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September 12, 2016

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Reference: U.S. Geological Survey TRIGA Reactor (GSTR), Docket 50-274, License R-113, Request for Additional Information (RAI) dated June 28, 2016

Subject: Responses to RAI questions

Mr. Wertz:

Responses to RAI questions are provided in the enclosed pages. Please contact me if further details, or corrections, are needed.

As per our phone conversations during September 2016, I have attached a "track changes" version of the proposed Technical Specifications as Attachment 1.

Sincerely,

Brycen R. Roy

USGS Reactor Supervisor

I declare under penalty of perjury that the foregoing is true and correct.
Executed on 9/12/2016

Attachment

Copy to:
Vito Nuccio, Reactor Administrator, MS 911
USGS Reactor Operations Committee

A020
NRR

Designated As
Original LT 10/3/16

1. The U.S. Department of Energy (DOE) Interagency Agreement for Enriched Uranium (Attachment 6), provided by your letter dated April 1, 2016, (ADAMS Accession No. ML16110A008), expired on September 31, 2015.

   Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982 states that the NRC will require, as a precondition to issuing or renewing an operating license for a research or test reactor, that the licensee has entered into an agreement with the DOE for the disposal of high-level radioactive wastes and spent nuclear fuel following cessation of operations.

   Provide an active DOE fuel contract suitable for the NRC staff to use to verify that the USGS facility has a means to ensure that the fuel will be removed from the facility after operations cease.

Response: We now have a new agreement from DOE so a new Interagency Fuel Lease Agreement is in effect. Appropriate portions of the agreement are attached as Attachment 2.

2. The response to RAI No. 3, provided in Attachment 1 of your letter dated April 1, 2016, indicated that the analysis for the k-effective (keff) for the fuel storage racks was performed with TRIGA fuel containing 8.5 weight percent (wt%) uranium. However, the GSTR License Renewal Application (LRA) SAR (ADAMS Accession No. ML092120136) indicates that GSTR uses both 8.5 wt% and 12 wt% uranium TRIGA fuel.

   NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors – Format and Content," Section 9.2, "Handling and Storage of Reactor Fuel," and American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.1-2007, "The Development of Technical Specifications for Research Reactors," provide guidance that the fuel and fueled devices shall be stored in a geometric array where keff is no greater than 0.90 for all conditions of moderation and reflection.

   Indicate, by analysis, that the keff of the fuel storage racks for all fuel types used at GSTR is less or equal to 0.90, or justify why no additional information is needed.

Response: We have performed an MCNP analysis on the limiting condition of 19 fresh 12 wt% elements stored in a hexagonal storage rack. That analysis shows a Keff of 0.88, which is less than 0.9. This condition is limiting because it assumes fresh (new) fuel of the highest loading allowable in a storage rack with the highest capacity allowable. The results of the MCNP run are attached as Attachment 3.

3. The proposed TSs do not appear to have a surveillance requirement (SR) to compare GSTR power pulse fuel temperatures and peak power levels, which is provided in the current GSTR TS D.5.

   NUREG-1537, Part 1, Chapter 14, Appendix 14.1, "Standard Format and Content for Technical Specifications for Research Reactors," Section 4.1," Reactor Core Parameters," item (3), "Pulse Limits," provides guidance that the relationship between pulse peak fuel temperature and inserted reactivity should be determined when changes are made to the core.

   Provide a proposed TS SR for pulsing, or justify why no change is needed.

We propose to add a specification 9. to TS 4.1 Reactor Core Parameters that states the following:

9. For each month during which pulsing is performed, the relationship between peak fuel temperature and inserted reactivity shall be determined.

4. The GSTR LRA SAR, Section 11.1.6, "Radiation Monitoring," provides information associated with the Radiation Area Monitors, but does not appear to indicate the setpoints, nor the basis for the setpoints.

   NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 3.7.1, "Monitoring Systems," item (3) "Area Monitors," provides guidance that the alarm and automatic action setpoints should be specified to ensure that personnel exposures and potential doses remain below the limits of Title 10 *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection Against Radiation."

   Provide the alarm or automatic action setpoints and a description of how the setpoints help ensure that personnel exposures and potential doses will remain below 10 CFR Part 20 limits, or justify why no change is needed.

Response: The alarm and automatic action setpoints for the area monitors have been set to be above the levels seen during typical reactor operation with a sufficient margin to prevent spurious alarms. The levels are set conservatively enough to allow time for an operator to leave the area if there were to be an area monitor alarm. The setpoints are also conservative enough to ensure the requirements of 10 CFR 20 are met for radiation workers and members of the public. We believe that putting numeric setpoint values in the Technical Specifications could actually be counter to the ALARA principle since the facility would not be able to adjust the setpoints to more conservative values without going through the very time-consuming and resource intensive process of a formal technical specification amendment. We believe no change is needed.

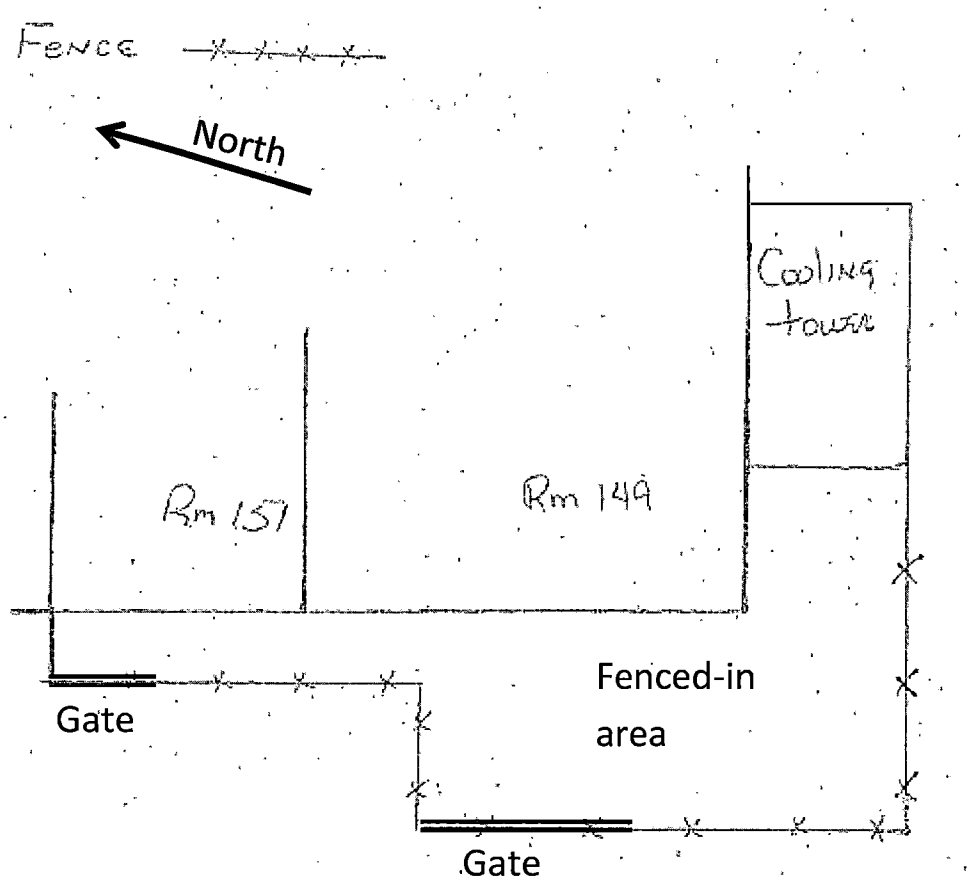
5. Proposed TS 1.2, "Definitions," has the following items identified by the NRC staff:

   a. The proposed definition of "Licensed Area" includes a change that added "the area inside the wrought iron fence and south cooling tower wall that is near the SW corner of Building 15."

      NUREG-1537, Part 1, Section 9.5, "Possession and Use of Byproduct, Source and Special Nuclear Material," provides guidance that "the applicant should clearly state the materials and areas of the facility requested to be authorized by the reactor license."

      Provide a description and justification for the area identified above, or justify why no additional information is needed.

Response: We would like to add the cooling tower and fenced-in area to the defined licensed area because access to this area is controlled by the reactor staff and there are facility Security Plan requirements regarding this area. Licensed materials are not normally located in this area other than temporary placement of packaged Class 7A shipments that are waiting to be loaded in a transportation vehicle. The fence was installed in the early 1990's to provide a "stand-off" distance from the reactor facility to assist in compliance with the 0.1 Rem/yr public dose limitation given in 10 CFR 20.1301(a)(1). The area inside the fence is posted as a "Restricted Area". Radiation surveys are made at the fence line and environmental dosimetry is posted on the fence. The drawing below shows the basic layout of the fenced area, and the location of the two fence gates, which are normally locked shut.



b. The proposed definition of "Experiment," Specification 1, "Secured Experiment" appears to be missing the criteria provided in the guidance in ANSI/ANS-15.1-2007, for Secured Experiment, which states "or by forces that can arise as a result of credible malfunctions."

Revise the proposed TS definition for Secured Experiment to include the guidance in ANSI/ANS-15.1-2007, as provided above, or justify why no change is needed.

Response: We will change the definition for "Secured Experiment" to meet the guidance provided in ANSI/ANS-15.1-2007.

c. The proposed definition of "Experiment," Specification 2, "Movable Experiment" indicates that it is one that is not secured. Proposed TS 3.1.1.1, "Shutdown Margin," uses the term "non-secured experiment." NRC staff is not clear if movable, not secured and non-secured experiments are all the same.

Revise the proposed TSs, as necessary, to use a consistent name associated with the definition of Movable Experiment, or justify why no changes are needed.

Response: We believe that "not secured" is clearly understood to mean that the experiment meets the definition of "Movable Experiment". We will change the definition of "Shutdown Margin" to use the phrase "movable experiment" instead of "non-secured experiment".

d. The proposed definition of "Control Rod" states "neutron absorbing material," whereas TS 5.3.2, "Control Rods," indicate "borated graphite, B_4C powder or boron." The NRC staff seeks consistent definitions to avoid confusion, and finds the information in TS 5.3.2 to better describe neutron absorbing material.

NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 1.2.2, "Format," provides guidance that any information used to support the TSs should be explicitly referenced.

Revise proposed definition in TS 1.2 "Control Rod" to be consistent with TS 5.3.2, or justify why no change is needed.

Response: We will change the definition of "Control Rod" to state "borated graphite, B4C powder or boron" instead of "neutron absorbing material".

e. The proposed definition of "Control Rod," Specifications 2, "Shim Rod" and Specification 3, "Transient Rod" does not indicate the means or mechanism for moving the control rod. In contrast, the Regulating Rod in Specification 1 indicated that is position may be varied manually or by the servo-controller. The NRC staff finds that additional detail would avoid future confusion.

NUREG-1537, Part 1, Chapter 14, Appendix 14:1, Section 1.2.2, "Format," provides guidance that any information used to support the TSs should be explicitly referenced.

Revise proposed definition in TS 1.2, "Control Rod," Specification 2, "Shim Rod," and Specification 3, "Transient Rod," to indicate the means for positioning the control rod, or justify why no change is needed.

Response: We will change the definitions of "Shim Rod" and "Transient Rod" (TS 1.2, Specifications 2 and 3) to indicate that the shim rod and transient rod positioning is done manually.

f. The proposed definition of "Shutdown Reactivity" indicates that all control rods are inserted. "The proposed definition of "Shutdown Margin" uses the term "shutdown reactivity." However, the NRC staff finds that the definition of Shutdown Margin includes the most reactive rod in its most reactive position, which conflicts with the definition of shutdown reactivity, which states that all control rods are inserted.

NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 1.2.2, "Format," provides guidance that any information used to support the TSs should be explicitly referenced.

Revise the proposed definition of Shutdown Reactivity to resolve the discrepancy with its use in the definition of Shutdown Margin, delete the definition of Shutdown Reactivity (not used anywhere else in the TSs), or justify why no change is needed.

Response: We will delete the definition of "Shutdown Reactivity".

6. Proposed TS 3.1.1.1, "Shutdown Margin," provides a value of $0.30 for the shutdown margin (SDM). The NRC staff has reviewed your responses to RAI No. 24.3 provided in your letters dated August 30 and November 16, 2012, and May 17, 2013 (ADAMS Accession Nos. ML12251A231, ML12334A001 and ML13162A662, respectively) and has been unable to substantiate the accuracy needed to determine the control rod worth to support a TS SDM value of $0.30. The NRC staff noted that the RAI response indicated that the maximum error in the SDM calculation was $0.159, or more than half of the proposed SDM limit. Given this error, a SDM of at least $0.46 would be required.

The guidance in NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 3.1, "Reactor Core Parameters," item (2), "Shutdown Margin," suggests a value of $0.50. The GSTR LRA SAR Section 13.2.6.2, "Accident Analysis and Determination of Consequences," states that the current GSTR TSs have a shutdown margin of $0.55 with the most reactive rod withdrawn.

Revise proposed TS 3.1.1.1 to use a reactivity value for the Shutdown Margin consistent with the guidance and current SAR value ($0.50 or $0.55, respectively); provide further information which demonstrates why the proposed change to $0.30 is acceptable; or justify why no change is needed.

Response: The guidance in NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 3.1 indicates that the shutdown margin should be large enough to be readily determined experimentally, and then it suggests $0.50. The GSTR staff has demonstrated the ability to experimentally determine the shutdown margin within $0.159. In the NUREG definition of "shutdown margin", it suggests a "... reactivity necessary to provide confidence that the reactor can be made subcritical ..." There is no requirement, contrary to what is stated at the end of the first paragraph of your RAI Item 6, to have the minimum shutdown margin at least $0.30 more than the maximum error of the reactivity determination. The $0.30 we have proposed is a reactivity value that will provide confidence that the reactor can be made subcritical, given the maximum $0.159 error in the experimental determination. Therefore the $0.30 value meets the guidance of NUREG-1537. In addition, we note that there are other U.S. NRC licensed TRIGA reactors that have shutdown margin limits of less than $0.30. Examples are the facilities at Washington State University, Pennsylvania State University, and Wisconsin State University. Therefore no change is needed to our proposed $0.30 value. To make the specification more concise, the formatting of the specification was changed.

7. Proposed TS 3.1.3, "Core Configuration Limitations," does not appear to include limitations on the core configuration for the removal or insertion of fuel or control rods.

   NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 3.1, "Reactor Core Parameters," item (4), "Core Configurations," provides guidance for TS reactivity limits associated with fuel and control rod relocation.

   Revise proposed TS 3.1.3 to include the guidance in NUREG-1537, Part 1, Chapter 14, Appendix 14.1 for the TS reactivity limits reference above, including appropriate SRs, or justify why no change is needed.

Response: We propose to add two additional specifications to TS 3.1.3 as shown below:

d./4. Fuel shall not be inserted or removed from the core unless the reactor is subcritical by more than the calculated worth of the most reactive fuel assembly being moved.

e./5. Control rods shall not be manually removed from the core unless the core has been shown to be subcritical with all control rods in the full-out position.

8. Proposed TS 3.1.4, "Fuel Parameters," Specification d., describes "significant bulges, pitting, or corrosion." It is not clear to the NRC staff how to quantify the term "significant" as it relates to visual inspection of the bulges, pitting or corrosion.

   NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 3.1, "Reactor Core Parameters," item (6), "Fuel Parameters," provides guidance for the visual inspection of fuel.

   Revise proposed TS 3.1.4 to follow the guidance in NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 3.1, "Reactor Core Parameters," item (6), "Fuel Parameters," or justify why no change is needed.

Response: We believe that our proposed TS 3.1.4 does follow the guidance of NUREG-1537. The General Atomics Mechanical Maintenance Manual, GA-8287, discusses fuel-moderator elements (Sections 2.4 and 2.5) as well as use of the Fuel-Element Inspection Tool (Section 5.4). At no point does it suggest, or even mention, performing a visual inspection of the fuel elements. Therefore the manufacturer's guidance on visual inspections is void. Since a visual inspection is, by its very nature, not a method of objective measurement, it is not possible for us to quantify the term "significant". We also note that objective criteria for visual inspections of TRIGA fuel are not included in other U.S. NRC-licensed TRIGA facilities. We believe no change is needed to TS 3.1.4.

9. Proposed TS 3.2.1, "Control Rods," has the following items identified by the NRC staff:

   a. The NRC staff finds an apparent typographical error as Specification b. is listed twice.

      Revise proposed TS 3.2.1, to correct the typographical error, or justify why no change is needed.

Response: We agree with the finding. The second "b." specification will be changed to "c.".

   b. Specification a. ends with an "or," but a semicolon follows Specification b., leaving it unclears to the relationship to the specification that follows (the 2nd Specification b.).

      Revise the wording that links Specifications a., b., and the 2nd Specification b. to indicate if they are "and," "or," or stand-alone.

Response: In our copy of the proposed TS 3.2.1, we find that specification a. and the first specification b. are both followed with a semicolon and then the word "or". This indicates that any one of the three subparts will make the control rods inoperable if they are not met. The use of a semicolon is grammatically acceptable and consistent with other usage in the Technical Specifications. We do not believe that a change is needed.

   c. The NRC staff finds that Specification b. is not clear and would be clearer if it was worded "1 second for any shim or regulating rod or..."

      Revised the proposed wording in Specification b., or justify why no change is needed.

Response: We do not find the wording to be unclear; however, we propose to change the wording to be: "b. The scram time exceeds 1 second for any shim or regulating rod or the scram time exceeds 2 seconds for the transient rod;"

d. The 2nd Specification b. describes a "standard control rod" which is not defined in the TS Definitions. The NRC staff is not clear if this definition differs from the definition of control rod in the TS Definitions section.

Revise the proposed wording in the 2nd Specification b. to clearly indicate what constitutes a "standard control rod," or justify why no change is needed.

Response: We propose to change the wording to be:
"c. The maximum reactivity insertion rate of any shim or regulating rod exceeds $0.29 per second."

We also propose to change the wording in TS 4.1, Specification 3 to coincide with TS 3.2.1, Specification c.

10. Proposed TS 3.2.3 has the following items identified by the NRC staff:

a. Proposed TS 3.2.3, "Applicability" and "Objective" statements indicate the safety systems are required during "reactor operation." The NRC staff finds that some of the Interlocks listed in Table 3.3 may be required in modes other than reactor operation.

Revise proposed TS 3.2.3, "Applicability" and "Objective" statements to ensure the proper reactor operating conditions are stated for the Interlocks listed in Table 3.3, or justify why no change is needed.

Response: We will add the phrase "and interlocks" following the mention of "reactor safety system channels" within the Applicability and Objective statements.

b. Proposed TS 3.2.3, Table 3.2, "Preset Timer" Function states "SCRAM (≤ 15 sec.)." The NRC staff finds that this is not a clear description and additional detail would alleviate potential future confusion.

Revise proposed TS 3.2.3, Table 3.2, "Preset Timer," to state "SCRAM after pulse initiation (≤ 15 sec)," or another similar description, or justify why no change is needed.

Response: We will revise proposed TS 3.2.3, Table 3.2, "Preset Timer" in the "Function" column to state, "SCRAM ≤15 sec after pulse initiation".

c. Proposed TS 3.2.3, Table 3.2, "High voltage," in the column "Function" states "SCRAM on loss of nominal operating voltage to required power channels." The NRC staff is not clear which power channels are required.

Revise proposed TS 3.2.3, Table 3.2, "High voltage" in the column "Function" description to indicate the required power channels, or justify why no change is needed.

Response: We will revise proposed TS 3.2.3, Table 3.2, "High voltage" in the "Function" column to state, "SCRAM on loss of nominal operating voltage to NP1000 and NPP1000 power channels".

11. Proposed TSs 3.3 and 4.3, "Reactor Primary Tank Water," does not appear to have a specification for radioactivity content of the primary tank water.

NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 3.3, "Coolant Systems," item (8)," Secondary and Primary Coolant Radioactivity Limits," and Section 4.3, "Coolant Systems," Item (4), "Analysis of Coolants for Radioactivity," provide guidance for TS limits associated with primary coolant radioactivity.

Revise proposed TS 3.3 and TS 4.3 to include a specification for reactor tank water radioactivity, or justify why no change is needed.

Response: We will revise the proposed TS 3.3 to add an item d as follows:
"d. The reactor shall not be operated if the radioactivity of pool water exceeds the limits of 10 CFR 20 Appendix B Table 3 for radioisotopes with half-lives >24 hours."

In addition, we will revise the proposed TS 4.3 to add an item 4 as follows:
"4. The pool water radioactivity shall be measured at least quarterly."

12. Proposed TS 3.5, "Ventilation and Confinement System," provides a specification for operation of the ventilation system when the reactor is operating, but does not appear to include other situations which could result in the airborne release of radioactivity, such as the movement of fuel, fueled experiments or other material associated with experiments with the potential to become airborne.

ANSI/ANS-15.1-2007, Section 3.4.1, "Operations that Require Containment or Confinement," items (2) through (4) provide guidance for activities other than reactor operation requiring operation of the ventilation system. Additionally, NUREG-1537, Part II, Section 9.2, "Heating, Ventilation, and Air Conditioning," provides guidance that the operation of the ventilation system will limit normal airborne radioactive material to the limits in 10 CFR Part 20.

Revise proposed TS 3.5 to be consistent with the guidance in ANSI/ANS-15.1-2007 and NUREG-1537, or justify why no changes are needed.

Response: We propose to add Specifications 3 through 5 to TS 3.5 which by their addition make our TS consistent with 15.1 and NUREG-1537. We propose Specifications 3 through 5 to read as follows:

3. Movement of irradiated fuel or fueled experiments with significant fission product inventory outside of containers, systems, or storage areas within the reactor bay shall not be performed unless the ventilation system is operating as described in TS 3.5, Specifications 1 and 2. If there is a failure of the ventilation system while movement of these materials is being performed, the material shall be placed in an appropriate location until the ventilation system is made operable.

4. Core or control rod work that could cause a change in reactivity of more than one dollar shall not be performed unless the ventilation system is operating as described in TS 3.5, Specifications 1 and 2. If there is a failure of the ventilation system while this work is being performed, the material that could cause the change in reactivity shall be placed in an appropriate location until ventilation system is made operable.

5. Movement of experiments within the core that could reasonably cause a change of total worth of more than one dollar shall not be performed unless the ventilation system is operating as described in TS 3.5, Specifications 1 and 2. If there is a failure of the ventilation system while movement of these

experiments is being performed, the material shall be placed in an appropriate location until the ventilation system is made operable.

13. The proposed TS 3.7.1, "Radiation Monitoring Systems," has the following items identified by the NRC staff:

    a. Proposed TS 3.7.1 states that each radiation monitoring channel in Table 3.4 has a readout in the control room and is capable of sounding an audible alarm. The environmental dosimeters are listed in Table 3.4, and the NRC staff is not clear if the environmental dosimeters have a readout in the control room or sound an audible alarm.

       Revise proposed TS 3.7.1 to account for the readout and audible alarm capability of the environmental dosimeters, or justify why no change is needed.

Response: The environmental dosimeters are passive dosimeters and do not have readout or alarm capability. We have added a statement to TS 3.7.1 to clarify that the Environmental Dosimeters are not required to have a readout in the control room and audible alarm capabilities.

    b. Proposed TS 3.7.1, "Radiation Monitoring Systems," Table 3.4, lists the Radiation Area Monitor as a required channel. SAR Section 11.1.7.2, "Fixed Area Monitors," indicates that there are five gamma-sensitive area monitors in the GSTR facility, and the reactor bay monitor must be operable to support GSTR operation. SAR Section 7.7.1, "Area Radiation Monitors," describes radiation levels being monitored at strategic areas through the GSTR facility.

       However, it is not clear to the NRC staff how the different terms "Radiation Area Monitor," "Fixed Area Monitors," and "Area Radiation Monitors," are related since they appear to describe the same system for monitoring the gamma radiation in various locations of the GSTR facility. It is also not clear where the Radiation Area Monitor is located, or what components constitute the Radiation Area Monitor Channel required in TS 3.7.1, Table 3.4. The footnote to Table 3.4, indicates that the "monitors" may be out of service, but only describes a temporary substitute for the Area Radiation Monitor. It is not clear if this includes the Radiation Area Monitor and/or Continuous Air Monitor.

       NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 3.7.1, "Monitoring Systems," item (3) "Area Monitors," provides guidance that a TS should require operable area radiation monitors in and near the reactor.

       i. Provide a clear (or consistent) description relating the "Radiation Area Monitor" "Fixed Area Monitors" and "Area Radiation Monitors" for NRC staff understanding of the different names, what constitutes a Radiation Area Monitor Channel, or justify why no change is needed.

Response: The description in SAR subsection 11.1.7.2 uses the word "fixed" to indicate that these area monitors are not portable, but are permanently mounted at their respective locations. This is in contrast to SAR subsection 11.1.7.1 which discusses "Portable Survey Meters". In that case, the word "portable" means that these survey meters may be moved from one location to another and they are not permanently mounted. These subsections are under the SAR Section 11.1, "Radiation Protection". A Radiation Area Monitor Channel is a standalone instrument, positioned in a fixed location, including the detector, high voltage, signal processing, alarm function, and display.

We propose to change the footnote last sentence to be "Radiation Area Monitor" instead of "Area Radiation Monitor".

ii. Revise proposed TS 3.7.1, Table 3.4, to clearly articulate the radiation monitor needed to support the GSTR operation, and its location, if needed to differentiate it from other area radiation monitors, or justify why no change is needed.

Response: In order to clarify the requirements, we propose to change Table 3.4 as follows:

Table 3.4: Minimum Radiation Monitoring Channels*	
Radiation Monitoring Channel	**Number**
Continuous Air Monitor sampling reactor bay air	1
Radiation Area Monitor in reactor bay	1
Environmental Dosimeter outside reactor facility	3

iii. Revise proposed TS 3.7.1, Table 3.4, Footnote, to clearly articulate those monitors which can be temporarily replaced with a substitute monitor.

Response: In order to clarify the footnote of Table 3.4, we propose to revise it as follows:

*The Continuous Air Monitor or the Radiation Area Monitor may be out-of-service for up to 2 hours for calibration, maintenance, troubleshooting, or repair. During this out-of-service time, no experiments or maintenance activities shall be conducted which could directly result in alarm conditions (e.g., airborne releases or high radiation levels), and the ventilation system shall be operating. A portable, gamma-sensitive ion chamber, with display visible from the control room, may be utilized as a temporary substitute for the required Radiation Area Monitor (but not for the Continuous Air Monitor) for a period up to 60 days.

c. Proposed TS 3.7.1, has a footnote, indicated by an "*" that states "Monitors..." The NRC staff is not clear how this note relates to the number of channels in Table 3.4, or if this note indicates that multiple monitors can be out-of-service simultaneously, including all monitors which comprise a channel as listed in Table 3.4.

   Revise proposed TS 3.7.1, footnote, to clearly indicate the number of monitors which may be out-of-service simultaneously, explain how that relates to the number of channels provided in Table 3.4, or justify why no change is needed.

Response: We believe our proposed revision of the footnote (shown in the response to RAI Item 13.b) also resolves this item.

14. Proposed TS 3.8.1, "Reactivity Limits," Specifications a. and b., provide reactivity limits for a single moveable experiment and a single secured experiment, respectively. There does not appear to be a TS that provides a limit on the sum of all experiments.

The guidance in NUREG-1537, Part 1, refers to ANSI/ANS-15.1-2007, and U.S. Nuclear Regulatory Commission Regulatory Guide (RG) 2.2, "Development of Technical Specifications for Experiments in Research Reactors," November 1973. ANSI-15.1-2007, 3.8.1, "Reactivity Limits," item (2) provides guidance that "the sum of the absolute values of the reactivity worths of all experiments," should be indicated as a limit. RG 2.2, C.1.A, "Reactivity Effects," item (5) provides guidance that "the sum of the magnitudes of the static reactivity worths of all unsecured experiments which coexist should not exceed the maximum value of potential reactivity worth authorized for a single secured removable experiment (e.g., proposed TS 3.8.1, Specification b., $3.00)."

Propose a TS limit for the sum of the absolute reactivity values for all experiment, or justify why no change is needed. Include an assessment of the reactivity value for the proposed TS limit consistent with the guidance described above.

Response: We propose to add a third item under section 3.8.1 to give a limit on the sum of absolute reactivity worth values to $5.00. This value is consistent with other 1 MW TRIGA facilities that have been relicensed in recent years by the U.S. NRC. We reconfirm that we must still meet TS 3.1.1.1, Specification 1.a and TS 3.1.1.2, Specification 1. The new specification will state:

"c. The sum of the absolute reactivity worth for all experiments shall be less than $5.00."

15. Proposed TS 4, "Surveillance Requirements," states "All bases for the following surveillance requirements can be found in the operating procedures within the Reactor Operations Manual or in Safety Analysis Report. The approved operating procedures are periodically reviewed and reapproved by the Reactor Operations Committee (ROC)." The NRC staff is not clear as to the purpose of this information, nor does it match the guidance in NUREG-1537 or ANSI/ANS-15.1-2007.

NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 1.2.2, "Format," provides guidance that any information used to support the TSs should be explicitly referenced.

Revise or remove the statement in TS 4.0 above, or justify why no change is needed.

Response: We will remove the paragraph immediately following the heading, "4. Surveillance Requirements"

16. Proposed TS 4.0, "General," has the following items identified by the NRC staff:

a. Proposed TS 4.0, Specification 1, defers some SRs during an "extended reactor shutdown." However, "extended reactor shutdown" is not defined.

NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 4, "Surveillance Requirements," provides guidance that surveillances that are not required for safety while the reactor is shutdown may be deferred.
Revise proposed TS 4.0, Specification 1, "extended reactor shutdown to match the guidance in NUREG-1537, Part 1, Chapter 14, Appendix 14.1, or justify why no change is needed.

Response: We propose to change TS 4.0, Specification 1, to remove the term "extended reactor shutdown" and to read:

"1. Surveillance requirements may be deferred during reactor shutdown (except TS 4.3 Specifications 1 and 3, and TS 4.7 Specifications 1, 2, 3, and 4). However, they shall be completed prior to reactor startup unless reactor operation is required for performance of the surveillance. Such surveillance shall be performed as soon as practical after reactor startup. Scheduled surveillance which cannot be performed with the reactor operating may be deferred until a planned reactor shutdown."

b. Proposed TS 4.0, Specification 1, allows SRs to be deferred during reactor shutdown, except those as stated, including "section (TS) 4.7, specification 2." The NRC staff is not clear why TS 4.7, specifications 1, 3 and 4, could be deferred during reactor shutdown since the possibility could exist that other activities within the GSTR facility involving the use or handling of radioactive material, could result in the need for the CAM, ARM and environmental monitors to be maintained.

   Provide a justification for the provision in TS 4.0, Specification 1 to defer the surveillance requirements of TS 4.7, Specifications 1, 3 and 4; propose a revision to TS 4.0, Speculations 1, to remove TS 4.7; or justify why no change is needed.

Response: This item is addressed by the change proposed in our response to RAI 16 a above.

17. Proposed TS 4.1, "Reactor Core Parameters," has the following items identified by the NRC staff:

    a. Specification 2, provides a SR for the total reactivity worth of each control rod. However, the periodicity appears limited to any significant change in core or control rod configuration, and the term "significant change" is not defined.

       NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 4.2, "Reactor Control and Safety Systems," provides guidance that the surveillance should be performed annually.

       Revise proposed TS 4.1, Specification 2, to include an annual SR period and include a definition of "significant change," or justify why no changes are needed.

Response: We propose to revise TS 4.1, Specification 2, to read,
"2. The total reactivity worth of each control rod shall be measured annually or following a change in core or control rod configuration that is expected to change the total reactivity worth of that control rod by more than $0.30 (not including transient fission product poison effects)."

b. Proposed TS 4.1, Specification 3, provides a SR for the maximum reactivity insertion rate following any significant change in core or control rod configuration. However, "significant change in core or control rod configuration," is not defined, and no other periodicity is provided.

   ANSI/ANS-15.1-2007, Section 4.2, "Reactor Control and Safety Systems," item (2) provides guidance that the rod insertion and withdraw speeds should also be measured annually.

   Revise proposed TS 4.1, Specification 3 to define "significant change in core or control rod configuration," and include an annual SR period, or justify why no changes are needed.

Response: We propose to revise TS 4.1, Specification 3, to read,
"3. The maximum reactivity insertion rate of a standard control rod shall be measured annually or following a change in core or control rod configuration that is expected to change the total reactivity worth of that control rod by more than $0.30 (not including transient fission product poison effects)."

We will also change TS 4.1, Specifications 4 and 5 to have wording consistent with Specifications 2 and 3.

c. Proposed TS 4.1, Specification 6, provides a SR on the transient control rod mechanical stop when pulsing is scheduled. However, no periodicity is provided by the TS, and the drive mechanism is not included in the SR.

   ANSI/ANS-15.1-2007, Section 4.2, "Reactor Control and Safety Systems," item (3) provides guidance that the transient rod and mechanism should be tested and inspected annually.

   Revise proposed TS 4.1, Specification 6, to include a SR for the transient rod and drive mechanism to be tested and inspected annually, or justify why no change is needed.

Response: TS 4.1, Specification 6 will be revised to include an annual periodicity requirement for the transient rod and drive mechanism.

d. Proposed TS 4.1, Specification 8, provides a SR for fuel element inspection at intervals of 5 years or 500 pulses.

   NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 4.1, "Reactor Core Parameters," item (6), "Fuel Parameters," provides guidance that fuel inspections for TRIGA reactor fuel which is used to pulse should be performed annually. The NRC staff notes the low pulsing activity at GSTR and, in accordance with guidance referenced above, would consider a relaxed requirement to be acceptable (e.g., 20 percent of the fuel elements inspected each year, including some peak power fuel elements, with all the fuel elements being inspected over a 5 year period along with a requirement that if a fuel element fails the annual 20 percent inspection, all fuel elements are inspected).

   Revise proposed TS 4.1, Specification 8, to include an annual SR period consistent with the guidance in ANSI/ANS-15.1-2007, or justify why no change is needed.

Response: Fuel movement operations are widely considered to be operations that pose a higher hazard risk and create more potential for safety problems than routine operations. As a result, the GSTR has always performed fuel inspections every 5 years (quinquennially) instead of performing a partial inspection every year. The GSTR has never had a fuel element fail a periodic inspection in over 47 years of operation. Our only 3 leaking elements (1 FFCR and 2 IFEs) were discovered during normal operation and not as a part of routine fuel element inspections. In addition, our pulsing operations are limited to $3.00 pulses and our average pulsing rate is <5 pulses per year. Our proposed specification is consistent with the guidance in ANSI/ANS-15.1-2007 which states, "Any requirements for periodic inspection of the fuel or limitations on fuel burnup shall be included as appropriate." We do not believe a change is needed.

18. Proposed TS 4.2, "Reactor Control and Safety Systems," has the following items identified by the NRC staff:

a. Proposed TS 4.2, Specification 2, provides a SR for the scram time following any repair or non-routine maintenance on the control rod drive. The NRC staff is not clear why the surveillance is limited to "non-routine" maintenance or to just the control rod drive.

ANSI/ANS-15.1-2007, Section 4.2, "Reactor Control and Safety Systems," item (4) provides guidance that scram time testing should be performed following any work on the control rods or control rod drives.

Revise proposed TS 4.2, Specification 2, to follow the guidance described above, or justify why no changes are needed.

Response: The standard TRIGA control rod drive mechanisms contain three limit switches that require very precise adjustment in order for the rod drive to function. The need for minor adjustment of one of these limit switches is not unusual and should not be justification for performing a scram time measurement. The standard TRIGA control rod drive mechanism performs a scram by turning off an electromagnet that disconnects the control rod from the drive mechanism, so there is no potential for a limit switch adjustment on the drive mechanism to affect the scram time. We propose to revise the wording of TS 4.2, Specification 2, to read:
"2. The scram time shall be measured at least annually or after any work (not including routine limit switch adjustments) is performed on a control rod or control rod drive.

b. Proposed TS 4.2, Specification 3, provides a SR, for the items listed in TS Table 3.2, to perform channel checks following modifications or repairs.

ANSI/ANS-15.1-2007, Section 4.2, "Reactor Control and Safety Systems," item (6) provides guidance that an operability (channel) test should be performed following modifications or repairs. The definitions in Section 1.3 of ANSI//ANS-15.1-2007 provide guidance for the definition of a channel test.

Revise proposed TS 4.2, Specification 3, to perform a channel test following modifications or repairs, or justify why no change is needed.

Response: We propose to revise TS 4.2, Specification 3, to require channel tests as follows,

"3. A channel test of each of the reactor safety system channels in Table 3.2 for the intended mode of operation shall be performed prior to each day's operation or prior to each operation extending more than one day. The same channel tests shall be performed after modifications or repairs to the scram channels to ensure operability of the respective channels." The term "checks" in the NOTE for TS 4.2 will be changed to "specifications" as part of this change.

c. Proposed TS 4.2, "Specification 4, provides a SR for items in TS 3.2.3, Table 3.2, "Minimum Reactor Safety Channels," related to pulsing, and for Table 3.2 and 3.3, except for the NM-1000.

The specification is not clear to the NRC staff. It appears to cover surveillances for items needed for pulsing and for other items on a semi-annual period.

Revise proposed TS 4.2, Specification 4, to clearly indicate which of the SRs are for the items in TS 3.2.3, Tables 3.2 and 3.3.

Response: We propose to revise TS 4.2, Specification 4, to read:
"4. A channel test of items in Tables 3.2 and 3.3 shall be performed at least semi-annually, except for those two items required solely for pulse mode operation, which shall be channel tested during each startup for pulse mode operation. The two items required solely for pulse mode operation are the Preset timer scram in Table 3.2 and the control rod interlock in Table 3.3 that prevents withdrawal of any rod except the Transient Rod. "

d. Proposed TS 4.2, Specification 4, provides an exception for the semi-annual test of the NM-1000 which is not clear to the NRC staff. Also, it is not clear which TS 4 SR is applicable to the NM-1000. Additionally, NM-1000 is not listed in TS 3.2.3, Table 3.2.

   Revise proposed TS 4.2, Specification 4, to clearly indicate the SR for the NM-1000, if it is only associated with the interlock in Table 3.3, or justify why no change is needed.

Response: We believe that the response to the prior RAI item (18.c) also addresses this item.

e. Proposed TS 4.2 does not appear to provide SRs for TS 3.2.3, Table 3.2.

   ANSI/ANS-15.1-2007, Section 4.2, "Reactor Control and Safety Systems," provides guidance for surveillances of reactor safety and control systems, like those listed in TS 3.2.3, Table 3.2.
   Provide SRs for TS 3.2.3, Table 3.2, or justify why no change is needed.

Response: We believe that the response to prior RAI item 18.c also addresses this item.

19. Proposed TS 4.3, "Reactor Primary Tank Water," has the following items identified by the NRC staff:

    a. Proposed TS 4.3, Specification 1, provides a SR for the reactor tank water level alarm. The NRC staff finds that proposed TS 3.3, "Reactor Primary Tank Water," provides a limiting condition of operation (LCO) for the tank level, but does not indicate an alarm.

       The regulations in 10 CFR 50.36(c)(2)(ii)(a), requires installed instrumentation to detect a significant abnormal degradation of the reactor coolant boundary.

       Revise proposed TS 3.3, to include an alarm for the reactor tank water level, or justify why no change is needed.

Response: The reference, 10 CFR 50.36(c)(2)(ii)(A) states:

"(ii) A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

(A) Criterion 1. *Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary."*

This reference does not apply to the GSTR because we do not have a reactor coolant pressure boundary, which is clearly defined in 10 CFR 50.2 as applying to boiling and pressurized water-cooled nuclear power plants. Despite this erroneous justification, we believe it is in the best interest of the GSTR to revise proposed TS 3.3 to read:

"c. The reactor shall not be operated if the tank water level is more than 24 inches below the top lip of the reactor tank, and an alarm which is audible to the reactor operator shall sound when the water level is too low."

b. Proposed TS 4.3, "Reactor Primary Tank Water," does not appear to have a channel test for the reactor tank water level alarm.

   NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 4.3, "Coolant Systems," item (8), "Primary Coolant Sensors and Channels," provides guidance for calibration of sensors associated with the primary coolant.

   Revise proposed TS 4.3 to include a channel test for the reactor tank level alarm, or justify why no change is needed.

Response: We will revise proposed TS 4.3 to read:
"1. A channel test of the reactor tank water level alarm shall be performed at least semi-annually. "

c. The "NOTE" is contained within proposed TS 4.3, Specification 3. The NRC staff finds that this may be a typographical error as the NOTE may be applicable to all three proposed TSs (TS 4.3, Specifications 1 through 3).

   Revise the placement of the NOTE to indicate its applicability to all three proposed TS 4.3, Specifications 1 through 3, or justify why no change is needed.

Response: The NOTE is intended to be applicable to all parts of TS 4.3, so it will be reformatted to be on a separate line as:

"NOTE: These specifications are not required if the reactor fuel has been removed from the tank."

d. Proposed TS 4.3, Specification 3, requires a monthly measurement of the conductivity. Proposed TS 3.3, Specification b., provides a conductivity limit when averaged over one month. The NRC staff is unclear how a monthly surveillance can produce a monthly average give only one data point.
   Revise proposed TS 4.3, Specification 3, to measure the conductivity more frequently in order to determine a monthly average, or justify why no change is needed.

Response: The intent is that an average of conductivity readings should be made in case more than one reading was taken during the month. In practice, these readings are frequently taken more often than required and averaging would be appropriate. In order to prevent further confusion, we propose to revise TS 4.3 item 3 to read:

"3. The reactor tank water conductivity shall be measured monthly. Multiple measurements taken in one month shall be averaged to determine the monthly value."

20. Proposed TS 4.5, "Ventilation and Confinement System," Specification 2, provides for a channel test of the reactor bay ventilation system's ability to automatically switch to the emergency mode upon actuation of the CAM high alarm. The LCO in TS 3.5, indicates that the emergency ventilation system is operable if the emergency exhaust fan is operating and if the reactor bay minimum differential pressure is 0.1 inches water column. The NRC staff finds that the SR in TS 4.5 does not appear to test the fan or differential air pressure, only the ventilation system's ability to switch.

NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 4.4.2, "Confinement," provides guidance that confinement systems should have a functional test of the overall system described in the SAR.

Revise proposed TS 4.5, Specification 2, to ensure the emergency ventilation system is channel tested consistent with the requirements in the TS 3.5 LCO, or justify why no change is needed.

Response: We propose revising TS 4.5, item 2, to read as follows:

"2. A channel test of the reactor bay ventilation system's ability to automatically switch to the emergency mode upon actuation of the CAM high alarm, and to provide a reactor bay minimum differential pressure of 0.1 inches water column, shall be performed quarterly. "

21. Proposed TS 4.7, "Radiation Monitoring System," has the following items identified by the NRC staff:

a. Proposed TS 3.7.2, "Effluents," has a value for Ar-41 that could allow the reactor room airborne radioactivity levels to potentially exceed the Derived Air Concentration provided in 10 CFR Part 20. As such, the NRC staff considers the proposed "monthly" periodicity for the channel check for TS 4.7, Specification 1, for the radiation area monitor to be too infrequent to verify the airborne radiation levels and to be able to respond in accordance with the requirements in 10 CFR 20.1702, as appropriate. The NRC staff considers a daily channel check of the continuous air monitor to be a more appropriate frequency for the assessment of any potential Ar-41 activity in the reactor room.

Revise proposed TS 4.7, Specification 1, to daily, or justify why no change is needed.

Response: Exceeding the 10 CFR 20 DAC value for ^{41}Ar in the reactor bay the GSTR is a routine event and is not a significant radiological issue for the following reasons:

1) The 10 CFR 20 DAC values bases, as stated in 10 CFR 20, Appendix B are: "The DAC values relate to one of two modes of exposure: either external submersion or the internal committed dose equivalents resulting from inhalation of radioactive materials. Derived air concentrations based upon submersion are for immersion in a semi-infinite cloud of uniform concentration and apply to each radionuclide separately." Since ^{41}Ar provides a submersion dose, the assumption of a semi-infinite cloud needs to be evaluated. The GSTR reactor bay can be modeled as a volume equivalent to a hemisphere of 5.5 meters (~18 feet) radius. This is significantly smaller than a semi-infinite hemisphere and it greatly reduces the dose rate a person would receive from being in a 1.0 DAC level of ^{41}Ar. In fact, the ratio is approximately a factor of 33 which means that a DAC level of 33 in the GSTR reactor bay would give the same radiation dose rate as being in a semi-infinite cloud of ^{41}Ar at a DAC level of 1.0. Therefore, limiting the allowable ^{41}Ar level in the GSTR reactor bay to 1.0 is unjustified.

2) The 10 CFR 20 DAC values assume an exposure at the DAC concentration for 2000 hours in a year. The GSTR reactor bay only has significant ^{41}Ar levels as a result of power operation and the highest annual power production at the GSTR is ~1400 hours, significantly less than the 2000 hour assumption.
3) GSTR staff members are normally not in the reactor bay during full power operation because there simply is no work to be done in the bay during operation. Since ^{41}Ar provides an external dose, that dose is detected and reported through the normal staff radiation dosimetry. Staff doses from ^{41}Ar are a small fraction of the staff's typical annual doses received at the facility. This removes the actual GSTR conditions even further from the 10 CFR 20 DAC assumptions.

Based on the above facts, we do not believe that a change is needed. However, we will propose to add the ^{41}Ar monitor to TS4.7, Specification 1 requiring a month channel check. We will also add the ^{41}Ar monitor to Table 3.4 as one of the minimum radiation monitoring channels. In addition, the ^{41}Ar reading is displayed on the console monitor, and facility procedure dictates that console indications should be scanned frequently by the operator.

b. Proposed TS 4.7. Specification 3, provides a SR for the Ar-41 monitor, but TS 3.7.2, "Effluents," does not appear to provide an LCO for the Ar-41 monitor.

   Provide an LCO for the Ar-41 monitor, or justify why no change is needed.

Response: The LCO (TS 3.7.2) is provided on the allowable released concentration of ^{41}Ar, which is appropriate. The SR is on the equipment that is used to evaluate the ^{41}Ar concentration, which is also appropriate. See the response to part a above for further clarification, and we propose to add to the footnote of the table that we are allowed by procedure to also determine ^{41}Ar releases by calculation, as a function of reactor operating history.

22. Proposed TS 4.8, "Experimental Limits," Specification 1, references "section 3.7.1," and Specification 2, references "section 3.8.2, section 3.8.3, and section 6.2.3." The NRC staff finds that a more definitive reference to the sections, such as "TS 3.7.1," would reduce potential confusion. The NRC staff finds other examples where TS sections are referenced without including the "TS," e.g., see TS 4.0, Specification 1.

    NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 1.2.2, "Format," provides guidance that any information used to support the TSs should be explicitly referenced.

    Revise proposed TS 4.8, to indicate that the sections being referenced are "TS," and review and revise similar examples in the TS, or justify why no change is needed.

Response: We will revise the proposed TS 4.8 to read:

"1. The reactivity worth of an experiment shall be estimated or measured, as appropriate, before routine reactor operation with that experiment to ensure that the limits of TS 3.7.1 are not exceeded.

2. An experiment shall not be installed in the reactor or its irradiation facilities unless a safety analysis has been performed and reviewed for compliance with TS 3.8.2 and TS 3.8.3 by the Reactor Supervisor or ROC in full accord with TS 6.2.3, and the procedures which are established for this purpose."

23. Proposed TS 5.1, "Site and Facility Description," has the following items identified by the NRC staff:

a. Proposed TS 5.1, Specification 1, states "The licensed area includes..." which the NRC staff finds could infer that there are other areas beyond those listed in TS 5.1, Specifications a. through c.

Revise porposed TS 5.1, Specification 1 to state "The licensed area shall be...," or justify why no change is needed.

Response: We will revise the proposed TS 5.1 item 1 to read:

"1. The licensed area shall be the following locations on the Denver Federal Center:

a. Building 15: Rooms 149 through 152, Rooms 154, 157, 158, B10, B10B, and B11;

b. Area inside the wrought iron fence and south cooling tower wall that is near the SW corner of Building 15;

c. Building 10: Room 2."

b. Proposed TS 5.1, Specification 2, states "The reactor bay volume is 12000 cubic feet, and is designed...," which is more precise than necessary and present compliance challenge during an NRC inspection. The NRC staff considers a nominal value for the reactor bay volume to be acceptable.

Revise proposed TS 5.1, Specification 2 to state "The reactor bay volume shall be a nominal 12000 cubic feet, and shall be designed...," or justify why no change is needed.

Response: Our proposed TS 5.1, item 2, includes the symbol "~" in front of the value 12000 to indicate that it is an approximate value. Since this symbol has caused confusion, we will revise the proposed TS 5.1, item 2, to include the phrase "a nominal" as shown below:

"2. The reactor bay volume shall be a nominal 12000 cubic feet, and shall be designed to restrict leakage."

24. Proposed TS 5.2, "Reactor Coolant System," contains a "NOTE:" that indicates that the specifications are not required if the reactor core is defueled. The NRC staff is unclear if this this TS Note is acceptable if the core could be unloaded but fuel could still be stored in the fuel storage racks in the pool.

Explain the adequacy of the TS Note if core could be unloaded but fuel could still be stored in the fuel storage racks in the pool.

Revise proposed TS 5.2, Note, if necessary to clarify the conditions which no longer require TS 5.2, Specifications 1 and 2, or justify why no change is needed.

Response: Pool water requirements that must be met if fuel is stored in the storage racks in the pool are given in proposed TS 3.3 and proposed TS 5.4. Per proposed TS 5.4, item 3, if fuel is stored in water, regardless of whether it is in the pool or not, the water quality must be maintained according to proposed TS 3.3, item b. In addition, the requirement for sufficient cooling medium is required by proposed TS 5.4, item 2, whether it is air or water. We believe no change is needed.

25. Proposed TS 5.3.1, "Reactor Core," Specification 2, states "The TRIGA core assembly may..." The NRC staff is not clear if this "may" should be "shall" in order to limit the GSTR core to the fuel described it the specification.

ANSI/ANS-15.1-2007, Section 1.3, "Definitions," provides a definition of "shall" as denoting a requirement.

Revise proposed TS 5.3.1, Specification 2, to use "shall", or justify why no change is needed.

Response: We will revise the proposed TS 5.3.1, specification 2, to read:

"2. The TRIGA core assembly shall consist of stainless-steel clad fuel elements (8.5 to 12.0 wt% uranium), aluminum-clad fuel elements (8.0 wt% uranium), or a combination thereof."

26. Proposed TS 5.3.3, "Reactor Fuel," Specifications 1.a and 2.a, describe "nominal 20% ^{235}U enrichment," which could include highly enriched uranium (HEU) fuel, as provided in the definition for HEU in 10 CFR 50.2, "Definitions." The NRC staff recognizes that GSTR uses low enriched fuel, as described in the SAR. However, for clarity, the description provided in TS 5.3.3 should preclude any possibility for the use of HEU fuel (equal to 20 percent or greater enriched uranium-235).

Revise proposed TS 5.3.3, Specifications 1.a and 2.a to indicate that HEU fuel is not used at GSTR, or justify why no change is needed.

Response: We will revise the proposed TS 5.3.3, specifications 1.a and 2.a as follows:

"1. Aluminum-clad TRIGA fuel. The individual unirradiated aluminum-clad fuel elements shall have the following characteristics:

a. Uranium content: nominally 8.0 wt% with a ^{235}U enrichment of less than 20%;

...

2. Stainless-steel clad TRIGA fuel. The individual unirradiated standard TRIGA fuel elements shall have the following characteristics:

a. Uranium content: nominal range of 8.5 to 12.0 wt% with a ^{235}U enrichment of less than 20%;"

27. Proposed TS 5.4, "Fuel Storage," has the following items identified by the NRC staff:

a. Specification 1 states "all fuel elements..." However, the NRC staff finds that USGS has the capability to perform fueled experiments and therefore should consider the need to store fueled experiments or devices.

   ANSI/ANS-15.1-2007, Section 5.4, "Fissionable Material Storage," provides guidance for the storage of fuel, including fuel, and fueled experiments and devices.

   Revise proposed TS 5.4 to include fueled experiments and devices, or justify why no change is needed.

Response: We will revise the proposed TS 5.4, specification 1, to read:

"1. All fuel elements and fueled devices shall be stored in a geometrical array where the k-effective is less than 0.9 for all conditions of moderation."

b. Specification 3 states "must" rather than "shall."

ANSI/ANS-15.1-2007, Section 1.3, "Definitions," provides a definition of "shall" as denoting a requirement.

Revise proposed TS 5.4, Specification 3, to use "shall," or justify why no change is needed.

Response: We will revise the proposed TS 5.4, specification 3, to read:

"3. If stored in water, the water quality shall be maintained according to TS 3.3.b."

28. Proposed TS 6.1, "Organization," references ANSI/ANS-15.4, "Standard for the Selection and Training of Personnel for Research Reactors." The NRC staff finds that the title referenced does not match the title of ANSI/ANS-15.4, which is titled "Selection and Training of Personnel for Research Reactors," and does not contain "Standard for the," in the title. Additionally, the NRC staff noted that ANSI/ANS-15.4 has been recently revised (ANSI/ANS-15.4-2016) and consideration should be given to referencing the latest version.

Revise proposed TS 6.1 to use the correct title and revision year (e.g., 2007 or 2016) for ANSI/ANS-15.4, or justify why no change is needed.

Response: We will revise the proposed TS 6.1, "Organization", to correct the title for ANSI/ANS-15.4, as shown below. However, the intent of the GSTR is that staff personnel selection process is that it will be guided by ANSI/ANS-15.4, regardless of the current revision date. As such, we do not believe adding the revision date would be in the best interest of the facility.

"... The minimum qualification for all members of the reactor operating staff shall be in accordance with ANSI/ANS 15.4, "Selection and Training of Personnel for Research Reactors." ..."

29. Proposed TS 6.1.1, "Structure," has the following issues identified by the NRC staff:

a. Proposed TS 6.1.1 references the "USNRC," which the NRC staff finds is not consistent with the guidance in NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 6.1.1, "Structure," and Figure 1, and ANSI/ANS-15.1-2007, Section 6.1.1, "Structure," and Figure 1.

NUREG-1537, Part 1, Chapter 14, Appendix 14.1 and ANSI/ANS-15.1-2007, provide guidance that does not include a reference to the USNRC.

Revise proposed TS 6.1.1 to remove USNRC, or justify why no change is needed.

Response: We will revise the proposed TS 6.1.1 to remove "and USNRC" as shown below:

"The reactor administration shall be related to the USGS structure as shown in Figure 1."

b. Proposed TS 6.1.1, refers to the reactor administration in (Figure) 1: Administrative Structure. However, the NRC staff finds that figure provided in the proposed TS is only labeled as "1: Administrative Structure," and does not include "Figure" in the title. The NRC staff finds that it would be appropriate to add "Figure" to the title (e.g., "Figure 1: Administrative Structure") to reduce potential confusion when referring to the administrative structure.

   ANSI/ANS-15.1-2007, Section 6.1.1, "Structure," provides guidance for the title as Figure 1.

   Revise proposed TS (Figure) 1: Administrative Structure to add Figure to the title, or justify why no change is needed.

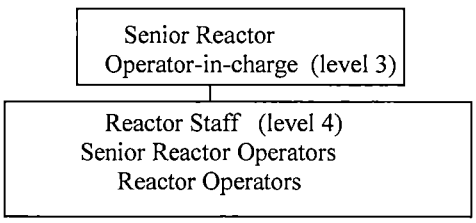
Response: We will revise the title to the Administrative Structure figure to include the word "Figure".

c. Proposed TS 6.1.1, (Figure) 1: Administrative Structure, provides a position (box) for the Senior Reactor Operator-in-charge. However, the NRC staff finds that the position does not have an indicated authority level in the box.

   ANSI/ANS-15.1-2007, Section 6.1.1., "Structure," provides guidance to indicate the responsibility levels for the staff responsible for the operation of the facility.

   Revise proposed TS (Figure) 1: Administrative Structure to indicate responsibility level of the Senior Reactor Operator-in-charge, or justify why no change is needed.

Response: We will revise proposed TS Figure 1 to indicate that the Senior Reactor Operator-in-charge is the Level 3 authority and the Reactor Staff is the Level 4 authority, as shown below.



d. The NRC staff finds that the Radiation Safety Committee Chairperson and Health Physics Staff have been replaced in the proposed TS (Figure) 1: Administrative Structure, with the Reactor Health Physicist.

   NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 6.1.1, "Structure," provides guidance that the TS should clearly indicate how and when the radiation safety staff communicates with facility management to resolve safety issues. Although the proposed change is consistent with the guidance in NUREG-1537, Part 1, the NRC is not clear as to the reason for the proposed change.

   Provide a basis or justification for the change to the Reactor Health Physicist, or explain why no additional information change is needed.

Response: The change in radiation safety communication is due to the reactor facility no longer possessing licensed materials that are under the purview of the Radiation Safety Committee Chairman or the associated USGS employee. This is a result of the GSTR license amendment number 12, issued

March 23, 2016. The oversight of the RSC was "transferred" to the ROC. In reality, prior to the license amendment, the audit functions were being performed by both the RSC and the ROC. These items include the radiation safety program annual review and procedure reviews. It also includes auditing the licensed material inventory, annual dosimetry, HP instrument calibrations, and waste handling and transfer. The licensed materials that were transferred to the reactor license are now inspected annually instead of every 3 years (at most) by regional NRC. The guidance of NUREG-1537, Part 1 is met and we believe no change is needed.

e. The NRC staff finds that the Reactor Health Physicist does not have a communication or a responsibility line to the Reactor Administrator (Level 1) or to the Senior Reactor Operator-in-charge.

   ANSI/ANS-15.1-2007, Section 6.1.1, Figure 1, provides guidance that Radiation Safety (Health Physicist) has a reporting line to the Level 1 or 2, and a communication line to the Level 3.

   Revise proposed TS (Figure) 1 to provide reporting and communication lines consistent with the guidance in ANSI/ANS-15.1-2007, or justify why no change is needed.

Response: We will revise Figure 1 to show a communication line from the Health Physicist to Level 3 (Senior Reactor Operator-in-charge). The current Figure 1 does show a reporting line to Level 2.

f. The proposed TS (Figure) 1: Administrative Structure, Reactor Operations Committee (ROC) illustration box contains information which includes: the ROC, the chairperson, the membership, and the reactor supervisor ex-officio. The NRC staff is not clear if this information constitutes requirements.

   ANSI/ANS-15.1-2007 provides guidance for the information needed in the Review/Audit illustration box of Figure 1 which does not contain information on the chairperson, the membership and the reactor supervisor ex-officio.

   Revise the TS (Figure) 1: Administrative Structure, for the ROC illustration box to be consistent with the guidance in ANSI/ANS-15.1-2007.

Response: Since we are not being allowed the option of justifying why not change is needed, we will revise Figure 1 to show a Review/Audit box comprised of the Reactor Operations Committee and no other details will be shown (see below).

Review/Audit Reactor Operations Committee

30. Proposed TS 6.1.2, "Responsibility," describes responsibilities for the various GSTR facility management levels. However, the NRC staff finds that the Senior Reactor Operator-in-charge is not described.
    ANSI/ANS-15.1-2007, Section 6.1.1, provides guidance that other organizational levels or staffing may be added to meet specific facility needs.

    Revise proposed TS 6.1.2, to add the responsibility for the Senior Reactor Operator-in-charge, or justify why no change is needed.

Response: We will revise proposed TS 6.1.2 specifications c and d to clarify the position of Senior Reactor Operator-in-charge, as shown below:

> "c. Senior Reactor Operator-in-charge (Level 3): The Senior Reactor Operator-in-charge reports to the Reactor Supervisor. This person is primarily involved in the oversight and direct manipulation of reactor controls, oversight and direct operation and maintenance of reactor related equipment, and oversight of recovery from unplanned shutdowns; and
>
> d. Reactor Staff (Level 4): Other Senior Reactor Operators and Reactor Operators report to the Senior Reactor Operator-in-charge and the Reactor Supervisor and are primarily involved in the direct manipulation of reactor controls, monitoring of instrumentation, and direct operation and maintenance of reactor-related equipment."

31. Proposed TS 6.1.3, "Staffing," has the following items identified by the NRC staff:

    a. Proposed TS 6.1.3, Specification 1.b., requires a second person present within the Denver Federal Center.

       ANSI/ANS-15.1-2007, Section 6.1.3, "Staffing," item (1)(b) provides guidance that the second individual be at the facility complex. However, the NRC staff finds that the Denver Federal Center appears to be much larger than the facility complex intended in the guidance provided in ANSI/ANS-15.1-2007, and as such, could substantially lengthen the time needed to respond to a GSTR emergency.

       Revise proposed TS 6.1.3, Specification 1.b, to indicate the facility complex (e.g., Building 15 or GSTR facility), or justify why no change is needed.

Response: The Denver Federal Center (DFC) is a controlled access facility that has minimal traffic and no significant impediments to easy transit from one building to another. Travel time to the reactor facility from any other building on the DFC would be no more than 5 minutes. The DFC has minimal traffic (vehicular or foot traffic) and the streets are always uncongested. In addition, there are no stop lights on the DFC and relatively few stop signs. There are times when a designated second person is needed to go to the USGS central shipping office on the DFC. This office is located on the opposite side of the DFC from Building 15. The designated second person might also be needed to go to Building 10 which is included in our licensed area but is not within Building 15. We note that other recently-issued educational institution NPR licenses allow the second individual to be at the "facility complex" and that term is not defined. Many university campuses are larger than the DFC and extremely more congested. As a result, we believe it is reasonable and consistent with recent licensing actions and consistent with the intent of the guidance not to change proposed TS 6.1.3.

b. Proposed TS 6.1.3, Specification 1.e., requires a list of facility personnel and contact information. However, the NRC staff finds that the types of support staff do not appear to be included by list.

ANSI/ANS-15.1-2007, 6.1.3 item (2) provides guidance that the list shall include: management personnel; radiation safety personnel; and other operations personnel.

Revise proposed TS 6.1.3, Specification 1.e., to include the support staff types provided in the ANSI/ANS-15.1-2007 guidance, or justify why no change is needed.

Response: TS 6.1.3, Specification 1.e. will be revised to read "A list of management personnel, radiation personnel, and reactor staff along with their contact information shall be available to the operator on duty."

c. Proposed TS 6.1.3, Specification 2.d, "Events requiring the direction of a Senior Reactor Operator," item d., includes "irradiation facility." However, the NRC staff is not clear if irradiation facility is equivalent to experiment.

ANSI/ANS-15.1-2007, Section 6.1.3, "Staffing," provides guidance that a Senior Reactor Operator is need for the relocation of an experiment.

Revise proposed TS 6.1.3, Specification 2.d., to include experiment, or justify why no change is needed.

Response: TS 6.1.3, Specification 2.d. will be revised to change the term "irradiation facility" to "experiment."

32. Proposed TS 6.1.4, "Selection and Training of Personnel," provides reference to ANSI/ANS-15.4-2007," and states that the title is "Standard for the Selection and Training of Personnel for Research Reactors." However, the NRC staff finds that this title does not match the title for ANSI/ANS-15.4-2007 (or the revised version, ANSI/ANS-15.4-2016).

ANSI/ANS-15.4 has the title "Selection and Training of Personnel for Research Reactors."

Revise proposed TS 6.1.4, reference to ANSI/ANS-15.4, to correct the title, or justify why no change is needed.

Response: The title of the referenced standard will be changed to read "Selection and Training of Personnel for Research Reactors" in TS 6.1.4.

33. Proposed TS 6.1.2, "Responsibility," Specification a., provides responsibilities for the Reactor Administrator. However, the NRC staff finds that the Reactor Administrator is not clearly defined by position or title within the USGS organization.

ANSI/ANS-15.1-2007, Section 6.1.2, "Responsibility," provides guidance that the individual responsible for the facility's license should be identified.

Revise proposed TS 6.1.2, Specification a. to clearly identify the individual within USGS responsible for the facility's license as the Reactor Administrator, or justify why no change is needed.

Response: The title for the Reactor Administrator can change with any change in the organizational structure within the USGS. It would be imprudent to specifically name the current person or their title. The Reactor Administrator is the direct supervisor of the Reactor Supervisor (Level 2 reports directly to Level 1). We propose to add the following sentences to the beginning of TS 6.1.2. "Responsibility for the safe operation of the reactor facility shall be with the chain of command established in Figure 1. Individuals at the various management levels, in addition to having responsibility for the policies and operation of the reactor facility, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license, the established charter, and the technical specifications.

34. Proposed TS 6.2.1, "Composition and Qualifications," and proposed TS 6.2.2, "Charter and Rules," provide a description of the ROC composition and qualifications, and charter and rules, and which uses the words "is" and "will" in several locations. The NRC staff finds the use of "is" and "will" to be less effective than "shall" and could result in potential confusion.

    ANSI/ANS-15.1-2007, Section 1.3, "Definitions," provides a definition of "shall" as denoting a requirement.

    Revise proposed TS 6.2.1 and 6.2.2 to use "shall" for requirements, or justify why no change is needed.

Response: TS 6.2.1 shall be revised to use the word "shall" in place of "is" and "will."

35. Proposed TS 6.2.2, "Charter and Rules," has the following items identified by the NRC staff:

    a. Proposed TS 6.2.2, states that "Criteria have been established for the conduct...," but the NRC staff finds that it doesn't state if the criteria is required.

       ANSI/ANS-15.1-2007, Section 6.2.2, "Charter and Rules," states that the review and audit functions shall be conducted in accordance with the charter or directive.

       Revise proposed TS 6.2.2 to indicate that the criteria is in effect for the conduct of the review and audit functions, or justify why no change is needed.

Response: The second sentence of TS 6.2.2 will be revised to read "The review and audit functions shall be conducted in accordance with an established charter for the Committee as written in the USGS Manual."

    b. Proposed TS 6.2.2, states "...non-Survey members...," which the NRC staff finds is not clearly defined. The NRC staff finds that a clear definition will avoid potential confusion in the future.

       Revise proposed TS 6.2.2 to clarify or define the term "non-Survey members," or justify why no change is needed.

Response: The term will be changed to be "non-USGS members."

c. Proposed TS 6.2.2, states, "A quorum for review, audit, and approval purposes shall consist of not less than one-half of the committee membership, provided that the operating staff does not constitute a majority of the committee membership." The NRC staff finds that the second reference to "committee membership" should be changed to "quorum" in order to reflect the limitations provided by the definition of quorum.

Revise proposed TS 6.2.2 to change second reference to committee membership to quorum, or justify why no change is needed.

Response: TS 6.2.2 will be revised to read "A quorum for review, audit, and approval purposes shall consist of not less than one-half of the voting membership where the operating staff does not constitute a majority."

36. Proposed TS 6.2.3, "Review and Audit Function," has the following items identified by the NRC staff:

a. Proposed TS 6.2.3, Specification c., states, "...near a technical specification limit..." The NRC staff is not clear as to what constitutes the term "near."

Revise proposed TS 6.2.3, Specification c., to clearly indicate the criteria for the review function.

Response: TS 6.2.3 Specification c will be changed to read "All new experiments or classes of experiments that could have reactivity or safety significance."

b. Proposed TS 6.2.3, Specifications a., d., and e. and use "charter." The NRC staff is not clear as to the definition of "charter."

Revise proposed TS 6.2.3, Specifications a., d., and e. and to remove or clarify the use of the term "charter," or justify why no change is needed.

Response: As stated in TS 6.2.2 Charter and Rules, "Criteria have been established for the conduct of the meetings and a charter for the Committee is written in the USGS Survey Manual." This charter within the USGS Survey Manual is the charter referenced in TS 6.2.3, Specifications d and e (contrary to RAI Question 36.a, there is no mention of "charter" within Specification a).

c. Proposed TS 6.2.3, does not appear to require an audit of the Security Plan.

NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 6.2.4, "Audit Function," provides guidance that includes an audit of the Security Plan.

Revise proposed TS 6.2.3, to include an audit of the Security Plan, or justify why no change is needed.

Response: TS 6.2.3 Specification d will be revised to include an audit of the Security Plan.

d. Proposed TS 6.2.3, states "A written report or minutes of the findings and recommendations of the review shall be submitted to the Reactor Administrator and the review and/or audit group members within 3 months after the audit has been completed."

The NRC staff is not clear as to the reference to the "review and audit group members" since it appears to be the responsibility of the ROC.

Revise proposed TS 6.2.3 to clarify the "review and audit group members," or justify why no change is needed.

Response: TS 6.2.3 will be changed from "review and audit group members" to "the ROC."

e. Proposed TS 6.2.3, states "These meetings will also include annual audits of the reactor facility and reactor records by the Committee."

The NRC staff finds it is not clear as to which "meetings" are referenced, and that this sentence may be better located within TS 6.2.3 to improve clarity.

Revise proposed TS 6.2.3 to indicate which "meetings" are referenced, or justify why no change is needed.

Response: The sentence under TS 6.2 Review and Audit has been revised to provide clarity for which meetings were required for review and audit. In addition TS 6.2.3 Review and Audit Function was divided into two sections, TS 6.2.3 Review Function and TS 6.2.4 Audit Function. This section division has been added to provide clarity and is recommended within ANSI/ANS-15.1.

In addition to the section division, the sentence "These meetings will also include annual audits of the reactor facility and reactor records by the Committee." will be removed. Finally, the first sentence of the newly added TS 6.2.4 Audit Function shall be modified to read as follows: "The audit function shall include selective (but comprehensive) examination of operating records, logs, other documents, and the reactor facility." We believe the above changes provide better clarity for the function of the ROC and their review function and audit function as performed during semi-annual meetings.

37. Proposed TS 6.4, "Procedures," has several items identified by the NRC staff:

a. Proposed TS 6.4 does not appear to state the responsible party for the review and approval of procedures.

ANSI/ANS-15.1-2007, Section 6.4, "Procedures," provides guidance for the review and approval of procedures.

Revise proposed TS 6.4 to indicate the responsible party for the review and approval of procedures, or justify why no change is needed.

Response: TS 6.4 will be modified to read as follows:

Written operating procedures shall be prepared, reviewed, and approved prior to initiating any of the activities listed in this section ~~to ensure the safety of operation of the reactor, but shall not preclude the use of independent judgment and action should the situation require such~~. The procedures shall be

reviewed by the ROC and approved by the Reactor Supervisor, and such reviews and approvals shall be documented in a timely manner. Substantive changes to the procedures shall be made effective only after documented review by the ROC and approval by the Reactor Supervisor. Minor modification to the original procedures that do not change their original intent may be made by the Reactor Supervisor. Temporary deviations from the procedures may be made by the responsible SRO or Reactor Supervisor in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and report within 24 hours or the next working day to the Reactor Supervisor. Procedures shall be in effect and in use for the following items:

b. Proposed TS 6.4 does not appear to provide any requirements associated with changes to procedures.

   NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 6.4, "Procedures," provides guidance that states that the method for procedure changes should be included in the TSs, and that minor modifications or temporary changes allowed by ANSI/ANS-15.1-2007 should not be spelled out in the TSs.

   Revise proposed TS 6.4 to include the method used for procedure changes, or justify why no change is needed.

Response: See response to 37.a

c. Proposed TS 6.4, Specification a. does not include all of the information provided in the guidance in ANSI/ANS-15.1-2007, Section 6.4, "Procedures," item (4) that states "or those that have an effect on reactor safety."

   Revise proposed TS 6.4, Specification a. to include the guidance in ANSI/ANS-15.1-2007, or justify why no change is needed.

Response: TS 6.4 Specification a will be modified to add the statement "or those that may have an effect on reactor safety" to the end of the specification

d. Proposed TS 6.4, Specification f. does not include all of the information provided in the guidance in ANSI/ANS-15.1-2007, Section 6.4, "Procedures," item (4) specifically that states "or core reactivity."

   Revise proposed TS 6.4, Specification f. to include the guidance provided in ANSI/ANS-15.1-2007, or justify why no change is needed.

Response: Assuming that you meant to refer to item (6) of ANSI/ANS-15.1, Section 6.4, TS 6.4 Specification f will be modified to add the statement "or core reactivity" to the end of the specification.

38. Proposed TS 6.5, "Experiment Review and Approval," has the following items identified by the NRC staff:

a. Proposed TS 6.5 uses "will" and "must" in several locations. The NRC staff considers the use of "will" and "must" could be subject to confusion and to be less effective than "shall."

ANSI/ANS-15.1-2007, Section 1.3, "Definitions," provides a definition of "shall" as denoting a requirement.

Revise proposed TS 6.5 to use "shall" to indicate requirements, or justify why no change is needed.

Response: TS 6.5 will be revised to use the word "shall" in place of "will."

b. Proposed TS 6.5 uses the phrase "as part of the 10 CFR 50.59 process," which is not clear to the NRC staff with respect to which requirements are associated with 10 CFR 50.59, "Changes, Tests and Experiments."

The regulation in 10 CFR 50.59 requires a review of any experiment not described in the final SAR in order to determine if NRC approval is required.

Revise proposed TS 6.5 to clearly indicate that all experiments require a review in accordance with the 10 CFR 50.59 review requirements, or justify why no change is needed.

Response: TS 6.5 will be revised to read the following:

"All experiments proposed for the reactor will be either Class I or Class II experiments and will be reviewed in accordance with the 10 CFR 50.59 review requirements. The review and classification of the proposed experiments will be the responsibility of the Reactor Supervisor.

Class I experiments include all experiments that have been run previously or that are minor modifications to a previous experiment. These are experiments which involve small changes in reactivity, no external shielding changes, and/or limited amounts of radioisotope production. The Reactor Supervisor has the authority to approve the following~~, as part of the 10 CFR 50.59 process~~:

.

.

.

Class II experiments include all new experiments and major modifications of previous experiments. These experiments must be reviewed and approved~~, as part of the 10 CFR 50.59 process,~~ by the ROC before being run.

c. Based on the definition of Class II experiments provided within TS 6.5, the NRC staff is not clear why the list of experiments is needed in TS 6.5, Specifications a through g, as Class II experiments appear to constitute all experiments other than Class I experiments. The list of experiments provided by TS 6.5, Specifications a through g, is in addition to the guidance provided in ANSI/ANS-15.1-2007, and may be more effective to maintain the criteria in an operating procedure.

Revise proposed TS 6.5 to remove the list of Specification a through g, or justify why no change is needed.

Response: TS 6.5 will be modified to remove the list of Specification a through g.

d. The "Radiation Safety Committee," is used in proposed TS 6.5, but not defined in the TS, or identified in TS Figure 1. NRC staff finds that the role and responsibility for the Radiation Safety Committee is not clearly defined.

Revise proposed TS 6.5 to provide the role and responsibility for the Radiation Safety Committee, or justify why no change is needed.

Response: TS 6.5 will be modified to remove the sentence "The USGS Radiation Safety Committee may also be consulted."

39. Proposed TS 6.6.2, "Actions to Be Taken in the Event of an Occurrence of the Type Identified in Section 6.7.2 Other than a Safety Limit Violation," and proposed TS 6.6.2, Specification d., both refer to "Section 6.7.2." The NRC staff finds this reference is not clearly stated as it should reference "TS Section 6.7.2," or "TS 6.7.2." Additionally, Specification d., refers to NRC, rather than U.S. NRC.

NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 1.2.2, "Format," provides guidance that any sources used to support the TSs should be explicitly referenced.

a. Revise proposed TS 6.6.2 to include "TS" in the section reference, or justify why no change is needed.

Response: TS 6.6.2 will be revised to include "TS" in the section reference.

b. Revise proposed TS 6.6.2, Specification d., to clearly refer to the "U.S. NRC," or justify why no change is needed.

Response: NRC was defined within the TS document to refer to the U.S. Nuclear Regulatory Commission. Use of the acronym NRC vs the phrase U.S. NRC to refer to the U.S. Nuclear Regulatory Commission is used within thousands of documents nationwide for licensing and relicensing of reactor facilities including but certainly not limited to the letter from the U.S. Nuclear Regulatory Commission to us the licensee on this very request for additional information. No change is needed.

c. Review and revise other examples referencing the U.S. NRC, as appropriate, or justify why no other changes are needed.

Response: As stated previously, NRC was defined within the TS document to refer to the U.S. Nuclear Regulatory Commission. Use of the acronym NRC vs the phrase U.S. NRC to refer to the U.S. Nuclear Regulatory Commission is used within thousands of documents nationwide for licensing and relicensing of reactor facilities including but certainly not limited to the letter from the U.S. Nuclear Regulatory Commission to us the licensee on this very request for additional information. No change is needed.

40. Proposed TS 6.7.2, "Special Reports," has the following items identified by the NRC staff:

a. Proposed TS 6.7.2, Specification a, does not appear to contain the US NRC address, as provided in the guidance in NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 6.7.2, "Special Reports."

Revise proposed TS 6.7.2 to include the address for the U.S. NRC as provided in the guidance in NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 6.7.2, or as provided in the proposed TS 6.7.2, b., or justify why no change is needed.

Response: TS 6.7.2 Special Reports, Specification a. will be revised to say "...followed by a report in writing to the NRC, Document Control Desk, Washington, D.C. ...".

b. Proposed TS 6.7.2, Specification a. iii, uses the term "LSSS" which is not consistent with the guidance provided in ANSI/ANS-15.1-2007, 6.7.2 (1)(c)(i).

Revise proposed TS 6.7.2, Specification a. iii, consistent with the guidance in ANSI/ANS-15.1-2007, or justify why no change is needed.

Response: TS 6.7.2 Special Reports, Specification a. iii. will be revised to say "Operation with the actual safety system setting less conservative than the LSSS;"

c. Proposed TS 6.7.2, Specification a. v, uses "failure" instead of "malfunction" and appears to omit "renders" and generally does not match the wording provided in the guidance in ANSI/ANS-15.1-2007, Section 6.7.2.

Revise proposed TS 6.7.2, consistent with the guidance provided in ANSI/ANS-15.1-2007, or justify why no change is needed.

Response: TS 6.7.2 Specification a. v. will be revised to read "Malfunction of a required safety system component which *renders or* could render the reactor safety system incapable of performing its intended safety function unless failure is discovered during maintenance tests or periods of reactor shutdown;"

d. Proposed TS 6.7.2, Specification a. vi, does not appear to be consistent with the wording provided in the guidance in ANSI/ANS-15.1-2007, Section 6.7.2, regarding "reactor trips." Revise proposed TS 6.7.2, Specification a. vi, consistent with the guidance in ANSI/ANS-15.1-2007, or justify why no changes are needed.

Response: A sentence will be added to the end of TS 6.7.2 Specification a. vi. that states "Reactor trips resulting from a known cause are excluded;"

e. Proposed TS 6.7.2, Specification a. vii, does not appear to be consistent with the wording provided in the guidance in ANSI/ANS-15.1-2007, Section 6.7.2, regarding the "causes" the existence or development of a condition.

   Revise proposed TS 6.7.2, Specification a. vii, consistent with the guidance in ANSI/ANS-15.1-2007, to include "causes," or justify why no changes are needed.

Response: TS 6.7.2, Specification a. vii. will be changed to read "An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy *causes or* could have caused the existence or development of a condition which *results or* could result in operation of the reactor outside the specified safety limits; or"

41. The NRC staff finds the numbering inconsistent in the proposed TSs. In some cases, the TS specifications are numerically listed (e.g., 1, 2, 3, etc.) and in other cases they are alpha-numerically listed (e.g., a, b, c, etc.). The NRC staff finds this confusing.

    NUREG-1537, Part 1, and ANSI/ANS-15.1-2007 provide guidance that uses a numerical listing. However, an alpha-numerical listing is also acceptable given it is used consistently.

    Consider revising the proposed TS to use either a consistent numerical or alpha-numerical listing, or justify why no changes are needed.

Response: Please see the attached complete USGS TSs. The following sections were modified to use a consistent format for numbering hierarchy: Sections 2.2, 3.1.1.1, 3.1.1.2, 3.1.2, 3.1.3, 3.1.4, 3.2.1, 3.2.2, 3.2.3, 3.3, 3.7.1, 3.7.2, 3.8.1, 3.8.2, 3.8.3, 6.1.2, 6.2.3, 6.2.4, 6.4, 6.5, 6.6.1, 6.6.2, 6.7.1, 6.7.2.

APPENDIX A

To

FACILITY LICENSE NO. R-113

DOCKET NO. 50-274

TECHNICAL SPECIFICATIONS AND BASES

FOR

THE UNITED STATES GEOLOGICAL SURVEY TRIGA RESEARCH REACTOR

SEPTEMBER 2016

1. Introduction

1.1 Scope

This document constitutes the Technical Specifications for the Facility License No. 113 as required by 10 CFR 50.36 and supersedes all prior Technical Specifications. This document includes the "Basis" to support the selection and significance of the specifications. Each basis is included for information purposes only. They are not part of the Technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere, except where they reference the USGS SAR or a specific Technical Specification. These specifications are formatted in a manner consistent with ANSI/ANS 15.1-2007.

1.2 Definitions

Audit: A quantitative examination of records, procedures or other documents.

Channel: A channel is the combination of sensing, signal processing, and outputting devices which are connected for the purpose of measuring the value of a parameter.

Channel Calibration: A 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 include a Channel Test.

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

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

Confinement: Confinement means an enclosure of the reactor bay which is designed to limit the release of effluents from the enclosure to the external environment through controlled or defined pathways.

Control Rod: A control rod is a device fabricated from borated graphite, B_4C powder or boron and/or fuel which is used to establish neutron flux changes and to compensate for routine reactivity losses. A control rod may be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged. Types of control rods shall include:

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

2. Shim Rod: A shim rod is a control rod having an electric motor drive and scram capabilities. It may have a fueled-follower section. Its position is varied manually.

3. Transient Rod: The transient rod is a control rod having an electric motor and pneumatic cylinder drive with scram capabilities. It can be rapidly ejected from the reactor core to produce a pulse or its position may be varied manually. It may have an air-filled follower.

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

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

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

2. Movable Experiment: A movable experiment is one that is not secured and intended to be moved while near or inside the core during reactor operation.

Instrumented Fuel Element: An instrumented fuel element is a special fuel element in which one or more thermocouples have been embedded for the purpose of measuring the fuel temperatures during reactor operation.

Irradiation Facilities: Irradiation facilities shall mean vertical tubes, rotating specimen rack, pneumatic transfer system irradiation tubes, sample-holding dummy fuel elements and any other in-tank device intended to hold an experiment.

Licensed Area: Rooms 149-152, 154, 157, 158, B10, B10B and B11 of Building 15, the area inside the wrought iron fence and south cooling tower wall that is near the SW corner of Building 15; and Room 2 of Building 10.

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

Operable: A system or component shall be considered operable when it is capable of performing its intended function.

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

Pulse Mode: Pulse mode shall mean any operation of the reactor with the mode selector in the pulse position.

Reactivity Worth of an Experiment: The reactivity worth of an experiment is the value of the reactivity change that results from the experiment being inserted into or removed from its intended position.

Reactor Operating: The reactor is operating whenever it is not secured or shut down.

Reactor Operator: An individual who is licensed to manipulate the controls of a reactor.

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

Reactor Secured: The reactor is secured when:

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

2. *Or* all the following conditions exist:

    a. All neutron-absorbing control devices are fully inserted or other safety devices are in their shutdown position, as required by technical specifications;

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

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

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

Reactor Shutdown: The reactor is shut down if it is subcritical by at least one dollar in the reference core condition with the reactivity worth of all installed experiments included.

Reference core condition: The condition of the core when it is at ambient temperature (cold, 18-25 °C) and the reactivity worth of xenon is negligible.

Review: A qualitative examination of records, procedures or other documents.

Safety Channel: A safety channel is a measuring channel in the reactor safety system.

Scram time: Scram time is the elapsed time between the initiation of a scram and the instant that the control rod reaches its fully-inserted position.

Senior Reactor Operator: An individual who is licensed to direct the activities of reactor operators. Such an individual is also a reactor operator.

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

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 and will remain subcritical without further operator action, starting from any permissible operating condition with the most reactive rod is in its most reactive position.

Square-Wave Mode (S.W. Mode): The square-wave mode shall mean any operation of the reactor with the mode selector in the square-wave position.

Steady-State Mode (S.S. Mode): Steady-state mode shall mean operation of the reactor with the mode selector in the manual or auto position.

Surveillance Intervals: Allowable surveillance intervals shall not exceed the following:

1. Quinquennial - interval not to exceed 70 months.
2. Biennial - interval not to exceed 30 months.
3. Annual - interval not to exceed 15 months.
4. Semi-annual - interval not to exceed 7.5 months.
5. Quarterly - interval not to exceed 4 months.
6. Monthly - interval not to exceed 6 weeks.
7. Weekly - interval not to exceed 10 days.

Unscheduled Shutdown: 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 that could adversely affect safe operation, not including shutdowns that occur during testing or checkout operations.

2. Safety Limits and Limiting Safety System Setting

2.1 Safety Limit-Fuel Element Temperature

Applicability. This specification applies to the reactor fuel.

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

Specifications.

1. The temperature in an aluminum-clad TRIGA fuel element shall not exceed 500 °C under any mode of operation.

2. The temperature in a stainless-steel clad TRIGA fuel element shall not exceed 1,150 °C.

Basis. The important parameter for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification especially since it can be measured. A loss of 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. 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 aluminum-clad TRIGA fuel element is based on data which indicate that the zirconium hydride will undergo a phase change at 535 °C. This phase change can cause severe distortion in the fuel element and possible cladding failure. Maintaining the fuel temperature below this level will prevent this potential mechanism for cladding failure (SAR 4.5).

The safety limit for the stainless-steel clad TRIGA fuel is based on data including the large mass of experimental evidence obtained during high performance reactor tests on this fuel. These data indicate that the stress in the cladding due to hydrogen pressure from the dissociation of zirconium hydride will remain below the ultimate stress provided that the temperature of the fuel does not exceed 1,150 °C (SAR 4.5.4.1).

2.2 Limiting Safety System Setting (LSSS)

Applicability. This specification applies to thermal reactor power.

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

Specifications.

1. The limiting safety system setting shall be a steady state thermal power of 1.1 MW.

Basis. The limiting safety system setting is a total core thermal power, which, if exceeded shall cause the reactor safety system to initiate a reactor scram. This setting applies to all modes of operation. In steady-state operation up to 1.1 MW, ample margins exist between this setting and the safety limits of

peak fuel temperature as specified in SAR 14.2.1, as long as the aluminum-clad fuel is restricted to the F and G rings of the core assembly (SAR 4.5.4.1).

Thermal and hydraulic calculations indicate that stainless-steel clad TRIGA fuel may be safely operated up to power levels of at least 1.9 MW with natural convection cooling (SAR 4.5.4.5).

3. Limiting Conditions of Operation

3.1 Reactor Core Parameters

3.1.1 Steady-state Operation

3.1.1.1 Shutdown Margin

Applicability. These specifications apply to the reactor at all times that it is in operation.

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

Specifications.

1. The reactor shall not be operated unless the shutdown margin provided by the control rods is at least $0.30 with the following conditions:

    a. Irradiation facilities and experiments in place and all movable experiments in their most reactive state;

    b. The most reactive control rod fully-withdrawn; and

    c. The reactor in the reference core condition where there is no ^{135}Xe poison present and the core is at ambient temperature. Calculations may be performed to determine a "no ^{135}Xe poison" reactivity condition.

Basis. The value of the shutdown margin assures that the reactor can be shut down from any operating condition even if the most reactive control rod should remain in the fully-withdrawn position. Since the reactor is seldom in a "no ^{135}Xe poison" condition, it is acceptable to perform calculations to determine the "no ^{135}Xe poison" reactivity condition.

3.1.1.2 Core Excess Reactivity

Applicability. This specification applies to the reactivity condition of the reactor and the reactivity worths of control rods and experiments. It applies for all modes of operation.

Objective. The objectives that must be simultaneously met are to assure that the reactor has sufficient reactivity to meet its mission requirements, be able to be shut down at any time, and not exceed its fuel temperature safety limit.

Specifications.

1. The maximum available excess reactivity shall not exceed $7.00 at reference core conditions.

Basis. This amount of excess reactivity will provide the capability to operate the reactor at full power with experiments in place and ^{135}Xe built up in the core.

3.1.2 Pulse Mode Operation

Applicability. This specification applies to the energy generated in the reactor as a result of a pulse insertion of reactivity.

Objective. The objective is to ensure that the fuel temperature shall not exceed 830 °C.

Specifications.

1. The reactivity to be inserted for pulse operation shall be determined and limited by a mechanical stop on the transient rod, such that the reactivity insertion shall not exceed $3.00.

Basis. The fuel temperature rise during a pulse transient has been estimated conservatively to not exceed any fuel temperature limits with a $3.00 pulse insertion.

3.1.3 Core Configuration Limitations

Applicability. This specification applies to mixed cores of aluminum-clad and stainless-steel clad types of fuel.

Objective. The objective is to ensure that the fuel temperature safety limit shall not be exceeded due to power peaking effects in a mixed core.

Specifications.

1. Aluminum-clad fuel shall only be loaded in the F and G rings of the core.

2, There shall be at least 110 fuel elements in the core (not including fuel-followed control rods).

3. There shall not be a fuel element in the central thimble.

4. Fuel shall not be inserted or removed from the core unless the reactor is subcritical by more than the calculated worth of the most reactive fuel assembly being moved.

5. Control rods shall not be manually removed from the core unless the core has been shown to be subcritical with all control rods in the full-out position.

Basis. The limitation of power peaking effects ensures that the fuel temperature safety limit shall not be exceeded in an operational core. Keeping aluminum-clad fuel in the F and G rings limits those fuel temperatures to safe values for aluminum-clad fuel (SAR 4.5.1.2). Keeping at least 110 fuel elements in the core helps reduce the power peaking in the core.

3.1.4 Fuel Parameters

Applicability. This specification applies to all fuel elements.

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

Specifications.

1. The reactor shall not operate with damaged fuel elements, except for the purpose of locating damaged fuel elements.

2. A fuel element shall be considered damaged and must be removed from the core if:

a. The transverse bend exceeds 0.0625 inches over the length of the cladding;

b. Its length exceeds its original length by 0.10 inch for stainless-steel clad fuel or 0.50 inch for aluminum-clad fuel;

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

d. Visual inspection identifies significant bulges, pitting, or corrosion; and

e. ^{235}U burnup is calculated to be greater than 50% of initial content.

Basis. Gross failure or obvious, significant visual deterioration of the fuel is sufficient to warrant declaration of the fuel as damaged. The elongation and bend limits are the values found acceptable to the USNRC (NUREG-1537).

3.2 Reactor Control and Safety System

3.2.1 Control Rods

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

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

Specifications.

1. The reactor shall not be operated unless all control rods are operable.

2. Control rods shall not be considered operable if:

a. Physical damage is apparent to the rod or rod drive assembly and it does not respond normally to control rod motion signals; or

b. The scram time exceeds 1 second for any shim or regulating rod or the scram time exceeds 2 seconds for the transient rod; or

c. The maximum reactivity insertion rate of any shim or regulating rod exceeds $0.29 per second.

Basis. This specification ensures that the reactor shall be promptly shut down when a scram signal is initiated. Experience and analysis have indicated that for the range of transients anticipated for a TRIGA reactor, the specified scram time is adequate to ensure the safety of the reactor (SAR 13.2.2.2.1).

3.2.2 Reactor Measuring Channels

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

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

Specifications.

1. The reactor shall not be operated in the specified mode unless the minimum number of power measuring channels listed in Table 3.1 is operable.

Table 3.1 Minimum Measuring Channels			
Measuring Channel	Effective Mode		
	S.S.	Pulse	S.W.
Power level (NP1000 and NPP1000)	2	-	2
Pulse power level (NPP1000)	-	1	-
Power level (NM1000)	1	-	1
Water temperature	1	1	1

Basis. The power level monitors ensure that the reactor power level is adequately monitored for steady-state, square wave and pulse modes of operation (SAR 7.2.3.1). The specifications on reactor power level indication are included in this section since the power level is directly related to the fuel temperature. The water temperature monitor ensures that water temperature will be kept within the specified limit.

3.2.3 Reactor Safety System

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

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

Specifications.

1, The reactor shall not be operated unless the minimum number of safety channels described in Table 3.2 and interlocks described in Table 3.3 are operable.

Table 3.2 Minimum Reactor Safety Channels				
Safety Channel	Function	Effective Mode		
		S.S.	Pulse	S.W.
Power level	SCRAM @ 1.1. MW(t) or less	2	-	2
Preset timer	SCRAM $\leq$15 sec after pulse initiation	-	1	-
Console SCRAM button	SCRAM	1	1	1
High voltage	SCRAM on loss of nominal operating voltage to the NP1000 and NPP1000 power channels	2	1	2
Watchdog SCRAMs	Scram within 8 seconds upon lack of response in DAC or CSC computer (one scram circuit per computer)	2	2	2

Table 3.3 Minimum Interlocks				
Interlock	Function	Effective Mode		
		S.S.	Pulse	S.W.
NM1000 Power level	Prevents control rod withdrawal at $<10^{-7}$% power	1	-	-
Transient Rod Cylinder	Prevents application of air unless fully inserted	1	-	-
1kW Pulse interlock	Prevents entering pulse mode above 1 kW	1	-	-
Shim and Regulating rod drive circuits	Prevents simultaneous manual withdrawal of two rods	1	-	1
Shim and Regulating rod drive circuits	Prevents withdrawal of any rod except Transient Rod	-	1	-

Basis. The power level scrams provide protection to ensure that the reactor can be shut down before the safety limit on the fuel element temperature will be exceeded. The manual scram allows the operator to shut down the system if an unsafe or abnormal condition occurs. The high voltage scram ensures that the required power measuring channels have sufficient high voltage as required for proper functioning of their power level scrams. The interlock to prevent startup of the reactor at count rates less than 10^{-7}% power ensures that the startup is not initiated unless a reliable indication of the neutron flux level in the reactor core is available. The interlock to prevent entering pulse mode above 1 kW is to ensure that the magnitude of the pulse will not cause the fuel element temperature safety limits to be exceeded. The interlock to prevent application of air to the transient rod unless the cylinder is fully inserted is to prevent pulsing the reactor in the steady-state mode. The interlock to prevent withdrawal of the shim, safety or regulating rod in the pulse mode is to prevent the reactor from being pulsed while on a positive period. The interlock to prevent simultaneous withdrawal of two control rods is to limit reactivity insertion rate from the standard control rods.

3.3 Reactor Primary Tank Water

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

Objective. The objective is to ensure that there is an adequate amount of high quality water in the reactor tank for fuel cooling and shielding purposes, and that the bulk temperature of the reactor tank water remains sufficiently low to guarantee ion exchanger resin integrity.

Specifications.

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

    a. The bulk tank water temperature shall not exceed 60 °C;

    b. The conductivity of the tank water shall be less than 5 μmhos/cm when averaged over a one month period;

    c. The reactor shall not be operated if the tank water level is more than 24 inches below the top lip of the reactor tank, and an alarm which is audible to the reactor operator shall sound when the water level is too low; and

    d. The reactor shall not be operated if the radioactivity of the pool water exceeds the limits of 10 CFR 20 Appendix B Table 3 for radioisotopes with half-lives >24 hours.

    NOTE: These specifications are not required to be met if the reactor fuel has been removed from the tank.

Basis. The bulk water temperature limit is necessary to ensure that the ion exchange resin does not undergo severe thermal degradation. Experience at many research reactor facilities has shown that maintaining the conductivity within the specified limit provides acceptable control of corrosion (NUREG-1537). The minimum water level of no more than 24 inches below the top lip of the reactor tank ensures sufficient cooling water both for normal operation and during the design reactor tank leak of 350 gpm for any aluminum clad fuel to cool to safe levels after a reactor shutdown. This water level (no more than 24 inches below the top lip of the tank) gives approximately 18 feet-4 inches of water above the top grid plate of the core.

3.4 This section intentionally left blank.

3.5 Ventilation and Confinement System

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

Objective. The objective is to ensure that the ventilation and confinement system shall be in operation to mitigate the consequences of possible releases of radioactive materials resulting from reactor operation.

Specifications.

1. The reactor shall not be operated unless a facility ventilation system is operating and the reactor bay pressure is maintained negative with respect to surrounding areas by at least 0.1" water pressure except for short periods of time (not to exceed 2 hours) for system troubleshooting, maintenance and movement of personnel or equipment through open doors, provided the CAM is operating. The normal mode ventilation system is considered operable if:

a. The normal exhaust fan is operating; and

b. The reactor bay is sufficiently confined to allow a minimum differential pressure of 0.1" water column to be maintained by the normal exhaust fan.

2. The reactor bay ventilation system shall operate in the emergency mode, with all exhaust air passing through a HEPA filter, whenever a high level continuous air monitor (CAM) alarm is present due to airborne particulate radionuclides emitted from the reactor or samples in the reactor bay. The emergency mode ventilation system is considered operable if:

a. The emergency exhaust fan is operating; and

b. The reactor bay is sufficiently confined to allow a minimum differential pressure of 0.1" water column to be maintained by the emergency exhaust fan.

3. Movement of irradiated fuel or fueled experiments with significant fission product inventory outside of containers, systems, or storage areas within the reactor bay shall not be performed unless the ventilation system is operating as described in TS 3.5, Specifications 1 and 2. If there is a failure of the ventilation system while movement of these materials is being performed, the material shall be placed in an appropriate location until the ventilation system is made operable.

4. Core or control rod work that could cause a change in reactivity of more than one dollar shall not be performed unless the ventilation system is operating as described in TS 3.5, Specifications 1 and 2. If there is a failure of the ventilation system while this work is being performed, the material that could cause the change in reactivity shall be placed in an appropriate location until ventilation system is made operable.

5. Movement of experiments within the core that could reasonably cause a change of total worth of more than one dollar shall not be performed unless the ventilation system is operating as described in TS 3.5, Specifications 1 and 2. If there is a failure of the ventilation system while movement of these experiments is being performed, the material shall be placed in an appropriate location until the ventilation system is made operable.

Basis. The worst-case maximum total effective dose equivalent is well below the 10 CFR 20 limit for individual members of the public. This has been shown to be true for scenarios where the ventilation system continues to operate during the MHA and where the ventilation system does not operate during the MHA (SAR 13.2.1). Therefore, operation of the reactor for short periods while the reactor bay underpressure is not maintained because of testing or reactor bay open doors, does not compromise

the control over the release of radioactive material to the unrestricted area nor should it cause occupational doses that exceed those limits given in 10 CFR 20.

3.6 This section intentionally left blank.

3.7 Radiation Monitoring Systems and Effluents

3.7.1 Radiation Monitoring Systems

Applicability. This specification applies to the radiation monitoring systems.

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

Specifications.

1. The reactor shall not be operated unless the minimum number of radiation monitoring channels listed in Table 3.4 is operating. Each channel except for the Environmental Dosimeters shall have a readout in the control room and be capable of sounding an audible alarm.

Table 3.4: Minimum Radiation Monitoring Channels*	
Radiation Monitoring Channel	**Number**
Continuous Air Monitor sampling reactor bay air	1
Radiation Area Monitor in reactor bay	1
Environmental Dosimeter outside reactor facility	3
^{41}Ar Monitor sampling the stack exhaust	1

*The Continuous Air Monitor or the Radiation Area Monitor may be out-of-service for up to 2 hours for calibration, maintenance, troubleshooting, or repair. During this out-of-service time, no experiments or maintenance activities shall be conducted which could directly result in alarm conditions (e.g., airborne releases or high radiation levels), and the ventilation system shall be operating. A portable, gamma-sensitive ion chamber, with display visible from the control room, may be utilized as a temporary substitute for the required Radiation Area Monitor (but not for the Continuous Air Monitor) for a period up to 60 days. Calculations may be performed to determine ^{41}Ar releases as a function of reactor operating history as a temporary substitute for the required ^{41}Ar monitor for a period up to 60 days.

Basis. The radiation monitors provide information to operating personnel regarding routine releases of radioactivity and any impending or existing danger from radiation. The alarm setpoints are chosen to be at levels higher than those normally encountered during routine reactor operations. Their operation will

provide sufficient time to evacuate the facility or take the necessary steps to prevent the spread of radioactivity to the surroundings (SAR 11.1.6).

3.7.2 Effluents

Applicability. This specification applies to the release rate of ^{41}Ar.

Objective. The objective is to ensure that the concentration of the ^{41}Ar in the unrestricted areas shall be below the applicable effluent concentration value in 10 CFR 20.

Specifications.

1. The annual average concentration of ^{41}Ar discharged into the unrestricted area shall not exceed 4.8×10^{-6} μCi/ml at the point of discharge.

Basis. If ^{41}Ar is continuously discharged at 4.8×10^{-6} μCi/ml, measurements and calculations show that ^{41}Ar released to the publicly accessible areas under the worst-case weather conditions would result in an annual TEDE of 0.5 mrem. This is only 5% of the applicable limit of 10 mrem. The calculation was performed with the Environmental Protection Agency's Comply code (SAR 11.1.1.1.4).

3.8 Limitations on Experiments

3.8.1 Reactivity Limits

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

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

Specifications.

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

    a. The absolute reactivity worth of any single movable experiment shall be less than $1.00; and

    b. The absolute reactivity worth of any single secured experiment shall be less than $3.00; and

    c. The sum of the absolute reactivity worth for all experiments shall be less than $5.00.

Basis. The worst event which could possibly arise is the sudden removal of a movable experiment immediately prior to, or following, a pulse transient of the maximum licensed reactivity insertion. Limiting the worth of a movable experiment to less than $1.00 will ensure that the additional increase of transient power and temperature is slow enough for the high power scram to be effective and, since this

transient is not a super-prompt pulse, we would not violate the 1 kW Pulse Interlock which prevents entering pulse mode above 1 kW (SAR 14.3.2.3).

The worst event that is considered in conjunction with a single secured experiment is the sudden removal of the experiment while the reactor is operating in a critical condition at a low power level. This is equivalent to pulse-mode operation of the reactor. Hence, the reactivity limitation for a single secured experiment at $3.00 is the same as that of a maximum allowed pulse, although a scram would be initiated much more quickly for the experiment removal accident (SAR 13.2.2.2.1 and 14.3.1.2).

3.8.2 Materials

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

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

Specifications.

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

    a. Explosive materials, such as gunpowder, TNT, or nitroglycerin, in quantities greater than 25 milligrams TNT-equivalent shall not be irradiated in the reactor or irradiation facilities. Explosive materials in quantities less than or equal to 25 milligrams TNT-equivalent may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than half the design pressure of the container;

    b. Each fueled experiment shall be controlled such that the total inventory of ^{131}I-^{135}I in the experiment is no greater than 1.5 curies and the total inventory of ^{90}Sr in the experiment is no greater than 5 millicuries; and

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

Basis. This specification is intended to prevent damage to reactor components resulting from failure of an experiment involving explosive materials (SAR 13.2.6.2). The 1.5-curie limitation on ^{131}I-^{135}I, and the 5 millicurie limit on ^{90}Sr, ensure that in the event of a failure of a fueled-experiment involving total release of the iodine, the dose in the reactor bay and in the unrestricted area will be considerably less than that allowed by 10 CFR 20 (SAR 13.2.6).

3.8.3 Failures and Malfunctions

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

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

Specifications.

1. Where the possibility exists that the failure of an experiment (except fueled experiments) under normal operating conditions of the experiment or reactor, credible accident conditions in the reactor, or possible accident conditions in the experiment could release radioactive gases or aerosols to the reactor bay or the unrestricted area, the quantity and type of material in the experiment shall be limited such that the airborne radioactivity in the reactor bay or the unrestricted area will not result in exceeding the applicable dose limits in 10 CFR 20, assuming that:

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

    b. If the effluent from an irradiation 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;

    c. If the effluent from an irradiation facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of these aerosols can escape; and

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

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

3.9 This section intentionally left blank.

4. Surveillance Requirements

4.0 General

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

Objective. The objective is to verify the operability of systems related to reactor safety.

Specifications.

1. Surveillance requirements may be deferred during reactor shutdown (except TS 4.3 Specifications 1 and 3, and TS 4.7 Specifications 1, 2, 3, and 4). However, they shall be completed prior to reactor startup unless reactor operation is required for performance of the surveillance. Such surveillance shall be performed as soon as practical after reactor startup. Scheduled surveillance which cannot be performed with the reactor operating may be deferred until a planned reactor shutdown.

2. Any additions or modifications to the ventilation system, the core and its associated support structure, the pool or its penetrations, the primary coolant system, the rod drive mechanism or the reactor safety system shall be made and tested to assure that the systems will meet their functional requirements in accordance with manufacturer specifications or specifications reviewed by the ROC. A system shall not be considered operable until after it is successfully tested.

3. The reactor control and safety systems, pool water level alarm, and radiation monitoring systems shall be tested to be operable after the completion of non-routine maintenance of the respective items.

Basis. These specifications relate to changes in reactor systems which could affect the safety of the reactor. These changes will be formally addressed by following the requirements of 10 CFR 50.59. As long as changes or replacements to these systems meet or exceed the original design specifications, then it can be assumed that they meet the presently accepted operating criteria. Additional requirements may be needed, based on the evaluation through the 10 CFR 50.59 process. This specification is not intended to circumvent or replace the regulations in 10 CFR 50.59.

4.1 Reactor Core Parameters

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

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

Specifications.

1. A channel calibration shall be made of the power level monitoring channels by the calorimetric method at least annually.

2. The total reactivity worth of each control rod shall be measured annually or following a change in core or control rod configuration that is expected to change the total reactivity worth of that control rod by more than $0.30 (not including transient fission product poison effects).

3. The maximum reactivity insertion rate of a standard control rod shall be measured annually or following a change in core or control rod configuration that is expected to change the total reactivity worth of that control rod by more than $0.30 (not including transient fission product poison effects).

4. The core shutdown margin shall be determined at least annually and following a change in core or control rod configuration that is expected to change the shutdown margin by more than $0.30 (not including transient fission product poison effects).

5. The core excess reactivity shall be determined annually or following a change in core or control rod configuration that is expected to change the excess reactivity by more than $0.30 (not including transient fission product poison effects).

6. The transient rod and drive mechanism shall be tested and inspected at least annually.

7. Verification of core configuration to include aluminum-clad fuel only in the F and G rings of the core and to have a minimum of 110 elements in the core shall be determined by visual means prior to each day of operation.

8. All fuel elements shall be inspected for damage or deterioration and measured for length and transverse bend at least at quinquennial intervals or if 500 pulses have been performed since the last fuel inspection.

9. For each month during which pulsing is performed, the relationship between peak fuel temperature and inserted reactivity shall be determined.

NOTE: These checks are not required if reactor fuel has been removed from the tank.

Basis. Experience has shown that the identified frequencies will ensure performance and operability for each of these systems or components. Movement of the core components could change the reactivity of the core and thus affect both the core excess reactivity and the shutdown margin, as well as affecting the worth of the individual control rods. Evaluation of these parameters is therefore required after any such movement. Without any such movement, the changes of these parameters over an extended period of time and operation of the reactor have been shown to be small, so that an annual measurement is sufficient to ensure compliance with the specifications. Experience at TRIGA reactors indicates that examination of a five-year cycle is adequate to detect problems. A five-year cycle reduces the handling of the fuel elements and thus reduces the risk of accident or damage due to handling.

4.2 Reactor Control and Safety Systems

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

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

Specifications.

1. The control rods shall be visually inspected for damage or deterioration at least biennially.

2. The scram time shall be measured at least annually or after any work (not including routine limit switch adjustments) is performed on a control rod drive.

3. A channel test of each of the reactor safety system channels in Table 3.2 for the intended mode of operation shall be performed prior to each day's operation or prior to each operation extending more than one day. The same channel tests shall be performed after modifications or repairs to the scram channels to ensure operability of the respective channels.

4. A channel test of items in Table 3.2 and 3.3 shall be performed at least semi-annually, except for those two items required solely for pulse mode operation, which shall be channel tested during each startup for pulse mode operation. The two items required solely for pulse mode operation are the Preset timer scram in Table 3.2 and the control rod interlock in Table 3.3 that prevents withdrawal of any rod except the Transient Rod.

NOTE: These specifications are not required if the reactor fuel has been removed from the tank.

Basis. Inspection of the control rods allows early detection of signs of deterioration indicated by signs of changes of corrosion patterns or of swelling, bending, or elongation.
The channel checks performed daily before operation and after any modifications or repairs provide timely assurance that the systems will operate properly during operation of the reactor.
Experience has shown that the identified frequencies will ensure performance and operability for each of these systems or components.

4.3 Reactor Primary Tank Water

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

Objective. The objective is to ensure that the reactor tank water level and the bulk water temperature monitoring systems are operating and to verify appropriate alarm settings.

Specifications.

1. A channel test of the reactor tank water level alarm setpoint shall be performed at least semi-annually.

2. A channel check of the reactor tank bulk water temperature alarm setpoint shall be performed quarterly. A channel calibration of the reactor tank bulk water temperature system shall be performed at least annually.

3. The reactor tank water conductivity shall be measured monthly. Multiple measurements taken in one month shall be averaged to determine the monthly value.

4. The pool water radioactivity shall be measured at least quarterly.

NOTE: These specifications are not required if the reactor fuel has been removed from the tank.

Basis. Experience has shown that the frequencies of checks on systems which monitor reactor primary water can adequately keep the tank water at the proper level and maintain water quality at such a level to minimize corrosion and maintain safety. Experience at the GSTR shows that the surveillance specification on the conductivity is adequate to detect the onset of degradation of the quality of the pool water in a timely fashion. Experience also indicates that the surveillance specification on pool water level and pool water temperature are adequate to detect losses of pool water in a timely manner and to enable operators to take appropriate action when the coolant temperature approaches the specified limit. The quarterly and annual surveillances of the temperature monitor are also adequate to assure operability of the temperature channel. The pool water level alarm system is a reliable unit and therefore the specification of a semi-annual test is sufficient to assure operability of the pool water level alarm.

4.4 This section intentionally left blank.

4.5 Ventilation and Confinement System

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

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

Specifications.

1. A channel check of the reactor bay ventilation shall be performed prior to each day's operation or prior to each operation extending more than one day.

2. A channel test of the reactor bay ventilation system's ability to automatically switch to the emergency mode upon actuation of the CAM high alarm and to provide a reactor bay minimum differential pressure of 0.1" water column shall be performed quarterly.

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

4.6 This section intentionally left blank.

4.7 Radiation Monitoring System

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

Objective. The objective is to ensure that the radiation monitoring equipment is operating properly and to verify the appropriate alarm settings.

Specifications.

1. A channel check of the radiation area monitor, continuous air monitor, and ^{41}Ar monitor shall be performed monthly.

2. A channel test of the continuous air monitor shall be performed quarterly.

3. A channel calibration of the radiation area monitor and continuous air monitor and ^{41}Ar monitor shall be performed annually.

4. The environmental dosimeters shall be changed and evaluated at least annually.

Basis. Experience has shown that an annual calibration is adequate to correct for any variation in the system due to a change of operating characteristics over a long time span. The frequency of changing and evaluating environmental dosimeters are also adequate to provide the required record based on past experience.

4.8 Experimental Limits

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

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

Specifications.

1. The reactivity worth of an experiment shall be estimated or measured, as appropriate, before routine reactor operation with that experiment to ensure that the limits of TS 3.8.1 are not exceeded.

2. An experiment shall not be installed in the reactor or its irradiation facilities unless a safety analysis has been performed and reviewed for compliance with TS 3.8.2 and TS 3.8.3 by the Reactor Supervisor or ROC in full accord with TS 6.2.3, and the procedures which are established for this purpose.

Basis. Experience has shown that experiments which are reviewed by the staff of the GSTR and the ROC can be conducted without endangering the safety of the reactor or exceeding the limits in the Technical Specifications.

4.9 This section intentionally left blank.

5. Design Features

5.1 Site and Facility Description

Applicability. This specification applies to the U.S. Geological Survey TRIGA Reactor site location and specific facility design features.

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

Specifications.

1. The licensed area shall be the following locations on the Denver Federal Center:

    a. Building 15: Rooms 149 through 152, Rooms 154, 157, 158, B10, B10B, and B11;

    b. Area inside the wrought iron fence and south cooling tower wall that is near the SW corner of Building 15;

    c. Building 10: Room 2.

2. The reactor bay volume shall be a nominal 12000 cubic feet and shall be designed to restrict leakage.

3. The reactor facility shall be equipped with a ventilation system designed to exhaust air and other gases from the reactor bay and release them from vertical level at least 21 feet above ground level.

4. Emergency controls for the ventilation system shall be located in the reactor control room.

Basis. The reactor building and site description are strictly defined (SAR Chapter 2). The facility is designed such that the ventilation system will normally maintain a negative pressure in the reactor bay with respect to the outside atmosphere so that there will be no uncontrolled leakage to the unrestricted environment. Controls for normal and emergency operation of the ventilation system are located in the reactor control room. Proper handling of airborne radioactive materials (in emergency situations) can be conducted from the reactor control room with minimum exposure to operating personnel (SAR 9.1 and 13.2.1).

5.2 Reactor Coolant System

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

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

Specifications.

1. The reactor core shall be cooled by natural convective water flow.

2. The tank water inlet and outlet pipes to the heat exchanger and to the demineralizer shall be equipped with siphon breaks 14 feet above the top of the core or higher.

NOTE: These specifications are not required to be met if the reactor core has been defueled.

Basis.

1. This specification is based on thermal and hydraulic calculations which show that the TRIGA core can operate in a safe manner at power levels up to 1.9 MW with natural convection flow of the coolant water (SAR 4.5.4.5).

2. In the event of accidental siphoning of tank water through inlet and outlet pipes of the heat exchanger or demineralizer system, the tank water level will drop to a level no less than 14 feet from the top of the core (SAR 5.2).

5.3 Reactor Core and Fuel

5.3.1 Reactor Core

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

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

Specifications.

1. The core shall be an arrangement of TRIGA uranium-zirconium hydride fuel-moderator elements positioned in the reactor grid plate.

2. The TRIGA core assembly shall consist of stainless-steel clad fuel elements (8.5 to 12.0 wt% uranium), aluminum-clad fuel elements (8.0 wt% uranium), or a combination thereof.

3. The fuel shall be arranged in a close-packed configuration except for single element positions occupied by in-core experiments, irradiation facilities, graphite dummies, aluminum dummies, stainless steel dummies, control rods, and startup sources. The core may also contain two separated experiment positions in the D through E rings, each occupying a maximum of three fuel element positions.

4. G-ring grid positions may be empty (water filled).

5. The reflector, excluding experiments and irradiation facilities, shall be graphite, water, or a combination of graphite and water. A reflector is not required if the core has been defueled.

Basis.

1. Standard TRIGA cores have been in use for years and their characteristics are well documented. Analytic studies performed at GSTR for a variety of mixed fuel arrangements

indicate that such cores with mixed loadings would safely satisfy all operational requirements (SAR 4.2).

2. The core will be assembled in the reactor grid plate which is located in a tank of light water. Water in combination with graphite reflectors can be used for neutron economy and the enhancement of irradiation facility radiation requirements (SAR 4.2).

5.3.2 Control Rods

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

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

Specifications.

1. The shim and regulating control rods shall have scram capability and contain borated graphite, B_4C powder or boron, with its compounds in solid form as a poison, in aluminum or stainless steel cladding. These rods may incorporate fueled followers.

2. The transient control rod shall have scram capability and contain borated graphite, B_4C powder or boron, with its compounds in a solid form as a poison in an aluminum or stainless steel cladding. The transient rod drive mechanism shall have an adjustable upper limit to allow a variation of reactivity insertions. This rod may incorporate an aluminum-or air-follower.

Basis. The poison requirements for the control rods are satisfied by using neutron absorbing borated graphite, B_4C powder or boron with its compounds in a solid form. These materials must be contained in a suitable clad material such as aluminum or stainless steel to ensure mechanical stability during movement and to isolate the poison from the tank water environment. Control rods (that are fuel-followed) provide additional reactivity to the core and increase the worth of the control rod. The use of fueled-followers has the additional advantage of reducing flux peaking in the water-filled regions vacated by the withdrawal of the control rods. 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 rapid withdrawal from the reactor core which results in a reactor pulse. The nuclear behavior of the air- or aluminum-follower, which may be incorporated into the transient rod, is similar to a void. A more detailed description of the control rods and their properties can be found in SAR 4.2.2.

5.3.3 Reactor Fuel

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

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

Specifications.

1. Aluminum-clad TRIGA fuel. The individual unirradiated aluminum-clad fuel elements shall have the following characteristics:

a. Uranium content: nominally 8.0 wt% with a ^{235}U enrichment of less than 20%;

b. Hydrogen-to-zirconium atom ratio nominally 1 to 1; and

c. Cladding is aluminum of a nominal 0.030 inch thickness.

2. Stainless-steel clad TRIGA fuel. The individual unirradiated standard TRIGA fuel elements shall have the following characteristics:

a. Uranium content: nominal range of 8.5 to 12.0 wt% with a ^{235}U enrichment of less than 20%;

b. Hydrogen-to zirconium atom ratio nominally between 1.6 to 1 and 1.7 to 1; and

c. Cladding is 304 stainless steel of a nominal 0.020 inch thickness.

Basis.

1. A nominal uranium content of 8 wt% in an aluminum-clad TRIGA element is less than the traditional stainless-steel clad element design value of 8.5 wt%. Such a decrease gives a lower power density. The nominal hydrogen-to-zirconium ratio of 1 to 1 could result in a phase change of the ZrH if fuel temperature is allowed to exceed 535 °C. Although this would not necessarily cause a rupture of the fuel cladding, it would cause distortion and stressing of the cladding.

2. A maximum nominal uranium content of 12 wt% in a standard TRIGA element is about 50% greater than the lower-loaded nominal value of 8.5 wt%. Such an increase in loading would result in an increase in power density of less than 50%. An increase in local power density of 50% reduces the safety margin by, at most, 10%. The maximum hydrogen-to-zirconium ratio of 1.7 to 1 could result in a maximum stress under accident conditions to the fuel element cladding of about a factor of 1.5 greater than the value resulting from a hydrogen-to-zirconium ratio of 1.6. However, this increase in the cladding stress during an accident would not exceed the rupture strength of the cladding.

5.4 Fuel Storage

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

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

Specifications.

1. All fuel elements and fueled devices shall be stored in a geometrical array where the k-effective is less than 0.9 for all conditions of moderation.

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

3. If stored in water, the water quality shall be maintained according to TS 3.3, Specification 1.b.

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

6. Administrative Controls

6.1 Organization

Individuals at the various management levels, in addition to being responsible for the policies and operation of the reactor facility, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license, technical specifications, and federal regulations. The minimum qualification for all members of the reactor operating staff shall be in accordance with ANSI/ANS 15.4, "Selection and Training of Personnel for Research Reactors."

6.1.1 Structure

The reactor administration shall be related to the USGS structure as shown in Figure 1.

6.1.2 Responsibility

The following specific organizational levels and responsibilities shall exist:

1. Reactor Administrator (Level 1): The Reactor Administrator is responsible to the USGS Director and is responsible for guidance, oversight, and management support of reactor operations;

2. Reactor Supervisor (Level 2): The Reactor Supervisor reports to the Reactor Administrator and is responsible for directing the activities of the Reactor Operators and Senior Reactor Operators and for the day-to-day operation and maintenance of the reactor;

3. Senior Reactor Operator-in-charge (Level 3): The Senior Reactor Operator-in-charge reports to the Reactor Supervisor. This person is primarily involved in the oversight and direct manipulation of reactor controls, oversight and direct operation and maintenance of reactor related equipment, and oversight of recovery from unplanned shutdowns; and

4. Reactor Operator (Level 4): Other Senior Reactor Operators and Reactor Operators report to Senior Reactor Operator-in-charge and the Reactor Supervisor and are primarily involved in the direct manipulation of reactor controls, monitoring of instrumentation, and direct operation and maintenance of reactor-related equipment.

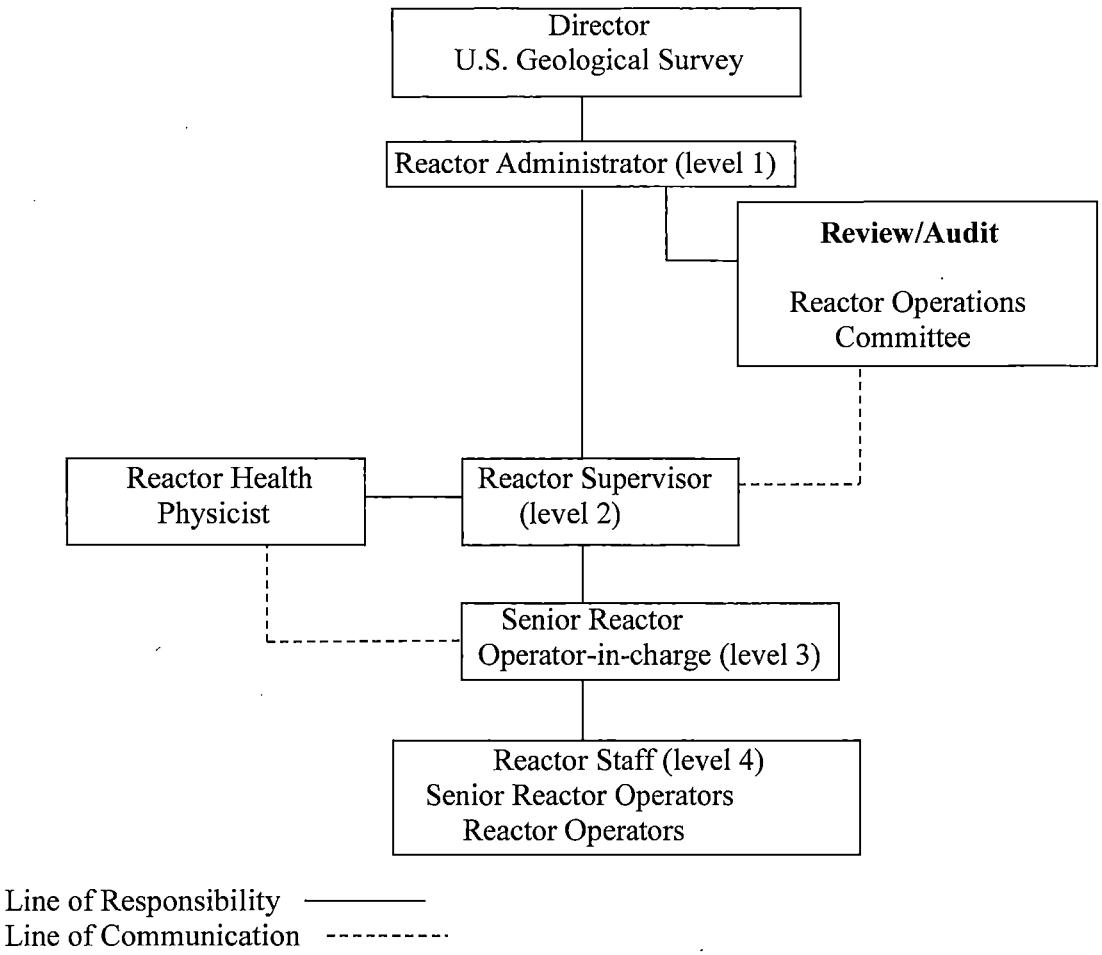


Figure 1: Administrative Structure

6.1.3 Staffing

1. The minimum staffing when the reactor is not secured shall be:

   a. A Licensed Operator in the control room;

   b. A second person present within the Denver Federal Center who is able to carry out prescribed instructions;

   c. If neither of these two individuals is a Senior Reactor Operator, a Senior Reactor Operator shall be readily available on call. Readily available on call means an individual who:

      i. Has been specifically designated and the designation is known to the operator on duty;

      ii. Can be contacted by phone, within 5 minutes, by the operator on duty; and

iii. Is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 minutes or within a 15-mile radius).

d. It is not necessary to have a SRO on call if the Reactor Operator in the control room is a SRO. If the Reactor Operator in the control room is a SRO, a second person shall be available at the facility or on call; and

e. A list of management personnel, radiation personnel, and reactor staff along with their contact information shall be available to the operator on duty.

2. Events requiring the direction of a Senior Reactor Operator

a. Initial approach to critical after each completed shutdown checklist;

b. Initial approach to power after each completed shutdown checklist;

c. All fuel or control rod relocations within the reactor core region;

d. Relocation of any in-core components (other than normal control rod movements) or experiment with a reactivity worth greater than one dollar; or

e. Recovery from an unscheduled shutdown or an unscheduled significant (>50%) power reduction.

6.1.4 Selection and Training of Personnel

The selection, training and requalification of operations personnel shall follow the guidance of ANSI/ANS 15.4, " Selection and Training of Personnel for Research Reactors."

6.2 Review and Audit

The ROC shall meet at least semi-annually for the purpose of providing their primary responsibility of review and audit of the safety aspects of reactor facility operations.

6.2.1 Composition and Qualifications

The ROC shall be composed of at least four voting members, including the Chairman. All members of the Committee shall be knowledgeable in subject matter related to reactor operations. To expedite Committee business, a Committee Chairman may be appointed. The Chairman of the ROC shall be listed by name on the Committee roster.

The Committee shall be appointed by the USGS Director. No definite term of service shall be specified; but should a vacancy occur in the Committee, the Director shall appoint a replacement. The remaining members of the Committee shall be available to assist the Director in the selection of new members. The Reactor Supervisor shall be an ex-officio member of the Committee, and the Reactor Supervisor shall be the only non-voting member of the Committee. The ROC shall report to the Reactor Administrator.

6.2.2 Charter and Rules

The ROC consists of USGS members and non-USGS members, and the Committee must meet at least semi-annually.

The review and audit functions shall be conducted in accordance with an established charter for the Committee as written in the USGS Manual. Dissemination and review of Committee minutes shall be done within 60 days of each respective Committee meeting.

A quorum for review, audit, and approval purposes shall consist of not less than one-half of the voting membership where the operating staff does not constitute a majority. The Chairperson or an alternate must be present at all meetings in which the official business of the committee is being conducted. Approvals by the committee shall require an affirmative vote by a majority of the non-USGS members present and an affirmative vote by a majority of the USGS members present.

6.2.3 Review Function

The following items shall be reviewed:

1. Determinations that proposed changes in the facility, procedures, tests, or experiments are allowed without prior authorization by the responsible authority, as detailed in 10 CFR 50.59;

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

3. All new experiments or classes of experiments that could have reactivity or safety significance;

4. Proposed changes in technical specifications, license, or charter;

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

6. Operating abnormalities having safety significance;

7. Reportable occurrences listed in TS 6.7.2; and

8. Audit reports.

A written report or minutes of the findings and recommendations of the review shall be submitted to the Reactor Administrator and the ROC within 3 months after the review has been completed.

6.2.4 Audit Function The audit function shall include selective (but comprehensive) examination of operating records, logs, other documents, and the reactor facility. Discussions with cognizant personnel

and observation of operations should be used also as appropriate. In no case shall the individual immediately responsible for the area perform an audit in that area. The following items shall be audited:

1. Facility operations for conformance to the technical specifications and applicable license or charter conditions: at least once per calendar year (interval between audits not to exceed 15 months);

2. The retraining and requalification program for the operating staff: at least once every other calendar year (interval between audits not to exceed 30 months);

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

4. The reactor facility emergency plan, implementing procedures, and security plan: at least once every other calendar year (interval between audits not to exceed 30 months).

Deficiencies uncovered that affect reactor safety shall immediately be reported to the Reactor Administrator. A written report of the findings of the audit shall be submitted to the Reactor Administrator and the ROC within 3 months after the audit has been completed.

6.3 Radiation Safety

The Reactor Supervisor, in coordination with the Reactor Health Physicist, shall be responsible for implementation of the radiation safety program. The requirements of the radiation safety program are established in 10 CFR 20. The program should use the guidelines of the ANSI/ANS 15.11-2009, ``Radiation Protection at Research Reactor Facilities.''

6.4 Procedures

Written operating procedures shall be prepared, reviewed, and approved prior to initiating any of the activities listed in this section. The procedures shall be reviewed by the ROC and approved by the Reactor Supervisor, and such reviews and approvals shall be documented in a timely manner. Substantive changes to the procedures shall be made effective only after documented review by the ROC and approval by the Reactor Supervisor. Minor modification to the original procedures that do not change their original intent may be made by the Reactor Supervisor. Temporary deviations from the procedures may be made by the responsible SRO or Reactor Supervisor in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported within 24 hours or the next working day to the Reactor Supervisor. Procedures shall be in effect and in use for the following items:

1. Surveillance checks, calibrations, and inspections that are required by Technical Specifications or those that may have an effect on reactor safety;

2. Startup, operation and shutdown of the reactor;

3. Implementation of emergency and security plans;

4. Core changes and fuel movement;

5. Performing maintenance on major components that could affect reactor safety;

6. Administrative controls for operations, maintenance, and experiments that could affect reactor safety or core reactivity;

7. Radiation protection, including ALARA requirements; and

8. Use, receipt and transfer of licensed radioactive material, if appropriate.

6.5 Experiment Review and Approval

1. All experiments proposed for the reactor will be either Class I or Class II experiments and shall be reviewed in accordance with the 10 CFR 50.59 review requirements. The review and classification of the proposed experiments shall be the responsibility of the Reactor Supervisor.

2. Class I experiments include all experiments that have been run previously or that are minor modifications to a previous experiment. These are experiments which involve small changes in reactivity, no external shielding changes, and/or limited amounts of radioisotope production. The Reactor Supervisor has the authority to approve the following:

a. Experiments for which there exists adequate precedence for assurance of safety;

b. Experiments which represent less than that amount of reactivity worth necessary for prompt criticality; or

c. Experiments in which any significant reactivity worth is stable and mechanically fixed, that is, securely fastened or bolted to the reactor structure.

3. Class II experiments include all new experiments and major modifications of previous experiments. These experiments must be reviewed and approved by the ROC before being run. These experiments may involve larger changes in reactivity, external shielding changes, and/or larger amounts of radioisotope production.

6.6 Required Actions

6.6.1 Actions to Be Taken in Case of Safety Limit Violation

In the event a safety limit is exceeded:

1. The reactor shall be shutdown and reactor operation shall not be resumed until authorized by the NRC;

2. An immediate notification of the occurrence shall be made to the Reactor Supervisor, Reactor Administrator, ROC; and

3. A report, and any applicable follow-up report, shall be prepared and submitted to the NRC. The report shall describe the following:

a. Applicable circumstances leading to the violation including, when known, the cause and contributing factors;

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

c. Corrective action to be taken to prevent recurrence.

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

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

1. The reactor shall be secured and the Reactor Supervisor notified;

2. Operations shall not resume unless authorized by the Reactor Supervisor;

3. The ROC shall review the occurrence at their next scheduled meeting; and

4. Where appropriate, a report shall be submitted to the NRC in accordance with TS 6.7.2.

6.7 Reports

6.7.1 Annual Operating Report

An annual report covering the previous calendar year shall be created and submitted, no later than March 31 of the year following the report period, by the Reactor Supervisor to the NRC consisting of:

1. A brief summary of operating experience including the energy produced by the reactor and the hours the reactor was critical;

2. The number of unplanned shutdowns, including corrective actions taken (when applicable);

3. A tabulation of major preventative and corrective maintenance operations having safety significance;

4. A brief description, including a summary of the safety evaluations, of changes in the facility or in procedures and of tests and experiments carried out pursuant to 10 CFR 50.59;

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

6. A summarized result of environmental surveys performed outside the facility;

7. A summary of exposures received by facility personnel and visitors where such exposures are greater than 25% of that allowed; and

8. Results of fuel inspections (when performed).

6.7.2 Special Reports

In addition to the requirements of applicable regulations, and in no way substituting therefore, reports shall be made by the Reactor Supervisor to the NRC as follows:

1. A report within 24 hours by telephone, digital submission, or fax to the NRC Operations Center followed by a report in writing to the NRC, Document Control Desk, Washington, D.C. within 14 days that describes the circumstances associated with any of the following:

    a. Any release of radioactivity above applicable limits into unrestricted areas, whether or not the release resulted in property damage, personal injury, or exposure;

    b. Any violation of a safety limit;

    c. Operation with the actual safety system setting less conservative than the LSSS;

    d. Operation in violation of a Limiting Condition for Operation;

    e. Malfunction of a required reactor safety system component which renders or could render the system incapable of performing its intended safety function unless the failure is discovered during maintenance tests or periods of reactor shutdown;

    f. Any unanticipated or uncontrolled change in reactivity greater than $1.00. Reactor trips resulting from a known cause are excluded;

    g. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy causes or could have caused the existence or development of a condition which results or could result in operation of the reactor outside the specified safety limits; or

    h. Abnormal and significant degradation in reactor fuel, cladding, or coolant boundary

2. A report within 30 days in writing to the NRC, Document Control Desk, Washington, D.C. of:

    a. Permanent changes in the facility organization involving Level 1-2 personnel; or

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

6.8 Records

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

1. Normal reactor operation (but not including supporting documents such as checklists, data sheets, etc., which shall be maintained for a period of at least two years);

2. Principal maintenance activities;

3. Reportable occurrences;

4. Surveillance activities required by the Technical Specifications;

5. Reactor facility radiation and contamination surveys;

6. Experiments performed with the reactor;

7. Fuel inventories, receipts, and shipments;

8. Approved changes to the operating procedures; and

9. ROC meetings and audit reports.

6.8.2 Records to be Retained for at Least One Operator License Term

1. Records of retraining and requalification of Reactor Operators and Senior Reactor Operators shall be retained for at least one license term; and

2. Records of retraining and requalification of licensed operators shall be maintained while the individual is employed by the licensee, or until that operator's license is renewed, whichever is shorter.

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

1. Gaseous and liquid radioactive effluents released to the environs;

2. Offsite environmental monitoring surveys;

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

4. Radiation exposures for all personnel monitored; and

5. Drawings of the reactor facility.

INTERAGENCY AGREEMENT

BETWEEN

UNITED STATES DEPARTMENT OF ENERGY

AND

UNITED STATES GEOLOGICAL SERVICES

FOR ENRICHED URANIUM

SNM INTERAGENCY AGREEMENT NUMBER: 1012
AMENDMENT: 0003

THIS INTERAGENCY AGREEMENT (sometimes referred to as the "Agreement"), entered into this 30th day of September, 2015 between the **UNITED STATES DEPARTMENT OF ENERGY** (hereinafter called "DOE") and **U.S. GEOLOGICAL SERVICES** (herein after called the "Agency"), an executive department or independent establishment of the Government of the United States of America or a bureau or office thereof;

WHEREAS, the parties hereto desire to establish the terms and conditions applicable to the distribution of special nuclear material to the Agency, pursuant to the Atomic Energy Act of 1954, as amended, whether ordered and received directly from a DOE facility or transferred from a lessee of DOE;

NOW, THEREFORE, the parties hereto mutually agree as follows:

ARTICLE 1-DEFINITIONS

As used in this Agreement, the term:

a. *Act* means the Atomic Energy Act of 1954, as amended.

b. *Base charge* means the dollar amount per unit of normal or depleted uranium or special nuclear material in standard form and specification in effect as of the time of any particular transaction under this Agreement takes place.

c. *Blending* means the altering of the isotopic composition of a quantity of an element by means other than through the irradiation of a material in a nuclear reactor.

d. *DOE* means the Department of Energy, exercising the authority and performing the functions of the Secretary, the statutory head of that agency, or any duly authorized representative thereof, including the Contracting Officer except for the purpose of deciding an appeal under the article entitled "Disputes."

e. *DOE facility* means a laboratory, plant, office, or other establishment operated by or on behalf of DOE.

f. *DOE established specifications* means the specifications for purity and other physical or chemical properties of normal or depleted uranium or special nuclear material.

b. Agency may, upon request to DOE, observe the weighing of the material and the taking of samples by DOE. The dates and places for the weighing and sampling will be established by DOE and communicated to Agency upon receipt of Agency request.

ARTICLE 15-TRANSFER OF MATERIAL

Transfer of material by Agency to a lessee with the approval of DOE as provided in this Agreement shall not have the effect of relieving Agency of any obligation hereunder, except as to return of or payment for material so transferred.

ARTICLE 16-OTHER CONTRACTS AND AGREEMENTS

This Agreement contemplates the possibility of separate agreements between Agency and DOE with respect to materials which are subject to this Agreement, which may provide for suspension, termination, or revision of matters hereunder; and for reimbursement of charges incurred pursuant to this Agreement. Except as provided in such agreements, Agency obligations under this Agreement for material subject to this Agreement shall continue notwithstanding the existence of such separate agreement or agreements.

ARTICLE 17-NOTICES

a. Any notices required by this Agreement of Agency shall be submitted in writing to DOE addressed to:

Manager for Nuclear Energy Oak Ridge Site Office
U.S. Department of Energy
Post Office Box 2001
Oak Ridge, Tennessee 37831

b. Any notices required by this Agreement of the DOE shall be submitted in writing to Agency addressed to:

Mr. Brycen Roy
Reactor Manager
United States Geological Services
Post Office Box 25046
Mail Stop 974
Denver, Colorado 80225-0046

On Behalf of the
U.S. GEOLOGICAL SERVICES

BrycenRRoy

Brycen Roy, Acting Reactor Manager

8/1/16

Date

On Behalf of the
U.S. DEPARTMENT OF ENERGY

Jamie Ford

Digitally signed by Jamie Ford
DN: cn=Jamie Ford, o=USDOE, ou, email=jamie.ford@science.doe.gov, c=US
Date: 2016.08.10 16:04:06 -04'00'

Jamie Ford, Contracting Officer

8/10/16

Date

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 the initial fission neutron source distribution used the   1 source points that were input on the ksrc card.
 the criticality problem was scheduled to skip  15 cycles and run a total of 1015 cycles with nominally     2000 neutrons per cycle.
 this problem has run  15 inactive cycles with    30315 neutron histories and 1000 active cycles with     2002795 neutron histories.

 this calculation has completed the requested number of keff cycles using a total of    2033110 fission neutron source histories.
 all cells with fissionable material were sampled and had fission neutron source points.

 the results of the w test for normality applied to the individual collision, absorption, and track-length keff cycle values are:

          the k( collision) cycle values appear normally distributed at the 95 percent confidence level
          the k(absorption) cycle values appear normally distributed at the 95 percent confidence level
          the k(trk length) cycle values appear normally distributed at the 95 percent confidence level

 ------------------------------------------------------------------------------------------------------------------------------------
 |                                                                                                                                  |
 | the final estimated combined collision/absorption/track-length keff = 0.88149 with an estimated standard deviation of 0.00059   |
 |                                                                                                                                  |
 | the estimated 68, 95, & 99 percent keff confidence intervals are 0.88090 to 0.88208, 0.88031 to 0.88266, and 0.87993 to 0.88305 |
 |                                                                                                                                  |
 | the final combined (col/abs/tl) prompt removal lifetime = 1.1250E-04 seconds with an estimated standard deviation of 8.2938E-08 |
 |                                                                                                                                  |
 | the average neutron energy causing fission = 2.3367E-02 mev                                                                     |
 | the energy corresponding to the average neutron lethargy causing fission = 8.0149E-08 mev                                       |
 |                                                                                                                                  |
 | the percentages of fissions caused by neutrons in the thermal, intermediate, and fast neutron ranges are:                        |
 |        (<0.625 ev):  92.78%         (0.625 ev - 100 kev):   6.19%          (>100 kev):   1.04%                                |
 |                                                                                                                                  |
 | the average fission neutrons produced per neutron absorbed (capture + fission) in all cells with fission = 1.6995E+00            |
 | the average fission neutrons produced per neutron absorbed (capture + fission) in all the geometry cells = 8.9779E-01            |
 |                                                                                                                                  |
 | the average number of neutrons produced per fission = 2.437                                                                     |
 |                                                                                                                                  |
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 the estimated average keffs, one standard deviations, and 68, 95, and 99 percent confidence intervals are:

       keff estimator       keff     standard deviation      68% confidence        95% confidence        99% confidence       corr

            collision     0.88097         0.00085          0.88012 to 0.88181    0.87928 to 0.88265    0.87873 to 0.88320
           absorption     0.88164         0.00062          0.88102 to 0.88226    0.88041 to 0.88287    0.88001 to 0.88327
         track length     0.88097         0.00086          0.88010 to 0.88183    0.87925 to 0.88268    0.87869 to 0.88324
          col/absorp      0.88149         0.00059          0.88090 to 0.88208    0.88031 to 0.88267    0.87992 to 0.88305    0.4608
         abs/trk len      0.88149         0.00059          0.88090 to 0.88208    0.88031 to 0.88266    0.87993 to 0.88304    0.4323
         col/trk len      0.88097         0.00085          0.88012 to 0.88181    0.87928 to 0.88265    0.87873 to 0.88320    0.9723
       col/abs/trk len    0.88149         0.00059          0.88090 to 0.88208    0.88031 to 0.88266    0.87993 to 0.88305
```