Department of Mechanical Engineering

THE UNIVERSITY OF TEXAS AT AUSTIN

Nuclear Engineering Teaching Laboratory · http://www.me.utexas.edu/~nuclear/index.php/netl
1 University Station, R9000 · Austin, Texas · 78712 · (512) 232-5370 · Fax (512) 471-4589

June 22, 2012

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

Subject: Request for Change to License Technical Specifications Incorporating 2008, 2010, and 2011 Requests

Ref:
1. The University of Texas at Austin Facility License R-129, Docket 50-602
2. Letter of March 28 2008 (ML080920755)
3. Letter of April 1, 2010 (ML101241147)
4. Letter of April 6, 2010 (ML101330271)
5. The University of Texas at Austin Request for Additional Information, Re: License Amendment request, Submission of changes to License Technical Specifications (TAC NO. ME8072) (June 14, 2012, ML121290640)

Ms. Torres:

This letter communicates the response to the request for information (ref. 5), and includes two parts. The 1st part provides an update to the request for Technical Specification revision based on (1) the status of NRC determining the application of "initial startup" to research and test reactors, and (2) the results of a technical editing review. The 2nd part provides specific response to individual RAIs.

Part 1

Pursuant to telephone conversation regarding expectations of how this request will be processed, we are withdrawing the request for a definition of *initial startup* and revising the request for change to an unlabeled section of 6.1.3 which is currently:

Events requiring the direction of a senior reactor operator shall be:

We request 6.1.3 be changed to:

Events requiring the presence at the facility of a senior operator shall be:

A020
NRR

which is consistent with ANSI/ANS-15.1-2007. Although the ANSI standard only requires the presence of a senior reactor operator *"at the facility,"* the UT Safety Analysis Report (10.1.3.1) states:

> *"Movement of fuel or control rods and relocation of experiments with greater than one dollar reactivity worth will require the presence of a license certified senior operator. Other activities, such as initial startup, recovery from unscheduled shutdowns and modifications to instrument systems, control systems, safety systems, radiation measurement equipment or engineered safety features, will require concurrence and documentation by a license certified senior operator."*

Therefore it is proposed to add to 6.1.3:

c. *Operating procedures shall specify when a senior operator is required to directly supervise licensed activities, including (but not limited to):*
   i. *The first startup (excluding control rod manipulations with the reactor shutdown for maintenance or testing) and ascension to power following:*
      (a). *Relocation of fuel or control rods,*
      (b). *Relocation of experiments with a reactivity worth of greater than one dollar,*
      (c). *Modifications to the reactor control or safety system,*
      (d). *Non-routine corrective maintenance to the reactor control and safety system, and/or*
      (e). *Recovery from an unscheduled shutdown.*
   ii. *Changes to the configuration of the reactor core:*
      (a). *Fuel element or control rod relocations within the reactor core region, and/or*
      (b). *Relocation of any experiment with a reactivity worth of greater than one dollar.*

The bases for the Technical Specifications have been augmented to indicate the source of the requirements in the specification, differences between the specification and the ANSI-15.1-2007, and clarification of the term "Relocation."

This added material preserves the basis for "the presence of a licensed certified senior operator" as provided in the UT Safety Analysis Report, augments requirements for additional conditions requiring supervision, and exceeds recommendation of ANSI-15.1-2007. Therefore the proposed change does have the potential for any negative impact on the safety basis.

Additional items for correction were identified while performing a final, technical editing review:

1. Reactivity units in 2.2.3 and 3.2.1 c were revised to include reactivity in both Δk/k and $. The initial submission requested that reactivity specifications be revised to include dollars ($) as the unit used in operations, but these sections were not included in the original submission. This is an editorial change, and does not possess any potential impact on safety.

2. Specification 3.3.3c was changed from mr/hr to mR/hr. This is an editorial change, and does not possess any potential impact on safety.

3. Specification 3.3.3 was changed to cm^3 from cm3. This is an editorial change, and does not possess any potential impact on safety.

4. Section 6.1.3 was formatted with an indexed paragraph leading each list where the original contained multiple bulleted lists with the same index letters. This is an editorial change, and does not possess any potential impact on safety.

5. The Reactor Oversight Committee meeting frequency (6.2.2) was changed from 6 months to "at least twice each year, with no more than 9 months between meetings." The intent is to conduct a review meeting two times per year, and not necessarily at exact 6 month intervals. Review frequency exceeds the recommendations of ANSI-15.1-2007, and is an editorial change with no safety significance.

6. The internal list of items required in report of a safety violation (6.5.d) was converted to a formatted, numbered list. This is an editorial change, and does not possess any potential impact on safety.

7. Corrections changing the name "College of Engineering" to "Cockrell School of Engineering" were found to have been inadequately implemented, and this was corrected. This is an editorial change, and does not possess any potential impact on safety.

Part 2

The RAI responses indicated below have been implemented in the attached, proposed Technical Specifications.

Response to RAI 1: The table of contents has been updated.

Response to RAI 2.a: The term "licensee" has been removed.

Response to RAI 2.b: "Certified Operators" has been removed.

Response to RAI 3: The term "senior reactor operator" has been replaced by "senior operator".

Response to RAI 4: "Senior Operator" and "Operator" have been revised to the 10CFR55.

Response to RAI 5.a: There are several "limiting conditions for operation" (LCOs) in the approved UT Technical Specifications which do not explicitly state that the reactor must be shutdown if the condition is not met. In fact, Technical Specifications, section 3.1., Reactor Core Parameters, contains three specifications (Excess Reactivity, Shutdown Margin, and Transient Insertions) of which only one specifies "The reactor shall not be operated unless" the condition is met. Regardless, if an LCO is not met then the reactor must be shutdown; the requirement is not based solely on LCO specification statements.

The requirement that the reactor be shutdown if an LCO is not met (or the condition restored) is in section 6.5.2, where it states "In the event of reportable occurrence...Reactor conditions shall be returned to normal or the reactor shutdown," with "Operation in violation of a limiting condition for operation established in technical specifications..." identified as a reportable event in 6.6.2.

Since failure to meet the LCO requires a reactor shutdown by 6.5.2, the phrase "The reactor shall not be operated unless" does not affect requirements or actions to be taken. As noted, the current Technical Specifications as approved by the USNRC contain multiple LCOs that do not include such a requirement explicitly in the LCO.

While the phrase ["The reactor shall not be operated unless"] is not precluded, the phrase is inconsistent with the structure of the other, similar reactivity LCO statements in 3.1. Therefore, removal of the phrase is strictly an editorial revision to make 3.1.2 more parallel to 3.1.1 and 3.1.3, and has no safety significance.

Response to RAI 5.b: The *reference core conditions* are used to verify reactivity limits are met. An accurate assessment of reactivity in the reference core conditions is significant. The original definition for reference core condition was taken from ANSI 15.1, and includes a considerable margin of error. However, the UT facility has never determined reference core conditions with significant xenon present, and does not consider a reference reactivity value with a large margin of error useful or desirable.

Calculations show that reactivity change from xenon following shutdown from full power equilibrium conditions cannot be detected using control rod positions after 72 hours, where the reactivity from xenon is less than the \$0.05 (which experience shows is a minimum detectable level based on a single control rod differential position). Therefore, a more conservative approach is proposed where the reference core condition is restricted to conditions where reactivity from xenon is below the level that would occur 72 hours after shutdown. This may be verified in determining the reference core condition by allowing a 72 shutdown hour interval or by calculation when desired (for instance, when the reactor has not achieved equilibrium conditions prior to shutdown).

The specification has been revised to reflect this proposed requirement, and supporting analysis has been completed and incorporated in a revision to the basis for section 3.1.

Response to RAI 5.c: The bases have been revised as follows:

A.3.1 – Analysis of xenon transients to justify limits on reactivity from xenon when determining reference core conditions.

A.3.1.2 – Incorporation of the restriction on using experiments to credit shutdown margin.

A.3.1.2 – Statement of basis for the pulse timer.

A.3.2.4 –Support for reduced redundancy in requiring two fuel temperature channels.

A.3.4.2 – Support for removal of the reference to the "Tables of Chemical Hazard Information" that are no longer supported in the "Handbook of Laboratory Safety," and identification of sources of information previously provided in the Table.

A.5 – Added bases for Design Features to reference sources for the Design Feature specifications and provide additional information where warranted to aid understanding the safety basis for the specification.

A.6 – Added bases for Administrative Controls to reference sources for Administrative Controls specifications and provide additional

information where warranted to aid understanding the safety basis for the specification.

Response to RAI 6: The wording has been revised to more clearly indicate the specification applies to operation in the pulse mode.

Response to RAI 7.a: Fuel temperature trip follows single channel logic, i.e., only one channel is required for a fuel temperature scram, and a second channel is redundant, i.e., two channels are not required to cause a scram. The basis has been revised.

Response to RAI 7.b: Diversity and redundancy in safety system protection against exceeding the fuel temperature safety limit is provided by a combination of fuel temperature and power level channels, as indicated in A.3.2.3. Two redundant percent power level channels monitor the power level for the limiting safety system. A digital wide range channel may also function as a safety channel but only by diversification as a supplemental channel to an analog linear power channel. The temperature trip provides a third level of redundancy above the two redundant power level channels, and a third level of diversity above the combination of analog and digital power level channels. The basis has been revised.

Response to RAI 8: The *Handbook of Laboratory Safety* no longer provides a "Table of Chemical Information." The Handbook states that using the MSDS for material compatibility and characterization is appropriate. The previous Table was based on data from various sources, which should not be precluded from use and are provided in the revised Basis.

Response to RAI 9: The words "high reactive" have been changed to "highly reactive"; this was a transcription error.

Response to RAI 10: The correct wording was inadvertently altered when preparing the proposal transmittal; no change is being sought based on the incorrect transcription of the current Technical Specifications.

Response to RAI 11: The deletion of the statement regarding release of radioactivity to the reactor room or to the environment was a transcription error as noted in the previous (RAI 10) response; the phrase has been restored.

Response to RAI 12: The request for change on this item is withdrawn.

Response to RAI 13: In 2001 the concept of this revision was discussed with the URNRC program manager, and the results documented by an internal memorandum (attached) in 2000. It was determined that:

> *"byproduct material produced in the NETL reactor is under the jurisdiction of the 10 CFR part 50 license while within the boundary of the NETL. The NETL site is defined in the Technical Specification and the Safety Analysis report and includes all areas within Building 159 at the J.J. Pickle research campus."*

It was decided at the time that a license change was not required. In 2008 it was decided that rather than rely on corporate knowledge of interpretation of the regulation, a clear "rule based" approach would be adopted and a Technical Specification change requested.

Therefore, this change does not increase the area of the Nuclear Engineering Teaching Laboratory under the reactor license and only converts an interpretation into the appropriate format. As indicated in earlier submissions of this proposed revision, the current Radiation Protection Program, Security Plan, Emergency Plan and facility procedures currently incorporate the interpretation and no change to plans or procedures is required. As clarification of conditions already in effect, this is strictly an editorial change with no safety significance.

Response to RAI 14.a: The national consensus standard ANSI/ANS-15.1-2007 does not specify position titles, and instead uses generic coding (Level 1, 2, 3, and 4) to identify generic types or examples of responsibilities. Within the standard, the terms Level 1, 2, 3, or 4 are used in 6.1.1 (Structure), 6.2.1 (Composition and Qualifications – i.e., of the review and audit group), 6.2.2 (Charter and Rules), 6.2.3 (Review Function), 6.2.4 (Audit Function), 6.3 (Radiation Safety), 6.4 (Procedures), and 6.5 (Experiments Review and Approval), 6.6.1 (Action to be taken in the Event of a Safety Limit Violation) and 6.6.2 (Action to be taken in the event of an Occurrence of the type identified in Secs. 6.7.2(1)(b) and 6.7.2.(1)(c)) and 6.7.2 (Special reports).

Level terminology was not used in the approved Safety Analysis Report, Safety Evaluation Report (NUREG 1135 and supplement), or the original UT Technical Specifications in any of the sections corresponding to ANSI/ANS-15.1 except in the Technical Specifications (1) as labels for specific position titles on the organization chart, and (2) to identify those positions that require notification of personnel changes in section 6.6.2. In every other section, the Level terminology was not used, in deference to specific UT positions or titles. In this proposal, specific positions for which a change requires a report to the NRC have been explicitly defined and the level terminology is no longer used.

This is an editorial change that makes terminology in the organization chart and section 6.6.2 consistent with the remainder of section 6 as previously approved by the USNRC. This change is strictly administrative, and does not have any potential safety significance.

Response to RAI 14.b: The description for 6.1.1 was revised to reflect current positions and responsibilities.

Response to RAI 14.c: "Operations Staff" in the original technical specifications explicitly refers to SRO and RO on the graphic; the "operations staff" is now replaced by operator and senior operator.

Response to RAI 14.d: The specified information has been added to the chart.

Response to RAI 14.e: Following a comprehensive review of subsection labeling of section 6, the procedures section label was corrected and one additional action taken.

Section 6.6.2 of the original technical specifications had three indexed lists without logical or numbered separators so that uniquely referencing a specific item on one of the lists was problematic. Therefore section was revised to include two numbered sections with 6.6.2 for 30 day notification requirements and 6.6.3 for immediate notification and follow-up requirements. The third list, "other events that will be considered reportable," was incorporated into 6.6.3 rather than as a separate list. This is strictly an editorial change, and does not have safety significance

Please contact me by phone at 512-232-5373 or email whaley@mail.utexas.edu if you require additional information or there is a problem with this submittal.

Thank you,

P. M. Whaley
Associate Director
Nuclear Engineering Teaching Laboratory
The University of Texas at Austin

Attachments:

1. Memorandum, S. O'Kelly (Associate Director, NETL) to J. White (UT Radiation Safety Officer) Re: Licensed By-Product Material at the NETL
2. Updated, Proposed Technical Specifications
3. Technical Specifications Basis Document

I declare under penalty of perjury that the foregoing is true and correct.
Executed on February 8, 2012.

Steven R. Biegalski
NETL Director

DEPARTMENT OF MECHANICAL ENGINEERING
THE UNIVERSITY OF TEXAS AT AUSTIN

Nuclear Engineering Teaching Laboratory · (512) 471-5787 · FAX (512) 471-4589

January 8, 2001

Memorandum

To: John White
Radiation Safety Officer

From: Sean O'Kelly
Associate Director, NETL

Subject: Licensed By-Product Material at the NETL

Ref: NRC Inspection Manual, Part 9900: 10 CFR Guidance

The NRC has confirmed in a phone conversation (A. Adams, Project Manager) that byproduct material produced in the NETL reactor is under the jurisdiction of the 10 CFR Part 50 license while within the boundary of the NETL. The NETL site is defined in the Technical Specifications and the Safety Analysis Report and includes all areas within Building 159 at the J.J. Pickle Research Campus.

NRC guidance documents issued 8/23/88 specify the following conditions for byproduct material used at a research reactor facility:

1. Byproduct material produced by the facility is under the facility license within the facility site until transferred for disposal or shipment to a state or NRC materials license.

2. Byproduct material transferred to the reactor facility for irradiation will remain on the State license until the time it is placed in the reactor.

3. Calibration sources and byproduct materials not specified on the Part 50 license shall remain under the jurisdiction of the State license.

This provides clear license separation of the byproduct material at the NETL and makes a license change unnecessary. There may be exceptions to the generic guidance but these may be considered case-by-case. Please contact me if you require additional information.

cc: CF#23
C. Beard
K. Ball

Appendix A
DRAFT Technical Specifications
Revision 2
Docket 50-602
The University of Texas at Austin
TRIGA Reactor
JUNE 2012

Table of Contents

TECHNICAL SPECIFICATIONS APPENDIX: BASES

1.0 DEFINITIONS

1.1 Operator

An individual licensed under 10CFR55 to manipulate the controls of the facility.

1.2 Senior Operator

An individual licensed under 10CFR55 to manipulate the controls of a facility and to direct the licensed activities of licensed operators.

1.3 Instrumentation Channel

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

1.3.1 Channel Test

Channel test is the introduction of a signal into the channel for verification that it is operable.

1.3.2 Channel Check

Channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

1.3.3 Channel Calibration

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

1.4 Confinement

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

1.5 Experiment

Any operation, component, or target (excluding devices such as detectors, foils, etc.), which is designed to investigate non-routine reactor characteristics or which is intended for irradiation within the pool, on or in a beam tube or irradiation facility and which is not rigidly secured to a core or shield structure so as to be part of their design.

1.5.1 Experiment, Moveable

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

1.5.2 Experiment, Secured

A secured experiment is any experiment, experiment facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining force 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 conditions.

1.5.3 Experimental Facilities

Experimental facilities shall mean rotary specimen rack, pneumatic transfer tube, central thimble, beam tubes and irradiation facilities in the core or in the pool.

1.6 Fuel Element, Standard

A fuel element is a single TRIGA element of standard type. Fuel is U-ZrH (<20% enriched uranium) clad in stainless steel. Hydrogen to zirconium ratio is nominal 1.6.

1.7 Fuel Element, Instrumented

An instrumented fuel element is a special fuel element fabricated for temperature measurement. The element shall have at least one thermocouple embedded in the fuel near the axial and radial midpoints.

1.8 Mode; Manual, Auto, Pulse, Square Wave

Each mode of operation shall mean operation of the reactor with the mode selection switches in the manual, auto, pulse or square wave position.

1.9 Steady-State

Steady-State mode operation shall mean any operation of the reactor with the mode selection switches in the manual, auto or square wave mode. The pulse mode switch will define pulse operation.

1.10 Operable

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

1.11 Operating

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

1.12 Protective Action

Protective action is the initiation of a signal or the operation of equipment within the reactor safety system in response to a variable or condition of the reactor facility having reached a specified limit.

1.12.1 Instrument Channel Level

At the protective instrument channel level, protective action is the generation and transmission of a trip signal indicating that a reactor variable has reached the specified limit.

1.12.2 Instrument System Level

At the protective instrument system level, protective action is the generation and transmission of the command signal for the safety shutdown equipment to operate.

1.12.3 Reactor Safety System Level

At the reactor safety system level, protective action is the operation of sufficient equipment to immediately shut down the reactor.

1.13 Reactivity, Excess

Excess reactivity is that amount of reactivity that would exist if all the control rods were moved to the maximum reactive condition from the point where the reactor is exactly critical.

1.14 Reactivity Limits

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

1.15 Reactor Core, Standard

A standard core is an arrangement of standard TRIGA fuel in the reactor grid plate and may include installed experiments.

1.16 Reactor Core, Operational

An operational core is a standard core for which the core parameters of excess reactivity, shutdown margin, fuel temperature, power calibration, and reactivity worths of control rods and experiments have been determined to satisfy the requirements set forth in the Technical Specifications.

1.17 Reactor Operating

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

1.18 Reactor Safety Systems

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

1.19 Reactor Secure

The reactor is secure when:

1.19.1 Subcritical:

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

1.19.2 The following conditions exist:

a. The minimum number of neutron absorbing control rods are fully inserted in shutdown position, as required by technical specifications.

b. The console key switch is in the off position and the key is removed from the lock.

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

d. No experiments are being moved or serviced that have, on movement, a reactivity worth equal to or exceeding one dollar.

1.20 Reactor Shutdown

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

1.21 Reference Core Condition

The condition of the core when:

a. Fuel is at ambient temperature (cold), and

b. The reactivity worth of xenon is not greater than the reactivity worth of xenon following shutdown (from steady state, full power operation) for 72 hours.

1.22 Research Reactor

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

1.23 Rod, Control

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

1.23.1 Shim Rod

A shim rod is a control rod with an electric motor drive that does not perform a special function such as automatic control or pulse control. The shim rod shall have scram capability.

1.23.2 Regulating Rod

A regulating rod is a control rod used to maintain an intended power level and may be varied manually or by a servo-controller. The regulating rod shall have scram capability.

1.23.3 Standard Rod

The regulating and shim rods are standard control rods.

1.23.4 Transient Rod

A transient rod is a control rod used to initiate a power pulse that is operated by a motor drive and/or air pressure. The transient rod shall have scram capability.

1.24 Safety Limits

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

1.25 Scram Time

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

1.26 Shall, Should and May

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

1.27 Shutdown Margin

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

1.28 Shutdown, Unscheduled

An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions which could adversely affect safe operation, not including shutdowns which occur during testing or check-out operations.

1.29 Value, Measured

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

1.30 Value, True

The true value is the actual value of a parameter.

1.31 Surveillance Activities

Surveillance activities (except those specifically required for safety when the reactor is shutdown), may be deferred during reactor shutdown, however, they must be completed prior to reactor startup unless reactor operation is necessary for performance of the activity. Surveillance activities scheduled to occur during an operating cycle which cannot be performed with the reactor operating may be deferred to the end of the cycle.

1.32 Surveillance Intervals

Maximum intervals are to provide operational flexibility and not to reduce frequency. Established frequencies shall be maintained over the long term. Allowable surveillance intervals shall not exceed the following:

1.32.1 5 years (interval not to exceed 6 years).
1.32.2 2 years (interval not to exceed 2-1/2 years).
1.32.3 Annual (interval not to exceed 15 months).
1.32.4 Semiannual (interval not to exceed 7-1/2 months).
1.32.5 Quarterly (interval not to exceed 4 months).
1.32.6 Monthly (interval not to exceed 6 weeks.
1.32.7 Weekly (interval not to exceed 10 days).
1.32.8 Daily (must be done during the calendar day).

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limit

Specification(s)

The maximum temperature in a standard TRIGA fuel element shall not exceed 1150°C for fuel element clad temperatures less than 500°C and shall not exceed 950°C for fuel element clad temperatures greater than 500°C. Temperatures apply to any condition of operation.

2.2 Limiting Safety System Settings

2.2.1 Fuel Temperature

Specification(s)

The limiting safety system setting shall be 550°C as measured in an instrumented fuel element. One instrumented element shall be located in the B or C ring of the reactor core configuration.

2.2.2 Power Level (Manual, Auto, Square Wave)

Specification(s)

The maximum operating power level for the operation of the reactor shall be 1100 kilowatts in the manual, auto and square wave modes.

2.2.3 Reactivity Insertion (Pulse)

Specification(s)

The maximum transient reactivity insertion for the pulse operation of the reactor shall be 2.2% $\Delta k/k$ (\$3.14) in the pulse mode.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Core Parameters

3.1.1 Excess Reactivity

Specification(s)

The maximum available core reactivity (excess reactivity) does not exceed 4.9% $\Delta k/k$ ($7.00) for reference core conditions with no negative reactivity worth credited to moveable experiments.

3.1.2 Shutdown Margin

Specification(s)

The magnitude of shutdown margin in reference core conditions, and with no reactivity from negative worth experiments, shall be greater than 0.2% $\Delta k/k$ ($0.29)

3.1.3 Transient insertions

Specification(s)

a. Total worth of the transient rod shall be limited to 4.00 dollars (2.8% $\Delta k/k$), and

b. While operating in the pulse mode, the total time the pulse rod is withdrawn after initiation of the pulse shall not exceed 15 seconds.

3.1.4 Fuel Elements

Specification(s)

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

a. In measuring the elongation, the length exceeds the original length by 2.54 mm (1/10 inch).

b. In measuring the transverse bend, the bend exceeds the original bend by 1.5875 mm (1/16 inch).

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

3.2 Reactor Control and Safety System

3.2.1 Control Assemblies

Specification(s)

The reactor shall not be operated unless the control rods are operable, and

a. Control rods shall not be operable if damage is apparent to the rod or drive assemblies.

b. The scram time measured from the instant a simulated signal reaches the value of a limiting safety system setting to the instant that the slowest scrammable control rod reaches its fully inserted position shall not exceed 1 second.

c. Maximum reactivity insertion rate of a standard control rod shall be less than 0.2% $\Delta k/k$ (\$0.29) per second.

3.2.2 Reactor Control System

Specification(s)

The reactor shall not be operable unless the minimum safety interlocks are operable. The following control system safety interlocks shall be operable:

	Control Rod Drive Interlock Function	Number Operable	Control Rod	Effective Mode*			
				M	A	S	P
a	Startup Withdrawal - prevent rod up movement if startup signal is less than 2 counts per second	3	Standard rods	X	X		
		1	Transient rod	X	X	X	X
b	Simultaneous Withdrawal - prevent rod up movement for two or more rods	3	Standard rods	X			
		2	Shim rods		X		
		1	Transient rod	X	X		
C	Non pulse condition – prevent air actuation if rod drive is not down	1	Transient rod	X	X		
d	Pulse Withdrawal – prevent withdrawal of non pulse rods	3	Standard rods			X	X
e	Transient Withdrawal - Prevent air actuation if linear power is more than 1 kilowatt	1	Transient rod			X	X

*Modes are: (M) Manual, (A) Auto, (S) Square Wave, and (P) Pulse

3.2.3 Reactor Safety System

Specification(s)

The reactor shall not be operable unless the minimum safety channels are operable. The following control rod scram safety channels shall be operable.

Safety System Function		Number Operable	Safety Channel	Effective Mode* M, A, S	P
a	Scram at ≤ 550°C	1	Fuel Temperature	X	X
b	Scram at ≤ 1.1 MW	2	Power Level	X	
	Scram at ≤ 2000 MW	1	Pulse Power		X
c	Scram on loss	2	High Voltage	X	X
d	Scram on loss	1	Magnet Current	X	X
e	Scram on demand	1	Manual Scram Console Button	X	X
f	Scram on loss of timer reset	2	Watchdog Trip Microprocessor scan rate	X	X

*Modes are: (M) Manual, (A) Auto, (S) Square Wave, and (P) Pulse

3.2.4 Reactor Instrument System

Specification(s)

A minimum configuration of measuring channels shall be operable. The following minimum reactor parameter measuring channels shall be operable:

Instrument System Function		Number Operable	Safety Channel	Effective Mode* M, A, S	P
a.	Temperature	1	Fuel Temperature	X	X
b.	Power	2	Power Level	X	
	Pulse	1	Pulse Power		X
c.	Pulse	1	Pulse Energy		X

*Modes are: (M) Manual, (A) Auto, (S) Square Wave, and (P) Pulse

3.3 Operational Support Systems

3.3.1 Water Coolant Systems

Specification(s)

Corrective action shall be taken or the reactor shut down if any of the following (a.-d.) reactor coolant conditions are observed:

a. The bulk pool water temperature exceeds 48 °C.

b. The water depth is less than 6.5 meters measured from the pool bottom to the pool water surface.

c. The water conductivity exceeds 5.0 µmho/cm for the average value during measurement periods of one month.

d. The pressure difference during heat exchanger operation is less than 7 kPa (1 psig) measured between the chilled water outlet pressure and the pool water inlet pressure to the heat exchanger.

e. Pool water data from periodic measurements shall exist for water pH and radioactivity. Radioactivity measurements shall include total alpha-beta activity and gamma ray spectrum analysis.

3.3.2 Air Confinement Systems

Specification(s)

Corrective action shall be taken or the reactor shut down if any of the following air confinement conditions do not exist:

a. Equipment shall be operable to isolate the reactor area by closure of room ventilation supply and exhaust dampers, and shutdown of system supply and exhaust fans.

b. The reactor room ventilation system shall have an automatic signal to isolate the area if air particulate radioactivity exceeds preset values.

c. An auxiliary air purge system to exhaust air from experiment systems shall have a high efficiency particulate filter.

d. Room ventilation shall require two air changes per hour or exhaust of pool areas by the auxiliary air purge system.

3.3.3 Radiation Monitoring Systems

Specification(s)

Radiation monitoring while the reactor is operating requires the following minimum conditions :

a. A continuous air monitor (particulate) shall be operable with readout and audible alarm. The monitor shall sample reactor room air within 5 meters of the pool at the pool access level. Alarm set point shall be equal to or less than a measurement concentration of 2×10^{-9} μCi/cm^3 with a two hour particulate accumulation.

The particulate continuous air monitor shall be operating when the reactor is operating. A set point of the monitor will initiate the isolation signal for the air ventilation system.

The particulate air monitor may be out of service for a period of 1 week provided the filter is evaluated daily, and a signal from the argon-41 continuous air monitor is available to provide information for manual shutdown of the HVAC.

b. A continuous air monitor (argon-41) shall be operable with readout and audible alarm. The monitor shall sample exhaust stack air from the auxiliary air purge system when the system is operating. Alarm set point shall be equal to or less than a measurement concentration of 2×10^{-5} μCi/cm^3 for a daily release.

The argon-41 continuous air monitor shall be operating when the auxiliary air purge system is operating. The average annual concentration limit for release at the stack shall be 2×10^{-6} μCi/cm^3.

If the argon-41 monitor is not operable, operating the reactor with the auxiliary air purge system shall be limited to a period of ten days.

c. Area radiation monitors (gamma) shall be operable with readout and audible alarm. Alarm set point shall be a measurement value equal to or less than 100 mR/hr.

One area radiation monitor shall be operating at the pool level when the reactor is operating. Two additional area radiation monitors shall be operating at other reactor areas when the reactor is operating.

3.4 Limitations on Experiments

3.4.1 Reactivity

Specification(s)

The reactor shall not be operated unless the following conditions governing experiment reactivity exist:

a. A moveable experiment shall have a reactivity worth less than 1.00 dollar.

b. The reactivity worth of any single secured experiment shall be less than 2.50 dollars.

c. The total of absolute reactivity worths of reactor core experiments shall not exceed 3.00 dollars, including the potential reactivity which might result from malfunction, flooding, voiding, or removal and insertion of the experiments.

3.4.2 Materials

Specification(s)

The reactor shall not be operated unless the following conditions governing experiment materials exist:

a. Experiments containing materials corrosive to reactor components, compounds highly reactive with water, potentially explosive materials, and liquid fissionable materials shall be doubly encapsulated. Guidance for classification of materials shall use the Material Safety Data Sheet (MSDS) or a similar source of information involving hazardous chemicals.

b. If a capsule fails and releases material which could damage the reactor fuel or structure by corrosion or other means, removal and physical inspection shall be performed to determine the consequences and need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by the Director, or his designated alternate, and determined to be satisfactory before operation of the reactor is resumed.

c. Explosive materials in quantities greater than 25 milligrams shall not be irradiated in the reactor or experimental facilities. Explosive materials in quantities less than 25 milligrams may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than the design pressure of the container.

d. Each fueled experiment shall be controlled such that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 750 millicuries and the maximum strontium inventory is no greater than 2.5 millicuries

e. Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under (1) normal operating conditions of the experiment or reactor, (2) credible accident conditions in the reactor, (3) possible accident conditions in the experiment shall be limited in activity such that if 100% of the gaseous activity or radioactive aerosols produced escaped to the reactor room or the atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the derived air concentration limits (DAC) of 10CFR20 Appendix B, and averaged effluent from the reactor room to the environment would not exceed effluent limits of Appendix B.

f. In calculations pursuant to e. above, the following assumptions shall be used:

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

(2) If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99% efficiency for 0.25 micron particles, at least 10% of these vapors can escape; and

(3) For materials whose boiling point is above 55°C and where vapors formed by boiling this material can escape only through an undisturbed column of water above the core, at least 10% of these vapors can escape.

4.0 SURVEILLANCE REQUIREMENTS

4.1 Reactor Core Parameters

4.1.1 Excess Reactivity

Specification(s)

Excess reactivity shall be determined annually or after significant control rod or reactor core changes.

4.1.2 Shutdown Margin

Specification(s)

Shutdown margin shall be determined annually or after significant control rod or reactor core changes.

4.1.3 Transient Insertion

Specification(s)

Transient rod function shall be evaluated annually or after significant control rod or reactor core changes. The transient rod drive and associated air supply shall be inspected annually, and the drive cylinder shall be cleaned and lubricated annually.

A comparison of pulse data with previous measurements at annual intervals or each time the interval to the previous measurement exceeds the annual interval.

4.1.4 Fuel Elements

Specification(s)

The reactor fuel elements shall be examined for physical damage by a visual inspection, including a check of the dimensional measurements, made at biennial intervals

4.2 Reactor Control and Safety System

4.2.1 Control Assemblies

Specification(s)

Control rod worths shall be determined annually or after significant control rod or reactor core changes, and

a. Each control rod shall be inspected at biennial intervals by visual observation.

b. The scram time of a scrammable control rod shall be measured annually or after maintenance to the control rod or drive.

c. The reactivity insertion rate of a standard control rod shall be measured annually or after maintenance to the control rod or drive.

4.2.2 Reactor Control System

Specification(s)

The minimum safety interlocks shall be tested at semiannual intervals or after repair or modification.

4.2.3 Reactor Safety System

Specification(s)

The minimum safety channels shall be calibrated annually or after repair or modifications. A channel test shall be done prior to each days operation, after repair or modifications, or prior to each extended period of operation.

4.2.4 Reactor Instrument System

Specification(s)

The minimum configuration of instrument channels shall be calibrated annually or after repair or modification. Calibration of the power measuring channels shall be by the calorimetric method. A channel check and channel test of the fuel temperature instrument channels and power level instrument channels shall be made prior to each days operation or prior to each extended period of operation.

4.3 Operational Support Systems

4.3.1 Water Coolant Systems

Specification(s)

The following measurements shall monitor the reactor coolant conditions:

a. The pool temperature channel shall have a channel calibration annually, channel check monthly and will be monitored during reactor operation.

b. The pool water depth channel shall have a channel calibration annually, channel check monthly and will be monitored during reactor operation.

c. The water conductivity channel shall have a channel calibration annually and pool water conductivity will be measured weekly.

d. The pressure difference channel shall have a channel test prior to each days operation, after repair or modifications, or prior to each extended period of operation of the heat exchanger and will be monitored during operation.

e. Measure pool water pH with low ion test paper or equivalent quarterly. Sample pool water radioactivity quarterly for total alpha-beta activity. Analyze pool water sample by gamma spectroscopy annually for isotope identification.

4.3.2 Air Confinement Systems

Specification(s)

The following actions shall demonstrate the air confinement conditions:

a. Annual examination of door seals and isolation dampers.

b. Monthly functional tests of air confinement isolation.

c. Monthly check of the auxiliary air purge system valve alignments for experimental areas.

d. Daily check of ventilation system alignment for proper exhaust conditions prior to reactor operation.

4.3.3 Radiation Monitoring Systems

Specification(s)

The following conditions shall apply to radiation monitoring systems:

a. Calibrate particulate air monitor at semiannual intervals and check operability weekly.

b. Calibrate argon-41 air monitor at biennial intervals and check operability monthly.

c. Calibrate area radiation monitors at semiannual intervals and check operability weekly prior to reactor operation.

4.4 Limitations on Experiments

4.4.1 Reactivity

Specification(s)

The reactivity of an experiment shall be measured before an experiment is considered functional.

4.4.2 Materials

Specification(s)

Any surveillance conditions or special requirements shall be specified as a part of the experiment approval.

5.0 DESIGN FEATURES

5.1 Site and Facility Description

5.1.1 Location

Specification(s)

a. The site location is in the northeast corner of The University of Texas at Austin J.J. Pickle Research Campus.

b. The TRIGA reactor is installed in room 1.104 of the Nuclear Engineering Teaching Laboratory.

c. The reactor core is assembled in an above ground shield and pool structure with horizontal and vertical access to the core.

d. Licensed areas of the facility for NRC-licensed materials shall consist of the entire facility designated as the Nuclear Engineering Teaching Laboratory.

5.1.2 Confinement

Specification(s)

a. The reactor room shall be designed to restrict leakage and will have a minimum enclosed air volume of 4120 cubic meters.

b. Ventilation system should provide two air changes per hour and shall isolate air in the reactor area upon detection of a limit signal related to the radiation level.

c. An air purge system should exhaust experiment air cavities and shall be filtered by high efficiency particulate absorption filters.

d. All exhaust air from the reactor area enclosure shall be ejected vertically upward at a point above the facility roof level.

5.1.3 Safety Related Systems

Specifications

Any modifications to the air confinement or ventilation system, the reactor shield, the pool or its penetrations, the pool coolant system, the core and its associated support structure, the rod drive mechanisms or the reactor safety system shall be made and tested in accordance with the specifications to which the systems were originally designed and fabricated. Alternate specifications may be approved by the Reactor Oversight Committee. A system shall not be considered operable until after it is tested successfully.

5.2 Reactor Coolant System

5.2.1 Natural Convection

Specification(s)

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

5.2.2 Siphon Protection

Specification(s)

Pool water level shall be protected by holes for siphon breaks in pool water system pipe lines.

5.3 Reactor Core and Fuel

5.3.1 Fuel Elements

Specification(s)

The standard TRIGA fuel element at fabrication shall have the following characteristics:

a. Uranium content: 8.5 Wt% uranium enriched to a nominal 19.7% Uranium-235.

b. Zirconium hydride atom ratio: nominal 1.6 hydrogen to zirconium, ZrH_x.

c. Cladding: 304 stainless steel, nominal .020 inches thick.

5.3.2 Control Rods

Specification(s)

The shim, regulating, and transient control rods shall have scram capability, and

a. Include stainless steel or aluminum clad and may be followed by air or aluminum, or for a standard rod may be followed by fuel with stainless steel clad.

b. Contain borated graphite, B_4C powder, or boron and its compounds in solid form as a poison.

c. The transient rod shall have a mechanical limit. An adjustable limit will allow a variation of reactivity insertions.

d. Two shim rods, one regulating rod and the transient rod are the minimum control rods.

5.3.3 Configuration

Specification(s)

The reactor shall be an arrangement of core single grid positions occupied by fuel elements, control rods, and graphite elements. Single element positions may be occupied by voids, water or experiment facilities. Special multielement positions or single element positions may be occupied by approved experiments.

5.4 Reactor Fuel Element Storage

Specification(s)

a. All fuel elements shall be stored in a geometric array where the effective multiplication is less than 0.9 for all conditions of moderation.

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 design values.

5.5 Reactor Pool Irradiator

Specification(s)

The irradiator assembly shall be an experiment facility.

a. A 10,000 Curie gamma irradiator may be located in the reactor pool. The irradiator isotope shall be cobalt-60.

b. Location of the assembly shall be at a depth of at least 4.5 meters and at a distance of at least 0.5 meters from the reactor core structure.

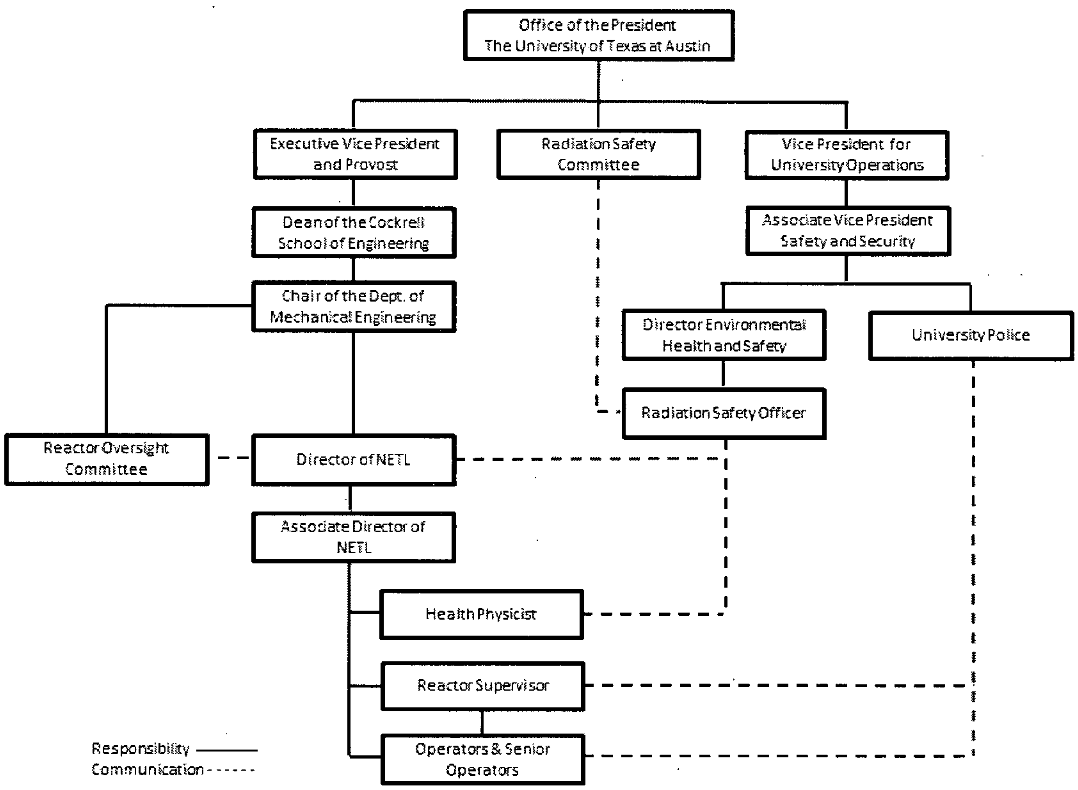
c. Pool water sample requirements shall monitor pool water for source leakage. At a pool water activity of $2.5x10^{-5}$ $\mu Ci/cm^3$ the gamma irradiator components shall be tested to locate and remove any leaking source.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

6.1.1 Structure

The facility shall be under the control of the Director, Associate Director or a delegated Senior Operator. The management for operation of the facility shall consist of the organizational structure as follows:



6.1.2 Responsibility

Line Management

Facility line management is responsible for the policies and operation of the facility, and responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to the operating license and technical specifications. Personnel changes in the following line management positions are subject to special reporting requirements (specified in Section 6.6.2 a):

- University of Texas President
- Executive Vice President and Provost

- Dean of the Cockrell School of Engineering
- Chair of the Mechanical Engineering Department
- Director of the NETL
- Associate Director of the NETL

Reactor operation and maintenance activities are conducted by senior operators and operators under direction of the Reactor Supervisor. Facility health physics and radiological control functions are integrated into operations by the NETL Health Physicist. The Reactor Supervisor and Health Physicist report to the NETL Associate Director, who reports to the NETL Director.

A member of Facility Management (Director or Associate Director) or a Senior Operator reviews and approves all experiments and experimental procedures prior to their use in the reactor.

The Director is responsible to the Chair of the Department of Mechanical Engineering and the Dean of the Cockrell School of Engineering for safe operation and maintenance of the reactor and its associated equipment. These responsibilities may be delegated to the Associate Director during the Director's absence from the Facility. A Reactor Oversight Committee chartered and appointed by the Cockrell School of Engineering monitors facility operations, provides oversight for the facility, reports to the Dean, and makes recommendations to the Facility Director as appropriate.

The Dean of Cockrell School of Engineering reports to the Executive Vice President and Provost, who reports to the President of The University of Texas at Austin. The President of the University of Texas at Austin charters and appoints a Radiation safety Committee to provide policies and direction. Radiation safety polices and directions are implemented by the University Radiation Safety Officer (RSO). The RSO is an ex-officio member of the Reactor Oversight Committee (described in Section 6.2), and has independent lines of communication with the NETL Director and Health Physicist.

Environmental Health and Safety Management Oversight

The University of Texas provides resources for radiological protection and facility security independent of facility line management. The reporting structure for the UT Police Department, Radiation safety Officer, and committees with facility review and audit are included on the organization chart.

The Vice President for University Operations (reporting to the President) is responsible for environment, safety and health management and other areas less germane to reactor facility safety. The Vice President for University Operations is supported by four (4) associates, including the Associate Vice President for Campus Safety and Security.

The Campus Safety and Security organization is composed of directorates for Environmental Health and Safety, UT Police Department, Parking and Transportation, Fire Prevention Services, and Emergency Preparedness. The Environmental Health and Safety directorate is responsible for coordination and management of the various EH&S functions (e.g., the UT Radiation Safety Officer). Parking and Transportation Services manages Pickle Research Campus site access during normal business hours. The UT Police Department provides local law enforcement support, including notification of the facility Reactor Supervisor and control room (operators and senior operators) of facility security alarm conditions and armed response.

6.1.3 Staffing

a. The minimum staffing when the reactor is not secure shall be:

   i. An operator or senior operator in the control room.

   ii. A second person in the facility area that can perform prescribed written instructions. Unexpected absence for two hours shall require immediate action to obtain an alternate person.

   iii. A senior operator readily available. The available senior operator should be within thirty minutes of the facility and reachable by telephone.

b. Events requiring the presence at the facility of a senior operator shall be:

   i. All fuel element or control rod relocations within the reactor core region.

   ii. Relocation of any experiment with a reactivity worth of greater than one dollar.

   iii. Recovery from an unscheduled shutdown or significant power reduction.

   iv. Initial startup and approach to power.

c. Operating procedures shall specify when a senior operator is required to directly supervise licensed activities, including (but not limited to):

   i. The first startup (excluding control rod manipulations with the reactor shutdown for maintenance or testing) and ascension to power following:

      (a). Movement of fuel or control rods,

(b). Relocation of experiments with a reactivity worth of greater than one dollar,

(c). Modifications to the reactor control or safety system,

(d). Non-routine corrective maintenance to the reactor control and safety system, and/or

(e). Recovery from an unscheduled shutdown.

ii. Changes to the configuration of the reactor core:

(a). Fuel element or control rod relocations within the reactor core region, and/or

(b). Relocation (i.e., within the core region) of any experiment with a reactivity worth of greater than one dollar.

d. A list of reactor facility personnel by name and telephone number shall be available to the operator in the control room. The list shall include:

i. Management personnel.

ii. Radiation safety personnel.

iii. Other operations personnel.

6.1.4 Selection and Training of Personnel

The selection, training and requalification of operators shall meet or exceed the requirements of American National Standard for Selection and Training of Personnel for Research Reactors ANSI/ANS -15.4. Qualification and requalification of licensed operators shall be subject to an approved NRC (Nuclear Regulatory Commission) program.

6.2 <u>Review and Audit</u>

6.2.1 Composition and Qualifications

A Reactor Oversight Committee shall consist of at least three (3) members appointed by

the Dean of the Cockrell School of Engineering that are knowledgeable in fields which relate to nuclear safety. The university radiation safety officer shall be a member or an ex-officio member. The committee will perform the functions of review and audit or designate a knowledgeable person for audit functions.

6.2.2 Charter and Rules

The operations of the Reactor Oversight Committee shall be in accordance with an established charter, including provisions for:

a. Meeting frequency (at least twice each year, with no more than 9 months between meetings).

b. Quorums (not less than one-half the membership where the NETL Director, Associate Director and Reactor Supervisor do not hold a majority).

c. Dissemination, review, and approval of minutes.

d. Use of subgroups.

6.2.3 Review Function

The review function shall include facility operations related to reactor and radiological safety. The following items shall be reviewed:

a. Determination in accordance with 10CFR50.59 that proposed changes in equipment, systems, tests, experiments, or procedures do not require a license amendment.

b. All new procedures and major revisions thereto, and proposed changes in reactor facility equipment or systems having safety significance.

c. All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity.

d. Changes in technical specifications or license.

e. Violations of technical specifications or license.

f. Operating abnormalities or violations of procedures having safety significance.

g. Other reportable occurrences.

h. Audit reports.

6.2.4 Audit Function

The audit function shall be a selected examination of operating records, logs, or other documents. An audit will be by a person not directly responsible for the records and may include discussions with cognizant personnel or observation of operations. The following items shall be audited and a report made within 3 months to the Director and Reactor Oversight Committee:

a. Conformance of facility operations with license and technical specifications at least once each calendar year.

b. Results of actions to correct deficiencies that may occur in reactor facility equipment, structures, systems, or methods of operation that affect safety at least once per calendar year.

c. Function of the retraining and requalification program for operators at least once every other calendar year.

d. The reactor facility emergency plan and physical security plan, and implementing procedures at least once every other year.

6.3 Operating Procedures

Written operating procedures shall be prepared, reviewed and approved by the Director or a supervisory Senior Operator and the Reactor Oversight Committee prior to initiation of the following activities:

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

b. Fuel loading, unloading and movement in the reactor.

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

d. Surveillance calibrations and tests required by the technical specifications or those that could have an effect on reactor safety.

e. Administrative controls for operation, maintenance: and the conduct of experiments or irradiations that could have an effect on reactor safety.

f. Personnel radiation protection, consistent with applicable regulations or guidelines,

and shall include a management commitment and programs to maintain exposures and releases as low as reasonably achievable.

g. Implementation of required plans such as the emergency plan or physical security plan.

Substantive changes to the above procedures shall be made effective after approval by the Director or a supervisory Senior Operator and the Reactor Oversight Committee. Minor modifications to the original procedures which do not change the original intent may be made by a senior operator but the modifications must be approved by the Director or a supervisory Senior Operator. Temporary deviations from the procedures may be made by a senior operator in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported to the Director or a supervisory Senior Operator.

6.4 Experiment Review and Approval

All new experiments or classes of experiments shall be approved by the Director or a Supervisory Senior Operator and the Reactor Oversight Committee.

a. Approved experiments shall be carried out in accordance with established and approved procedures.

b. Substantive changes to previously approved experiments shall require the same review as a new experiment.

c. Minor changes to an experiment that do not significantly alter the experiment may be made by a supervisory senior operator..

6.5 Required Actions

6.5.1 Action to be taken in case of a Safety Limit Violation

In the event of a safety limit violation, the following section shall be taken:

a. The reactor shall be shut down and reactor operation shall not be resumed until a report of the violation is prepared and authorization to restart by the Nuclear Regulatory Commission (NRC) is issued.

b. The safety limit violation shall be promptly reported to the Director of the facility or a designated alternate.

c. The safety limit violation shall be subsequently reported to the NRC.

d. A safety limit violation report shall be prepared and submitted to the Reactor Oversight Committee. The report shall describe:

(1) Applicable circumstances leading to the violation including, when known the cause and contributing factors,

(2) Effect of the violation on reactor facility components, systems, or structures and on the health and safety of the public,

(3) Corrective actions taken to prevent recurrence.

6.5.2 Action to be taken in the Event of an Occurrence that is Reportable.

In the event of a reportable occurrence, the following action shall be taken:

a. Reactor conditions shall be returned to normal or the reactor shutdown. If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed unless authorized by the Director or his designated alternate.

b. Occurrence shall be reported to the Director or his designated alternate and to the Nuclear Regulatory Commission as required.

c. Occurrence shall be reviewed by the Reactor Oversight Committee at the next regularly scheduled meeting.

6.6 Reports

All written reports shall be sent within the prescribed interval to the NRC, Washington D.C. 20555, Attention: Document Control Desk.

6.6.1 Operating Reports

Routine annual reports covering the activities of the reactor facility during the previous calendar year shall be submitted within three months following the end of each prescribed year. Each annual operating report shall include the following information:

a. A narrative summary of reactor operating experience including the energy produced by the reactor or the hours the reactor was critical, or both.

b. The unscheduled shutdowns including, where applicable, corrective action taken to preclude recurrence.

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

d. Tabulation of major changes in the reactor facility and procedures, and tabulation of new tests or experiments, or both, that are significantly different from those performed previously, including conclusions that no unreviewed safety questions were involved.

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

f. A summary of exposures received by facility Personnel and visitors where such exposures are greater than 25% of that allowed or recommended.

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

6.6.2 30-Day Special Reports

A written report shall be submitted within 30 days to the NRC of:

a. Permanent changes in the facility organization for positions including:

- University of Texas President
- Executive Vice President and Provost
- Dean of the Cockrell School of Engineering
- Chair of the Mechanical Engineering Department
- Director of the NETL
- Associate Director of the NETL

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

6.6.3 Immediate Notification & Follow-up Reports

A report to NRC Operations Center by telephone not later than the following working day and confirmed in writing by telegraph or similar conveyance to be followed by a written report within 14 days that describes the circumstances of the event of any of

the following:

a. Violation of fuel element temperature safety limit.

b. Release of radioactivity above allowable limits.

c. Operation with actual safety-system settings for required systems less conservative than the limiting safety system settings specified in the technical specifications.

d. Operation in violation of limiting conditions for operation established in technical specifications unless prompt remedial action is taken.

e. Reactor safety system component malfunctions which render or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns.

   However, where components or systems are provided in addition to those required by the technical specifications, the failure of components or systems is not considered reportable provided that the minimum number of components or systems specified or required performs their intended reactor safety function.

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

g. Abnormal and significant degradation in reactor fuel, or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks) where applicable which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both.

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

6.7 RECORDS

The records may be in the form of logs, data sheets , or other suitable forms. The required information may be contained in single or multiple records, or a combination thereof.

6.7.1 Records to be Retained for the Lifetime of the Reactor Facility:

(Note: Applicable annual reports, if they contain all of the required information, may be used as records in this section.)

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

b. Offsite environmental monitoring surveys required by technical specifications.

c. Events that impact or effect decommissioning of the facility

d. Radiation exposure for all personnel monitored.

e. Updated drawings of the reactor facility.

6.7.2 Records to be Retained for a Period of at Least Five Years or for the Life of the Component Involved Whichever is Shorter:

a. Normal reactor facility operation (supporting documents such as checklists, log sheets, etc. shall be maintained for a period of at least one year).

b. Principal maintenance operations.

c. Reportable occurrences.

d. Surveillance activities required by technical specifications.

e. Reactor facility radiation and contamination surveys where required by applicable regulations.

f. Experiments performed with the reactor.

g. Fuel inventories, receipts, and shipments.

h. Approved changes in operating procedures.

i. Records of meeting and audit reports of the review and audit group.

6.7.3 Records to be Retained for at Least One Licensing Cycle:

Retraining and requalifications of licensed operations personnel. Records of the most recent complete cycle shall be maintained at all times the individual is employed.

A.1.0 DOCKET 50-602 INFORMATION

The Technical Specifications of this document depend on the analysis and conclusions of the Safety Analysis Report. Descriptive information important to each specification is presented in the form of the applicability, objective and bases. This information defines the conditions effective for each technical specification for the Docket 50-602 facility.

A.1.1 Applicability

The applicability defines the conditions, parameters, or equipment to which the specification applies.

A.1.2 Objective

The objective defines the goals of the specification in terms of limits, frequency, or other controllable item.

A.1.3 Bases

The bases presents information important to the specification, including such things as justification, logical constraints and development methodology.

A.2.0 SAFETY LIMITS & LIMITING SAFETY SYSTEM SETTINGS APPLICABILITY, OBJECTIVES AND BASES

A.2.1 Safety Limit

Applicability

This specification applies to the temperature of the reactor fuel in a standard TRIGA fuel element.

Objective

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

Bases

The important parameter for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification since it can be measured directly. A loss in the integrity of the fuel element cladding could arise from a build-up of excessive pressure between the fuel-moderator and the cladding if the fuel temperature exceeds the safety limit. The pressure is caused by the presence of air, fission product gases, and hydrogen from the dissociation of the hydrogen and zirconium in the fuel-moderator. Hydrogen pressure is the most significant component. The magnitude of this pressure is determined by the fuel-moderator temperature and the ratio of hydrogen to zirconium in the alloy.

The safety limit for the standard TRIGA fuel is based on calculations and experimental evidence. The results indicate that the stress in the cladding due to hydrogen pressure from the dissociation of zirconium hydride will remain below the ultimate stress provided that the temperature of the fuel does not exceed 1150°C and the fuel cladding does not exceed 500°C. For conditions that might cause the clad temperatures to exceed 500°C the safety limit of the fuel should be set at 950°C.

A.2.2 Limiting Safety System Setting

A.2.2.1 Fuel Temperature

Applicability

This specification applies to the protective action for the reactor fuel element temperature.

Objective

The objective is to prevent the fuel element temperature safety limit from being reached.

Bases

For non pulse operation of the reactor, the limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated preventing the safety limit from being exceeded. A setting of 550°C provides a safety margin at the point of measurement of at least 400°C for standard TRIGA fuel elements in any condition of operation. A part of the safety margin is used to account for the difference between the true and measured temperatures resulting from the actual location of the thermocouple. If the thermocouple element is located in the hottest position in the core, the difference between the true and measured temperatures will be only a few degrees since the thermocouple junction is near the center and the mid-plane of the fuel element. For pulse operation of the reactor, the same limiting safety system setting will apply. However, the temperature channel will have no effect on limiting the peak powers generated because of its relatively long time constant (seconds) as compared with the width of the pulse (milliseconds). In this mode, however, the temperature trip will act to limit the energy release after the pulse if the transient rod should not reinsert and the fuel temperature continues to increase.

A.2.2.2 Power Level (Manual, Auto, Square Wave)

Applicability

This specification applies to the protective action for the reactor during non pulse operation.

Objective

The objective is to prevent the fuel element temperature safety limit from being reached.

Bases

Thermal and hydraulic calculations indicate that standard TRIGA fuel elements may be safely operated at power levels in excess of 1500 kilowatts with natural convection cooling. Conservative estimates indicate that a departure from nucleate boiling ratio of approximately two will occur at about 1900 kilowatts. A limiting setting for the power 1evel measurement at 1.1 megawatts assures sufficient margin for safety to allow for calibration errors. The power calibration goal is a measurement accuracy of 5% although an error of 10% may be representative of some measurements.

A.2.2.3 Reactivity Insertion (Pulse)

Applicability

This specification applies to the reactivity insertion for the reactor during pulse operation.

Objective

The objective is to prevent the fuel element temperature safety limit from being reached.

Bases

Calculations indicate that standard TRIGA fuel elements may be safely operated at transient conditions in excess of 2.2% Δk/k with ambient cooling conditions. Conservative estimates indicate that a substantial safety margin exists for the rise of peak fuel temperature with reactivity insertions as large as 2.8% Δk/k.

A.3.0 OBJECTIVES AND BASES FOR THE LIMITING CONDITIONS FOR OPERATION

A.3.1 Reactor Core Parameters

Limiting core reactivity specifications are predicated on the reference core conditions. The reference core condition requires radioactive xenon to be negligible, i.e. with reactivity effects from xenon at or less than a detectable level. Evaluation that xenon reactivity is negligible is based on (1) assessment of the minimum detectable level for reactivity changes, (2) calculation concentration following reactor operation, and (3) evaluation of the reactivity attributed to xenon.

(1) Detectable Level for Reactivity Changes Based on Changes in Rod Position

Reactivity changes are evaluated based on differential rod worth positions using a calibration of position versus reactivity worth associated with the position of each rod. The control rod move nominally 15 in. from full in to full out, while the position indication changes 960 units. A unit of movement is therefore 1/960th of nominally 15 in., or 0.0156 units of movement per in. The June 29, 2011 calibration shows a change in the linear portion of the reactivity curve from 400 to 500 units of \$1.2603 to \$1.70322, or \$0.00443 per unit. Because of mechanical linkages to the control rod, the control rod drive, and the position indicator, repeatability for movements less than 10 units is challenging. Therefore \$0.04-\$0.05 is a practical limit for evaluating reactivity changes using movement of a single rod, a lower limit of detection for reactivity changes based on movement of a single rod.

(2) Time Dependent Xenon Concentration

Nuclear Reactor Engineering, 3rd Ed. (Glasstone and Sessonke), section 5.72 indicates xenon levels following shutdown after operation at maximum steady state power level varies as:

$$X(t_S) = \frac{\lambda_I}{\lambda_X - \lambda_I} \cdot I_0 \cdot \left(e^{-\lambda_I \cdot t_s} - e^{-\lambda_X \cdot t_s}\right) + X_0 \cdot e^{-\lambda_X \cdot t_s}$$

Where:

$X(t_S)$ is the time dependent xenon concentration (atoms/cm^3)

X_0 is the initial, steady state xenon concentration (atoms/cm^3)

t_S is the time after shutdown (s)

λ_I is the decay constant of iodine 135 (s^{-1})

λ_X is the decay constant of xenon 135 (s^{-1})

I_0 is the initial concentration of iodine 135 and tellurium 135 (atoms/cm^3)

The initial steady state level of xenon concentration is:

$$X_0 = \frac{(\gamma_I + \gamma_X)\cdot\Sigma_f\cdot\phi}{\lambda_X + \sigma_X\cdot\phi}$$

Where

γ_I is the cumulative thermal U^{235} fission product yield of iodine 135 and tellurium 135

γ_X is the cumulative thermal U^{235} fission product yield of xenon 135

σ_X is the microscopic cross section for thermal neutron absorption of xenon 135

Σ_f is the microscopic cross section for fission in the reactor

And the initial steady state level of iodine concentration is:

$$I_0 = \frac{\gamma_I\cdot\Sigma_f\cdot\phi}{\lambda_I}$$

Therefore, the ratio of equilibrium xenon concentration to time variant xenon following shutdown is:

$$\frac{X(t_S)}{X_0} = \frac{\gamma_I\cdot(\lambda_X + \sigma_X\cdot\phi)}{(\lambda_X - \lambda_I)\cdot(\gamma_I + \gamma_X)}\cdot\left(e^{-\lambda_I\cdot t_s} - e^{-\lambda_X\cdot t_s}\right) + e^{-\lambda_X\cdot t_s}$$

Cross section and yield values used in the equation are taken from Evaluated Nuclear Data Files (ENDF/B-VII.1). Half-lives used to calculate decay constants are taken from the Chart of the Nuclides as provided on the National Nuclear Data Center web site. Relevant properties for ^{135}Xe and ^{135}I are tabulated below (the short-lived tellurium precursor yield of 0.032162 is included in the iodine yield).

	$t_{½}$ (s)	λ	σ_a (b)	Γ
Iodine 135	2.37E4	2.924E-5	8.00E1	0.061436
Xenon 135	3.29E4	2.106E-5	2.67E6	0.0045

The buildup and decay of xenon 135 based on the isotope characteristics is displayed in Fig. A.3.1.1-1 and A.3.1.1-2, with the first graph on a traditional linear display and the second showing the ratio on a log scale. Calculations for the lower three flux values are not separable on the scale used.

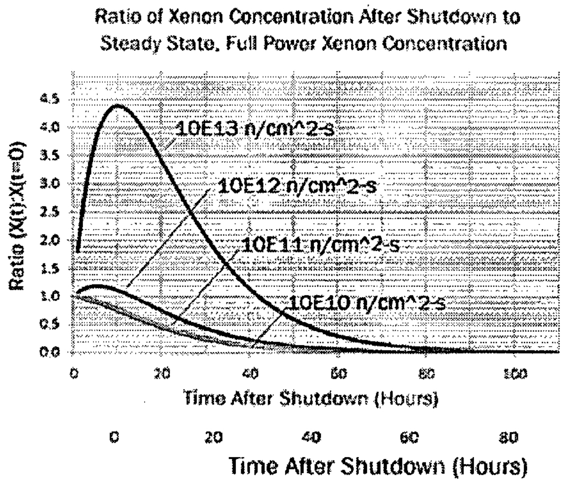


Figure A.3.1.1-1, Xenon Concentration Following Shutdown (Linear Scale)

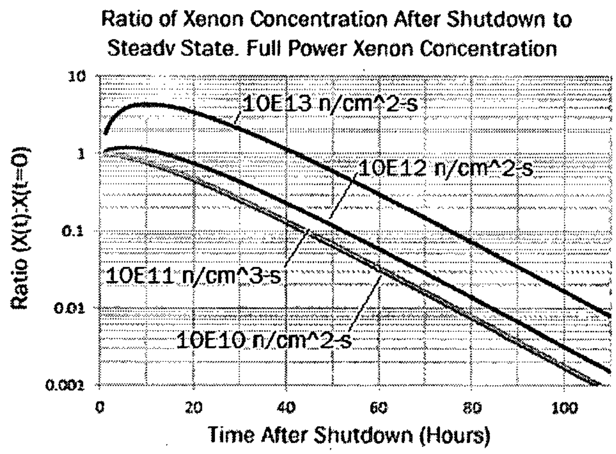


Figure A.3.1.1-2, Xenon Concentration Following Shutdown (Log-Linear Scale)

Historical calculations (GA-4361, Calculated Fluxes and Cross Sections for TRIGA Reactors, G. B. West, 1963) show core-average full power neutron flux approximately 10^{13} for 1 MW TRIGA reactors. Therefore at 72 hours after reactor shutdown following equilibrium full power operations, xenon concentration is 13% of the steady state value and 10% at 75 hours.

(3) Time dependent Xenon Reactivity

Using the definition of reactivity, a change in reactivity is calculated:

$$\Delta\rho = -\rho_1 - \rho_1 \equiv \Delta\rho_{1,2} = \frac{k_1 - 1}{k_1} - \frac{k_2 - 1}{k_2} = \frac{1}{k_1}\cdot\left(1 - \frac{k_1}{k_2}\right)$$

Using $k = k_\infty \cdot P_{nl}$:

$$\Delta\rho_{1,2} = \frac{1}{k_1} \cdot \left(1 - \frac{k_{1,\infty} \cdot P_{nl,1}}{k_{2,\infty} \cdot P_{nl,2}} \right)$$

A KENOVI model of the UT TRIGA was generated based on the initial core configuration. Xenon inventory was calculated using the T-6 depletion sequence of SCALE 6.1. Three days of operation at 1.1 MW was modeled to build in steady state xenon inventory, followed by 4 days of shutdown with 85 data points. For operation at 600 K, the average mean free path was determined to have a maximum value of 0.93117 cm, a minimum value of 0.925077 cm, an average value of 0.928111 and a standard deviation of 0.00157. It can be inferred therefore that the materials buckling does not change appreciably; and since the reactor geometry was not changed in the modeling, the probability of non leakage (P_{nl}) which is a function of materials and geometric buckling, the non leakage factor does not change appreciably with xenon. Therefore the change in reactivity simplifies to:

$$\Delta\rho_{1,2} = \frac{1}{k_1} \cdot \left(1 - \frac{k_{1,\infty}}{k_{2,\infty}} \right)$$

Of the 4 factors in the formula for k_∞, thermal utilization factor is the only one that is appreciably affected by poison. Over reasonable periods of time, the fuel remains relatively constant so compared to a condition without xenon:

$$\Delta\rho_{1,2} = \frac{1}{k_1} \cdot \left(1 - \frac{f_1}{f_2} \right) = \frac{1}{k_1} \cdot \left(1 - \frac{N_{Xe,1} \cdot \sigma_{Xe,1} + \Sigma_{a,f}}{\Sigma_{a,f}} \right)$$

$$\Delta\rho_{1,2} = \frac{1}{k_1} \cdot \left(- \frac{N_{Xe,1} \cdot \sigma_{Xe,1}}{\Sigma_{a,f}} \right)$$

The reactivity from xenon is therefore approximately proportional to the xenon concentration, and the ratio of concentrations previously calculated provides a ratio of reactivity from xenon following shutdown.

The KENO calculations provide k_{eff}, fission product and fuel isotope concentrations, and the fraction of absorptions at each interval (for this data, 1 hour time steps). Therefore the reactivity change can be calculated directly. Fig. A.3.1.1-3 shows (1) the reactivity change attributed to fission product positions (as discrete points), and (2) the concentration of xenon 135 (in comparison to steady state, full power operation, as a line). Data acquired during a 12 hour operation in 2012 is included on the graph as a black line, showing good agreement

between the model and the system. The reactivity contribution from xenon is clearly negligible from 72 hours following shutdown.

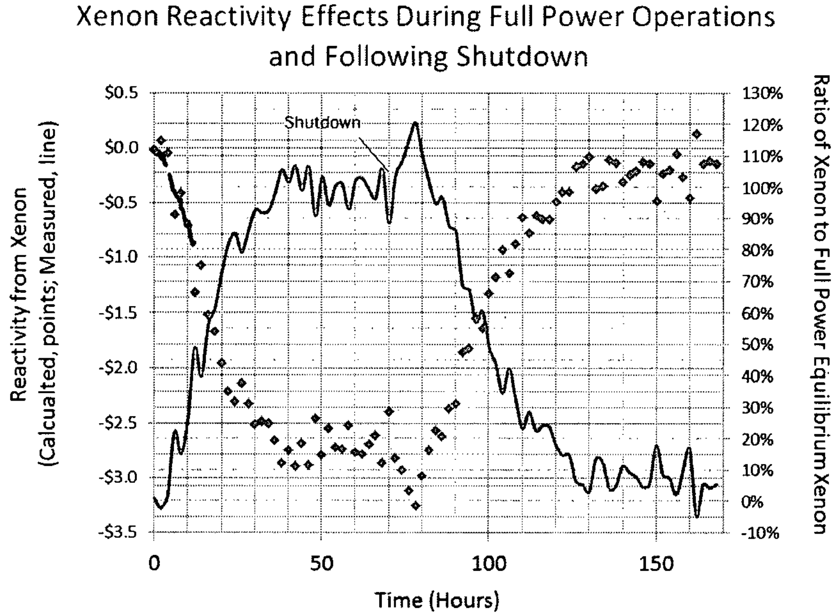


Figure A.3.1.1-3, Xenon reactivity Effects During Full Power Operations & Following Shutdown

Data for neutron absorption has less "noise" and correlates directly to concentration. A second graph correlates the fraction of neutrons absorbed by xenon to the reactivity deficit at each time step.

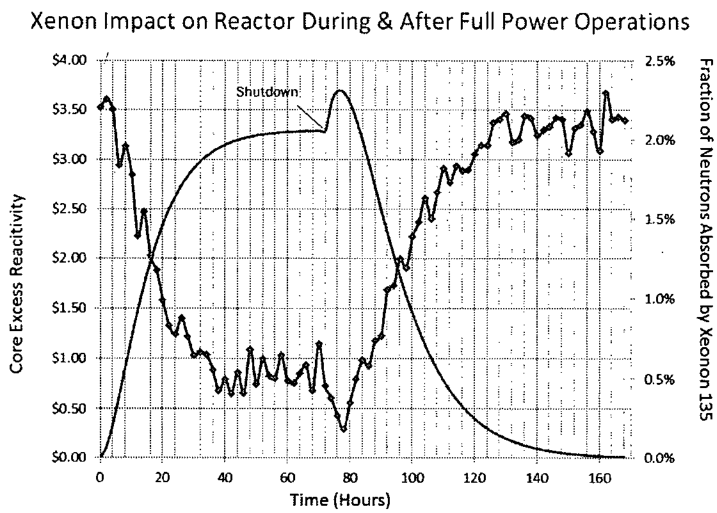


Figure A.3.1.1-4, Xenon Impact on Reactor During & After Full Power Operations

The ratio of the fraction of neutrons absorbed in xenon during steady state full power operation to the fraction at intervals after shutdown was developed. These ratios were applied to the reactivity existing during steady state full power operation. As shown below, the reactivity form xenon is reduced to less than $0.05 at aboOut 72 hours after shutdown.

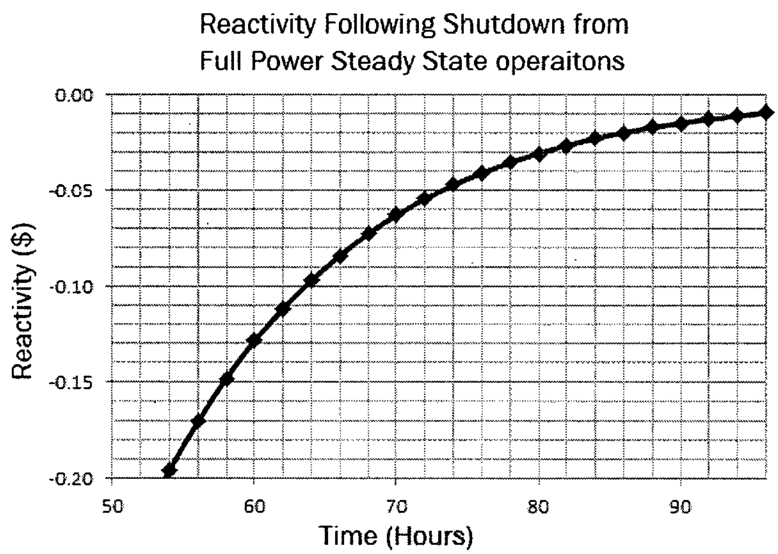


Figure A.3.1.1-3, Reactivity Following Shutdown from Full Power Steady State Operations

The operating schedule of UT reactor does not ever result in full power, steady state xenon concentrations; as shown in Fig. A.3.1.1-3, 12 hours at full power operations produces a fraction of the effect from xenon as compared to equilibrium conditions. Consequently 72 hours is an extremely conservative estimate of time to reach negligible xenon level. The shutdown interval to achieve negligible xenon may be calculated using the actual operating history to support determining reference core condition.

A.3.1.1 Excess Reactivity

Applicability

This specification applies to the reactivity condition of the reactor core in terms of the available excess above the cold xenon free, critical condition.

Objective

The objective is to prevent the fuel element temperature safety limit from being reached by limiting the potential reactivity available in the reactor for any condition of operation.

Bases

Maximum excess core reactivity is sufficient to provide the core rated power, xenon compensation and reactivity for shutdown. Analysis of the reactor core demonstrates that no single component represents sufficient potential reactivity to reach the fuel element temperature safety limit during any condition of operation.

A.3.1.2 Shutdown Margin

Applicability

This specification applies to the reactivity margin by which the reactor core will be considered shutdown when the reactor is not operating.

Objective

The objective is to assure that the reactor can be shut down safely by a margin that is sufficient to compensate for the failure of a control rod or the movement of an experiment.

Bases

The value of the shutdown margin assures that the reactor can be shut down from any operating condition. These conditions include the assumption that no credit is given to negative worth experiments when determining shutdown margin.

A.3.1.3 Transient Insertions

Applicability

This specification applies to the total potential worth of the transient rod and the allowable reactivity insertion for reactor pulse operation.

Objective

The objective is to limit the reactivity available for pulse insertion to a value that will not cause the fuel temperature safety limit to be exceeded.

Bases

Calculations demonstrate that the total insertion of all the transient rod worth will not exceed the fuel temperature safety limit. For a 2.8% Δk/k pulse a safety margin would exist between the fuel element safety limit and the rise of peak fuel temperature above an assumed ambient pool temperature of 50°C. Experiments with pulsed operation of TRIGA reactors by the manufacturer indicate that insertions up to 3.5% Δk/k have not exceeded the fuel temperature safety limit.

Power level following a pulse achieves a maximum (peak) when the reactivity from elevated fuel temperature compensates for the reactivity inserted to initiate the pulse. After fuel temperature feedback regulates power, power level may exceed the steady state power level limit for some interval until the reactivity from temperature reaches equilibrium. Therefore,

the amount of time that the reactivity from the pulse rod is allowed to remain in the core is limited. A preset timer insures that the transient rod will not remain in the pulse position for an extended time after the pulse.

A.3.1.4 Fuel Elements

Applicability

This specification applies to the measurement parameters for the fuel elements.

Objective

The objective is to verify the physical condition of the fuel element cladding.

Bases

The elongation limit has been specified to assure that the cladding material will not be subjected to stresses that could cause a loss of integrity in the fuel containment and to assure adequate coolant flow. The limit of transverse bend has been shown to result in no difficulty in disassembling the reactor core. Analysis of the removal of heat from touching fuel elements shows that there will be no hot spots resulting in damage to the fuel caused by this touching. Experience with TRIGA reactors has shown that fuel element bowing that could result in touching has occurred without deleterious effects.

A.3.2 <u>Reactor Control and Safety System</u>

A.3.2.1 Control Assemblies

Applicability

This specification applies to the function of the control rods.

Objective

The objective is to determine that the control rods arc operable by specification of apparent physical conditions, the scram times for scrammable control rods and the reactivity insertion rates for standard control rods.

Bases

The apparent condition of the control rod assemblies will provide assurance that the rods will continue to perform reliably and as designed. The specification for rod scram time assures that the reactor will shut down promptly when a scram signal is initiated. The specification for rod reactivity insertion rates assures that the reactor will start up at a controllable rate when rods are withdrawn. Analysis has indicated that for the range of transients anticipated for a TRIGA reactor the specified scram time and insertion rate is adequate to assure the safety of the reactor.

A.3.2.2 Reactor Control System

Applicability

These specifications apply to logic of the reactor control system.

Objective

The objective is to determine the minimum control system interlocks operable for operation of the reactor.

Bases

Interlocks are specified to prevent function of the control rod drives unless certain specific conditions exist. Program logic of the digital processors implement the interlock functions.

Two basic interlocks control all rod movements in the manual mode. The interlock to prevent startup of the reactor at power levels less than 2 neutron cps, which corresponds to approximately 4 milliwatts, assures that sufficient neutrons are available for controlled reactor startup. Simultaneous withdrawal of more than one control rod is prevented by an interlock to limit the maximum positive reactivity insertion rate available for steady state operation.

Interlocks applicable to the transient rod determine the proper rod operation during manual mode and pulse mode operation. The non pulse condition interlock determines the allowable position of the rod drive for actuation of the FIRE switch. Actuation of the switch applies the air impulse for removal of the transient rod from the reactor core.

Auto mode applies the same interlock controls as the manual mode to the shim and transient rods. Servo calculations limit reactivity insertions by controlling regulating rod drive speed. One limit, a reactor period of four decades per minute, restricts simultaneous up motion of the regulating rod with any other rod.

Two basic interlocks control rod movements for the pulse mode. The interlock to prevent withdrawal of the motor driven rods in the pulse mode is designed to prevent changing the critical state of the reactor prior to the pulse. A power level interlock controls potential fuel temperature changes by setting a limit of less than 1 kilowatt for initiation at any pulse.

Square wave mode applies the same interlock controls as the pulse mode to all control rods. A pulse transient terminates the mode by changing to auto or manual mode. The change to auto or to manual mode becomes effective when a preset condition (demand power) occurs or a preset time (ten seconds) expires.

A.3.2.3 Reactor Safety System

Applicability

These specifications apply to operation of the reactor safety system.

Objective

The objective is to determine the minimum safety system scrams operable for the operation of the reactor.

Bases

Safety system scram functions consist of three types. These scram types are the limiting safety system settings, operable system conditions, and the manual or program logic scrams. The scrams cause control rod insertion and reactor shutdown.

Scrams for limiting safety system settings consist of signal trip levels that monitor fuel temperature and power level. The trip levels are conservative by a significant margin relative to the fuel element temperature safety limit.

Operation without adequate control and safety system power supplies is prevented by scrams on neutron detector high voltage and control rod magnet current.

Manual action of the scram switch, key switch, or computer actuation of watchdog timers will initiate a protective action of the reactor safety system. Either of two watchdog circuits provide updating timers to terminate operation in the event that key digital processing routines fail, such as a display system. Each watchdog circuit with four resettable timers contains one trip relay and monitors one microcomputer.

A.3.2.4 Reactor Instrument System

Applicability

These specifications apply to measurements of reactor operating parameters.

Objective

The objective is to determine the minimum instrument system channels to be operable for continued operation of the reactor.

Bases

The minimum measuring channels are sufficient to provide signals for automatic safety system operation. Signals from the measuring system provide information to the control and safety system for a protective action. Instruments provide redundancy by measurements of the same parameters and diversification by measurements of different parameters. Diversity and redundancy in safety system protection against fuel temperature limits is provided by a combination of fuel temperature and power level channels as previously noted (A.3.2.3). Two redundant percent power channels monitor the power level limiting safety system. A digital wide range channel may also function as a safety channel but only by diversification as a supplemental channel to an analog linear power channel. Pulse parameters of peak power and energy release are measurements of a single detector chamber. There are, however, two separate peak and energy monitoring circuits.

A.3.3 Operational Support System

A.3.3.1 Water Coolant Systems

Applicability

This specification applies to the operating conditions for the reactor pool and coolant water systems.

Objective

The objective is to assure that adequate conditions are maintained to provide shielding of the reactor radiation, protection against corrosion of the reactor components, cooling of the reactor fuel, and prevent leakage from the primary coolant.

Bases

The specifications for conditions of the pool water coolant system provide controls that are to control the radiation exposures and radioactive releases associated with the reactor fission product inventory.

a. The bulk water temperature constraint assures that sufficient core cooling exists under all anticipated operating conditions and protects the resin of the water purification system from deterioration.

b. A pool water depth of 6.5 meters is sufficient to provide more than 5.25 meters of water above the reactor core so that radiation levels above the reactor pool are at reasonable levels.

c. Average measurements of pool coolant water conductivity of 5.0 µmho/cm assure that water purity is maintained to control the effects of corrosion and activation of coolant water impurities.

d. A pressure difference at the heat exchanger chilled water outlet and the pool water inlet of 7 kPa will be sufficient to prevent loss of pool water from the primary reactor coolant system to the secondary chilling water system in the event of a leak in the heat exchanger.

e. Periodic sampling of pool water pH and radioactivity are supplemental measurements that assist evaluation of the overall conditions of the reactor pool. Protection of aluminum components requires a pH range of 5 to 8.5. Measurements of radioactivity in the pool water provide information to evaluate working hazards for personnel, leakage indications for radioactive sources in the pool, and monitoring for activation of unknown components in the water.

A.3.3.2 Air Confinement Systems

Applicability

This specification applies to the air ventilation conditions in the reactor area during reactor operation.

Objective

The objective is to control the release of air in the reactor area or experimental facilities.

Bases

The specifications for exhaust ventilation and isolation of the reactor bay provide control for radioactive releases for both routine and non routine operating conditions.

a. Air confinement of the reactor bay includes a provision for isolation of the air flow of the ventilation system. Dampers in the room supply air ducts and room return air ducts limit the leakage rate and total release of radioactive airborne materials to a fraction of the available volume.

b. A signal from a particulate air monitor in the vicinity of the reactor pool initiates the automatic isolation of the supply air dampers and return air dampers. The isolation process takes less than one minute and includes the shutdown of supply fan and exhaust fan. An equivalent to one maximum permissible concentration is the set point.

c. Air from experiment areas within the neutron flux regions of the core will ventilate separately from room air by way of a filter bank that includes a high efficiency particulate filter. Space is available to install a charcoal filter for special experiment conditions.

d. Control of concentrations of argon-41 in reactor room air depends on ventilation of the room air at a rate of two air changes per hour or operation of the auxiliary purge air system. Operation and isolation of the purge system is by manual control of damper and fan switches.

A.3.3.3 Radiation Monitoring Systems

Applicability

This specification applies to the radiation monitoring conditions in the reactor area during reactor operation.

Objective

The objective is to monitor the radiation and radioactivity conditions in the reactor area to control exposures or releases.

Bases

The radiation monitors provide information to operating personnel of impending or existing hazards from radiation so that there will be sufficient time to take the necessary steps to control the exposure of personnel and release of radioactivity or evacuate the facility. Alarm

setpoints do not include measurement uncertainty. These setpoints are measured values and not true values.

a. Air particulate radioactivity accumulates on the filter of a continuous monitor that records the radiation levels. An alert and alarm set point including remote readouts at the reactor control console inform the operator of the monitor status and activity levels. An alarm limit at two thousand picocurie/milliliter detects particulate activity concentrations at the occupational values of 10CFR20. The alarm set point exceeds occupational values for any single fission product nuclide in the ranges 84-105 and 129-149. Seventy percent of the particulate isotopes are also detectable at the reference concentrations within two hours. The gaseous argon-41 monitor can provide fission product gas monitoring during repair of the particulate monitor.

b. Air gaseous radioactivity of argon-41 concentrations require monitoring of the levels for effluent release and occupational exposure. The alarm setpoint detects a release concentration that will not exceed ten times either the occupation value at the stack or the reference concentration at the ground. Calculations of a stack release concentration of 1.2 $\mu Ci/cm^3$ indicate that the equivalent ground level concentration is equivalent to $1x10^{-8}$ $\mu Ci/cm^3$. A license limit for the average annual concentration is necessary to fix the amount of allowable release. Periods of inoperable argon-41 monitoring equipment of up to 10 days limit the amount of release without measurement to a fraction of the total annual release.

c. Several area radiation monitors (six) are part of the permanent installation. Some locations are experiment areas in which shield configurations determine the levels of radiation during reactor operation. At the pool access area radiation levels substantial enough to be a high radiation level may occur. Alarm levels at 100 mR/hr will monitor radiation areas if the limit of 2 or 5 mR/hr is not reasonable.

A.3.4 Limitations on Experiments

A.3.4.1 Reactivity

Applicability

This specification applies to the reactivity of experiments located in the reactor core.

Objective

The objective is to control the amount of reactivity associated with experiments to values that will not endanger the reactor safety limit.

Bases

a. The worth of single moveable experiment is limited so that sudden removal movement of the experiment will not cause prompt criticality. Worth of a single unsecured experiment will not cause a reactivity insertion that would exceed the core temperature safety limit.

b. The maximum worth of a single experiment is limited so that the fuel element temperature safety limit will not be exceeded by removal of the experiments. Since experiments of such worth must be secured in place, removal from the reactor operating at full power would result in a relatively slow power increase such that the reactor protective systems would act to prevent excessive power levels from being attained.

c. The maximum worth of all experiments is limited so that removal of the total worth of all experiments will not exceed the fuel element temperature safety limit.

A.3.4.2 Materials

Applicability

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

Objective

The objective is to prevent the release of radioactive material in the event of an experiment failure, either by failure of the experiment or subsequent damage to the reactor components.

Bases

a. Double encapsulation requirements lessen the leakage hazards of some types of experiment materials.

ANSI 15.4 references the "Tables of Chemical Hazard Information" in the Handbook of Laboratory Safety to support material characterization. However, the 5th edition (2000) of the Handbook does not include the table, The Handbook instructs the user to check the Material Safety Data Sheet for chemical compatibility, citing that "experience in many areas of safety has demonstrated that to be effective, systems must be kept as simple to implement as possible." Older editions of the handbook which contain the tables state that "hazard data were compiled from a wide variety of publications, including a privately circulated manual, and several other sources of information." Specific sources cited in different sections of the tables include:

- American Conference of Governmental Industrial Hygienists
- International Union of Pure and Applied Chemistry
- Chemical Abstracts Service, American Chemical Society
- Handbook of Chemistry and Physics
- National Bureau of Standards
- Union Carbide Corporation
- American Industrial Hygiene Association
- Manufacturing Chemists Association
- National Safety Council
- National Fire Protection Association
- Matheson Company
- Merck & Company
- Ansul Company
- Eastman Kodak Company
- American Society for Testing and Materials
- National Cancer Institute
- National Institutes of Health

b. Operation of the reactor with the reactor fuel or structure damaged is prohibited to avoid release of fission products.

c. Encapsulation requirements for explosive materials set a reference condition for the amount of material allowable for any reactor experiment. Damage from the explosive reaction depends on the available energy release and resultant gas creation. Approximate conditions for 25 milligrams of explosive material are the release of 25 calories (104 joules) of energy and 25 milliliters of gas. If a 1 milliliter volume is available for the reaction of an explosive material (density 1.654 gm/cm^3), the energy will represent an instantaneous pressure of 1032 atmospheres and the gas release adds another 25 atmospheres. Stress calculations for a thin wall, cylindrical capsule specify the requirements for the wall thickness and diameter of the encapsulation. The relationship determines the stress limit as one fourth the product of the pressure times the capsule diameter to wall thickness ratio. An aluminum capsule with a 1 milliliter volume requires a ratio that does not exceed 5.2. At a volume of 5 milliliters capsule dimensions with a diameter of 2.6 cm requires a wall thickness of 1 mm. These limiting values are within the constraints of aluminum tubular construction components for experiment facilities and experiments.

d. Fission product inventory limits of 750 millicurie iodine and 2.5 millicurie strontium fix the potential accident release concentrations. These two isotopes represent the radioactive exposure risk to individuals for fission product nuclides with short (iodine) and long (strontium) half-lives. If the isotope iodine-131 represents the total inventory release of 750 millicuries, the facility annual average release, including building wake dilution of the total inventory, will be equivalent to the reference level concentration of

$2x10^{-10}$ μCi/cm^3. In the case of strontium-90 the release is less than 1/5 the reference level concentration of $5x10^{-12}$ μCi/cm^3. Proper shutdown of the ventilation system by manual or automatic operation substantially reduces the effective total release. Any release of the total experiment inventory within the facility, however, in the form of iodine-131 or strontium-90 will exceed the occupational values within the facility for the oral ingestion or air inhalation of the radionuclides. As an extreme case the evacuation times to maintain the average annual concentration are 1 hour for iodine-131 and 1 month for strontium-90.

e. Accidental release of radioactive materials that cause airborne concentrations must meet 10CFR20 average annual limits. Concentration limits apply to occupational values that cause exposure within the facility and reference level concentrations that may exist as a release from the facility. Calculations assume a complete release of the material but also must define release rates and frequencies that are conservative or reasonable estimates of accident conditions.

f. This specification provides guidance for the calculation of conditions in part (e).

A.4.0 OBJECTIVES & BASES FOR THE SURVEILLANCE REQUIREMENTS

A.4.1 Reactor Core Parameters

A.4.1.1 Excess Reactivity

Applicability

This specification applies to the measurement of reactor excess reactivity.

Objective

The objective is to periodically determine the changes in core excess reactivity available for power generation.

Bases

Annual determination of excess reactivity and measurements after reactor core or control rod changes are sufficient to monitor significant changes in the core excess reactivity.

A.4.1.2 Shutdown Margin

Applicability

This specification applies to the measurement of reactor shutdown margin.

Objective

The objective is to periodically determine the core shutdown reactivity available for reactor shutdown.

Bases

Annual determination of shutdown margin and measurements after reactor core or control rod changes are sufficient to monitor significant changes in the core shutdown margin.

A.4.1.3 Transient Insertion

Applicability

This specification applies to surveillance of the transient rod mechanism and to observation of the reactor transient response.

Objective

The objective is to assure the function of the transient rod drive and to compare the reactor pulse insertion parameters.

Bases

Annual inspections of the pulse rod drive system should be sufficient to detect and correct changes in the system that could impair operability. Comparison of pulse parameter data should detect characteristic changes of reactor core transients.

A.4.1.4 Fuel Elements

Applicability

This specification applies to the inspection requirements for the fuel elements.

Objective

The objective is to inspect the physical condition of the fuel element cladding.

Bases

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

A.4.2 Reactor Control and Safety System

A.4.2.1 Control Assemblies

Applicability

This specification applies to the surveillance of the control rods.

Objective

The objective is to inspect the physical condition of the reactor control rods and establish the operable condition of the rod by periodic measurement of the scram tines and insertion rates.

Bases

Annual determination of control rod worths or measurements after significant core changes provide information about changes in reactor total reactivity and individual rod worths. The frequency of inspection for the control rods will provide periodic verification of the condition of the control rod assemblies. Verification will be by measurement of fueled sections and visual observation of absorber sections plus examination of linkages and drives. The specification intervals for scram time and insertion rate assure operable performance of the rods. Deviations that are significant from acceptable standards will be promptly corrected.

A.4.2.2 Reactor Control System

Applicability

This specification applies to the tests of the logic of the reactor control system.

Objective

The objective is to specify intervals for test, check or calibration of the minimum control system interlocks.

Bases

The periodic test of the interlock logic at semiannual intervals provides adequate information that the function of the control system interlocks are functional. Changes to the interlock logic consist of revisions to the microcomputer algorithms (hardware, software or firmware) and repair of input or output circuits including devices that are sensors for the interlocks. Calibrations or checks of the control system logic are not considered applicable functions.

A.4.2.3 Reactor Safety System

Applicability

This specification applies to tests of the function of the reactor safety system.

Objective

The objective is to specify intervals for test, check or calibration of the minimum safety system scrams.

Bases

The periodic calibration at annual intervals provides adequate information that the setpoints of the safety system scrams are functional. Tests of the safety system prior to each planned operation assure that each intended scram function is operable.

A.4.2.4 Reactor Instrument System

Applicability

These specifications apply to calibrations, checks, and tests of reactor measurement channels.

Objective

The objective is to specify intervals for test, check or calibration of the minimum instrument channels.

Bases

Annual calibration of instrument channels are scheduled to allow adjustments for changes in reactor and instrumentation parameters. Checks and tests prior to each system operation verify the function of key channels and systems.

A.4.3 Operational Support Systems

A.4.3.1 Water Coolant Systems

Applicability

This specification applies to surveillance conditions for the reactor pool and coolant water systems.

Objective

The objective is to maintain the reactor coolant conditions within acceptable specifications.

Bases

Conditions for the reactor coolant are monitored by visual observation of measurements or automatic action of sensors. Periodic checks and tests of measurement devices for the reactor coolant system parameters assure that the coolant system will perform its intended function. Measurement frequencies of pool parameters relate to the time periods appropriate to detection of abnormal conditions. Pool temperature, depth, and heat exchanger pressure differences have an immediate effect on system operation. Water conductivity, pH as a supplemental indicator, and pool radioactive concentrations are conditions that develop at rates detectable at monthly to annual intervals.

A.4.3.2 Air Confinement Systems

Applicability

This specification applies to surveillance conditions for the air ventilation in the reactor area.

Objective

The objective is to demonstrate the function of confinement and release of air from the reactor bay.

Bases

Periodic tests and checks of air confinement conditions verify appropriate ventilation functions. Monitoring frequencies verify performance of the confinement system exhaust daily by an alignment check that includes observation of negative pressures. Tests of the isolation feature at monthly intervals assure the acceptable operation of the system.

A.4.3.3 Radiation Monitoring Systems

Applicability

This specification applies to the surveillance conditions of the radiation monitoring channels.

Objective

The objective is to assure the radiation monitors are functional.

Bases

Periodic calibrations and frequent checks are specified to maintain reliable performance of the radiation monitoring instruments. Calibration and check frequencies follow the general recommendations of guidance documents.

A.4.4 Limitations on Experiments

A.4.4.1 Reactivity

Applicability

This specification applies to surveillance of the reactivity of experiments.

Objective

The objective is to assure the reactivity of an experiment does not exceed the allowable specification.

Bases

The measured reactivity or determination that the reactivity is not significant will provide data that configuration of the experiment or experiments is allowable.

A.4.4.2 Materials

Applicability

This specification applies to the surveillance requirements for materials inserted into the reactor.

Objective

The objective is to prevent the introduction of materials that could damage the reactor or its components.

Bases

A careful evaluation of all experiments is performed to classify the experiment as an approved experiment.

A.5.0 OBJECTIVES & BASES FOR DESIGN FEATURES

A.5.1 Site and Facility Descriptions

A.5.1.1 Location

Applicability

This specification applies to the TRIGA reactor site location and specific facility design features.

Objective

The objective is to specify those features related to the Safety Analysis evaluation.

Bases

a. The TRIGA facility site is located in an area controlled by The University of Texas at Austin. (Safety Analysis Report, 2.0)

b. The room enclosing the reactor has been designed with characteristics related to the safe operation of the facility. (Safety Analysis Report, 7.2.2)

c. The shield and pool structure have been designed for radiation levels of less than 1 rem/hr at locations that are not access ports to the reactor structure. (Safety Analysis Report, 7.2.1)

d. Identification of licensed areas assures that proper controls are established for the safety of the public and for the security of special nuclear materials. (Safety Analysis Report, 9.1.4, 9.2.1, 10.3)

A.5.1.2 Confinement

Applicability

This specification applies to the boundary for control of air in the area of the reactor.

Objective

The objective is to assure that provisions are made to control or restrict the amount of release of radioactivity into the environment.

Bases

a. Calculations of the concentrations of released radionuclides within the reactor area depend on the available enclosed air volume to limit the concentrations to acceptable levels. (Safety Analysis Report, 7.4)

b. Control of the reactor area air exchange is by fan motors and isolation dampers for the supply and exhaust air which are controlled by a logic signal from a radiation sensor to provide automatic air confinement. ((Safety Analysis Report, 7.2.2)

c. Emergency air ventilation is filtered to control the release of particulates and a pressure difference relative to the external ambient pressure is intended to prevent leakage of air without filtration. (Safety Analysis Report, 7.2.2)

d. Exhaust air during reactor operation is released at an elevated level for dispersion and is designed to provide a relative pressure difference to the external ambient pressure. (Safety Analysis Report, 7.2.2)

A.5.1.3 Safety Related Systems

Applicability

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

Objective

The objective is to assure the proper function of any system related to reactor safety.

Bases

This specification relates to changes in reactor systems which could affect the safety of the reactor operation. Changes or substitutions to these systems that meet or exceed the original design specifications are assumed to meet the presently accepted operating criteria. Questions that may include an unreviewed safety question are referred to the Reactor Oversight Committee. (10CFR50.59)

A.5.2 Reactor Coolant System

Applicability

This specification applies to the reactor coolant system composed of deionized water.

Objective

The objective is to assure that adequate water is available for cooling and shielding during reactor operation.

Bases

a. This specification is based on thermal and hydraulic calculations which show that a standard 85 clement TRIGA core can operate in a safe manner at power levels up to 1,900 kW with natural convection flow of the coolant water and a departure from nucleate boiling ratio of 2.0 (Safety Analysis Report 4.1, 5.1)

b. Siphon breaks set the subsequent pool water level for loss of coolant without an associated water return caused by inadvertent pumping or accidental siphon of water from the pool. (Safety Analysis Report, 5.2.1)

A.5.3 Reactor Core and Fuel

A.5.3.1 Fuel Elements

Applicability

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

Objective

The objective is to assure that the fuel elements are of such a design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Bases

The design basis of the standard TRIGA core demonstrates that 1.5 megawatt steady or 8400 MW peak pulse power presents a conservative limitation with respect to safety limits for the maximum temperature generated in the fuel. The fuel temperatures are not expected to exceed 550°C during any condition of normal operation. (Safety Analysis Report, 1.0)

A.5.3.2 Control Rods

Applicability

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

Objective

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

Bases

The poison requirements for the control rods are satisfied by using neutron absorbing borated graphite, B4C powder, or boron and its compounds. These materials must be contained in a suitable clad material, such as aluminum or stainless steel, to insure mechanical stability during movement and to isolate the poison from the pool water environment. Scram capabilities are provided for rapid insertion of the control rods which is the primary safety feature of the reactor. The transient control rod is designed for a reactor pulse. (Safety Analysis Report 4.2, 4.4.8)

A minimum configuration of control rods consist of two shim rods, a regulating rod and the transient rod. (Safety Analysis Report 4.2, 4.4.8)

The configuration of rods is necessary for the reactor to be operable. If the appropriate adjustments to the core reactivity are made, removal of one or more -of the control rods will facilitate the necessary inspection and repair activities. Definitions for shutdown and subcritical require the reactor core to meet the subcritical constraint if any rod is out of the core end the reactor is to be shutdown. (ANSI-15.1-2007)

A.5.3.3 Configuration

Applicability

This specification applies to the configuration of fuel elements, control rods, experiments and other reactor grid plate components.

Objective

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

Bases

Standard TRIGA cores have been in use for years and their characteristics are well documented.

A.5.4 Reactor Fuel Element Storage

Applicability

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

Objective

The objective is to assure that fuel storage will not achieve criticality and will not exceed design temperatures.

Bases

The limits imposed by these specifications are considered sufficient to provide conservative fuel storage and assure safe storage. (ANSI-15.1-2007)

A.5.5 Gamma Pool Irradiator

Applicability

This specification applies to the gamma irradiator experiment facility in the reactor pool.

Objective

The objective is to assure that the use of the irradiator does not cause any threat to the reactor or safety question.

Bases

Location of the irradiator is at a distance from the reactor sufficient to avoid interference with reactor operation. Depth of the pool water for adequate shielding of the irradiator is also a constraint of the location. (Safety Analysis Report 8.2.2)

A.6.0 Administrative Controls

Note: The bases for administrative controls contains only objective of the specification and the applicable bases.

A.6.1 Organization

A.6.1.1 Structure

The objective is to specify the organization for management and operation of the reactor.

Bases

The basis for Technical Specification 6.6.1 is the Safety Analysis Report, 10.1.1.1. A reorganization at The University of Texas in 2000 established the Vice President for University Operations (reporting to the President) responsible for environment, safety and health management and other areas less germane to reactor facility safety. The structure of the University Operations group (6/19/2010) is provided at:

http://www.utexas.edu/operations/about/uo_orgchart.pdf

In 2001 the position of Associate Director of the Nuclear Engineering Teaching Laboratory was added by Amendment 4 to the Technical Specifications.

ANSI/ANS-15.1 describes the position of the facility organization as Levels; corresponding organizational titles applicable to The University of Texas is provided below.

LEVEL	Responsibilities	Position	UT Title
1	Individual responsible for the reactor facility's licenses or charter	Unit or Organization Head	President, The University of Texas at Austin Executive Vice President and Provost Cockrell School of Engineering Dean
2	Individual responsible for reactor operation	Facility Director or Administrator	Department of Mechanical Engineering Chair NETL[1] Director NETL[1] Associate Director
3	Individual responsible for day-to-day operation or shift	Senior Reactor Operator or Supervisory SRO	Reactor Supervisor
4	Operating staff	SRO, Supervisory SRO, RO, Trainee	Reactor Operator Senior Operator

NOTE [1]: NETL is the Nuclear Engineering Teaching laboratory

A.6.1.2 Responsibility

Objective

The objective is to identify responsibilities for safe operation of the facility with the chain of command specified in 6.1.1.

Bases

The basis for Technical Specification 6.6.2 is the Safety Analysis Report, 10.1.1.2-4, Safety Analysis Report, 10.1.1.2-4

Amendment 4 to the Technical Specifications, and ANSI/ANS-15.1-2007, the Development of Technical Specifications for Research Reactors, Section 6.1.2.

A.6.1.3 Staffing

Objective

The objectives are to:

a. Specify the minimum staffing required when the reactor is not secured

b. Specify when the presence of a senior operator is required at the facility

c. Specify when direct supervision of a senior operator is required

d. Require a list of reactor facility personnel be readily available in the control room

Bases

The basis for 6.1.3 is the Safety Analysis Report 10.1.3.1 and ANSI/ANS-15.1-2007, the Development of Technical Specifications for Research Reactors, Section 6.1.3.

Safety Analysis Report, 10.1.3.1 (Staffing) The UT Safety Analysis Report (10.1.3.1) states:

> *"Movement of fuel or control rods and relocation of experiments with greater than one dollar reactivity worth will require the presence of a license certified senior operator. Other activities, such as initial startup, recovery from unscheduled shutdowns and modifications to instrument systems, control systems, safety systems, radiation measurement equipment or engineered safety features, will require concurrence and documentation by a license certified senior operator."*

UT has augmented the requirements to provide direct supervision by a senior operator for other activities conducted under unusual conditions, including supervision of the initial startup following activities with the potential to affect operating characteristics of the reactor control and safety systems and recovery from non-routine shutdowns as well as in-core activities.

The term "unplanned or unscheduled significant power reduction" in ANSI/ANS-15.1-2007, the Development of Technical Specifications for Research Reactors, Section 6.1 is not defined or used in the approved Safety Analysis Report or the applicable Safety Evaluation report (NURGE-1135 and Supplement). Power reductions at the UT facility are typically based on variable experiment program demands and not subject to programmed schedules in the sense of how the term "unplanned power changes" is used by the NRC (e.g., NRC Inspection Manual 0313, Industry Trends Program). Actions taken in accordance with approved Technical Specifications may involve reductions in power (for example, corrective action associated with specification 3.3.1 d). Since the term does not describe an identifiable condition for the UT facility and is not in the SAR or SER, it is not included in the specification.

"Control rod relocation within the reactor core region" refers to moving a control rod from one grid plate location to a different grid plate location.

"Relocation of any experiment with a reactivity worth of greater than $1.00" means movements within the core region.

A.6.1.4 Selection and Training of Personnel

Objective

The objective is to identify the standard for training, selection, and qualification of operations personnel.

Bases

The basis for section 6.1.4 is Safety Analysis Report 10.1.2, and Safety Analysis Report 10.2, and ANSI/ANS-15.1-2007, the Development of Technical Specifications for Research Reactors, Section 6.1.4.

A.6.2 Review and Audit

A.6.2.1 Composition and Qualifications

Objective

The objective is to establish a method for independent review and audit to advise management of safety aspects for facility operations.

Bases

The basis for section 6.2.1 is Safety Analysis Report 10.1.1.5 and ANSI/ANS-15.1-2007, the Development of Technical Specifications for Research Reactors, Section 6.2.

The original committee title (Nuclear Reactor Committee) is replaced by reactor Oversight Committee in order that the acronym provided by committee abbreviation does not create confusion.

The Reactor Oversight Committee is charted by and reports to the Dean of the Cockerel School of Engineering, identified as Level 1 personnel in A.6.1.1.

A.6.2.2 Charter and Rules

Objective

The objective is to specify meeting frequency, quorum requirements, use of subgroups, and management of minutes

Bases

ANSI/ANS-15.1-2007, the Development of Technical Specifications for Research Reactors, Section 6.2.1. The meeting frequency specified in section 6.2.2 exceeds the recommended frequency in ANSI-15.1. The NETL Director and NETL Associate Director are identified as Level 2 facility director or administrator respectively in A.6.1.1. Therefore the quorum requires that voting membership not have a majority of members comprised of the NETL Director, NETL Associate Director, and Reactor Supervisor.

A.6.2.3 Review Function

Objective

The objective is to identify items to be reviewed.

Bases

ANSI/ANS-15.1-2007, the Development of Technical Specifications for Research Reactors, Section 6.2.2.

A.6.2.4 Audit Function

Objective

The objective is to identify items to be audited and methods for auditing.

Bases

The basis for section 6.2.4 is Safety Analysis Report 10.6.4.1 and ANSI/ANS-15.1-2007, the Development of Technical Specifications for Research Reactors, Section 6.2.3. Specific organizational titles at the University of Texas that correspond to the ANSI Level terminology as identified in A.6.1.1 are provided and referred to where appropriate.

A.6.3 Operating Procedures

Objective

The objective is to identify required procedures and the procedure review and approval process.

Bases

The basis for section 6.2.3 is Safety Analysis Report 10.1.3.2, Safety Analysis Report 10.3.5 (radiological safety), Safety Analysis Report 10.5, (emergency and security procedures), and ANSI/ANS-15.1-2007, the Development of Technical Specifications for Research Reactors, Section 6.2.4. Specific organizational titles at the University of Texas that correspond to the ANSI Level terminology as identified in A.6.1.1 are provided and referred to where appropriate.

A.6.4 Experiment Review and Approval

Objective

The objective is to specify experiment (administrative) review and approval requirements.

Bases

The basis for 6.4 is the Safety Analysis Report 10.1.3.3 and ANSI/ANS-15.1-2007, the Development of Technical Specifications for Research Reactors, Section 6.5. Specific organizational titles at the University of Texas that correspond to the ANSI Level terminology as identified in A.6.1.1 are provided and referred to where appropriate.

A.6.5 Required Actions

A.6.5.1 Actions to be Taken in the Case of Safety Limit Violation

Objective

The objective is to specify actions to be taken in the case of Safety Limit violation.

Bases

The basis for 6.5 is the Safety Analysis Report 10.1.4.2 and ANSI/ANS-15.1-2007, the Development of Technical Specifications for Research Reactors, Section 6.6.1. The ANSI reference to Level 2 or designate is implemented as the NETL Director or designated alternate, as indicated in A.6.1.1.

A.6.5.2 Action to be taken in the Event of an Occurrence that is Reportable

Objective

The objective is to specify actions to be taken in the Event of an Occurrence that is Reportable.

Bases

The basis for 6.5.2 is the Safety Analysis Report 10.4.1.2, Safety Analysis Report 10.1.4.3, Safety Analysis Report 10.1.4.4, and ANSI/ANS-15.1-2007, the Development of Technical Specifications for Research Reactors, Section 6.6.2. . The ANSI reference to Level 2 or designate is

implemented as the NETL Director or designated alternate. The ANSI reference to review group is implemented as the Reactor Oversight Committee.

A.6.6 Reports

A.6.6.1 Operating Reports

Objective

The objective is to specify routine operating report requirements.

Bases

(a). Safety Analysis Report 10.1.4.1, and

(b). ANSI/ANS-15.1-2007, the Development of Technical Specifications for Research Reactors, Section 6.7.1,

A.6.6.2 30-Day Special Reports

Objective

The objective is to identify 30-day reporting requirements.

Bases

The basis for 6.6.2 is the Safety Analysis Report 10.1.4.5 and ANSI/ANS-15.1-2007, the Development of Technical Specifications for Research Reactors, Section 6.7.2. The ANSI reference to Level 1 or 2 personnel is implemented as the President of The University of Texas, Executive Vice President and Provost, Dean of the Cockrell School of Engineering, NETL Director, and NETL Associate Director as indicated in A.6.1.1.

A.6.6.3 Immediate Notification & Follow-up Reports

Objective

The objective is to identify events and situations that require immediate (24) notification and (14 day) reporting.

Bases

The basis for 6.6.3 is:

(a). Safety Analysis Report 10.1.4.2 (with respect to Safety Limit Violation),

(b). Safety Analysis Report 10.1.4.3, (with respect to release of radioactivity)

(c). Safety Analysis Report 10.1.4.4 (with respect to other events), and

(d). ANSI/ANS-15.1-2007, the Development of Technical Specifications for Research Reactors, Section 6.7.2.

   i. Specific organizational titles at the University of Texas that correspond to the ANSI Level terminology as identified in A.6.1.1 are provided and referred to where appropriate.

   ii. The information in the Standard related to conditions requiring immediate notification is provide in a single-list format.

A.6.7 RECORDS

A.6.7.1 Records to be Retained for the Lifetime of the Facility

Objective

The objective is to specify lifetime-record retention requirements.

Bases

The basis for 6.7 is the Safety Analysis Report 10.1.5.1 and ANSI/ANS-15.1-2007, the Development of Technical Specifications for Research Reactors, Section 6.7.1.

A.6.7.2 Records to be Retained for a Period of at Least Five Years or for the Life of the Facility Whichever is Shorter

Objective

The objective is to specify 5-year record-retention requirements.

Bases

The basis for 6.7.2 is the Safety Analysis Report 10.1.5.2 and ANSI/ANS-15.1-2007, the Development of Technical Specifications for Research Reactors, Section 6.7.1.

A.6.7.3 Records to be Retained for at Least One Licensing Cycle

Objective

The objective is to specify licensing-cycle record-retention requirements.

Bases

The basis for 6.7.3 is the Safety Analysis Report 10.1.5.3 and ANSI/ANS-15.1-2007, the Development of Technical Specifications for Research Reactors, Section 6.7.1.