manual, and preset timer safety channels SHALL be made on each day that the reactor is to be operated, or prior to each operation that extends more than one day.

b. A channel test of the detector power supply SCRAM functions for both the wide range and the power range and the watchdog circuit SHALL be performed annually, not to exceed 15 months.

c. Channel checks for operability SHALL be performed daily on fuel element temperature, wide range linear power, wide range log power, wide range reactor period/SUR, and power range linear power when the reactor is to be operated, or prior to each operation that extends more than one day.

d. The power range channel SHALL be compared with other independent channels for proper channel indication, when appropriate, each time the reactor is operated.

e. The pulse peak power channel SHALL be compared to the fuel temperature each time the reactor is pulsed, to ensure proper peak power channel operation.

Basis

System components have proven operational reliability.

a. Daily channel tests ensure accurate SCRAM functions and ensure the detection of possible channel drift or other possible deterioration of operating characteristics.

b. An annual channel test of the detector power supply SCRAM will ensure that this system works, based on past experience as recorded in the operation log book. An annual channel test of the watchdog circuit is sufficient to ensure operability.

c. The channel checks will make information available to the operator to ensure safe operation on a daily basis or prior to an extended run.

d. Comparison of the percent power channel with other independent power channels will ensure the detection of channel drift or other possible deterioration of its operational characteristics.

e. Comparison of the peak pulse power to the fuel temperature for each pulse will ensure the detection of possible channel drift or deterioration of its operational characteristics.

4.2.4 Reactor Interlocks

Applicability

These specifications apply to the surveillance requirements for the reactor control system interlocks.

Objective

The objective is to ensure performance and operability of the reactor control system interlocks.

Specifications

a. A channel check of the source interlock SHALL be performed each day that the reactor is operated or prior to each operation that extends more than one day except when the neutron signal is greater than the setpoint when the source is removed from the core.

b. A channel test SHALL be performed semi-annually, not to exceed 7 1/2 months, on the pulse mode inhibit interlock which prevents pulsing from power levels higher than one kilowatt.

c. A channel check SHALL be performed semi-annually, not to exceed 7 1/2 months, on the transient rod interlock which prevents application of air to the transient rod unless the cylinder is fully inserted.

d. A channel check SHALL be performed semi-annually, not to exceed 7 1/2 months, on the rod drive interlock which prevents movement of any rod except the transient rod in pulse mode.

e. A channel check SHALL be performed semi-annually, not to exceed 7 1/2 months, on the rod drive interlock which prevents simultaneous manual withdrawal of more than one rod.

Basis

The channel test and checks will verify operation of the reactor interlock system. Experience at the PSBR indicates that the prescribed frequency is adequate to ensure operability.

After extended operation, the photo neutron source strength may be high enough that removing the source may not drop the neutron signal below the setpoint of the source interlock. With a large intrinsic source there is no practical way to channel check the source interlock. In which case there is no need for a source interlock.

4.2.5 Overpower SCRAM

Applicability

This specification applies to the high power and fuel temperature SCRAM channels.

Objective

The objective is to verify that high power and fuel temperature SCRAM channels perform the SCRAM functions.

Specification

The high power and fuel temperature SCRAM's SHALL be tested annually, not to exceed 15 months.

Basis

Experience with the PSBR for more than a decade, as recorded in the operation log books, indicates that this interval is adequate to ensure operability.

4.2.6 Transient Rod Test

Applicability

These specifications apply to surveillance of the transient rod mechanism.

Objective

The objective is to ensure that the transient rod drive mechanism is maintained in an operable condition.

Specifications

a. On each day that pulse mode operation of the reactor is planned, a functional performance check of the transient rod system SHALL be performed. The transient rod drive cylinder and the associated air supply system SHALL be inspected, cleaned, and lubricated as necessary annually, not to exceed 15 months.

b. The reactor SHALL be pulsed annually, not to exceed 15 months, to compare fuel temperature measurements and peak power levels with those of previous pulses of the same reactivity value or the reactor SHALL NOT be pulsed until such comparative pulse measurements are performed.

Basis

Functional checks along with periodic maintenance ensure repeatable performance. The reactor is pulsed at suitable intervals and a comparison made with previous similar pulses to determine if changes in transient rod drive mechanism, fuel, or core characteristics have taken place.

4.3 Coolant System

4.3.1 Fire Hose Inspection

Applicability

This specification applies to the dedicated fire hoses used to supply water to the pool in an emergency.

Objective

The objective is to ensure that these hoses are operable.

Specification

The two (2) dedicated fire hoses that provide supply water to the pool in an emergency SHALL be visually inspected for damage and wear annually, not to exceed 15 months.

Basis

This frequency is adequate to ensure that significant degradation has not occurred since the previous inspection.

4.3.2 Pool Water Temperature

Applicability

This specification applies to pool water temperature.

Objective

The objective is to limit pool water temperature.

Specification

The pool temperature alarm SHALL be calibrated annually, not to exceed 15 months.

Basis

Experience has shown this instrument to be drift-free and that this interval is adequate to ensure operability.

4.3.3 Pool Water Conductivity

Applicability

This specification applies to surveillance of pool water conductivity.

Objective

The objective is to ensure that pool water mineral content is maintained at an acceptable level.

Specification

Pool water conductivity SHALL be measured and recorded daily when the reactor is to be operated, or at monthly intervals when the reactor is shut down for this time period.

Basis

Based on experience, observation at these intervals provides acceptable surveillance of limits that ensure that fuel clad corrosion and neutron activation of dissolved materials will not occur.

4.3.4 Pool Water Level Alarm

Applicability

This specification applies to the surveillance requirements for the pool level alarm.

Objective

The objective is to verify the operability of the pool water level alarm.

Specification

The pool water level alarm SHALL be channel checked monthly, not to exceed 6 weeks, to ensure its operability.

Basis

Experience, as exhibited by past periodic checks, has shown that monthly checks of the pool water level alarm ensures operability of the system during the month.

4.4 Confinement

Applicability

This specification applies to reactor bay doors.

Objective

The objective is to ensure that reactor bay doors are kept closed as per Specification 3.4.

Specification

Doors to the reactor bay SHALL be locked or under supervision by an authorized keyholder.

Basis

A keyholder is authorized by the Director or his designee.

4.5 Facility Exhaust System and Emergency Exhaust System

Applicability

These specifications apply to the facility exhaust system and emergency exhaust system.

Objective

The objective is to ensure the proper operation of the facility exhaust system and emergency exhaust system in controlling releases of radioactive material to the uncontrolled environment.

Specifications

a. It SHALL be verified monthly, not to exceed 6 weeks, whenever operation is scheduled, that the emergency exhaust system is operable with correct pressure drops across the filters (as specified in procedures).

b. It SHALL be verified monthly, not to exceed 6 weeks, whenever operation is scheduled, that the facility exhaust system is secured when the emergency exhaust system activates during an evacuation alarm (See Technical Specification 3.6.2 and 5.5).

Basis

Experience, based on periodic checks performed over years of operation, has demonstrated that a test of the exhaust systems on a monthly basis, not to exceed 6 weeks, is sufficient to ensure the proper operation of the systems. This provides reasonable assurance on the control of the release of radioactive material.

4.6 Radiation Monitoring System and Effluents

4.6.1 Radiation Monitoring System and Evacuation Alarm

Applicability

This specification applies to surveillance requirements for the area radiation monitor, the Beamhole Laboratory radiation monitor, the air radiation monitor, and the evacuation alarm.

Objective

The objective is to ensure that the radiation monitors and evacuation alarm are operable and to verify the appropriate alarm settings.

Specification

The area radiation monitor, the Beamhole Laboratory radiation monitor, the continuous air (radiation) monitor, and the evacuation alarm system SHALL be channel tested monthly not to exceed 6 weeks. They SHALL be verified to be operable by a channel check daily when the reactor is to be operated, and SHALL be calibrated annually, not to exceed 15 months.

Basis

Experience has shown this frequency of verification of the radiation monitor set points and operability and the evacuation alarm operability is adequate to correct for any variation in the system due to a change of operating characteristics. Annual channel calibration ensures that units are within the specifications defined by procedures.

4.6.2 Argon-41

Applicability

This specification applies to surveillance of the Argon-41 produced during reactor operation.

Objective

To ensure that the production of Argon-41 does not exceed the limits specified by 10 CFR Part 20.

Specification

The production of Argon-41 SHALL be measured and/or calculated for each new experiment or experimental facility that is estimated to produce a dose greater than 1 mrem at the exclusion boundary.

Basis

One (1) mrem dose per experiment or experimental facility represents 1% of the maximum 10 CFR Part 20 annual dose. It is considered prudent to analyze the Argon-41 production for any experiment or experimental facility that exceeds 1% of the annual limit.

4.6.3 ALARA

Applicability

This specification applies to the surveillance of all reactor operations that could result in occupational exposures to radiation or the release of radioactive effluents to the environs.

Objective

The objective is to provide surveillance of all operations that could lead to occupational exposures to radiation or the release of radioactive effluents to the environs.

Specification

As part of the review of all operations, consideration SHALL be given to alternative operational modes that might reduce staff exposures, release of radioactive materials to the environment, or both.

Basis

Experience has shown that experiments and operational requirements can, in many cases, be satisfied with a variety of combinations of facility options, core positions, power levels, time delays, and effluent or staff radiation exposures. Similarly, overall reactor scheduling achieves significant reductions in staff exposures. Consequently, ALARA must be a part of both overall reactor scheduling and the detailed experiment planning.

4.7 Experiments

Applicability

This specification applies to surveillance requirements for experiments.

Objective

The objective is to ensure that the conditions and restrictions of Specification 3.7 are met.

Specification

Those conditions and restrictions listed in Specification 3.7 SHALL be considered by the PSBR authorized reviewer before signing the irradiation authorization for each experiment.

Basis

Authorized reviewers are appointed by the facility director.

5.0 DESIGN FEATURES

5.1 Reactor Fuel

Specifications

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

a. The total uranium content SHALL be either 8.5 wt% or 12.0 wt% nominal and enriched to less than 20% uranium-235.

b. The hydrogen-to-zirconium atom ratio (in the ZrHx) SHALL be a nominal 1.65 H atoms to 1.0 Zr atom.

c. The cladding SHALL be 304 stainless steel with a nominal 0.020 inch thickness.

Basis

Nominal values of uranium loading, U-235 enrichment, hydrogen loading and cladding thickness are taken to mean those values specified by the manufacturer as standard values for TRIGA fuel. Minor deviations about these levels may occur due to variations in manufacturing and are not considered to be violations of this specification.

5.2 Reactor Core

Specifications

a. The core SHALL be an arrangement of TRIGA uranium-zirconium hydride fuel-moderator elements positioned in the reactor grid plates.

b. The reflector, excluding experiments and experimental facilities, SHALL be water, or D_2O, or graphite, or any combination of the three moderator materials.

Basis

The arrangement of TRIGA fuel elements positioned in the reactor grid plates ensures that adequate space is maintained for effective cooling. The Mark III TRIGA reactor is an open design without provision for reflector except in the form of natural water used for cooling and graphite elements which may be placed in the grid array. Restrictions on the reflector in this specification ensure any changes are analyzed against the criteria for experiments consistent with TS 3.7.

5.3 Control Rods

Specifications

a. The shim, safety, and regulating control rods SHALL have SCRAM capability and contain borated graphite, B_4C powder, or boron and its compounds in solid form as a poison in stainless steel or aluminum cladding. These rods may incorporate fueled followers which have the same characteristics as the fuel region in which they are used.

b. The transient control rod SHALL have SCRAM capability and contain borated graphite, B_4C powder, or boron and its compounds in a solid form as a poison in an aluminum or stainless steel clad. When used as a transient rod, it SHALL have an adjustable upper limit to allow a variation of reactivity insertions. This rod may incorporate a voided or a solid aluminum follower.

Basis

The poison requirements for the control rods are satisfied by using neutron-absorbing borated graphite, B4C powder, or boron and its compounds. These materials must be contained in a suitable cladding material, such as aluminum or stainless steel, to insure mechanical stability during movement and to isolate the poison from the pool water environment. SCRAM capabilities are provided by the rapid insertion of the control rods, which is the primary operational safety feature of the reactor. The transient control rod is designed for use in a pulsing TRIGA reactor and does not by design have a fuel follower.

5.4 Fuel Storage

Specifications

a. All fuel elements SHALL be stored in a geometrical array where the keff is less than 0.8 for all conditions of moderation.

b. Irradiated fuel elements SHALL be stored in an array which SHALL permit sufficient natural convection cooling by water such that the fuel element temperature SHALL NOT reach the safety limit as defined in Section 2.1 of the Technical Specifications.

Basis

The limits imposed by this specification are conservative and ensure safe storage and handling of nuclear fuel. GA-5402 "Criticality Safety Guide" places a general limitation on well-moderated U-235 to 300 grams per square foot. A rack of new 12 wt% elements would have no more than 288 grams per square foot (assuming 2" spacing and 60 grams U-235 per element). Additional work by General Atomics in 1966 showed that a 2x10 array of 12 wt% elements with no separation would have a keff = 0.7967. The fuel racks used for storage have an actual spacing of 2.0 inches and 2.5 inches and vertically offset by 20 inches, the calculations are conservative.

5.5 Reactor Bay and Exhaust Systems

Specifications

a. The reactor SHALL be housed in a room (reactor bay) designed to restrict leakage. The minimum free volume (total bay volume minus occupied volume) in the reactor bay SHALL be 1900 m^3.

b. The reactor bay SHALL be equipped with two exhaust systems. Under normal operating conditions, the facility exhaust system exhausts unfiltered reactor bay air to the environment releasing it at a point at least 24 feet above ground level. Upon initiation of a building evacuation alarm, the previously mentioned system is automatically secured and an emergency exhaust system automatically starts. The emergency exhaust system is also designed to discharge reactor bay air at a point at least 24 feet above ground level.

Basis

The value of 1900 m3 for reactor bay free volume is assumed in the SAR 13.1.1 Maximum Hypothetical Accident and is used in the calculation of the radionuclide concentrations for the analysis.

The SAR analysis 13.1.1 Maximum Hypothetical Accident does not take credit for any filtration present in the emergency exhaust system. The height above the ground of the release helps to ensure adequate mixing prior to possible personnel exposure.

5.6 Reactor Pool Water Systems

Specification

The reactor core SHALL be cooled by natural convective water flow.

Basis

Thermal and hydraulic calculations and operational experience indicate that a compact TRIGA reactor core can be safely operated up to power levels of at least 1.15 Mw (thermal) with natural convective cooling.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

6.1.1 Structure

The University Vice President for Research Dean of the Graduate School (level 1) has the responsibility for the reactor facility license. The management of the facility is the responsibility of the Director (level 2), who reports to the Vice President for Research, Dean of the Graduate School through the office of the Dean of the College of Engineering. Administrative and fiscal responsibility is within the office of the Dean.

The minimum qualifications for the position of Director of the PSBR are an advanced degree in science or engineering, and 2 years experience in reactor operation. Five years of experience directing reactor operations may be substituted for an advanced degree.

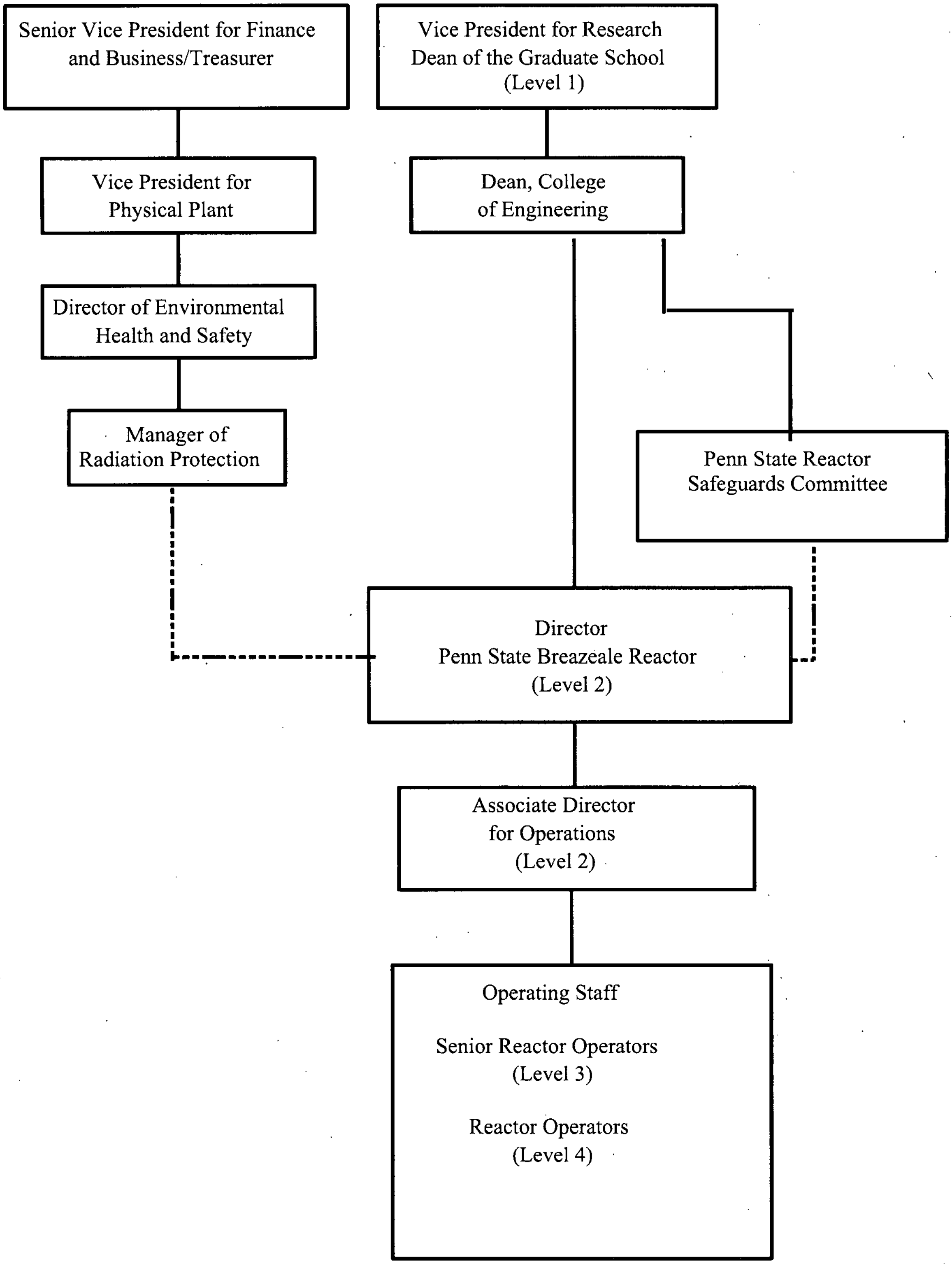
The Manager of Radiation Protection reports through the Director of Environmental Health and Safety, the assistant Vice President for Safety and Environmental Services, and to the Senior Vice President for Finance and Business/Treasurer. The qualifications for the Manager of Radiation Protection position are the equivalent of a graduate degree in radiation protection, 3 to 5 years experience with a broad byproduct material license, and certification by The American Board of Health Physics or eligibility for certification.

6.1.2 Responsibility

Responsibility for the safe operation of the reactor facility SHALL be within the chain of command shown in the organization chart. Individuals at the various management levels, in addition to having responsibility 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 and technical specifications.

In all instances, responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon appropriate qualifications.

ORGANIZATION CHART



6.1.3 Staffing

a. The minimum staffing when the reactor is not secured SHALL be:

1) A licensed operator present in the control room, in accordance with applicable regulations.

2) A second person present at the facility able to carry out prescribed written instructions.

3) If a senior reactor operator is not present at the facility, one SHALL be available by telephone and able to be at the facility within 30 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) Management personnel.

2) Radiation safety personnel.

3) Other operations personnel.

c. Events requiring the direction of a Senior Reactor Operator SHALL include:

1) All fuel or control-rod relocations within the reactor core region.

2) Relocation of any in-core experiment with a reactivity worth greater than one dollar.

3) Recovery from unplanned or unscheduled shutdown (in this instance, documented verbal concurrence from a Senior Reactor Operator is required).

6.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel SHALL meet or exceed the requirements of all applicable regulations and the American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1988, Sections 4-6.

6.2 Review and Audit

6.2.1 Safeguards Committee Composition

A Penn State Reactor Safeguards Committee (PSRSC) SHALL exist to provide an independent review and audit of the safety aspects of reactor facility operations. The committee SHALL have a minimum of 5 members and SHALL collectively represent a broad spectrum of expertise in reactor technology and other science and engineering fields. The committee SHALL have at least one member with health physics expertise. The committee SHALL be appointed by and report to the Dean of the College of Engineering. The PSBR Director SHALL be an ex-officio member of the PSRSC.

6.2.2 Charter and Rules

The operations of the PSRSC SHALL be in accordance with a written charter, including provisions for:

a. Meeting frequency - not less than once per calendar year not to exceed 15 months.

b. Quorums - at least one-half of the voting membership SHALL be present (the Director who is ex-officio SHALL NOT vote) and no more than one-half of the voting members present SHALL be members of the reactor staff.

c. Use of Subgroups - the committee chairman can appoint ad-Hoc committees as deemed necessary.

d. Minutes of the meetings - SHALL be recorded, disseminated, reviewed, and approved in a timely manner.

6.2.3 Review Function

The following items SHALL be reviewed:

a. 10 CFR Part 50.59 reviews of:

   1) Proposed changes in equipment, systems, tests, or experiments.

   2) All new procedures and major revisions thereto having a significant effect upon safety.

   3) All new experiments or classes of experiments that could have a significant effect upon reactivity or upon the release of radioactivity.

b. Proposed changes in technical specifications, license, or charter.

c. Violations of technical specifications, license, or charter. Violations of internal procedures or instructions having safety significance.

d. Operating abnormalities having safety significance.

e. Special reports listed in 6.6.2.

f. Audit reports.

6.2.4 Audit

The audit function SHALL be performed annually, not to exceed 15 months, preferably by a non-member of the reactor staff. The audit function SHALL be performed by a person not directly involved with the function being audited. The audit function SHALL include selective (but comprehensive) examinations of operating records, logs, and other documents. Discussions with operating personnel and observation of operations should also be used as appropriate. Deficiencies uncovered that affect reactor safety SHALL promptly be reported to the office of the Dean of the College of Engineering. The following items SHALL be audited:

a. Facility operations for conformance to Technical Specifications, license, and procedures (at least once per calendar year with interval not to exceed 15 months).

b. The requalification program for the operating staff (at least once every other calendar year with the interval not to exceed 30 months).

c. The results of action taken to correct deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operations that affect reactor safety (at least once per calendar year with the interval not to exceed 15 months).

d. The reactor facility emergency plan and implementing procedures (at least once every other calendar year with the interval not to exceed 30 months).

6.3 Operating Procedures

Written procedures SHALL be reviewed and approved prior to the initiation of activities covered by them in accordance with Section 6.2.3. Written procedures SHALL be adequate to ensure the safe operation of the reactor, but SHALL NOT preclude the use of independent judgment and action should the situation require such. Operating procedures SHALL be in effect and SHALL be followed for at least the following items:

a. Startup, operation, and shutdown of the reactor.

b. Core loading, unloading, and fuel movement within the reactor.

c. Routine maintenance of major components of systems that could have an effect on reactor safety.

d. Surveillance tests and calibrations required by the technical specifications (including daily checkout procedure).

e. Radiation, evacuation, and alarm checks.

f. Release of Irradiated Samples.

g. Evacuation.

h. Fire or Explosion.

i. Gaseous Release.

j. Medical Emergencies.

k. Civil Disorder.

l. Bomb Threat.

m. Threat of Theft of Special Nuclear Material.

n. Theft of Special Nuclear Material.

o. Industrial Sabotage.

p. Experiment Evaluation and Authorization.

q. Reactor Operation Using a Beam Port.

r. D_20 Handling.

s. Health Physics Orientation Requirements.

t. Hot Cell Entry Procedure.

u. Implementation of emergency and security plans.

v. Radiation instrument calibration

w. Loss of pool water.

6.4 Review and Approval of Experiments

a. All new experiments SHALL be reviewed for Technical Specifications compliance, 10 CFR Part 50.59 analysis, and approved in writing by level 2 management or designated alternate prior to initiation.

b. Substantive changes to experiments previously reviewed by the PSRSC SHALL be made only after review and approval in writing by level 2 management or designated alternate.

6.5 Required Action

6.5.1 Action to be Taken in the Event the Safety Limit is Exceeded

In the event the safety limit (1150°C) is exceeded:

a. The reactor SHALL be shut down and reactor operation SHALL NOT be resumed until authorized by the U.S. Nuclear Regulatory Commission.

b. The safety limit violation SHALL be promptly reported to level 2 or designated alternates.

c. An immediate report of the occurrence SHALL be made to the Chairman, PSRSC and reports SHALL be made to the USNRC in accordance with
Specification 6.6.

d. A report SHALL be prepared which SHALL include an analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report SHALL be submitted to the PSRSC for review.

6.5.2 Action to be Taken in the Event of a Reportable Occurrence

In the event of a reportable occurrence, (1.1.34) the following action SHALL be taken:

a. The reactor SHALL be returned to normal or shutdown. If it is necessary to shutdown the reactor to correct the occurrence, operations SHALL NOT be resumed unless authorized by level 2 or designated alternates.

b. The Director or a designated alternate SHALL be notified and corrective action taken with respect to the operations involved.

c. The Director or a designated alternate SHALL notify the office of the Dean of the College of Engineering and the office of the Vice President for Research, Dean of the Graduate School.

d. The Director or a designated alternate SHALL notify the Chairman of the PSRSC.

e. A report SHALL be made to the PSRSC which SHALL include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report SHALL be reviewed by the PSRSC at their next meeting.

f. A report SHALL be made to the Document Control Desk, USNRC Washington, DC 20555.

6.6 Reports

6.6.1 Operating Reports

An annual report SHALL be submitted within 6 months of the end of The Pennsylvania State University fiscal year to the Document Control Desk, USNRC, Washington, DC 20555, including at least the following items:

a. A narrative summary of reactor operating experience including the energy produced by the reactor, and the number of pulses $\geq$ \$2.00 but less than or equal to \$2.50 and the number greater than \$2.50.

b. The unscheduled shutdowns and reasons for them including, where applicable, corrective action taken to preclude recurrence.

c. Tabulation of major preventive and corrective maintenance operations having safety significance.

d. Tabulation of major changes in the reactor facility and procedures, and tabulation of new tests and experiments, that are significantly different from those performed previously and are not described in the Safety Analysis Report, including a summary of the analyses leading to the conclusions that no unreviewed safety questions were involved and that 10 CFR Part 50.59 was applicable.

e. A summary of the nature and amount of radioactive effluents released or discharged to environs beyond the effective control of the owner-operator as determined at or before 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, only a statement to this effect need be presented.

f. A summarized result of environmental surveys performed outside the facility.

6.6.2 Special Reports

Special reports are used to report unplanned events as well as planned major facility and administrative changes. These special reports SHALL contain and SHALL be communicated as follows:

a. There SHALL be a report no later than the following working day by telephone to the Operations Center, USNRC, Washington, DC 20555, to be followed by a written report to the Document Control Desk, USNRC, Washington, DC 20555, that describes the circumstances of the event within 14 days of any of the following:

   1) Violation of safety limits (See 6.5.1)

   2) Release of radioactivity from the site above allowed limits (See 6.5.2)

   3) A reportable occurrence (Section 1.1.34)

b. A written report SHALL be made within 30 days to the USNRC, and to the offices given in 6.6.1 for the following:

   1) Permanent changes in the facility organization involving level 1-2 personnel.

   2) Significant changes in the transient or accident analysis as described in the Safety Analysis Report.

6.7 Records

To fulfill the requirements of applicable regulations, records and logs SHALL be prepared, and retained for the following items:

6.7.1 Records to be Retained for at Least Five Years

a. Log of reactor operation and summary of energy produced or hours the reactor was critical.

b. Checks and calibrations procedure file.

c. Preventive and corrective electronic maintenance log.

d. Major changes in the reactor facility and procedures.

e. Experiment authorization file including conclusions that no unreviewed safety questions were involved for new tests or experiments.

f. Event evaluation forms (including unscheduled shutdowns) and reportable occurrence reports.

g. Preventive and corrective maintenance records of associated reactor equipment.

h. Facility radiation and contamination surveys.

i. Fuel inventories and transfers.

j. Surveillance activities as required by the Technical Specifications.

k. Records of PSRSC reviews and audits.

6.7.2 Records to be Retained for at Least One Training Cycle

a. Requalification records for licensed reactor operators and senior reactor operators.

6.7.3 Records to be Retained for the Life of the Reactor Facility

a. Radiation exposure for all facility personnel and visitors.

b. Environmental surveys performed outside the facility.

c. Radioactive effluents released to the environs.

d. Drawings of the reactor facility including changes.