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December 22nd 2015

US Nuclear Regulatory Commission
Document Control Desk
Washington DC 20555-0001
Attention: Michael Balazik, Project Manager

Re: Docket 50-326, License R-116
Re: ML14135A503

Dear Mr. Balazik,
Please find enclosed the response regarding the referenced technical RAIs (dated July 15, 2014). There are also two attachments to the response.

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

Executed on December 22nd 2015

G. E. Miller

Dr. George E. Miller
Supervisor

cc. Prof. A.J. Shaka, Director, UCI Nuclear Reactor Facility
J. T. Wallick, Associate Reactor Supervisor
G. L. Bosgraaf, UCI Radiation Safety Officer

A020
NRR

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION DATED July 15 2014

REGARDING LICENSE RENEWAL

FOR THE UNIVERSITY OF CALIFORNIA, IRVINE NUCLEAR REACTOR FACILITY

LICENSE NO. R-116

DOCKET NO. 50-326

The U.S. Nuclear Regulatory Commission (NRC) is continuing its review of the University of California, Irvine (UCI) application for the renewal of Facility Operating License No. R-116 for the UCI Nuclear Reactor Facility (UCINRF), dated October 18, 1999 (a redacted version of the application is available on the NRC's public Web site at www.nrc.gov under Agencywide Documents Access and Management System (ADAMS) Accession No. ML083110112), as supplemented. The NRC staff review of your safety analysis report (SAR) and proposed technical specifications (TSs) identified the following items which need additional clarification or information. Your responses are requested within 30 days from the date of this letter.

This request for additional information (RAI) is, in part, based on a comparison of the UCINRF safety analysis report and the proposed TSs with NUREG-1537, "Guidance for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," dated February 1996 and the American Nuclear Standards Institute/American Nuclear Society (ANSI/ANS)-15.1-2007, "The Development of Technical Specifications for Research Reactors."

1) *In NUREG-1537, Part 1, Section 1.4, "Shared Facilities and Equipment," it states that "[c]omplete descriptions and any safety implications that result from sharing facilities or systems should be evaluated in and referenced to the appropriate chapter of the SAR."*

   a) *The UCINRF RAI response dated August 1, 2011 (ADAMS Accession No. ML11255A073), Section 3.6, "Ventilation System," pages 6-9, describes the UCINRF ventilation system and Section 13.2, "Maximum Hypothetical Accident," pages 12-19, expands the description by predicting doses during a maximum hypothetical accident. During normal operations argon-41 is produced and released. The UCINRF ventilation system shares components with the chemistry building exhaust system.*

      i) *Could airborne radioactive material being exhausted from the reactor room during accident or normal conditions be discharged into public spaces because of system failures?* We believe no backflow into public spaces is possible.

      ii) *If not, explain why not.*

As indicated in the diagrams presented earlier, the shared system does not integrate until the roof level of Rowland Hall. Three main fans there run continuously with only one required to provide sufficient flow for normal building exhaust and prevent back-flow. Positive flow is thus assured from all directions at the mixing point. The limiting conditions for reactor operation require that ventilation operation meets flow criteria. Thus reactor operation would be halted or not begin if ventilation failure occurred. Furthermore, the ventilation system operation is monitored continuously for flow and pressure (24/7) by UCI Central Plant. In the event of an alarm or failure condition, Central Plant calls the facility. In the event of no response, investigation and repair proceeds and if access is needed, the UCIPD will call the reactor call list for assistance. This is evidenced by the need for the

facility staff to notify Central Plant if a door is opened for more than a couple of minutes, creating an alarm condition in the pressure differential monitor at the facility and at Central Plant.

iii) If this backflow is possible, what are potential doses to members of the public?

None is anticipated, see above.

iv) As airborne radioactive material is discharged through the ventilation system ducting during accidents or normal operation, could members of the public adjacent to the ducting be exposed to radiation?

Public exposure is minimal from portions of the ducts leading to the roof. This is unchanged from the earlier design. Normal operations result in only minimal exposures within the facility and even less to anyone in adjacent laboratories or areas adjacent to the ventilation ducts.

TLD dosimeters placed at facility and ventilation systems perimeters confirm this by exhibiting little or no exposure above background (or even below!) within quarterly measurement limits. These have been reported in annual reports.

In the revised SAR, Sections 13.2.5 and 13.2.7 exposures to building occupants in adjacent laboratories were discussed. The same arguments pertain to those adjacent to the ventilation ducts but with a lower source term. SAR 13.2.7 (November 2011) states:

> "There is no pathway for airborne activities to reach the adjacent areas of the building, so that the deep dose rates established in 13.2.5 for individuals within the facility may be utilized. Such individuals will not be immersed in an infinite cloud of radioactive volatiles, so the facility may be considered as a single source. Further only gamma-ray exposures would be experienced since the walls and windows are thick enough to absorb all beta radiation. There are no doors directly opening from the facility to adjacent building areas, so no leakage to interior building areas is anticipated."

The ventilation duct contains only a fraction of the source term so any exposure is much reduced. In any accident release situation, the purge exhaust passes a small fraction of the effluent gases only (there is a particulate filter) through the duct in passing occupied rooms. In such event the building will be evacuated ASAP so occupancy will be eliminated quickly. Areas nearby the ducts are mostly of low occupancy level (e.g., research labs and stockroom).

v) If not, explain why not. If exposure to radioactive material is possible, what are the potential doses to members of the public?

See above. Again this is not new and the situation has existed since the original license.

SAR Section 13.2.7 states

"Section 13.2.5 established a maximum dose rate to an individual within the facility of about 50 mrem/hr. Since this is an immersion calculation, it is reasonable to assume a distance factor before estimating the dose rate through a wall. The closest a person can stand outside the facility is in the adjacent hallway at the "window" wall. Assuming a whole body distance on each side of the wall, and including the wall thickness, it is reasonable to credit a distance factor of 5 feet. Making a further assumption that the source is an effective point source inside the facility, this could provide a distance factor of 5^2 or 25. Thus the external dose rate estimate is reduced by 50/25 or to 2 mrem/hr. making no allowances for attenuation by air or wall materials, This is at the acceptable rate[1] for public individuals over short time periods. In such an accident, the building will be cleared within less than 1 hour of the accident so that extensive exposure at this rate (which would need to exceed 25 hours to attain the 50 mrem annual limit[2]) would not be realized. Since there is no internal exposure, this exposure is the TEDE estimate. As it is so low, even if there are large uncertainties in the estimate, this appears to be entirely acceptable. Anyone at a further distance, such as in nearby laboratories or offices, would receive less as a result of additional distance factors and absorption reductions. It is important to emphasize that the building ventilation design is such that no mechanism exists for mixing of air between the reactor area and adjacent rooms, so no internal exposure can result from the MHA except to persons in the facility.

The above estimates were presented for adjacent laboratories based on the reactor room (volume 6.6 E8 cm^3) as a source. A passing duct is a more limited source. Assuming a duct of 1.5 feet diameter and 10 feet high passing a space, the source volume is now 5E5 cm^3. Assuming the same radioactivity concentration as the reactor room with the same intervening wall shielding and personnel proximity assumptions, exposures to personnel in rooms near the ducts would be (conservatively) only 1/1000 of the 2 mr/h estimated for the labs.

b) The UCINRF RAI response dated January 27, 2010 (ADAMS Accession No. ML100290365), Attachment C, SAR Chapter 5, "Reactor Coolant System," pages 43-47, provides an updated SAR describing the water cooling system and water purification system. The UCI chilled water system supplies cooling water for the shell side of the heat exchanger.

i) Discuss the radiological impact of a leak in the heat exchanger. Is leakage flow from the chilled water side into the primary side of the heat exchanger under all operating conditions of both the primary system and chilled water system?

Leakage flow within the heat exchanger is from secondary into the primary as established by pressure differentials under all conditions. The facility has recently installed pressure sensors in both primary and secondary systems to enable ongoing verification of this claim.

ii) If not, what is the maximum radiological impact of primary coolant entering the chilled water system?

[1] 10CFR20.1302(2)(ii)
[2] 10CFR20.1302(2)(ii)

None is anticipated (see above)

2) *In NUREG-1537, Part 1, Section 4.3, "Reactor Tank or Pool," it states that "the applicant should assess the possibility of uncontrolled leakage of contaminated primary coolant and should discuss preventive and protective features."*

a) *The UCINRF RAI response dated July 14, 2010 (ADAMS Accession No. ML101970039), Section 13.4, "Loss of Reactor Pool Water (Coolant)," pages 6-12, describes pool water leakage. The response states that rain and irrigations of areas outside the reactor building forced the drilling of wells adjacent to Rowland Hall. There seems to be a discrepancy in two statements: (1) that automatic sump pumps are used to drain these wells and (2) that samples are taken of the water being removed and are assayed by gamma-ray spectrometry. Clarify how the process exists for sampling radioactive contaminants in the discharge of the automatic sump pump system prior to returning water to the campus sewage system.*

Sampling is done on a scheduled (monthly) basis as a check on potential leakage, not prior to any release. Use of the pump would cease if reactor contaminated water were detected. As the reactor pool water is itself also checked monthly and found to be only very low contamination, the degree of contamination after soil filtering would be so low that modest release of water would be well below allowable sewer discharge levels.

b) *Previous annual reports identified that ground water has been observed in fuel element storage pits in the floor of the reactor room. Confirm whether or not these pits are used to store fuel. If they are or will be used to store fuel, discuss actions that would be taken to prevent water ingress prior to using the pits to store fuel and consider modifying TSs to reflect these actions, or explain why leakage into the storage pits would not make them unsuitable for fuel element storage. Alternatively, consider incorporating a TS that states, "fuel storage shall be limited to in pool storage only."*

These pits were originally designated for either wet or dry storage in very low criticality arrangements. Fuel stored in them will be secured in racks designed to limit reactivity as well as to raise the elements above any potential ground water levels. Off the five pits, only two have ever been subject to water incursion, and these would be used for fuel only as a last resort. In addition the pumping out of adjacent ground water will reduce the probability of future incursion. We do wish to be able continue use the pits as they provide additional security for new fuel, as well as options for used fuel.

c) *The UCINRF RAI response dated July 14, 2010, Section 13.4.1, "Mechanisms and Rate of Loss of Coolant," estimates that it would take 19.3 hours to drain the pool.*

i) *What is the maximum water loss that can occur without detection; that is, if the pool started leaking after the reactor staff left for the night, what is the maximum amount of pool water that could be lost without staff becoming aware?*

The pool alarm is set at just less than 12 inches below the tank rim. The normal water level is maintained at about 6 inches below the tank rim. Each inch corresponds to approximately 80 gallons of water. 6 x 80 = 480 gallons before an alarm would be reported to staff. This is less than 2% of the water (25,000).

ii) How many gallons per week would be the minimum detectable leakage rate based on UCINRF historical primary makeup rates for normal operations?

Normal loss rate based on history since we included a pool water level measurement gauge is taken to be between 10 and 30 gallons per week with a mean value of 20 gallons/week. The value varies with operating schedules and temperature variations. Data for pool level is recorded on each daily start-up check sheet, and rates of loss are calculated monthly. A rate of 1 inch drop in 1 week, or four times that mean value would be considered highly abnormal and cause for investigation. The pool water at this facility has a very low level of residual (long-lived) radioactivity, and ground-water in this area is not utilized in any way for public consumption, so that a slow leakage rate does not present a significant risk to the public. Requirements for checks and calibrations of the pool water level monitoring system have been incorporated in the revised Technical Specifications.
It can be noted that a loss of 1 foot of water above the core results in a negligible increase in direct radiation level in any rooms over the reactor. SAR 13.3 shows that 10 hours or more will elapse after an alarm condition before dose rates become of concern.

3) *In NUREG-1537, Part 1, Section 4.5.1, "Normal Operating Conditions," it requires establishment of a limiting core configuration. According to ANSI/ANS-15.1-2007, the reference core should consider Xenon and other poisons in calculating a shutdown margin (SDM). The UCINRF proposed TS 3.1.2, "Shutdown Margin," states, "[t]he shutdown margin provided by the control rods shall be greater than $0.55…." Given the reference core condition established in RAI response dated June 7, 2011 (ADAMS Accession No. ML111950452), the resulting SDM reactivity could be at least $0.30 less. Provide justification for this variance or revise proposed TS 3.1.2 to reflect a $0.85 SDM to support the inclusion of Xenon in the definition of reference core condition or consider incorporating the definition* ***Reference Core Condition:*** *The reference core condition is the condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible.*

Reference Core Condition has been revised to eliminate the ($0.30) modifier.

4) *NUREG-1537, Part 1, Section 11.1.1.1, "Airborne Radiation Sources," states that the SAR should discuss the production of airborne particles, aerosols, vapors, and nitrogen-16 (N-16) or other radionuclides. Describe the change or difference in radiation fields within the confinement space with the N-16 diffuser ON versus with the N-16 diffuser OFF or state the transport time of N-16 with and without the diffuser in operation.*

The following radiation levels in mr/h have been measured within the facility with the reactor at 250 kilowatts steady state power level:

Location	With diffuser	Without diffuser
Immediately over pool	1.6	20
Ceiling over pool	0.8	1.8
At facility corridor wall	0.03	0.3
Background - reactor OFF	0.03	0.03

5) *NUREG-1537, Part 1, Section 11.1.1.2, "Liquid Radiation Sources," states that the applicant should discuss compliance with the applicable sections of Title 10 of the Code of Federal Regulations (10 CFR) Part 20, "Standards for Protection Against Radiation." There are two places in the UCINRF proposed TSs that discuss liquid effluent, proposed TS 3.7.2, "Effluents," Specification b. and proposed TS 4.7, "Radiation Monitoring Systems and Effluents," Specification e. The regulations in 10 CFR 20.2003 require that liquid effluents disposed of by release into sanitary sewerage be readily soluble. There is no mention in the RAI responses or Bases discussions that these propose TSs limit releases to readily soluble radionuclides. Confirm that these proposed TSs limit release to readily soluble radionuclides, or provide justification as to why not.*

Yes all references to liquids are to soluble radionuclides. Others are disposed as "dry waste". The appropriate TS sections have been revised to require demonstration of solubility.

6) *In NUREG-1537, Part 1, Section 11.1.4, "Radiation Monitoring and Surveying," it requires a description of all radiation monitoring equipment.*

   a) *State the set points of the radiation area monitors and continuous air radiation monitor (CAM) listed in UCINRF proposed TS 3.7.1, "Radiation Monitoring Systems," and provide the bases for these set points.*

Set points for the radiation monitors and CAM are as follows:

Instrument	Alert	Alarm	Count rate
Radiation Area Monitors (2)	2 mr/h	10 mr/h	none
Continuous Air Particulate Monitor	none	*1E-8 μCi/ml	5000 cpm

Bases: RAM – 2 mr/h is the normal standard for short –term exposures to the public. 10 mr/h alarms at more serious but still safe for short term use. It also functions in after-hours as an alarm for inappropriate activity within the facility.

CAM – Based on instrument manufacturer recommendations. *DAC is from manufacturer recommendation for I-131. This is a factor of 2 lower than the 10 CFR 20 Appendix B, Table 2 limit value of DAC for I-131 for occupancy situations of 2E-8 μCi/ml. As the ventilation system provides for dilution in normal mode of x 100, and emergency node of x 1600, this provides an alarm level below the public release limit (Table 2) of 2E-10 μCi/ml.

   b) *The use of the acronym CAM is inconsistent. In the UCINRF proposed TS 3.5.1, "Ventilation System," and proposed TS 3.5.2, "Ventilation During Emergency Situations," it is defined as "continuous particulate air monitor" while proposed TS 3.7.1, "Radiation Monitoring Systems," defines it as "continuous air radiation monitor." Provide consistent nomenclature.*

The acronym CAM has now been identified as Continuous Air Particulate Monitor in TS 3.7.1., and utilized throughout the TS.

7) *NUREG-1537, Part 1, Section 13.1.2, "Insertion of Excess Reactivity," lists insertion-of-excess-reactivity events which includes a ramp insertion of reactivity by drive motion of the most reactive control or shim rod, or ganged rods. Provide details of the ramp insertion analysis including the reactor trips in order to demonstrate the protection of the safety limit.*

The TRIGA® core has been analyzed for a pulse excess reactivity insertion of up to $3.00 within 100 milliseconds using the ATR and FTR control rods. A ramp rate of $30 per second is the average in such an operation. Periods in the range of milliseconds are predicted and experienced.
Maximum fuel temperatures of 350 °C - well within the safety limit were projected and have been confirmed by measurements. Fuel temperatures in the pulse are also below the proposed scram setting (425 °C). In the original safety analysis, Chapter 7, and the relicense submittal (Section 13.3), model analysis suggests a fuel temperature condition following a pulse insertion of $3.00 in an accident situation at a steady state power level of 250 kw at a maximum of 570 °C. (240 °C steady state fuel temperature + 330 °C additional from the pulse). At UCI an accident of that magnitude is impossible as significant reactivity from the $3.00 license TS limit would be utilized in reaching 250 kilowatts, leaving only a reduced amount for accidental insertion.
While a slower ramp might be expected to produce higher fuel surface temperatures in a fuel with smaller or slower negative temperature feedback, these are not of concern for a homogeneous TRIGA® fuel which has a safety limit based on fuel meat temperature. Wherever and whenever the fuel meat is hot the negative feedback will occur, reducing power density rapidly at that location.
However, in response to this request a slow drive ramp insertion analysis has been conducted and the results are presented and discussed in an attachment to this document.

The following RAIs are based on a review of the proposed UCINRF TSs, as provided in your RAI response dated March 1, 2012 (ADAMS Accession No. ML12087A215), and follow up RAI response dated September 11, 2012 (ADAMS Accession No. ML12256A897).

8) *10 CFR Part 50.36(a)(1) states, in part, that the bases are not part of the TSs. Provide a TS statement to support the requirement of 10 CFR 50.36 (a)(1), or justify why one is not required. An example of TS statements that have been approved by the NRC staff include the following: "Included in this document are the Technical Specifications and the 'Bases' for the Technical Specifications. These bases, which provide the technical support for the individual Technical Specifications, are included for informational purposes only. They are not part of the Technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere."*

We have included the above statement verbatim on our introductory page.

9) *The UCINRF proposed TS 1.0, "Definitions," provides a definition for CHANNEL CALIBRATION that is not consistent with the guidance found in NUREG-1537 and ANSI/ANS-15.1-2007. Revise the proposed TS definition of CHANNEL CALIBRATION to be consistent with the guidance in NUREG-1537 and ANSI-15.1, or provide a justification for the proposed definition.*

This has been revised.

10) *The UCINRF proposed TS 1.0, "Definitions," contains a formatting inconsistency under CONTROL ROD, where the subparts are not spaces or indented, similar to other parts of the proposed TSs. Provide justification for formatting the subparts or revise for consistency.*

This has been corrected.

11) *The UCINRF proposed TS 1.0, "Definitions," CONTROL ROD contains an inconsistency in which subpart, "c. Adjustable Transient (ATR)," describes how control rod position is adjusted using an electric motor drive, while subpart, "d. Fast Transient (FTR)," does not have an equivalent description of how control rod position is adjusted. The SAR describes the use of pneumatic control for the FTR. Provide a justification for this difference or revise the definition of FTR.*

The descriptions have been revised to better parallel the other rods.

12) *The UCINRF proposed TS 1.0, "Definitions," INITIAL STARTUP states, "...before operation during the day at steady-state power level above 1 kilowatt, or by pulsing the reactor." Clarification is required on the significance of 1 kilowatt. Explain the purpose for the 1 kilowatt threshold or provide a basis describing the reason why 1 kilowatt is included in this proposed TS definition. Include in your discussion the need to have a senior reactor operator (SRO) present during the initial startup as required by 10 CFR 50.54(m)(1) if power does not reach 1 kilowatt.*

This has been revised to read:

"INITIAL STARTUP is the first start-up from reactor secured condition on any day when the reactor is to be operated in order to verify core excess and other instrument parameters."

The need for a senior operator remains in accordance with the regulations.

13) *The UCINRF proposed TS 1.0, "Definitions," REACTOR SECURED, subpart (2)(d) states, "No experiments are being moved or serviced that have a reactivity worth exceeding the maximum value allowed for a single experiment, or $1.00." Based on this definition, the exact reactivity limit is not clear. Only the most conservative, or that, which is applicable to the UCI reactor should constitute this definition (i.e., either $1.00 <u>or</u> the maximum value allowed for a single experiment). Provide justification for the proposed TS or revise the definition to identify only one condition.*

The REACTOR SECURED definition has been revised accordingly.

14) *The UCINRF proposed TS 1.0, "Definitions," SCRAM TIME states, "is the elapsed time between the initiation of a scram signal and a specified movement of a control or safety device." This definition is not clear as applied to the UCI reactor. As applicable to the UCI reactor, describe the specified movement (i.e., 85 percent of reactivity, all rods on the bottom, etc.) that constitutes this definition. For example, SCRAM TIME is the elapsed time between reaching a limiting safety system set point and the instant the slowest scrammable control rod reaches its fully-inserted position or SCRAM TIME is the elapsed time from the initiation of a scram signal to the time the slowest scrammable control rod is fully inserted. Also define*

the control or safety device specific to the UCI reactor. Provide justification for the proposed TS or revise the definition to identify one specific movement of a control or safety device.

The definition of SCRAM TIME has been revised accordingly.

15) *The UCINRF proposed TS 1.0, "Definitions," SUBSTANTIVE CHANGES may cause confusion in implementing 10 CFR 50.59, "Changes, tests, and experiments." Provide justification as to why this definition is required and is consistent with 10 CFR 50.59 or delete the definition.*

This definition has been removed, based on discussions on February 5th 2015 with NRC personnel.

16) *The UCINRF proposed TS 1.0, "Definitions," section does not include a definition for MOVEABLE EXPERIMENTS as endorsed in the ANSI/ANS-15.1-2007. The term "moveable" can mean "unsecured," but not all "unsecured" experiments are "moveable." Provide a definition of moveable experiments, and include this definition in UCINRF proposed TS 3.8.1, "Reactivity Limits," or provide justification as to why this definition should not be included.*

This has been included within the definition of EXPERIMENT.

17) *The UCINRF proposed TS 3.1.1, "Steady-state Operation," Applicability section, contains a typographical error in which a period is missing. Clarify or correct the typographical error. Also, review your proposed TSs for typographical or formatting errors and propose corrections as needed.*

This has been attempted through use of an "external" reviewer.

18) *NUREG-1537, Appendix 14.1, Section 3.1.6, "Fuel Element Inspection Parameters," Specification b., identifies limits on TRIGA fuel, including burnup. The UCINRF proposed TS 3.1.6 does not include a burnup limit. Provide a TS limit for fuel element burnup or provide a justification as to why this limitation should not apply to the UCINRF reactor.*

A burn-up limit has been included as requested as TS 3.1.6.e. The basis is NUREG 1282.

19) *The UCINRF proposed TS 3.2.3, "Reactor Safety System," "Table 2. Minimum Reactor Safety Channels," requires clarification as identified below.*
a) Second column, first row titled, "Function and trip level ***maximum*** *setting," may be misinterpreted. Provide clarity for this table by adjusting the title to read, "Function and trip level settings," and labeling corresponding set points using universal symbols <, >, ≥, ≤ or provide justification for the use of the proposed labeling system.*

These changes have been incorporated as requested.

b) Second column, third row lists, "Scram-425°C (IFE)" but does not indicate the range of acceptability. Identify if this should read "Scram ≤ 425°C (IFE)," or provide justification for proposed labeling system.

See above.

c) *Second column, fourth row listing, "Scram-110% of 250kw" should identify, "Scram-≤ 110% of 250kw **(275kw(t))**," or provide justification for proposed labeling system.*

See above.

d) *Second column, eighth row listing, "Scram-if motion of 3% g (0.03g) is exceeded" should identify, "Scram-if motion of ≤ 3% g (0.03g)," or provide justification for proposed labeling system.*

See above.

20) *The UCINRF proposed TS 3.2.3, "Reactor Safety System," "Table 3. Minimum Interlocks," first column, fifth row lists the interlock, "REG, SHIM, ATR Control Rod Drives" and the second column, fifth row lists the function, "Prevent movement of REG and SHIM rods in pulse mode." Confirm that there is no interlock for the ATR control rod drive that prevents its movement in pulse mode or make the appropriate corrections to the table?*

A revision has been made to clarify the interlocks present in the pulse mode.

21) *The UCINRF proposed TS 3.3.1, "Pool Water Level," Specification b., requires clarification for the following:*

a) *The use of the term "...shall operate **24/7**..." is colloquial and may be misinterpreted. Provide clarity to this requirement by stating, "...shall operate **continuously**..." or provide a justification for the use of the proposed language.*

This revision has been made as requested.

b) *The condition states, "Visual checking of water level shall be substituted every 10 hours during periods when the alarm is found to be inoperable and no substitute level device has been implemented." How long can visual checking every 10 hours exist before the pool water level alarm is restored to operation? How long can a substitute level device be implemented? Justify your proposed time periods.*

Since observation is decidedly painful, the restoration will be important. Depending on the availability of spares and delivery thereof, we expect the restoration to be completed with 7 working days at the outside, and hopefully within 1-2 days.

22) *The UCINRF proposed TS 3.8.2., "Materials," Specification b., states, "Explosive materials shall not be irradiated in quantities greater than 25 milligrams of TNT equivalent. Explosive materials in lesser quantities may be..." Clarification is required because a legal loop hole exists in this statement that does not specify the requirements for exactly 25 milligrams of TNT. The TS should identify, "...Explosive materials **in the amount of 25 milligrams or** lesser quantities may be..." Provide justification for the proposed language, or modify the proposed TS to conform to current NRC guidance.*

This change has been made as requested.

23) *The UCINRF proposed TS 3.8.2., "Materials," Specification b., requires that the pressure of accidental detonation of explosive material has been calculated and/or experimentally*

determined to be less than half the design pressure of the container the explosive is irradiated in. Provide calculations of the design of an example container to demonstrate the capability to meet the requirements of TS 3.8.2., Specification b.

This would be a requirement of the experimenter proposing such a device, and would verify this by use of outside consultants who have expertise in this arena.

Calculators of thickness of sealed containers to meet ASME design pressures can be found on-line at such sites as http://www.checalc.com/calc/vesselThick.html

As to explosive analysis, for combustibles, NRC 1805 provides a calculator. For others the type and design of the containment are crucial, and DOT guidance is helpful.
A useful general reference for explosives seems to be http://fas.org/man/dod-101/navy/docs/es310/chemistry/chemistry.htm

There is minor disagreement in the literature as to the proper equation to be used for explosion of TNT. This depends on oxygen availability. Using the conservative value of 11 mol of gas per mol (227.13g) of TNT, we find that 23 mg of TNT (1/10,000 mol) would yield, on explosion, 11 x 1/10,000 x 24.6 L of gas at NTP = 0.027 L = 27 mL, assuming ideal gas conditions. Confined to a volume of 10 mL (for the purpose of analysis), this would result conservatively, in a pressure of 27/10 x 1 atm or 2.7 atm. Confining a pressure of >= 5.4 atm would be required to meet the specification. However this ignores any heating effect. TNT releases 2175 J/g, so 23 mg will release only 50 J. Assuming a steel capsule of mass 100g and 0.5 J/g-K for specific heat, this would yield a temperature rise of only 1K. Such effect is minimal. Even a 100°C rise would only raise the calculated pressure from 2.7 to 3.7 atm, for a required strength of 7.4 atm.

24) The UCINRF proposed TS 3.8.2, "Materials," Basis section references a letter dated December 2, 2011, which describes the dose limits in terms of total effective dose equivalent (TEDE) and thyroid individual organ doses. Some confusion exists in that the dose limits should only be in terms of TEDE for this licensing action. Consider removing the individual organ dose references in the bases, or justify how this limit is applicable to establishing the TS limit.

These revisions have been made as requested.

25) The UCINRF proposed TS 3.8.3, "Failure or Malfunction," requires clarification. The wording of the specification is not standard and results in some confusion. Provide justification for the proposed language, or modify the proposed TS to conform to current NUREG-1537 guidance. An example TS reads as follows:

TS 3.8.3 Experiment Failure and Malfunctions

Specifications

Where the possibility exists that the failure of an experiment under (a) normal operating conditions of the experiment or the reactor (b) credible accident conditions in the reactor or (c) possible accident conditions in the experiment could release

radioactive gases or aerosols to the reactor room or the unrestricted area, the quantity and type of material in the experiment shall be limited in activity such that exposures of the reactor personnel to the gaseous activity or radioactive aerosols in the reactor room or control room will not exceed the occupational dose limits in 10CFR 20.1201. Additionally, exposures to members of the public to these releases in the unrestricted areas will not exceed the dose limits in 10CFR 20.1301, assuming that:

a. 100% of the gases or aerosols escape from the experiment;

b. If the effluent from an experimental facility exhausts through a holdup tank, which closes automatically on high radiation levels, the assumption shall be used that 10% of the gaseous activity or aerosols produced will escape;

c. If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, the assumption shall be used that 10% of the aerosols produced escape;

d. For materials whose boiling point is above 55°C and where vapors formed by boiling this material could escape only through an undisturbed column of water above the core, the assumption shall be used that 10% of these vapors escape; and

e. If an experiment container fails and releases material which could damage the reactor fuel or structure by corrosion or other means, physical inspection shall be performed to determine the consequences and the need for corrective action.

These revisions have been made as requested.

26) The UCINRF proposed TS 4.2, "Reactor Control and Safety Systems," Specification e., states, "On each day that pulse mode operation of the reactor is planned, a functional performance check of the transient (pulse) rod system…" Functional performance check is not a defined term. Is this a channel check? If so, use the properly defined term, or define and justify the term functional performance check.

TS 4,2 has been rewritten to utilize the term "OPERABLE" which has been defined.

27) The UCINRF proposed TS 4.2, Reactor Control and Safety Systems," Specification h., states, "A channel check of the pool water temperature measuring channel…" Confirm that a channel check is appropriate as this surveillance. The NUREG-1537 guidance identifies a calibration of measuring channels. Explain the use of a channel check versus a calibration or modify the proposed TS to conform with NRC guidance.

An annual calibration requirement for the pool water temperature measuring channel has been included in TS.4.2, replacing the daily "check". In addition, the requirement for an annual calibration of the pool water level measuring channel, has been moved from 4.3 to 4.2 to better cluster the "annual calibration" requirements. The pool temperature "alarm" is advisory, as it alerts operators to a procedural requirement not directly connected to safety, and provides no automatic scram, or outside agency notifications.

28) The UCINRF proposed TS 4.3, "Reactor Pool Water," Specification b., is not included in the list

of surveillances that cannot be deferred, found in TS 4.0, "General," Specification a., which states "Surveillance requirements may be deferred during prolonged (periods greater than 1 month) reactor shutdown (except Technical Specifications 4.3.a, 4.3.c, 4.3.e, 4.3.f and 4.3.g)." Why is TS 4.3, Specification b. allowed to be deferred for long term shutdown? Provide justification for this deferral, or modify the proposed TS to include TS 4.3, Specification b. on this list of exceptions.

This has been revised as requested and to acknowledge other changes made in numbering items.

29) The UCINRF proposed TS 6.1.2, "Responsibilites," "Figure 1. UCI Reactor Organization Chart," requires clarification. The solid-line and dashed-line relationships are not identified in a leger or within TS description. Confirm and indicate what these connections mean.

This has been included as requested.

30) The UCINRF proposed TS 6.1.3, "Staffing," item a.1., states, "The minimum staffing when the reactor is not secured shall include: 1. a licensed operator with direct access to the reactor controls." Does this statement mean a licensed operator must be present in the control room? If so, confirm and modify the proposed TS to reflect that a licensed operator must be present in the control room, or provide justification as to why not.

This has been revised as requested.

31) The UCINRF proposed TS 6.1.3, "Staffing," item d., states, "...Restart following any unplanned or unscheduled shutdown, or significant power reduction." The use of "...or significant power reduction," is unclear and may cause confusion in implementing the proposed TS. Modify the proposed TS to remove this phrase, define and justify what significant means, or provide justification why this terminology cannot be misunderstood.

Wording has been changed to clarify this concern.

32) The UCINRF proposed TS 6.2.4, "ROC Audit function," is missing two conditions prescribed by ANSI/ANS-15.1-2007. Modify the proposed TS to include the below sections, or provide justification to deviate from guidance.

e. The audit shall be performed by one or more persons appointed by the ROC. At least one of the auditors shall be familiar with reactor operations. No person directly responsible for any portion of the operation of the facility shall audit that operation; and

f. Any deficiencies that may affect reactor safety shall be immediately reported to ROC Chair, Level 1, and a written full report of the audit shall be submitted to the ROC within three months of the audit.

These revisions have been made as requested.

*33) The UCINRF proposed TS 6.3, "Radiation Safety," states, "As delineated in section 6.1.2.e, the UCI... The program **shall** use the guidelines of ANSI/ANS-15.11-2004." The NRC has accepted a "**should**" statement.*

This revision has been made.

34) The proposed UCINRF TS 6.4, "Operating Procedures," states, "Written procedures, reviewed and approved by the ROC, shall be in effect and implemented..." The NUREG-1537 guidance recommends the facility director be included for review and approval (e.g., "Written procedure, reviewed and approved by the ROC ***and facility director****, shall be in effect and implemented..."). Provide justification for this deviance from guidance, or modify the proposed TS to conform with the guidance.*

This revision has been made as requested.

35) The UCINRF proposed TS 6.4, "Operating Procedures," final paragraph states, "Substantive changes to procedures...but shall be reported to the RSO within 30 days." The definition of substantive changes may cause confusion and misinterpretation about what needs to be considered under 10 CFR 50.59. Provide justification why this cannot be misinterpreted or modify the proposed TS to meet the requirements of 10 CFR 50.59.

This has been revised as requested.

36) The UCINRF proposed TS 6.5, "Experimental Review and Approval," item b., states, "Substantive changes to existing experiments or classes shall be made only after review by the ROC and RSO or their designees. Minor changes that do not significantly...(SRO)." The definition of substantive changes may cause confusion and misinterpretation about what needs to be considered under 10 CFR 50.59. Provide justification for the use of this term, or modify the proposed TS to remove the section on substantive changes and minor changes to read, "Changes to existing experiments or classes shall be made only after review by the ROC and RSO or the RSO designee."

This has been revised as requested.

37) The UCINRF proposed TS 6.6.1, "Actions To Be Taken in Case of a Safety Limit Violation," items b. and c. are inconsistent. In TS 6.6.1, item b. there is a requirement to report the event ***immediately*** *while TS 6.6.1, item c. specifies that this event shall be reported within* ***24 hours****. Provide justification for the non-congruency, or modify the proposed TS to be consistent.*

TS 6.6 has been revised by moving material from 6.7.2 into this section. We believe this clarifies the action and reporting requirements to accord with 10CFR 50.36 as requested

38) The UCINRF proposed TS 6.7.1, "Annual Operating Report," states, "...each 12-month period for operations for the preceding year's activities between July 1st through June 30th." In addition, specify, when the report is actually due (e.g., "An annual report shall be created and submitted, by the Facility Director, to the Document Control Desk, U.S. NRC, Washington, DC, 20555, by September 30th of each year").

Revisions have been made as requested.

39) The UCINRF proposed TS 6.7.1, "Annual Operating Report," item d., differs slightly from the NUREG-1537 guidance, which states that a tabulation of changes in the reactor facility and procedures, and tabulation of new tests or experiments, including a summary of the analyses leading to the conclusions that they are allowed without prior authorization by the NRC and that 10 CFR 50.59 was applicable. The proposed TS is not as precise. Provide an explanation as to why this

deviation is justifiable or revise the proposed TS accordingly.

This has been revised as requested.

40) The UCINRF proposed TS 6.8.3, "Records to be retained for the lifetime of the reactor facility," is missing a condition prescribed by ANSI/ANS-15.1-2007. Modify the proposed TS to include the below section or provide justification to deviate from guidance.

1) Reviews and reports pertaining to a violation of the safety limit, the limiting safety system setting, or a limiting condition of operation.

This has been revised as requested.

41) Provide updated proposed TSs, including the revisions requested in e-mail dated September 11, 2012 (ADAMS Accession No. ML12256A897).

These revisions have been incorporated as requested.

ADDITIONAL REVISIONS.

Based on discussions with NRC personnel on February 5th 2015, we are requesting removal of former TS 3.3.4 Pool Water pH as an un-needed specification for an open pool reactor, with adequate monitoring of water conductivity and temperature. This has been done in the latest TS version. References to this specification in TS 4.0 and 4.3 have also been removed.

Further editorial revisions to the Technical Specifications have been made as a result of exchanges with NRC licensing personnel in the period February 2015 through November 2015. An updated version of the TS, dated December 18th, 2015 is submitted herewith as an attachment.

APPENDIX A
To
Facility License R-116 Docket 50-326

Technical Specifications
for the

U. C. Irvine

TRIGA® Mark I Nuclear Reactor

December 18, 2015

TABLE OF CONTENTS

Note: Included in this document are the Technical Specifications and the "bases" for the Technical Specifications. These bases, which provide the technical support for the individual Technical Specifications, are included for informational 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. DEFINITIONS

The following frequently used terms are defined to aid in the uniform interpretation of these specifications.

AUDIT is an examination of records, logs, procedures, or other documents to ascertain that appropriate specifications and guidelines are being followed in practice. An audit report is written to detail findings and make recommendations.

CHANNEL is a combination of sensor, lines, amplifier and output device which are connected for the purpose of measuring the value of a parameter.

CHANNEL CALIBRATION is an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter that the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm or trip, and shall be deemed to include a channel test.

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.

CHANNEL TEST is an introduction of a signal into the channel to verify that it is operable.

CLOSE-PACKED ARRAY is a fuel loading pattern in which the fuel elements are arranged in the core by filling the inner rings first.

CONFINEMENT is the enclosure of the overall facility designed to limit release of effluents between the enclosure and the external environment through controlled or defined pathways.

CONTROL ROD is a device fabricated from neutron absorbing material or fuel or both which is used to establish neutron flux changes and to compensate for routine reactivity changes. A control rod may be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged. Types of control rods shall include:

a. Regulating (REG): a rod having electric motor drive and scram capabilities. Its position may be varied manually or by an electronic controller. It shall have a fueled-follower section.

b. Shim (SHIM): a rod having electric motor drive and scram capabilities. Its position shall be varied manually. It shall have a fueled-follower section.

c. Adjustable Transient (ATR): a rod with scram capabilities that can be rapidly ejected from the reactor core using a pneumatic drive to produce a pulse. It has an electric motor drive to adjust its position or length of travel. It shall have a void follower.

d. Fast Transient (FTR): a rod with scram capabilities that can be rapidly ejected from the reactor core using a pneumatic drive to produce a pulse. Only fully UP or DOWN positions are available. It shall have a void follower.

CORE CONFIGURATION is a particular arrangement of fuel, control rods, graphite reflector elements, and experimental facilities inserted within the core grid plates.

CORE LATTICE POSITION is defined by a particular hole in the top grid plate of the core designed to hold a standard fuel element. It is specified by a letter, indicating the specific ring in the grid plate and a number indicating a particular position within that ring.

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

EXPERIMENT is any operation, hardware or target (excluding devices such as detectors or foils) which is designed to investigate non-routine reactor characteristics or which is intended for irradiation within an irradiation facility. Hardware rigidly secured to a core or shield structure so as to be part of their design to carry out experiments is not normally considered an experiment. Specific experiments shall include:

a. SECURED EXPERIMENT is any experiment or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible malfunctions.

b. UNSECURED EXPERIMENT is any experiment or component of an experiment that does not meet the definition of a secured experiment.

c. MOVEABLE EXPERIMENT is any experiment where it is intended that the entire experiment may be moved in or near the core or into or out of the core while the reactor is operating.

FUEL ELEMENT is a single TRIGA® fuel rod.

INITIAL STARTUP is the first start-up from reactor secured condition on any day when the reactor is to be operated in order to verify core excess and other instrument parameters.

INSTRUMENTED FUEL ELEMENT is an element in which one or more thermocouples are embedded for the purpose of measuring fuel temperature during reactor operation.

IRRADIATION FACILITIES are pneumatic transfer systems, central tube, rotary specimen rack, and the in-core facilities (including single element positions, three-element positions, and the seven element position) and any other facilities in the tank designed to provide locations for neutron or gamma ray exposure of materials.

MEASURED VALUE is the value of a parameter as it appears on the output of a channel.

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

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

OPERATIONAL CORE is a core configuration that meets all license requirements, including Technical Specifications.

PULSE MODE is any operation of the reactor with the mode switch in the PULSE position that satisfies all instrumentation and license requirements, including technical specifications, for pulse operation of the reactor.

REACTIVITY WORTH OF AN EXPERIMENT is the value of the reactivity change that results from the experiment being inserted into or removed from its intended position.

REACTOR FACILITY is the physical area defined by rooms B64, B64A, B54, B54A, and B54B in the service level of Rowland Hall on the campus of the University of California Irvine.

REACTOR OPERATING is any time at which the reactor is not secured or shutdown.

REACTOR SAFETY SYSTEMS are those systems, including their associated input channels that are designed to initiate automatic reactor scram or to provide information for the manual initiation of a scram for the purpose of returning the reactor to a shutdown condition.

REACTOR SECURED is when:

Either

(1) there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection;

Or

(2) the reactor is shutdown and the following conditions exist:

a. all neutron-absorbing control rods are fully inserted; and

b. the console key switch is in the OFF position and the key is removed from the console lock; and

c. no work is in progress involving fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods; and

d. no experiments are being moved or serviced that have a reactivity worth exceeding $1.00.

REACTOR SHUTDOWN is when the reactor is subcritical by at least $1.00 in the reference core condition with the reactivity worth of all installed experiments included.

REFERENCE CORE CONDITION is when the core is at ambient temperature (cold) and the reactivity worth of xenon is zero.

REVIEW means a qualitative examination of audits, reports and records, procedures or other documents from which appropriate recommendations for improvements are made.

RING is one of six concentric bands in the grid plate locations surrounding the central opening of the core. The rings are designated by the letters B through G, with the letter B used to designate the innermost band.

SAFETY CHANNEL is a measuring channel in the reactor safety system.

SCRAM TIME is the elapsed time between the initiation of a scram signal and all control rods reaching their bottom limit.

SEVEN ELEMENT POSITION is a hexagonal section which can be removed from the upper grid plate for insertion of specimens up to 4.4 in. in diameter after relocation of all six B-ring elements and removal of the central tube irradiation facility.

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

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

STEADY-STATE MODE is whenever the reactor is operating with the mode selector switch in the STEADY-STATE position.

SURVEILLANCE INTERVALS that are permitted are established as follows:

a. Quinquennial – interval not to exceed 6 years
b. Biennial – interval not to exceed 2-1/2 years
c. Annual – interval not to exceed 15 months
d. Semi-annual – interval not to exceed 7-1/2 months
e. Quarterly – interval not to exceed 4 months
f. Monthly - interval not to exceed 6 weeks
g. Daily –each day when the reactor is to be operated or before any operation extending more than one day

THREE ELEMENT POSITION is one of two triangular-shaped removable sections of the upper grid plate, one encompassing core lattice positions D5, E6 and E7 and the other D14, E18 and E19, designed to accommodate experiments. When fuel elements are placed in these locations, a special fixture shall be inserted to provide lateral support.

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limit - Fuel Element Temperature

Applicability. This specification applies to the fuel element temperature.

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

Specification(s). The temperature in a stainless steel clad, high hydride fuel element shall not exceed 1000°C under any condition of operation.

Basis. The important parameter for a TRIGA® reactor is the fuel element temperature, since it can be measured. The loss in the integrity of the fuel element cladding could arise from an excessive build-up of pressure in the fuel element. The safety limit for high hydride TRIGA® fuel is based on data including the experimental evidence obtained during high performance reactor tests of this fuel. These data indicate that the stress will remain below the ultimate stress provided the fuel temperature does not exceed 1150°C and the fuel cladding is water cooled.

The safety limit for the stainless steel clad, high hydride ($ZrH_{1.7}$) fuel element is based on NRC accepted limits in NUREG 1537 section 14.1 which also indicates that the stress in the cladding due to the hydrogen pressure from the dissociation of the zirconium hydride will remain below the yield stress provided the temperature of the fuel does not exceed 1150°C and the fuel cladding is water cooled.

2.2 Limiting Safety System Settings

Applicability. This specification applies to the scram setting for the fuel element temperature channel.

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

Specification(s). For a core composed entirely of stainless steel clad, high hydride fuel elements, a limiting safety system setting applies to the standard instrumented fuel element (IFE) which shall be located in the B- or C-ring as indicated in the following table:

Location	Limiting Safety System Setting
Core lattice positions B2, B4, C5, C6, or C7	≤ 425°C

Basis. Fuel temperature is measured by a fuel element designed for this purpose (IFE) in a system designed to initiate a reactor scram if a limit is exceeded. The limiting setting is conservatively chosen for five possible core positions that calculations in the SAR, as supplemented by letter dated June 7th 2011, indicate are similar in expressing the highest power density and thus the highest fuel temperatures attained in the core. In addition, the maximum recorded temperatures for the UCI reactor IFE for the period since 1969 are 250°C at steady state power operation, and 350°C for pulse operation. The LSSS is extremely conservative compared to the fuel temperature safety limit.

3. LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Core Parameters

3.1.1 Steady-state Operation

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

Objective. The objective is to assure that the fuel temperature safety limit is not exceeded.

Specification(s). The reactor power level in steady-state operation shall not exceed 250 kilowatts.

Basis. Calculations have been performed which show that for operation at 250 kW, the maximum fuel temperature is 253°C and the minimum DNB ratio is greater than 7.27. In addition, experience at other TRIGA® reactors and thermal and hydraulic calculations for this core (SAR, as supplemented by letter dated June 7th 2011) indicates that these power levels can be safely used with natural convection cooling of the fuel elements in the designed core configuration.

3.1.2 Shutdown Margin

Applicability. These specifications apply to reactivity condition of the reactor and the reactivity worths of the control rods and experiments. They apply for all modes of operation.

Objective. The objective is to assure that the reactor can be shut down at all times.

Specification(s). The reactor shall not be operated unless the shutdown margin provided by the control rods is greater than $0.55 with:

a. irradiation facilities and experiments in place and the total worth of all unsecured experiments in their most reactive state; and

b. the most reactive control rod fully withdrawn; and

c. the reactor in the reference core condition.

Basis. The value of the shutdown margin and limits on experiments assure that the reactor can be shut down from any operating condition even if the most reactive control rod should remain in the fully-withdrawn position.

3.1.3 Core Excess Reactivity

Applicability. These specifications apply to reactivity condition of the reactor and the reactivity worth of the control rods and experiments. They apply for all modes of operation.

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

Specification(s). The maximum available core excess reactivity based on the reference core condition shall not exceed $3.00.

Basis. An excess reactivity limit of $3.00 allows for flexibility in operating the reactor in steady state mode while limiting the reactivity addition for pulse operation. Computations presented in the SAR (Chapter 13.3) establish that a sudden insertion of $3.00 results in a fuel temperature of approximately 350°C, well below the established safety limit for this fuel (Technical Specification 2.1). Such calculations are conservative, being based on a purely adiabatic model. The specifications assure that no insertion of reactivity above this value shall be possible, even under non-normal operating conditions.

3.1.4 Pulse Mode Operation

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

Objective. The objective is to assure that the fuel temperature safety limit shall not be exceeded.

Specification(s). The reactor shall not be operated in the pulse mode unless in addition to the requirements of Technical Specification 3.1.2 and 3.1.3:

a. the steady-state power level of the reactor is less than 1 kilowatt; and

b. the total reactivity worth of the two transient control rods (ATR + FTR) is measured to not exceed $3.00.

Basis. The fuel temperature rise during a pulse transient has been calculated conservatively using an adiabatic model. Pulse reactivity insertion with the power level below 1 kW assures that the starting temperature for a pulse rise is below 25°C. The temperature rise from a $3.00 reactivity insertion pulse is thus calculated to bring the peak fuel temperature to less than 400°C, well below the safety limit and well below the recommended maximum fuel temperature limit of 830°C. (GA Report, A16613, 1981.)

3.1.5 This section intentionally left blank.

3.1.6 Fuel Element Inspection Parameters

Applicability. The specifications apply to all fuel elements, including fuel follower control rods.

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

Specification(s). The reactor shall not be operated with any fuel element identified to show damage. An exception is made for operation up to a power level at which a leak becomes detectable solely in order to be able to identify the leaking element. A fuel element shall be identified as showing damage and be removed from core if:

a. the transverse bend exceeds $1/16^{th}$ inches (0.0625 in) over the length of the element; or

b. the growth in length over original measurements exceeds $1/8^{th}$ inch (0.125 in); or

c. a cladding defect is suspected by a finding of release of any fission products; or

d. visual inspection identifies unusual pitting, bulging, or corrosion; or

e. burn-up of the element is estimated to exceed 50%.

Basis. These criteria are established by NUREG 1537 section 14.1. NUREG 1282 describes experiments indicating TRIGA® fuel can safely be used to at least 50% burn-up.

3.1.7 Core Configuration

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

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

Specification(s).

a. The core shall be an arrangement of TRIGA® 8.5/20 Low Enriched Uranium (LEU) fuel.

b. The core fuel elements shall include at least one 8.5/20 LEU fuel element with embedded thermocouples to enable monitoring of fuel element temperature.

c. The core fuel elements shall be kept in a close-packed array except for control rods, single- or three-element or seven-element positions occupied by in-core experiments, irradiation facilities (including transfer system termini), and a central dry tube.

d. The reflector, excluding experiments and experimental facilities, shall be graphite or a combination of graphite and water.

e. A control rod shall not be manually removed from the core unless calculations show that the core will be subcritical by $\geq$ \$0.55 excluding the worth of the rod being worked on and the worth of the most reactive remaining control rod.

Basis.

a. TRIGA® cores have been in use for years and their characteristics are well documented. LEU cores including 8.5/20 fuel have also been operated successfully at many facilities. In addition, analysis indicates that the low uranium loading, LEU 8.5/20 core will safely satisfy all operational requirements. See chapters 4 and 13 of the SAR as supplemented by letter dated June 7th 2011.

b. The IFE provides a signal to the fuel temperature safety channel.

c. Inner core lattice positions contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core. Vacancies are permitted only on the periphery of the core, where reactivity worths are lower.

d. Graphite and water reflectors are used for neutron economy and the enhancement of experimental facility radiation characteristics.

e. Manual manipulation of control rods will be allowed only when a single manipulation cannot result in inadvertent criticality.

3.2 Reactor Control and Safety Systems

3.2.1 Control Rods

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

Objective. To assure control rods are operable and that prompt reactor shut down following a scram is accomplished.

Specification(s). The reactor shall not be operated unless the REG, SHIM and ATR control rods are operable. Control rods shall not be considered operable if:

a. damage or deterioration is apparent to any rod or drive assembly that could affect operation; or

b. the scram time for any control rod is greater than 1 second for 90% reactivity insertion; or

c. the total reactivity worth of the two transient control rods (ATR and FTR) is greater than $3.00.

Basis. Experience has shown that rod movement is assured in the absence of damage and that scram times of less than 1 second are more than adequate to reduce reactivity and fuel temperatures rapidly to assure safety in view of known transient behavior of TRIGA® reactors. The FTR rod is not vital to safe scram as a result of its low worth value (less than $1.00) and can remain in core if not needed for pulsing. The total worth of the two transient rods is limited so as to restrict the pulse size.

3.2.2 Reactor Measuring Channels

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

Objective. To specify that minimum number of measuring channels that shall be available to the operator to assure safe operation of the reactor.

Specification(s). The reactor shall not be operated in the specified mode unless the measuring channels described in Table 1 are operable.

Table 1. Minimum Measuring Channels

Measuring Channel	Operating Mode	
	Steady-state	Pulse
Fuel Element Temperature	1	1
Linear Power Level	1	-
Log Power Level	1	-
Power Level (%)	1	1 (peak power)
nvt circuit (integrated pulse energy)	-	1

Note 1. Any single power level channel may be inoperable while the reactor is operating solely for the purpose of calibration and/or channel tests or checks on that channel.

Note 2. Any single power level channel that is not required for safety scram purpose by TS 3.2.3 and ceases to be operable during reactor operation shall be returned to operating condition within 5 minutes or the reactor shall be shut down. For channels required by TS 3.2.3 the reactor shall be shut down immediately if the channel becomes inoperable.

Basis. The fuel temperature displayed at the control console gives continuous information on the parameter which has a specified safety limit. The power level monitors assure that measurements of the reactor power level are adequately covered at both low and high power ranges in appropriate modes. Notes 1 and 2 allow for necessary tests for resolving of problems or recalibration while maintaining sufficient information for safe operation.

3.2.3 Reactor Safety System

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

Objective. To specify the minimum number of reactor safety system channels that shall be operable in order to assure that the fuel temperature safety limit is not exceeded.

Specification(s). The reactor shall not be operated unless the safety system channels described in Table 2 and the interlocks described in Table 3 are operable in the appropriate operating modes.

Table 2. Minimum Reactor Safety Channels

Safety Channel	Function and trip level setting	Operating Mode	
		Steady-state	Pulse
Fuel Element Temperature	Scram – ≤425°C (IFE)	1	1
Reactor Power level	Scram – ≤110% of 250 kw(t) (≤ 275 kw)	2	-
Loss of HV and/or signal on any required channel	Scram	1	1
Manual Bar	Scram	1	1
Preset Timer	Scram pulse rods < 15 seconds after pulse	-	1
Seismic Switch	Scram – if motion of ≥3% g (≤ 0.03g)	1	1
Pool Water Temperature	Manual Scram if > 25°C	1	1

Table 3. Minimum Interlocks

Interlock	Function	Operating Mode	
		Steady-state	Pulse
Wide Range Power Level Channel (Log)	Prevent control rod withdrawal when power level is ≤ 1 x 10^{-7} % of full power	1	-
REG, SHIM, ATR Control Rod Drives	Prevent application of air to fast transient rod when all other rods are not fully inserted	1	-
REG, SHIM, ATR Control Rod Drives	Prevent simultaneous withdrawal of more than one rod	1	-
REG, SHIM, ATR Control Rod Drives	Prevent movement of REG and SHIM rods and ATR drive in pulse mode	-	1
ATR Cylinder Drive	Prevent application of air to adjustable transient rod unless cylinder is fully down	1	-
Wide Range Linear Power Channel	Prevent ATR or FTR withdrawal unless power level ≤ 1 kilowatt	-	1

Basis.

Scrams. The fuel temperature scram provides the protection to assure that if a condition results in which the safety limit is approached, an immediate shutdown will occur to keep the fuel temperature well below. The justification basis is described in Technical Specification 2.2. The power level scrams are provided as added protection against abnormally high fuel temperature. The manual scram allows the operator to shut down the system if an unsafe or abnormal condition occurs. A high voltage scram on each channel assures that detector response is operating at all times. The seismic switch will scram the reactor if earth movement in any dimension exceeds 3% g (0.03g) in case the operator is prevented from operating the manual scram at the time. This level corresponds to movement noticeable by most persons, but (by MM scale) results in no damage to structures. The preset timer scram provides pulse "clipping" to reduce energy production at the tail of a pulse. The pool water level temperature limit is the value used for the thermal hydraulic analysis input coolant temperature in the SAR as supplemented by letter dated June 7th 2011, and also is designed to reduce stress between the aluminum tank liner and its concrete surround.

Interlocks. The interlock to prevent startup of the reactor with less than 10^{-7} % power indication assures that indication of neutron multiplication is present as reactivity is inserted. The interlocks on control rod drives are provided to prevent withdrawal of more than one control rod at a time avoiding multiple simultaneous reactivity insertions by operators. The interlocks which prevent the firing of the transient rods in the steady-state mode or if the power level is greater than 1 kilowatt prevent inadvertent pulses or pulsing when fuel temperature is too high.

3.3 Coolant Systems

3.3.1 Pool Water Level

Applicability. These specifications apply to the water level in the reactor pool at all times.

Objective. To assure there is sufficient water in the reactor pool to provide cooling and shielding for radiation from the core, and to check for potential pool leakage.

Specification(s).

a. The reactor shall not be operated unless the pool water level is at least 24 feet above the tank floor (1 foot below the tank edge).

b. An audible alarm, with reporting to a monitored remote location if not locally silenced by an operator, shall operate continuously to alert personnel if the water level in the reactor pool falls below the above limit. Visual checking of water level shall be substituted every 10 hours during periods when the alarm is found to be inoperable and no substitute level device has been implemented.

c. Records shall be maintained of the date, time and quantity of all make up water added to the pool.

Basis. Facility design calculations and subsequent measurements show that these water levels are sufficient to reduce full power operational radiation levels to acceptable levels within the facility and in any occupied areas above or surrounding the reactor. This is also true for shut down levels. The alarm will notify appropriate responders well before any increase in radiation levels to the

surroundings occurs. The alarm and the operational make up water records will, if it occurs unusually frequently, alert operators to the possibility that pool leakage might be occurring. The pool level is normally maintained at approximately 10 inches below the tank edge. Thus the alarm level is at 2 inches below the normal level corresponding to evaporation or leakage of only 160 gallons (640 liters, or 0.6% of total pool water). The maximum water leak rate calculated (SAR) indicates that the water would be at a sufficient level over the core for at least 10 hours to prevent unsafe release of radiation to the surrounding areas. Procedures call for a substitute pool level alarm system to be implemented during any extended period of failure.

3.3.2 Pool Water Temperature

Applicability. This specification applies to the water temperature in the reactor pool at all times.

Objective. To assure the water in the reactor pool stays within limits that provide sufficient cooling of the fuel and that minimizes stresses to the tank and reactor components.

Specification(s). The pool water temperature shall be maintained between 17°C and 25°C

Basis. These temperature limits are easily maintained using the available cooling system and guard against temperatures that might produce undue stresses on tank components or water purification systems. The thermal hydraulic analysis was based on an inlet core temperature of 25°C.

3.3.3 Pool Water Conductivity

Applicability. This specification applies to the conductivity of water in the reactor pool at all times.

Objective. To assure the water in the reactor pool is maintained at high purity to minimize potential corrosion of reactor components.

Specification(s). The pool water conductivity level shall be maintained less than 3 micromhos/cm. Make-up water shall meet this specification before being added to the pool.

Basis. Experience at other reactor facilities indicates that maintaining the conductivity within 5 micromhos/cm (μS/cm) is adequate to provide acceptable control of corrosion (NUREG 1537). An additional margin of assurance is provided by this lower specification. Degradation from this conductivity also aids in assessing possible leakage of treated secondary coolant water into the primary coolant water.

3.3.4 Pool Water Radioactivity

Applicability. This specification applies to the radioactivity of water in the reactor pool at all times.

Objective. To assure the water in the reactor pool is maintained at high purity.

Specification(s). The average pool water radioactivity level shall be maintained within limits for sewer disposal as established by 10 CFR § 20, Appendix B, Table 3 for radionuclides with half-lives longer than 24 hours.

Basis. Maintenance at this level will assure that any disposal of pool water, either planned or inadvertent, will be within appropriate and significant radioactivity limits. It also will provide verification of absence of fission product leakage.

3.4 This section intentionally left blank

3.5 Ventilation Systems

3.5.1 Ventilation System.

Applicability. This specification applies to the operability and operation of the facility ventilation system.

Objective. To assure that the ventilation system is operable to mitigate the consequences of possible releases of radioactive materials resulting from reactor operation.

Specification(s).

a. The reactor shall not be operated unless the ventilation system is operating as indicated by:

   i. a minimum of 0.10 inches of water negative pressure difference between the reactor room and the control room and between the reactor room and the air outside the building; and

   ii. a minimum total exhaust flow rate from the reactor room area of 3600 cubic feet per minute (cfm) is present.

   Note: The ventilation system may be inoperable for periods of time not to exceed two hours to allow repair, maintenance or testing of the system. During such an exception, no pulses shall be fired.

b. The reactor shall not be operated unless it is verified that the ventilation system goes into the emergency mode upon manual actuation or a signal of high radiation activity from a Continuous Air Particulate Monitor (CAM) measuring air from above the pool as described in Technical Specification 3.5.2. Verification shall be by observing the emergency flow rate is at least 240 cubic feet per minute (cfm), the absence of regular exhaust flow, and the pressure differential reading between the reactor room and the outside is negative.

c. Radioactive by-product materials shall not be handled in the facility unless the ventilation system is operating as described in Technical Specifications 3.5.1.a. and 3.5.1.b. above.

Basis. Through a combination of inflow dampers and outflow exhaust, facility design establishes and exceeds these pressure differentials and flows. The differential pressure assists in confinement of radioactive materials. The SAR establishes that normal operation effectively dilutes ^{41}Ar levels below 10 CFR § 20 limits and as detailed in facility annual reports. An automatic emergency mode with a small filtered purge exhaust is provided to limit release of radioactivity to the environment. Operation of the normal system adequately dilutes the ^{41}Ar released during experimental operations. The two hour exemption should not diminish the effectiveness of the CAM in detecting any release of radioactivity. The requirement not to pulse while the ventilation system is undergoing repair reduces the likelihood of fuel element failure during such times. The ventilation system should be required for by-product handling activities to reduce potential impact of release of volatile chemicals.

3.5.2 Ventilation During Emergency Situations

Applicability. This specification applies to the ventilation system provided for emergency situations.

Objective. To assure there is confinement of radioactive releases by closing of normal ventilation and establishing emergency ventilation.

Specification(s). A signal of high radiation activity alarm from a Continuous Air Particulate Monitor (CAM) measuring air from above the pool or manual operation from the control room shall carry out the following functions:

a. close off inflow air by closing dampers; and

b. close off outflow air by closing dampers in exhaust ducts and removing power from relevant exhaust fans and fume hood; and

c. remove power from pneumatic transfer system so it can no longer operate to transfer air through any core region; and

d. open outflow damper in a small "purge" exhaust duct system equipped with a HEPA filter.

Basis. These actions will result in confinement of any released radioactive materials, while beginning to purge contaminated air through a high grade filter. Experience has shown that fission product release from fuel elements is most rapidly detected by a CAM operating in this manner. The SAR establishes that the emergency purge system will, in the event of a radioactive gas release, be effective in limiting release to the environment and also providing personnel with sufficient time to evacuate before experiencing serious exposure. It is shown in Chapter 13 of the SAR, as supplemented by letter dated Dec 2nd 2011, that operation of the emergency exhaust system reduces off-site doses to below 10 CFR § 20 limits in the event of a TRIGA® fuel element failure. It is shown also that, if the reactor were to be operating at full steady-state power, fuel element failure will not occur even if all the reactor tank water were to be lost immediately.

3.6 This section intentionally left blank

3.7 Radiation Monitoring Systems and Effluents

3.7.1 Radiation Monitoring Systems

Applicability. This specification applies to monitoring of radiation levels.

Objective. To assure information is available to provide assurance of radiological safety of personnel at the facility, and of the absence of excessive releases beyond the facility.

Specification(s).

a. The reactor shall not be operated unless the following minimum radiation monitoring instruments are operating:

	Minimum Number Operating
Radiation Area Monitors (RAM)	2
Continuous Air Particulate Monitor (CAM)	1

b. Environmental monitoring dosimeter packs, exchanged at least quarterly, shall be in place at the primary exhausts of the facility at all times, except when undergoing exchange. Additional packs shall be located in adjacent buildings, and in a more remote control location for comparison.

Basis. These instruments through warnings and alarms will provide adequate notification of abnormal levels that could result in exposures or uncontrolled releases. The CAM provides a signal to actuate the ventilation emergency mode (Technical Specifications 3.5.1.b, 3.5.2, and 5.5.e). The CAM alarm set point is based on limiting air concentrations to below the DAC level for I-131 as listed in Appendix B of 10 CFR § 20. The environmental dosimeters provide information that can be used to track long term trends that might need attention.

3.7.2 Effluents

Applicability. This specification applies to the release rate of ^{41}Ar gas and liquid effluents.

Objective. To assure that concentration of ^{41}Ar in air and any liquid effluents released to accessible unrestricted areas shall be below the applicable limits of 10 CFR § 20.

Specification(s).

a. The annual average concentration of ^{41}Ar released to the environment shall not exceed 1×10^{-8} microcuries per milliliter (μCi/mL).

b. The quantity of radioactivity in liquid effluents released from the facility to the sewer system shall be soluble and not exceed the limits of 10 CFR § 20, Appendix B, Table 3.

Basis.

a. The analysis presented in Chapter 13.2 of the SAR, as supplemented by letter dated Oct 3rd 2011, concludes that the building exhaust and room ventilation system normally provides a dilution factor of 100 at the point of external release for airborne concentrations in the facility area. Under extremely unlikely conditions involving system failures, it provides a dilution factor of at least 30, reducing the facility room concentration predicted from calculations and measurements as a result of normal reactor operation to be well below 10 CFR § 20 Appendix B, Table 2 requirements (1×10^{-8} μCi/mL). This will be further assured since releases are permitted to be averaged over a one year period. The exposure risk to the public is reduced since the discharge plume is at a high level above the roof. Annual reports from this facility have shown that levels released have been well below this value.

b. This specification establishes assurance that any release of radioactive materials contained in liquids released to the sewer system does not exceed the limits required by regulations.

3.8 Limitations on Experiments

3.8.1 Reactivity Limits

Applicability. This specification applies to experiments placed in the reactor and its experimental facilities.

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

Specification(s). The reactor shall not be operated unless the following conditions governing reactivity worths exist:

a. the reactivity worth of any unsecured or moveable experiment shall not exceed $1.00; and

b. the reactivity worth of an individual experiment shall not exceed $3.00; and

c. the sum of the absolute values of reactivity worths of all experiments shall not exceed $3.00.

Basis. The limit on an unsecured experiment is to prevent an inadvertent pulse, and to maintain shutdown margin limitations. The insertion of $3.00 pulses has been analyzed as a safe operating condition for this reactor (SAR Chapter 13). Limitation of experiments such that a pulse larger than this value could not occur is prudent and stays well within safe limits. The limitations also assure that achievement of margins for shutdown is assured.

3.8.2 Materials

Applicability. This specification applies to experiments placed in the reactor and its experimental facilities.

Objective. To prevent damage to the reactor and to minimize excessive release of radioactive materials in the event of an experiment failure.

Specification(s). The reactor shall not be operated unless the following conditions governing experiments exist:

a. fueled experiments shall be limited such that the total inventory of iodine isotopes 131 through 135 in the experiment is not greater than 0.02 curies, and the strontium-90 inventory does not exceed 1 microcurie; and

b. explosive materials in the amount of 25 milligrams of TNT (or equivalent) or lesser quantities may be irradiated provided that the pressure produced upon accidental detonation of the explosive has been calculated and/or experimentally determined to be less than half the design pressure of the container; and

c. experiments containing corrosive materials shall be doubly encapsulated. The failure of an encapsulation of material that could damage the reactor shall result in removal of the sample and physical inspection of potentially damaged components.

Basis. In specification a, an assumption is made that complete release of volatile material (iodine) from a fueled experiment is possible. It is shown in the SAR, Chapter 13, as supplemented by letter dated Dec. 2nd 2011, that a release of 0.02 curies of iodine activity would result in a maximum dose to a person in the facility who evacuates in 5 minutes of 0.4 rem, or less than 1/75th of the 50 rem limit in 10 CFR § 20. An individual directly exposed in the exhaust in an unrestricted area under highly unlikely conditions would receive (TEDE) less than 0.54 mrem. The maximum projected TEDE to a person in the nearest living area will be less than 0.22 mrem, far less than the 100 mrem limit in 10 CFR § 20. These computations are extremely conservative as they assume no pool water is present. In the event of a failed experiment the pool water would be present to absorb/dissolve halogens and reduce the airborne concentrations.

Specifications b and c reduce the likelihood of damage to reactor components resulting from experiment failure and use information from NRC Reg. Guide 2.2.

3.8.3 Failures or Malfunctions

Applicability. This specification applies to experiments placed in the reactor and its experimental facilities.

Objective. To prevent damage to the reactor as well as to minimize release of radioactive materials in the event of an experiment failure.

Specification(s).

a. Where the possibility exists that the failure of an experiment under (a) normal operating conditions of the experiment or the reactor, (b) credible accident conditions in the reactor, or (c) possible accident conditions in the experiment could release radioactive gases or aerosols to the reactor room or any unrestricted area, the quantity and type of material in the experiment shall be limited in activity such that exposures of the reactor personnel to the gaseous activity or radioactive aerosols in the reactor room or control room will not exceed the occupational dose limits in 10 CFR § 20.1201. Additionally, exposures to members of the public to these releases in the unrestricted area will not exceed the dose limits in 10 CFR § 20.1301. In calculating potential consequences, one or more of the following assumptions shall be made, as applicable:

    i. 100% of the gases or aerosols escape from the experiment.

    ii. If the effluent from an experimental facility exhausts through a hold-up tank, which closes automatically on high radiation levels, 10% of the gaseous activity or aerosols will escape.

    iii. If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, 10% of the aerosols will escape.

    iv. For materials whose boiling point is above 55°C and where vapors formed by boiling this material are radioactive and could escape only through an undisturbed column of water above the core, 10% of these vapors escape.

b. If an experiment container fails and releases material which could damage reactor fuel or structure by corrosion or other means, physical inspection shall be performed to determine the consequences and the need for corrective action before further operation of the reactor.

Basis.

a. This specification is intended to assist experiment review and design in meeting the goals of 10 CFR § 20 by reducing the likelihood of excessive facility personnel or public exposure by gases or aerosols as a result of experiment failure.

b. This specification is intended to assure that any facility damage as a result of experiment failure is appropriately corrected before operations are resumed.

3.9 This section intentionally left blank.

4. SURVEILLANCE REQUIREMENTS

4.0 General

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

Objective. To assure the proper operation of any system related to reactor safety.

Specification(s).

a. Surveillance requirements may be deferred during prolonged (periods greater than 1 month) reactor shutdown (except Technical Specifications 4.2.j, 4.3.a, 4.3.b, 4.3.d, and 4.3.e). However, they shall be completed prior to reactor start-up unless reactor operation is required for performance of the surveillance. Such surveillance shall be performed as soon as practicable after reactor start-up. Scheduled surveillance which cannot be performed with the reactor operating may be deferred until a planned reactor shutdown.

b. All replacements, modifications, and changes to systems having a safety related function including the ventilation system, the core and its associated support structure, the pool, the pool coolant system, the control rod drive mechanisms, and the reactor safety system shall meet or exceed the requirements of the original system or component. A safety system shall not be considered operable until it has been properly tested to meet specifications.

Basis. Changes or maintenance can affect reactor operation parameters. This specification will assure that safety systems function according to established criteria before any reactor operation.

4.1 Reactor Core Parameters

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

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

Specification(s).

a. The total reactivity worth of each control rod shall be measured annually or following any significant change (>$0.25) in core configuration.

b. The core excess reactivity shall be determined using control rod position data prior to each day's operation, or prior to each operation extending more than one day, or following any significant change (>$0.25) in core configuration.

c. The shutdown margin shall be determined at each day's startup, or following any significant change (>$0.25) in core configuration.

d. All core fuel elements shall be inspected (under water) to assure compliance with Technical Specifications 3.1.6.a through 3.1.6.d quinquennially, but at intervals separated by not more than 500 pulses of magnitude greater than $1.00 of inserted reactivity. Fuel follower control rods shall be visually inspected and measured for bend at the same time interval. Such surveillance shall also be performed for elements in the B and C rings in the event that there is indication that fuel temperatures greater than the limiting safety system setting on temperature may have been exceeded.

e. Prior to resumption of routine pulse mode operations following a period of no pulse mode operations for more than 1 year, a test of pulsing performance with a pulse insertion of $1.50 shall be performed to assure pulsing power and fuel temperature response is as predicted from prior experience.

f. Full core, fuel and control rod surveillance shall be conducted before further reactor operation if significant changes are observed in any measured parameters such that it could be concluded that fuel element or control rod integrity has been compromised or fuel element or control rod damage has occurred.

Basis. Experience has shown that the identified frequencies are more than adequate to ensure performance and operability for this reactor. The value of significant change is measureable and will assure sufficient shutdown margin even taking into account decay of poison.
For fuel elements, the most severe stresses induced in the fuel elements result from pulse operation of the reactor, during which differential expansion between the fuel and the cladding occurs and the pressure of the gases within the elements increases sharply. The surveillance interval is selected based on the past history of more frequent, uneventful, inspections for over 40 years at this facility and experience at other TRIGA® facilities with similar power levels, fuel type, and operational modes. It is also designed to reduce the possibilities of mechanical failures as a result of handling elements, and to minimize potential radiation exposures to personnel.

4.2 Reactor Control and Safety Systems

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

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

Specification(s).

a. A channel calibration shall be made of the power level monitoring channels by the calorimetric method annually or immediately following any significant (>$0.25) core configuration change.

b. Control rod scram times for all four control rods shall be determined annually or for individual rods immediately following any maintenance work involving that control rod or drive mechanism that may have affected rod scram performance.

c. All control rods shall be visually inspected for deterioration quinquennially.

d. The transient (pulse) rod pneumatic cylinders and the associated air supply systems shall be inspected annually, and cleaned and lubricated if necessary.

e. On each day that pulse mode operation of the reactor is planned, the transient (pulse) rod system shall be verified to be operable before pulse operation is initiated.

f. A channel test of each of the reactor safety system channels and interlocks in Technical Specification 3.2 Tables 2 and 3, except for the pool water temperature measuring channel, shall be performed prior to each day's operation or prior to each operation extending more than one day.

g. A channel check of the functions of the seismic switch shall be performed annually or as soon as possible after an observed seismic event or one reported to be of sufficient magnitude to trip the switch.

h. A calibration of the pool water temperature measuring channel shall be performed annually to include verification of the alarm set point.

i. A calibration of the fuel temperature measuring channel shall be performed annually.

j. A calibration of the pool water level measuring channel shall be performed annually to include verification of the alarm set point.

Basis. The control rods are inspected and scram times checked to assure safe scram operations. The surveillance intervals for those and the channel surveillances are selected based on the past history for over 40 years at this facility and are adequate to correct for long term drifts and other instrument problems. The manufacturer of the seismic switch makes no recommendation for recalibration and believes the accelerometer settings remain effective for the life of the device. The channel test of the seismic switch involves simulation of a seismic event by tapping the switch to initiate a reactor scram, establishing operational functionality.

4.3 Reactor Pool Water

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

Objective. The objective is to assure that the reactor pool water level channel is operable, that alarm settings are verified and alarm reporting is functional. In addition, that the water level and purity is being maintained within acceptable limits.

Specification(s).

a. A channel check of the pool water level measuring channel shall be performed monthly to include channel verification of the alarm reporting system.

b. The pool water conductivity shall be measured at the end of each operating day or at shutdown for a period of operation extending more than one day. For periods of extended shutdown the conductivity measurement shall be made monthly.

c. The pool water temperature shall be monitored each hour during reactor operation.

d. The pool water radioactivity shall be measured quarterly.

e. The pool water loss rate shall be evaluated on each occasion when make-up water is added to the pool. Any unusual increase in loss rate shall be investigated as a possible pool leak before any further reactor operation.

Basis. These verifications will assure that a continued warning system for an unexpected loss of pool water is maintained, and that any perturbation of pool water quality noted then allows for corrective action to minimize corrosion, or build-up of radioactivity in the water. The frequent check on conductivity monitors possible leakage into the pool from the secondary water system. Temperature measurements will assure the pool water is maintained within operating limits. Radioactivity measurements will enable assessments of long term impacts of pool leaks and/or fission product leaks from a fuel element.

4.4 This section intentionally left blank.

4.5 Ventilation Systems

Applicability. This specification applies to the surveillance requirements for the reactor room ventilation system.

Objective. To verify performance is adequate to provide for normal and emergency mode ventilation for the facility to control and confine releases of airborne radioactive materials.

Specification(s).

a. A channel check of the existence of negative air pressure between the reactor room and the control room, and the reactor room and the outside air in both normal and emergency modes shall be performed daily.

b. A channel check of the exhaust flow rates from the reactor area in both normal and emergency modes shall be performed daily, to demonstrate that the ventilation system is operable in both normal and emergency modes by observation of flow rates, and valve/damper action.

c. A channel test of the function of the Continuous Air Particulate Monitor (CAM) alarm and the control room manual switch to properly set the ventilation system into emergency mode shall be performed daily.

Basis. Based on experience, these surveillances will assure that the ventilation system is functioning as specified. (Technical Specification 3.5).

4.6 This section intentionally left blank.

4.7 Radiation Monitoring System and Effluents

Applicability. This specification applies to the surveillance requirements for the radiation monitoring instrumentation required by Technical Specification 3.7.1.a and the effluent releases specified by Technical Specification 3.7.2.

Objective. The objective is to assure that the radiation monitoring system is operating properly and to verify the appropriate alarm settings and amounts of radioactivity in effluent releases.

Specification(s).

a. A channel test of the area radiation monitoring systems required by Technical Specification 3.7.1.a shall be performed monthly. This shall include verification of the alarm set points.

b. A channel check of the Continuous Air Particulate Monitor (CAM) required by Technical Specification 3.7.1.a shall be performed daily. This shall include verification of the alarm set point.

c. A channel calibration of the radiation monitoring systems required by Technical Specification 3.7.1.a shall be performed annually.

d. The environmental monitoring dosimeters required by Technical Specification 3.7.1.b including those monitoring exhaust effluents, shall be evaluated quarterly.

e. Any liquid effluents to be released to the sewer system from the facility shall be verified to contain only soluble materials and analyzed for radioactive content prior to release.

Basis. Surveillance of the equipment and effluents will assure that sufficient protection against excessive radiation or release of excessive radioactive materials is available. Past experience has shown that these practices and frequencies are adequate to assure proper operation.

4.8 Experiment Limits

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

Objective. The objective is to assure that experiments to be conducted do not damage the reactor or release excessive amounts of radioactive materials as a result of experiment failure.

Specification(s).

a. No experiment shall be installed in the reactor unless a safety analysis has been performed and reviewed in accordance with Technical Specifications 3.8 and 6.5.

b. The reactivity worth of a new experiment shall be verified at a power level less than 2 watts, before reactor operation at higher power with the experiment.

Basis. Past experience has shown that adherence to requirements described in Technical Specifications 3.8 and 6.5 are adequate to assure safe experimentation at this facility.

5. DESIGN FEATURES

5.1 Site and Facility Description

Specification(s).

The site shall be the reactor facility as described below.

The reactor facility shall be a restricted access area consisting of a main area, two associated laboratory areas, and a control room on a single level in the basement of Rowland Hall, on the University of California Irvine campus. The minimum free air volume of the reactor area including the two associated laboratories shall be 23,000 cubic feet. Normal entry to these areas shall be restricted to a single doorway from the control room. Large doors shall be provided to the adjacent loading dock to provide emergency egress and/or access for incoming or outgoing large items. Full visibility shall be provided between the control room and the reactor area.
The reactor shall be housed in a closed area designed to restrict leakage.

Basis. The extent of the site and facility is specified to define the controlled access area and the means of access. The closed area is designated to assist in mitigation of potential radioactive releases.

5.2 Reactor Coolant System

Specification(s).

a. The reactor core shall be cooled by natural convection water flow.

b. All piping and other equipment for pool water systems shall be above normal pool level. Inlet and outlet pipes that lead to the heat exchanger or demineralizer shall be equipped with siphon breaks not less than 14 feet above the upper core grid plate, unless those pipes end more than 14 feet above the upper core grid plate.

c. A pool water level indication is provided at the control console with an alarm at the control console and an alarm to a central monitoring station.

d. A pool water temperature indication shall be provided at the control console.

e. A pool water conductivity measurement instrument shall be provided in the reactor room.

f. Gamma and beta radiation spectrometry equipment shall be provided for water sample radioactivity assay.

Basis. Pool water quantity and quality is controlled so as to limit radiation and/or radioactivity release, and corrosion of components. Information is necessary to provide staff with indications of change in pool water characteristics.

5.3 Reactor Core and Fuel

5.3.1 Reactor Core.

Specification(s).

a. The core assembly shall consist of TRIGA® standard 8.5/20 stainless steel clad fuel elements.

b. The core fuel shall be kept in a close-packed array in core lattice positions except for control rods, single- or three-element or seven-element positions occupied by in-core experiments, irradiation facilities (including transfer system termini), graphite dummy elements, and a central dry tube.

c. Reflection of neutrons shall be provided by combinations of graphite and water, with the graphite in sealed containment with aluminum cladding, either in the form of rods occupying grid positions, or in a larger reflector structure surrounding the core.

d. An Am-Be neutron source shall be provided in one of two specific locations provided in the upper grid plate to provide start-up neutrons. It may be removed for maintenance purposes.

Basis. Standard TRIGA® fuel and reactor core design has a long and successful history of use. Model calculations in the SAR as supplemented by letter dated June 7th 2011, indicate acceptable neutronic and thermal hydraulic conditions for the core design under extended use and burn-up. The Am-Be source is in a sealed capsule and has a long useful life.

5.3.2 Control Rods.

Specification(s).

a. The SHIM and REG rods shall be motor driven with scram capability and solid boron compounds in a poison section, with fuel followers of standard TRIGA® fuel meeting the same specifications as in Technical Specification 5.3.3.

b. The ATR transient rod shall be motor and pneumatically driven, have scram capability, and contain solid boron compounds in a poison section. The ATR shall have an adjustable upper travel limit to provide variable pulse insertion capability. The FTR transient rod shall be pneumatically driven and have scram capability, and contain solid boron compounds in a poison section. The ATR and FTR shall incorporate air filled followers.

Basis. These control rods have been shown by model calculations and a history of use to be effective for assuring prompt shut-down and control of the reactor.

5.3.3 Reactor Fuel

Specification(s). Standard TRIGA® fuel elements shall have the following characteristics:

a. The total uranium content shall be nominally 8.5 % by weight, enriched to less than 20% ^{235}U.

b. The hydrogen to zirconium atom ratio in the zirconium hydride shall be a nominal 1.65 hydrogen atoms to 1.0 zirconium atom.

c. The cladding shall be 304 stainless steel, nominally 0.020 inches thick.

d. An upper fitting with an engraved unique serial number shall be designed to fit a latching tool for fuel movement.

Basis. TRIGA® fuel elements meeting these manufacturer's specifications have a long history of successful use with minimal failures. Minor deviations about these levels due to manufacturing variations are not to be considered violations of this specification.

5.4 Fuel Storage

Specification(s).

a. All fuel elements shall be stored in a geometrical array where the k_{eff} is less than 0.80 for all conditions of moderation and reflection.

b. Irradiated fuel elements and fueled devices shall be stored in an array which will permit sufficient natural convection cooling by water or air such that the fuel element or fueled device temperature will not exceed 80°C.

c. Fuel showing evidence of damage (see Technical Specification 3.1.6) shall be stored separately from fuel not suspected to be damaged, and shall be checked for fission product leakage.

Basis. These specifications establish a sufficient reactivity margin to guard against accidental criticality of elements in storage, and that heat dissipation does not create excess corrosion or other problems. Damaged fuel is more likely to have or develop fission product leakage and so must be monitored and kept separately.

5.5 Ventilation System

Specification(s).

a. The ventilation system shall operate in either normal or emergency mode. The ventilation system shall consist of ducts, blowers, dampers, flow and pressure measurement devices, and exhaust points above the roof of Rowland Hall.

b. During normal operations, the ventilation system shall be capable of exhausting air or other gases from the reactor area at a minimum rate of 3600 cubic feet per minute (cfm).

c. During normal operation the ventilation system shall be capable of maintaining a minimum of 0.10 inches of water pressure differential between the reactor area and the control room, and between the reactor area and the outside air.

d. During emergency situations involving release of radioactive materials into the air, an emergency exhaust with a high efficiency particulate arrestance (HEPA) filter shall be provided to exhaust a minimum of 240 cfm from the reactor area.

e. Shutdown of the normal reactor area exhaust system and start-up of the emergency exhaust system shall be initiated by a high radioactive particulate count rate alarm signal originating in the reactor room, or a manual switch in the control room.

f. During all modes of operation, the ventilation system shall exhaust at a minimum height of 90 feet above ground level.

Basis. The ventilation system assists in mitigating the effects of radioactive releases to the environment by providing dilution and control of such releases either during normal or emergency circumstances.

6. ADMINISTRATIVE CONTROLS

6.1 Organization and Structure

6.1.1 Structure.

The reactor facility is housed in the School of Physical Sciences of the University of California, Irvine. The reactor is related to the University structure of positions shown in the organization chart, Figure 1, where solid lines represent direct reporting responsibility, dashed lines indicate working relationships.

6.1.2 Responsibilities.

a. The licensee of the reactor is the Board of Regents of the University of California, which has delegated authority for license matters to the Provost and Executive Vice Chancellor (Level 1) of the University of California, Irvine.

b. The reactor facility is under the direction of a Reactor Director (Level 2) who shall be a tenure member of the University of California Irvine faculty. The Reactor Director shall report to the Chair of the Chemistry Department (Level 1), who, in turn, shall be responsible to the Dean of the School of Physical Sciences.

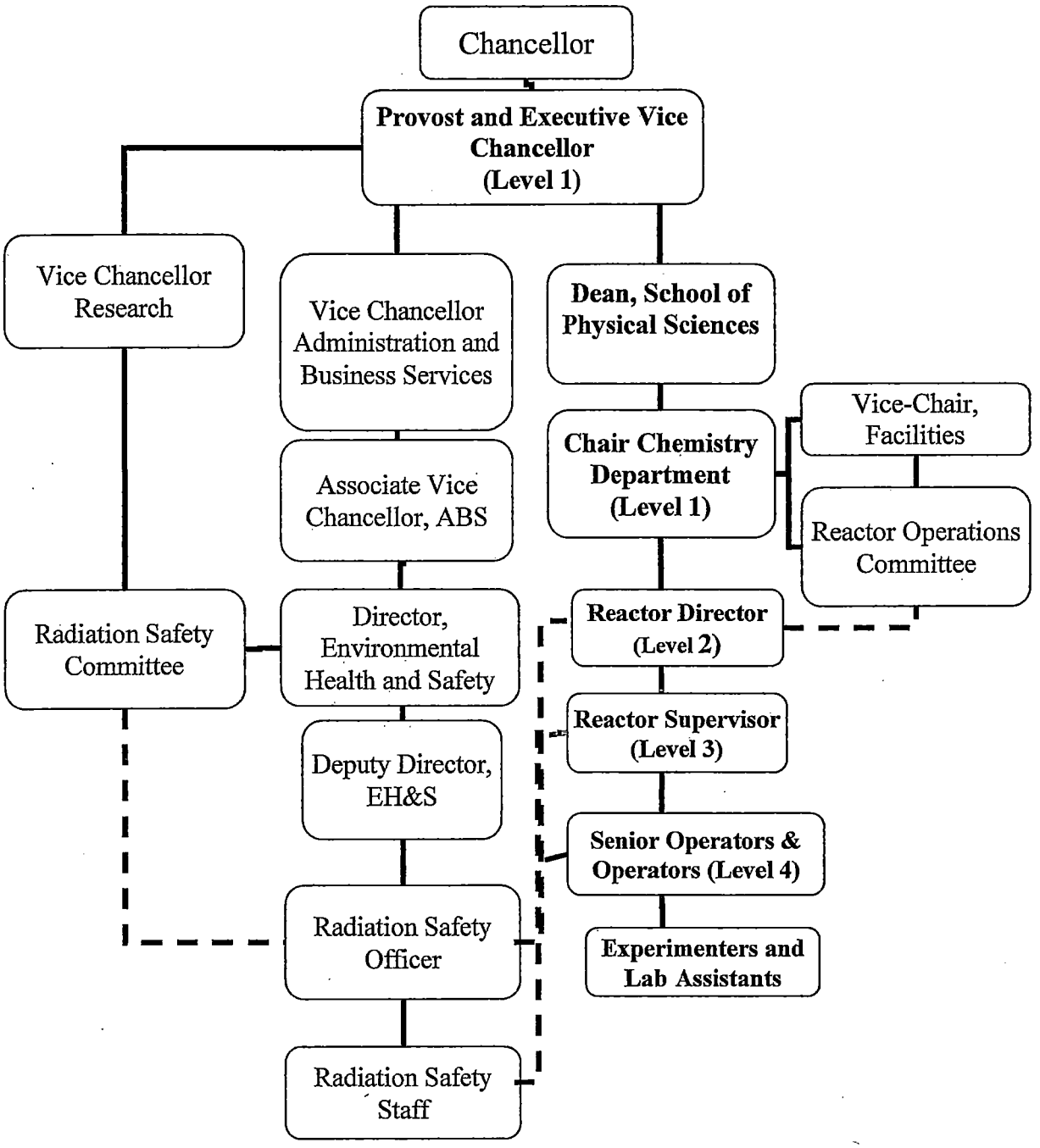
c. Operations shall be supervised by the Reactor Supervisor (Level 3) who shall hold a valid senior operator's license for the facility. This position shall be responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, the provisions of the Reactor Operations Committee and the provisions of the Radiation Safety Committee.

d. Reactor operators (Level 4) shall be responsible for operation of the reactor and performing needed maintenance and surveillance, including radiological safety and necessary supervision of experimenters. Senior reactor operators shall assume duties for supervision of operators as required by the US Nuclear Regulatory Commission (NRC) in Part 55 of 10 CFR, and Technical Specification 6.1.3.

e. There shall be a University of California, Irvine campus Radiation Safety Officer (RSO) responsible for the safety of operations from the standpoint of radiation protection. This position reports to the Office of Environmental Health and Safety which is an organization independent of the reactor operations organization as shown in Figure 1. An independent campus-wide Radiation Safety Committee (RSC) is responsible for establishment and review of all policies involving radiation and radioactivity. Routine radiological safety requirements within the reactor facility shall be carried out by reactor operators and/or individual experimenters, all of whom shall be required, by regulations, to have received training in radiological safety and be authorized for radiation use by the campus Radiation Safety Officer.

f. In the event of absence, or during filling of appointments to specific positions, temporary duties and responsibilities may be carried out by the person next higher or lower in line in the organization chart, provided the individual meets the basic qualifications for both positions.

Figure 1. UCI Reactor Organization Chart



6.1.3 Staffing

a. The minimum staffing when the reactor is not secured shall include:

   i. a licensed operator present in the control room; and

   ii. a second designated individual present within Rowland Hall able to carry out prescribed instructions and with the ability to check on the safety of the licensed operator and to act in the event of emergency; and

   iii. a licensed Senior Operator (SRO) readily available on call. Readily available on call means the SRO has been specifically designated, the designation is known to the operator on duty, the SRO can be rapidly contacted by phone by the operator on duty, and the SRO is capable of arriving at the reactor facility within 30 minutes under normal conditions.

b. A list of reactor facility personnel and other persons responsible for radiological safety and security on campus shall be kept in the reactor control room for use by an operator or experimenter. The list shall include telephone numbers of the Reactor Director, the Reactor Supervisor, the campus Radiation Safety Officer and other back-up radiological safety personnel, reactor operators, senior reactor operators, and personnel with responsibilities for maintenance in Rowland Hall.

c. Experimenters using the facility shall be certified by the campus Radiation Safety program as trained and authorized to use radioactive materials. The training shall include both general radiological training, including features of the ALARA program and specialized training in procedures for using reactor auxiliary experimental equipment (such as transfer systems), carrying out necessary surveys and record-keeping necessary for proper handling of radioactive materials within the reactor facility. Experimenters so trained and authorized are responsible for their own personal and sample/apparatus monitoring.

d. The following events require the presence in the facility of a licensed Senior Reactor Operator.

   i. Initial start-up and approach to power and final daily shutdown.

   ii. Fuel or control-rod relocations within the core region.

   iii. Insertion, removal, or relocation of any experiment worth more than $1.00.

   iv. Restart following any unplanned or unscheduled shutdown, or unexpected power decrease of >10%.

6.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall meet the requirements of ANSI/ANS-15.4 – 2007.

6.2 Review and Audit

A Reactor Operations Committee (ROC) shall review reactor operations to assure that the facility is operated in a manner consistent with public safety and within the terms of the facility license. Review and audit of radiological safety at the facility shall be carried out by the Radiation Safety Committee (RSC).

6.2.1 ROC Composition and Qualifications

The ROC shall have at least five voting members, at least one of whom shall be a health physicist designated by the Office of Environmental Health and Safety of the University. The Committee as a whole shall be knowledgeable in nuclear science and issues related to reactor and/or radiological safety. The membership shall include at least two members who are not associated with the Department of Chemistry. Approved alternates may serve in the absence of regular members. Members and alternates and a chairperson for the committee shall be appointed by the Chair of the Department of Chemistry (Level 1) or higher authority. The Reactor Director and Reactor Supervisor shall be non-voting members of the committee.

6.2.2 ROC Charter and Rules

The following responsibilities constitute the charter of the ROC.

a. Meeting at least annually, with provision for additional meetings when circumstances warrant to assure safety at the facility.

b. A quorum shall consist of not less than a majority of the voting members and shall include the chairperson or his/her designee. A quorum shall not consist of a majority of operations staff.

c. Review and audit of facility staff and operations as indicated in Technical Specifications 6.2.3 and 6.2.4.

d. Designation of individuals to perform audits of facility operations and records.

e. Preparation, approval, and dissemination of minutes of meetings.

f. Preparation and dissemination of findings and other reports as needed to assure safe operations of the reactor.

g. Approval of individuals for the supervision and operation of the reactor.

6.2.3 ROC Review Function

The following review functions shall be the responsibility of the ROC.

a. Review and approval of all proposed changes to the facility, its license, procedures, ROC charter, and Technical Specifications, including those made under provisions of 10 CFR § 50.59, and the determinations leading to decisions relating to 10 CFR § 50.59 approvals.

b. Review and approval of new or changed procedures, experiments, components, or instrumentation having safety significance.

c. Review of the quality assurance program implementation applicable to the reactor components.

d. Review of new experiments or changes in experiments that could have reactivity or safety significance.

e. Review of violations of technical specifications, license, or violations of procedures or instructions having safety significance.

f. Review of operating abnormalities that have safety significance.

g. Review of actions and reports listed in Technical Specifications 6.6.1, 6.6.2, or 6.7.2.

h. Review of audit reports, including reports from the campus Radiation Safety Officer, regarding the radiation protection program.

6.2.4 ROC Audit Function

The ROC shall perform audits or review audits performed by designated individuals on its behalf at least annually. The audit shall include, but not be limited to, the following items.

a. Facility operations for conformance to the Technical Specifications and applicable license conditions.

b. Retraining and requalification of operators according to the Requalification Plan.

c. The result of action taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, procedures or methods of operation that affect reactor safety.

d. The facility Emergency Plan (EP) and implementing procedures including written reports of any drills or exercises carried out.

e. At least one of the auditors shall be familiar with reactor operations but not directly responsible for any portion of reactor operations.

f. Any deficiencies identified in an audit that affect reactor safety shall be immediately reported to the chairperson of ROC, and to the Level 1 administrator. A written full report shall be submitted to ROC within 3 months of any audit.

6.3 Radiation Safety

As delineated in Technical Specification 6.1.2.e, the campus Radiation Safety Officer (RSO) is responsible for implementation of the radiological safety program at the reactor facility in accordance with applicable federal and state of California standards and regulations. The program should use the guidelines of ANSI/ANS 15.11-2004.

The RSO shall be responsible for an annual audit of the radiation safety program.

6.4 Operating Procedures

Written procedures, reviewed and approved by the ROC and the Reactor Director, shall be in effect and implemented for the following listed items. The procedures shall be adequate to assure the safety of the reactor but not preclude the use of independent judgment and action should the situation require such. Any changes to procedures shall be made in accordance with the requirements of 10 CFR § 50.59.

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

b. Installation or removal of fuel elements, control rods, experiments, and experimental facilities.

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

d. Surveillance checks, calibrations and inspections required by the technical specifications or that could have an effect on reactor safety.

e. Personnel radiation protection, including provisions to maintain personnel exposures as low as reasonably achievable (ALARA).

f. Administrative controls for operations and maintenance, and for the conduct of irradiations or experiments that could affect reactor safety.

g. Implementation of required plans including Emergency (EP) and Physical Security (PSP) plans.

h. Use, receipt and transport of by-product materials.

6.5 Experiment Review and Approval

Approved experiments shall be carried out in accordance with established and approved procedures. Procedures for experiment review and approval shall include the following requirements.

a. All new experiments or class of experiment shall be reviewed and approved by the ROC and approved in writing by the Reactor Director. The review shall include analysis by the RSO or other designated radiation safety personnel.

b. Changes to existing experiments or classes shall be made only after review by the ROC and RSO or the RSO designee, following performance of a 10 CFR § 50.59 evaluation and the conclusion that the proposed changes do not require prior NRC approval.

6.6 Required Actions

This specification is intended to assure compliance with requirements of 10 CFR § 50.36.

6.6.1 Actions to Be Taken in Case of a Safety Limit Violation

In the event the safety limit on fuel temperature is exceeded:

a. the reactor shall be shut down and the event reported immediately to the Reactor Director, the ROC chairperson, and the RSO. Reactor operation shall not be resumed until authorized by the NRC; and

b. the event shall be reported within the next working day to the NRC Headquarters Operations Center; and

c. a follow-up written report sent shall be reviewed by the ROC and sent within 14 days to the NRC (according to provisions of Technical Specification 6.7) describing:

   i. applicable circumstances leading to the violation including, where known, the cause and contributing factors; and

   ii. effects of the violation upon reactor facility components, systems, or structures, and on the health and safety of personnel and the public; and

   iii. the basis for corrective action taken to preclude recurrence.

6.6.2 Actions to Be Taken in the Case of Events Other than a Safety Limit Violation

In the event of an occurrence of the type listed in items 1-7 below:

a. the reactor shall be secured and the Reactor Director and Supervisor notified; and

b. operation shall not be resumed until authorized by the Reactor Director and the ROC; and

c. a follow-up written report shall be reviewed by the ROC and sent within 14 days to the NRC (according to provisions of Technical Specification 6.7) describing:

   i. applicable circumstances leading to the violation including, where known, the cause and contributing factors; and

   ii. effects of the violation upon reactor facility components, systems, or structures, and on the health and safety of personnel and the public; and

   iii. the basis for corrective action taken to preclude recurrence.

1. Release of radioactivity from the site above allowed limits.

2. Operation with actual safety system settings for required systems less conservative than the limiting safety system settings in these specifications.

3. Operation in violation of limiting conditions for operation unless prompt remedial action is taken as permitted in Technical Specification section 3.

4. A required reactor safety system component malfunction that renders or could render the safety system incapable of performing its intended safety function. If the malfunction or condition is caused by maintenance, then no report is required.

5. An unanticipated or uncontrolled change in reactivity greater than one dollar. Reactor trips resulting from known cause are excluded.

6. Abnormal or significant degradation in reactor fuel or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks) where applicable.

7. An observed inadequacy in implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

6.7 Reports

In addition to the requirements of applicable regulations, and in no way substituting for them, reports shall be made to the NRC as listed below. All written reports shall be directed to the Document Control Desk, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555-0001.

6.7.1 Annual Operating Report

A routine written annual report shall be submitted by the Reactor Director to the NRC, by September 30th each year regarding operations for the preceding academic year (July 1st through June 30th). The report shall include, but not be limited to:

a. a brief narrative summary of operating experience (including experiments performed) and a tabulation showing the energy generated by the reactor (in megawatt hours), the amount of pulse operation, and the number of hours the reactor was critical; and

b. the number of unplanned shutdowns and inadvertent scrams, including the reasons therefore, and corrective actions taken (if any) to reduce recurrence; and

c. a tabulation of major preventive and corrective maintenance operations having safety significance; and

d. a tabulation of changes in the reactor facility and procedures, and tabulations of new experiments, including a summary of the analyses leading to the conclusions that they are allowed without prior authorization by NRC and that 10 CFR § 50.59 was applicable; and

e. a summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the facility as measured at or prior to the point of such release or discharge. The summary shall include, to the extent practicable, an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed, a statement to this effect is sufficient; and

f. a summarized result of environmental surveys performed outside the facility; and

g. a summary of radiation exposures received by facility personnel and visitors, where such exposures are greater than 25% of that allowed.

6.7.2 Special Reports

In addition to reports required according to Technical Specification 6.6, a report shall be made in writing to the Document Control Desk, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555-0001 within 30 days of:

a. permanent changes in facility organization involving Level 1 or Level 2 personnel; and

b. significant changes in the transient or accident analyses as described in the SAR.

6.8 Records

In addition to the requirements of applicable regulations, and in no way substituting therefore, records and logs shall be prepared and retained for periods as described here. Records may be in a variety of formats.

6.8.1 Records to Be Retained for a Period of at Least 5 Years or for the Life of the Component Involved If Less than 5 Years.

a. Normal reactor facility operation, but not including supporting documentation such as checklists, log sheets, etc., which shall be retained for one year.

b. Principal maintenance activities.

c. Reportable occurrences.

e. Surveillance activities required by the Technical Specifications.

f. Reactor facility radiation and contamination surveys.

g. Experiments performed with the reactor.

h. Fuel inventories, receipts and shipments.

i. Approved changes in operating procedures.

j. ROC records of meetings and audit reports.

6.8.2 Records to Be Retained for at Least One Certification Cycle

Records of retraining and requalification of all licensed operators shall be retained at all times each individual has duties as an operator or until his or her license is renewed.

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

The following records shall be retained for the lifetime of the facility. Applicable annual reports containing this information may be used as records.

a. Reviews and reports pertaining to a violation of a safety limit, the limiting safety system setting, or a limiting condition for operation as described in Technical Specification 6.6.

b. Gaseous and liquid radioactive effluents released to the environs.

c. Results of off-site environmental monitoring surveys.

d. Radiation exposures for all personnel that were monitored.

e. Drawings of the reactor facility and safety related components.

UCI TRIGA® Reactor Reactivity RAMP Insertion Simulation.

J. T. Wallick, Associate Reactor Supervisor

G. E. Miller, Reactor Supervisor

December 2015

Conclusions

Parameters of interest achieved the following values during the simulation:

- Maximum reactor power was 1,118,874 watts.
- Peak fuel temperature was 40.24 °C.
- Highest core reactivity was 7.87 E-03 $\frac{dk}{k}$.
- Shortest positive reactor period was 1.52 E-03 seconds.
- Prompt criticality was never achieved.
- Total energy released was 0.39 kWh.

Overall, the transient behaved as expected given the conditions and accident analyzed. It provides fairly conclusive evidence that a reactor designed for far harsher transients, such as rod ejections using compressed air achieving transient times on the order of milliseconds, can withstand an excessively conservative uncontrolled continuous rod withdrawal scenario.

Methodology.

The program used was developed in spreadsheet software. The methodology used in modeling the reactor's predicted behavior during this transient is calculated in an iterative fashion as follows:

1. Given input parameters for the simulation:
   a. Effective delayed neutron precursor decay constant (λ_{eff}) is held constant at 0.1 sec^{-1}.
   b. Temperature coefficient of reactivity (δk_{α}) is -1.00 E-04 $\frac{dk}{k}/{}^{\circ}C$
   c. Effective delayed neutron fraction (β_{eff}) is 0.0079
   d. Prompt neutron generation lifetime (l^*) is 9.85 E-05 sec
   e. Reactivity addition rate of the rods($\delta k'_{rods}$) is 4.80 E-04 $\frac{dk}{k}/sec$
   f. Initial reactor power is 1.5 watts.
   g. Initial reactor fuel temperature is 25.0 °C.
   h. Single element total heat capacity is 1088 $\frac{W*sec}{C}$ + 1.5 * Fuel Temp $\frac{W*sec}{C}$
   i. Number of fuel elements in core is 80.
   j. The total fuel element heat capacity is 87040 $\frac{W*sec}{C}$ + 120 * Fuel Temp $\frac{W*sec}{C}$
   k. A high power scram will occur one second after power reaches 110% power (275 kW) to allow for instrumentation response time.
   l. The scram will insert all rods except the highest worth rod, which will stop outward motion.
   m. The total negative reactivity insertion of the scram will be 1.9039 E-02$\frac{dk}{k}$, which will be inserted over the course of one second.

2. Reactor parameters are calculated every iteration in the following manner:

   a. Time is incremented in 100 µsec steps, equal to the prompt neutron lifetime in the University of California, Irvine TRIGA® Mk I Reactor as calculated by General Atomics.
   b. Power is calculated based on the previous power, difference in time, and previous reactor period.

$$P = P_o * e^{\frac{t}{\tau}}$$

   c. Reactivity change of the control rods is added to the core.
   d. Energy generated is calculated based on reactor power and difference in time.

$$E = P * t$$

   e. Temperature change is calculated based on energy generated, total fuel element heat capacity, and previous fuel temperature.

$$\Delta T = \frac{E}{C}$$

   f. Current temperature is calculated based on the temperature change and previous temperature.

$$T = \Delta T + T_o$$

   g. Reactivity change of the temperature coefficient of reactivity is calculated based on the temperature change and the temperature coefficient of reactivity.

$$\Delta\delta k_\alpha = \Delta T * \alpha$$

   h. Current core reactivity is calculated based on the previous core reactivity, rod reactivity change, and reactivity change of the temperature coefficient of reactivity.

$$\delta k_{current} = \delta k_{previous} + \Delta\delta k_{rods} + \Delta\delta k_\alpha$$

   i. Period is calculated using the traditional in-hour equation and the current core reactivity.

$$\tau = \frac{l^*}{\rho} + \frac{\overline{\beta}_{eff} - \rho}{\lambda_{eff} * \rho - \dot{\rho}}$$

   j. Calculations repeat in this manner over the course of 30 seconds.

   k. Discontinuities arise as parameters switch negative to positive changes, prompt vs. delayed "switch over" and no allowance being made for "down power" effective delayed neutron precursor decay constant.

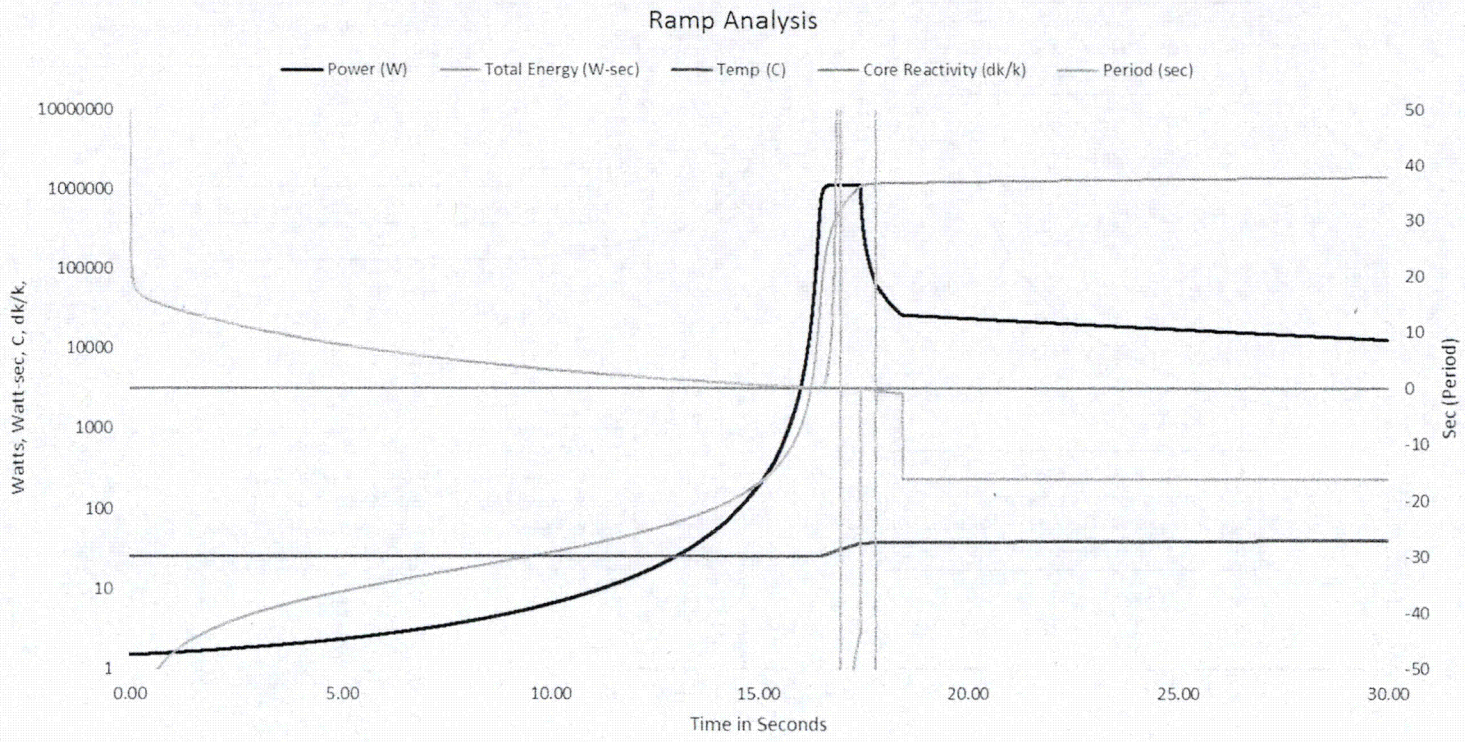
The attached graphs show:

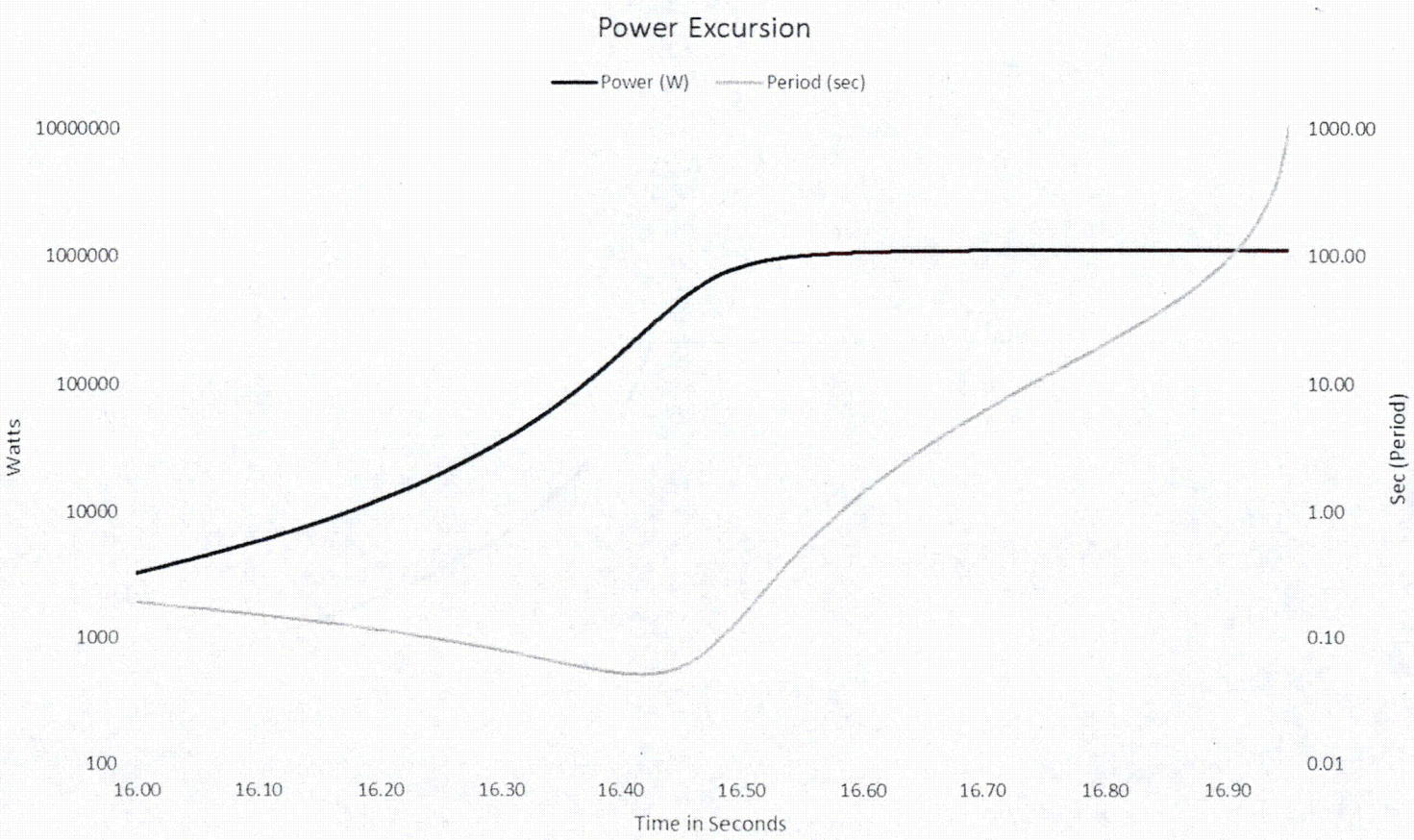
1. Power, Energy, Fuel Temperature, Reactivity and Period over the 30 second time of a RAMP.

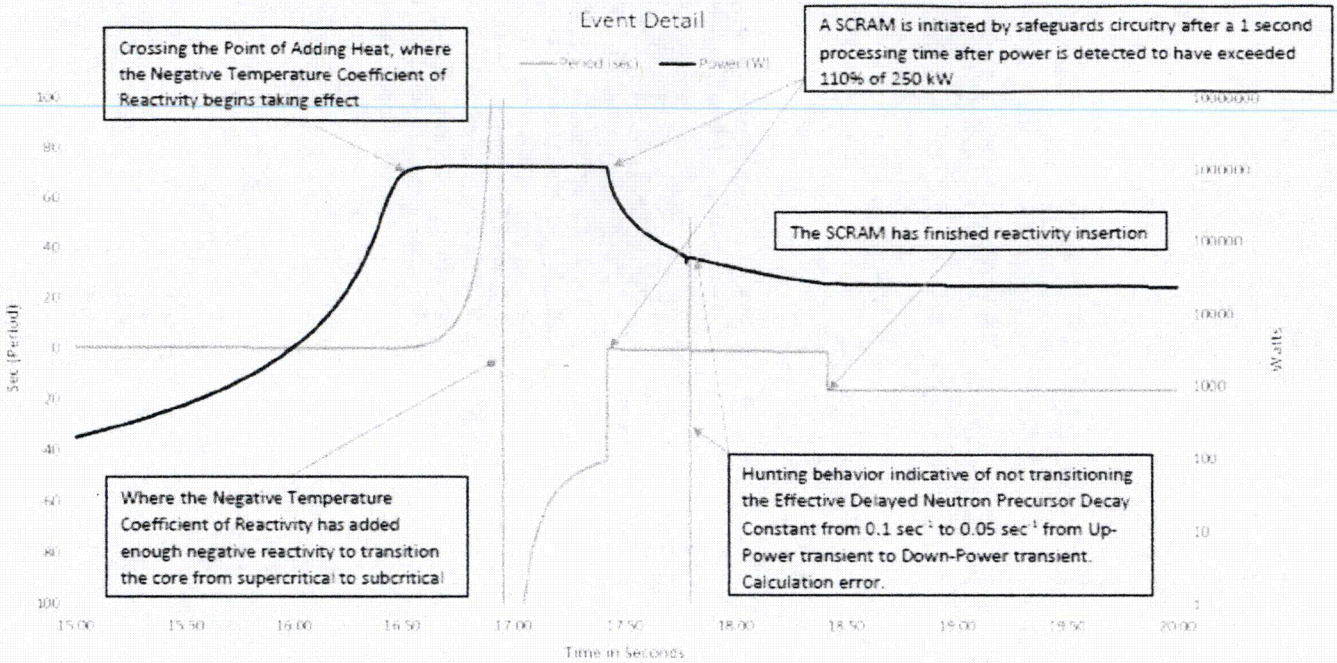
2. Power and period changes determined over the significant time period 16-17 seconds into the RAMP.

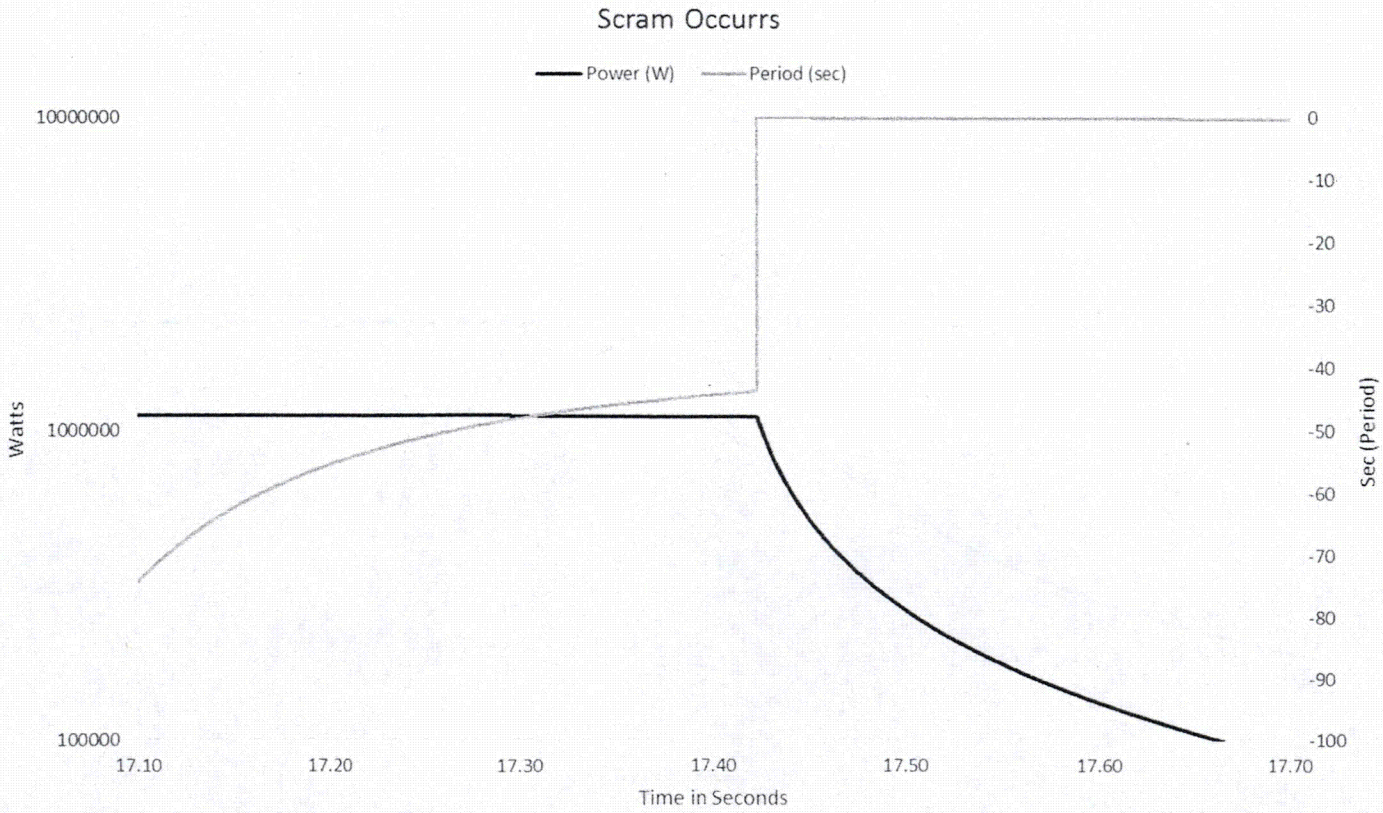
3. An annotated plot of the simulation results over the "excursion" time period of the event from 15.0 to 20. 0 seconds.

4. A detail of power and period through the predicted scram at 17.42 seconds into the RAMP.


Ramp Analysis
Power (W)
Total Energy (W-sec)
Temp (C)
Core Reactivity (dk/k)
Period (sec)
Watts, Watt-sec, C, dk/k,
Sec (Period)
Time in Seconds
10000000
1000000
100000
10000
1000
100
10
1
50
40
30
20
10
0
-10
-20
-30
-40
-50
0.00
5.00
10.00
15.00
20.00
25.00
30.00


Power Excursion
Power (W)
Period (sec)
Watts
Sec (Period)
Time in Seconds
10000000
1000000
100000
10000
1000
100
1000.00
100.00
10.00
1.00
0.10
0.01
16.00
16.10
16.20
16.30
16.40
16.50
16.60
16.70
16.80
16.90


Event Detail
Period (sec)
Power (W)
Crossing the Point of Adding Heat, where the Negative Temperature Coefficient of Reactivity begins taking effect
A SCRAM is initiated by safeguards circuitry after a 1 second processing time after power is detected to have exceeded 110% of 250 kW
The SCRAM has finished reactivity insertion
Where the Negative Temperature Coefficient of Reactivity has added enough negative reactivity to transition the core from supercritical to subcritical
Hunting behavior indicative of not transitioning the Effective Delayed Neutron Precursor Decay Constant from 0.1 sec⁻¹ to 0.05 sec⁻¹ from Up-Power transient to Down-Power transient. Calculation error.
Sec (Period)
Watts
Time in Seconds


Scram Occurrs
Power (W)
Period (sec)
10000000
1000000
100000
Watts
0
-10
-20
-30
-40
-50
-60
-70
-80
-90
-100
Sec (Period)
17.10
17.20
17.30
17.40
17.50
17.60
17.70
Time in Seconds