TEXAS A&M ENGINEERING
EXPERIMENT STATION

NUCLEAR SCIENCE CENTER

March 2, 2015 2015-0013

Document Control Desk
ATTN: Geoffrey Wertz
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Response to U.S. NRC Request for Additional Information Regarding the Renewal of Facility Operating License No. R-83 for the Nuclear Science Center TRIGA Reactor (TAC No. ME 1584), from the Texas A&M University System, Texas Engineering Experiment Station, Nuclear Science Center Reactor (NSCR, License No. R-83, Docket 50-128)

To Whom It May Concern:

The Texas A&M University System, Texas Engineering Experiment Station (TEES), Nuclear Science Center (NSC, License No. R-83) operates a LEU, 1MW, TRIGA reactor under timely renewal. In December, 2003 the NSC submitted a Safety Analysis Report (SAR) as part of the license renewal process. In December, 2005 a conversion SAR (Chapter 18) was submitted resulting in an order to convert from the USNRC. In July 2009, the NSC submitted an updated SAR, dated June 2009, to the USNRC. This updated 2009 version of our SAR incorporated the information from the conversion SAR and the startup of the new LEU reactor core. On November 5, 2014 the U.S. NRC submitted a Request for Additional Information as a part of the review process. Attached is our reply to this request.

If you have any questions, please contact Dr. Sean McDeavitt or Mr. Jerry Newhouse at 979-845-7551.

A020
NRR

TEL. 979.845.7551 | FAX 979.862.2667
nsc.tamu.edu
1095 Nuclear Science Rd. | 3575 TAMU | College Station, TX 77843-3575

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 2, 2015.

Dr. Sean M. McDeavitt
Director, Nuclear Science Center
Associate Professor of Nuclear Engineering
Department of Nuclear Engineering
Dwight Look College of Engineering
Texas A&M University
mcdeavitt@tamu.edu

Xc: 2.11/Central File
Duane Hardesty, USNRC Project Manager

Texas A&M Engineering Experiment Station
Nuclear Science Center
License No. R-83
Docket No. 50-128

Responses to the U.S. Nuclear Regulatory Commission's Requests for Information 1 – 57 Dated November 5, 2014

Submitted March 2, 2015

Texas A&M Engineering Experiment Station
Nuclear Science Center
License No. R-83
Docket No. 50-128

1. *The TEES/TAMUS updated SAR, dated May 2011 (ADAMS Accession No. ML111950376), Section 11.2.2, "Radioactive Liquid Waste," provides information about radioactive liquid waste, but does not describe the types or quantities of chemicals that may be used to treat the primary or secondary coolant systems, or used in the conduct of experiments. NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," Chapter 11.2.3, "Release of Radioactive Waste," provides guidance that the use and disposal of chemicals in the facility should be described. Provide a description of the TEES/TAMUS use of chemicals and their disposal, or justify why no additional information is necessary.*

NSC Response: The TEES/TAMUS updated SAR, Section 11.2.2, "Radioactive Liquid Waste," reads "Low-level liquid waste originates from four primary sources at the Nuclear Science Center. These sources are: floor drains, laundry, showers, and laboratories on the lower research level; the demineralizer room filter and ion bed; condensate from air handling units on mechanical chase; and the valve pit sump in cooling equipment room." It does not describe in detail or provide examples of what radioactive waste is expected to flow through those sources.

The expected waste from the lower research level sources and the valve pit sump are the same: Low-level byproduct material that is either in solution or suspension in liquid water. Pursuant to 10 CFR 20.2003 "Disposal by release into sanitary sewerage," waste that is in suspension is filtered from the liquid water prior to release, leaving only material that is in solution.

Liquid waste is generated from the demineralizer room filter and ion bed when the filter and ion bed is regenerated. Throughout this process, approximately 30,000 gallons of liquid water pass through the filter and ion bed, and go to the liquid waste holdup tanks. Again, the expected waste is low-level byproduct material that is either in solution or suspension in liquid water. Pursuant to 10 CFR 20.2003 "Disposal by release into sanitary sewerage," waste that is in suspension is filtered from the liquid water prior to release, leaving only material that is in solution.

While the mechanism for generating radioactive waste in liquid condensate in our air handling units is different than the demineralizer room filter and ion bed, the expected waste is the same: low-level byproduct material. While this condensate currently flows to the liquid waste holdup tanks, we are investigating methods to regularly sample it and potentially divert it for use in pool fills.

As stated in the SAR, all of this liquid waste flows through the same path: to the hold-up tanks and then the sanitary sewer system. However, it is not clear in the SAR that 10 CFR 20.2003 is followed. It reads that releases must be below 10 CFR 20, Appendix B, Table 3 limits, but that is only one portion of 10 CFR 20.2003. This response serves as clarification that the entirety of

10 CFR 20.2003 is and will continue to be satisfied by radioactive liquid waste disposal by release into sanitary sewerage at the NSC.

No chemicals are used to treat the primary coolant system. The water in this system flows through a filtration system to maintain purity, but no chemicals are added. Three chemicals are used to treat the secondary coolant system: 93% sulfuric acid to maintain pH, 12.5% sodium hypochlorite (bleach, approximately double concentration of home cleaning agents) as a biocide, and ChemCal Cooling Water Treatment 1040 as a scale inhibitor. These chemicals are highly diluted in the water of the secondary coolant system. Water from the secondary coolant system is periodically "blown down" to the sanitary sewerage and replaced with fresh city water to maintain acceptable conductivity (purity). These disposals are within university permit limits via the university water treatment plant. This path is the only disposal path for these chemicals.

Two chemicals are used to regenerate the ion exchange resin bed: 93% sulfuric acid, and caustic soda (sodium hydroxide). These chemicals are heavily diluted in water, approximately 20,000 gallons, and flows to the liquid waste holdup tanks. There, the pH is neutralized, as well as the solution going through the normal sampling process, and is then discharged to the sanitary sewerage. These disposals are within university permit limits via the university water treatment plant.

Small volumes of standard laboratory grade chemicals are used in chemical labs. These laboratory chemicals are disposed through an established procedure with the university Environmental Health and Safety office.

2. *The TEES/TAMUS updated SAR, Section 2.2.1.2, "Boundary and Zone Area Maps," indicated that the NSC site includes a linear accelerator and associated laboratory. NUREG-1537, Part 1, Chapter 1.4, "Shared Facilities and Equipment," provides guidance that the SAR should discuss any effect of the shared facility on the subject (licensed) facility. The TEES/TAMUS updated SAR does not provide any information relative to the licensing or safety impacts of the linear accelerator and associated laboratory to the NSC TRIGA reactor. Provide a description of the licensing and safety impacts of the linear accelerator or justify why no additional information is necessary.*

NSC Response: The linear accelerator and associated laboratory should have no licensing or safety impact on the NSC TRIGA reactor. The accelerator is approximately 45 feet from the nearest required reactor support facility, the liquid waste hold up tanks, and approximately 125 feet from the confinement building. The maximum accident at the accelerator and associated laboratory is no more severe than in any common research laboratory or building, such as those located within the confinement building. Given the distance from the confinement building and reactor support equipment, such accidents should not affect the NSC TRIGA reactor.

The building that houses the accelerator is downstream from the confinement building in electricity distribution, and it has its own transformer to prevent feedback from affecting upstream facilities. Similarly, the building that houses the accelerator is downstream from the confinement building in water distribution, and the valve can be closed to prevent leaks from affecting upstream facilities.

The linear accelerator is regulated by the State of Texas through Texas A&M University. However, because it is on the NSC site, changes in the accelerator facility are reviewed in accordance with 10 CFR 50.59.

3. *The TEES/TAMUS proposed Technical Specification (TS) 5.3, "Reactor Core," Specification 2 (ADAMS Accession No. ML12321A321), states, in part, that the reflector includes heavy water. Neither the TEES/TAMU updated SAR nor the Basis for TS 5.3, Specification 2, describes heavy water as a reflector material. NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 3.1, "Reactor Core Parameters," Item (4), "Core Configurations," provides guidance that the reflector material be described in the TSs. Provide a revised description of the reflector material in TS 5.3, Specification 2 or justify why no change is necessary.*

NSC Response: While heavy water is not currently used as a reflector material, it has been historically. Its use was evaluated in Experiment Authorization 20. The safety discussion is, in part, reprinted below:

> During handling of the D_2O, the initially tritium loaded D_2O and the continued production of tritium in the D_2O device is the major radiation hazard associated with this experiment. In the event of water leakage indication a gas sample shall be taken from inside the outer box through the sample tubes on the device and analyzed for tritium before venting or purging. (**Note:** a previous section of the experiment authorization describes the D_2O holder. It is essentially a robust and monitored double walled box.) HP authorization and supervision shall be required before any entries/penetrations of the inner box of the irradiation device and all applicable HP procedures shall be followed.
>
> The only anticipated effect the device could have on the reactor during operation would be the reactivity effect caused by the failure of the inner box containing D_2O, placing the D_2O two inches closer to the reactor. The computer program Exterminator-2 has been run for such a scenario and the results indicate no significant reactivity effect. In the event of a total failure of the device resulting in the release of the entire contents of D_2O, the D_2O would be so diluted by pool water as to have no significant reactivity effect on the reactor. Subsequent experiments have measured the device to be worth less than $0.40.

Any reintroduction of heavy water as a reflector will go through the known safety analysis. Any departure from what was done historically will be evaluated in accordance with 10 CFR 50.59.

The Basis for the TEES/TAMUS proposed Technical Specification 5.3, "Reactor Core," Specification 2 will be revised to describe heavy water as a reflector material.

> Revision: The core will be assembled in the reactor grid plate that is located in a pool of light water. Light water in combination with graphite or heavy water reflectors can be used for neutron economy and the enhancement of experimental facility radiation requirements.

4. *The TEES/TAMUS proposed TS 5.3, "Reactor Core," Specification 4, states, in part, that the instrumented fuel element shall be located adjacent to the central bundle. The Basis for TS 5.3, Specification 4, refers to the Basis of TS.2.2, "Limiting Safety System Setting" (LSSS). However, the Basis of TS 2.2 does not provide a description for the location of the instrumented fuel element. NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 1.2.2, "Format," provides guidance that the format should include the basis for each specification. Provide a basis for TS 5.3, Specification 4, which describes the location of the instrumented fuel element, or justify why no change is necessary.*

NSC Response: This question is linked with question 52. Question 52 rearranges the contents of TEES/TAMUS proposed Technical Specifications 3.1.4, "Reactor Core Configuration," and 5.3, "Reactor Core." The specification referenced in this question has been moved to Specification 3.1.4 and will be referred to in that way.

The Basis for the TEES/TAMUS proposed Technical Specification 3.1.4, "Reactor Core Configuration," Specification 3 will be revised to include the required information.

> Revision: SAR 13.3.1 Steady State Mode provides an evaluation of the LSSS in steady state mode. It states in part, "The location of the fuel cluster containing the instrumented fuel element shall be chosen to be as close as possible to the hottest fuel element in the core." The location(s) as close as possible to the hottest fuel element in the core are those adjacent to the central bundle with the exception of the corner positions. These adjacent positions are: C5 east, D4 north, E5 west, D6 south.

5. *The TEES/TAMUS proposed TS 1.3, "Definitions," Reference Core Condition, provides a value of ~$0.30 for xenon reactivity (negligible). Given that the reactivity required to satisfy TS 3.1.3, "Shutdown Margin" (SDM), is ~$0.50, with Specification 3 requiring the reactor in the reference core condition, the resulting SDM reactivity could be as low as ~$0.20, which is not consistent with the guidance in NUREG-1537, which provides a value of ~$0.50 SDM reactivity. Revise the TS 1.3 definition of Reference Core Condition for Xenon reactivity that results in a SDM reactivity that is consistent with the SDM guidance in NUREG-1537, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 1.3, "Definitions," Reference Core Condition will be revised.

Revision: The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible.

6. *The TEES/TAMUS proposed TS 4.1.3, "Shutdown Margin," Specification, states, in part, that the reactivity worth shall be determined following significant core configuration and/or control rod changes. However, significant core configuration is not defined. NUREG-1537, Chapter 14, Part 1, Appendix 14.1, Section 4.1, "Reactor Core Parameters," provides guidance that reactivity measurement should be done following changes in the core, in-core experiments, or control rods. Revise TS 4.1.3 to follow the guidance in NUREG-1537, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 4.1.3, "Shutdown Margin," Specification will be revised to follow the guidance of NUREG-1537, Chapter 14, Part 1, Appendix 14.1, Section 4.1, "Reactor Core Parameters."

Revision: The reactivity worth of each control rod and the shutdown margin shall be determined annually and following changes in the core, in-core experiments, or control rods.

7. *The TEES/TAMUS proposed TS 4.1.6, "Maximum Excess Reactivity," states, in part, that the excess reactivity shall be determined biennially and following core configuration and/or control rod changes. NUREG-1537, Chapter 14, Appendix 14.1, Section 4.1, "Reactor Core Parameters," Item (1), "Excess Reactivity," provides guidance that the measurement should be performed annually. Additionally, the use of the "and/or" in the TS specification is unclear. Revise TS 4.1.6 to follow the guidance in NUREG-1537, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 4.1.6, "Maximum Excess Reactivity," will be revised both to follow the guidance of NUREG-1537, Chapter 14, Appendix 14.1, Section 4.1, "Reactor Core Parameters," Item (1), "Excess Reactivity," and to remove the ambiguity of the "and/or."

> Revision: The excess reactivity shall be determined annually and following changes in the core, in-core experiments, or control rods for which the predicted change in reactivity exceeds the absolute value of the specified shutdown margin.

8. *The TEES/TAMUS proposed TS 4.2.2, "Reactor Safety Systems and Interlock," Specification 3, and TS 4.8.3, "Primary Coolant Temperature," Specification 2, provide requirements for calibration of the pool temperature measuring channel, and appears redundant, which could complicate TS compliance. Review TS 4.2.2, Specification 3, and TS 4.8.3, Specification 2, and revise to eliminate unnecessary redundancy, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 4.2.2, "Reactor Safety Systems and Interlock," Specification 3 will be deleted to remove the redundancy in pool temperature measuring channel calibration. The specification for this calibration will only be found in TS 4.8.3, Specification 2.

9. *The TEES/TAMUS proposed TS 4.1.2, "Pulse Mode Operation," Specification, states, in part, that the reactor shall not be declared operation for pulsing until such pulse measurements are performed. However, the TS does not provide a requirement as to the acceptability of the measurements. NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 4.1, "Reactor Core Parameters," Item (3), "Pulse Limits," provides guidance that the fuel temperature and pulse reactivity relationship should be determined to be acceptable. Revise TS 4.1.2 to include a requirement that the results of the pulse measurements are acceptable, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 4.1.2, "Pulse Mode Operation," Specification will be revised to follow the guidance of NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 4.1, "Reactor Core Parameters," Item (3), "Pulse Limits."

> Revision: The reactor shall be pulsed semiannually to compare fuel temperature measurements and core pulse energy with those of previous pulses of the same reactivity value. The reactor shall not be declared operational for pulsing until such pulse measurements are performed and are determined to be acceptable.

10. *The TEES/TAMUS proposed TS 4.2.3, "Scram Time," states, in part, that the scram time shall be measured annually. However, the guidance in the American National Standard Institute/American Nuclear Society (ANSI/ANS)-15.1-2007, "The Development of Technical Specifications for Research Reactors," includes additional criteria to perform the surveillance following work done to the rods or rod drive system. Revise TS 4.2.3 to include the additional surveillance criteria in ANSI/ANS-15.1-2007, or justify why no change is needed.*

NSC Response: The TEES/TAMUS proposed TS 4.2.3, "Scram Time," Specification 4 will be revised to follow the guidance of ANSI/ANS-15.1-2007.

Revision: The scram time shall be measured annually or whenever any work is done on the rods or the rod drive system.

11. *The TEES/TAMUS response to RAI 4 by letter dated November 21, 2011 (a redacted version can be found in ADAMS Accession No. ML113410067), provided a calculated radiation dose estimate to a worker in confinement for 2000 hours from Argon-41 (Ar-41) of 416 milli-Roentgens Equivalent Man using the assumption that the TS limit of 30 curies of Ar-41 was released uniformly over a year resulting in an average concentration of 2.5E-7 micro-curies per milliliter. As such, the calculated dose estimate may not represent an upper limit as the production and release of Ar-41 results from the operation of the reactor, which is not typically operated uniformly over a year. For example, a better estimate of the maximum dose to a radiation worker from the Ar-41 released during reactor operations would assume the 30 curies of Ar-41 was released uniformly during the 2000 hours that the worker was in confinement. Provide a revised dose estimate for a worker in confinement using a concentration of Ar-41 that could be released during reactor operation, or justify why no change is necessary.*

NSC Response:
At a stack rate of 8000 ft^3/minute and with 30 Curies released uniformly over the course of 2000 hrs. The dose estimate is as follows: It is assumed that the concentration in the building is equal to the concentration in the stack.

Dose to an occupational worker in confinement for 2000 hrs.

$$Stack\ rate * time = concentration$$

$$\left(\frac{Ci\ amount}{hr\ concentration}\right) * converstion\ factor = concentration\ in\ \mu Ci/cc/hr$$

$$\left(\frac{\mu Ci/cc/hr}{Ar^{41} DAC}\right) = per\ DAC$$

$$per\ DAC * time = DAChrs$$

$$\frac{DAChrs}{2000hrs} * occupational\ limit = dose$$

$$Dose * W_T = Whole\ body\ dose$$

$$(W_T = tissue\ weighting\ factor = 1.0\ for\ whole\ body)$$

$$8000\ ft^3 * 28316.8466\ cc\ per\ ft^3 = 2.27x10^8\ cc$$

$$(Converts\ the\ stack\ volume\ in\ ft^3 to\ cc)$$

$$2.27x10^{8}\frac{cc}{min} * 60\ min = 1.36x10^{10}\frac{cc}{hr}$$

(this is the amount of cc the stack puts out per hour)

$$\left(\frac{30\ Ci}{2.72x10^{10}\frac{cc}{hr}}\right) * 1x10^{6} = 1.10x10^{-6}\ \mu Ci/cc/hr$$

(this takes the amount of curies and converts that to the number of microcuries per cc the stack puts out hour)

$$\left(\frac{1.10x10^{-6}\ \mu Ci/cc/hr}{3.0x10^{-6}}\right) = 0.36667\ per\ DAC$$

(this is the number of DAC's per hour a worker would reiceve from the volume put out by the sta

$$0.3667 per\ DAC * 2000\ hrs$$

(here we multiply the DAC'sper hour by the number of hours a worker will be exposed to get DAC

$$\frac{0.36667 * \cancel{2000hrs}}{\cancel{2000hrs}} * 5\ rem\ = 1.84\ rem$$

(this calculation takes the DAChrs and divded by the standard working year and is then mulitplied by the annual limit to get the dose.)

$$1.84\ rem * 1.0W_T = 1.84\ rem\ Effective\ Dose\ Equivalent$$

(this is the dose we can expect an occupational worker in confinment to reicieve over the course o year if the maximum amount of $Ar^{41}$is released through the stack)

$1x10^{6}$ = *conversion factor from Ci to* μCi
$\frac{cc}{min}$ = *cc released from the stack at* $8000\frac{ft^3}{min}$
$3.0x10^{-6} = Ar^{41}DAC$
$5\ rem$ = *occupational limit*
W_t = *weighting factor for whole body*

Correction for the Dose to the general public:

$$Stack\ rate * time = concentration$$

$$\left(\frac{Ci\ amount}{concentration}\right) * converstion\ factor = \mu Ci/cc/year$$

$$\left(\frac{\left(\frac{\mu Ci/cc/year}{dilution\ factor}\right)}{effluent\ concentration}\right) * dose\ limit\ to\ general\ public$$

$$= Dose\ to\ the\ general\ public\ at\ the\ fence$$

$$2.27x10^{8}\ \frac{cc}{min} * 60\ min * 24\ hrs * 365.25\ days$$
$$= 1.19x10^{14}\ cc/year\ stack\ concentration$$

$$\left(\frac{30\ Ci}{1.19x10^{14}\ cc/year}\right) * 1x10^{6} = 2.52x10^{-7}\ \mu Ci/cc/year$$

(in this calculation we take the maximum aount of curies and divide by the cc per year put out by the stack, then we multiply by the conversion factor to go from Ci to μCi)

$$\left(\frac{\frac{1.10x10^{-6}\ \mu Ci/cc/year}{200}}{1x10^{-8}}\right) * 0.1\ rem = 0.0126\ rem\ whole\ body\ dose$$

(In this calculation we take μCi per cc put out by the stack per year and divide by the dilution factor. That total is then divided by the effluent concentration and then multiplied by the annual dose limit to the general public as per 10 CFR 20)

$$12.6\ mrem * 1.0\ W_T = 12.6mrem\ Effective\ Dose\ Equivalent$$

(This is the maximum dose a member of the general public could receive if they stood outside the fence for an entire year as a whole body dose)

$1x10^{6}$ = *conversion factor from Ci to μCi*
200 = *dilution factor.*
$1x10^{-8}$ = *effluent concentration*
0.1 = *dose to the general public*
W_T = *weighting factor for the whole body*

Note:

The maximum dose to a member of the general public is above the specified limits in 10 CFR 20.1101, 10 mrem, but the NSCR monitors the release of Ar^{41} periodically throughout the year in order to ensure compliance with the Technical Specification 3.5.2, Argon Discharge Limit. Additionally, the limit set forth in 10CFR20.1101 is not a requirement, but a constraint. Under the NSCR's ALARA program, the 10 mrem constraint will not be exceeded. This is accomplished by analysis of the Ar^{41}discharges on a monthly and quarterly basis, which is compiled to generate the discharge report on the annual report. The Radiation Safety office also has the capability to obtain data if in the event an abnormal occurrence happens. Therefore, an Ar^{41} released can be analyzed at any given time.

12. *The TEES/TAMUS proposed TS 3.7, "As Low As Reasonable Achievable (ALARA) Radioactive Effluents Released," Specification 4, states, in part, requirements for the discharge of radioactive material into the sanitary sewerage, but does not indicate a solubility requirement, or indicate how it would be satisfied. Title 10 of the Code of Federal Regulations (10 CFR) 20.2003(a)(1) states that the material must be readily soluble in water. Revise TS 3.7, Specification 4, to include a solubility requirement, including verification, or justify why no change is necessary.*

NSC Response: In practice, the NSC already has a solubility requirement and verification procedure in place. The TEES/TAMUS proposed TS 3.7, "As Low As Reasonable Achievable (ALARA) Radioactive Effluents Released," Specification 4 will be revised to include a solubility requirement which complies with Title 10 of the Code of Federal Regulations (10 CFR) 20.2003(a)(1).

Revision: The facility liquid effluents collected in the holdup tanks shall be discharged in accordance with 10 CFR 20.2003 "Disposal by release into sanitary sewerage." The liquid effluent shall also meet local sanitary sewer discharge requirements.

13. The TEES/TAMUS proposed TS 3.7, "As Low As Reasonable Achievable (ALARA) Radioactive Effluents Released," Specification 3, states, in part, requirements for a significant fission product leak or airborne radioactive release. However, the qualitative meaning of significant is not defined in the TSs. Provide a definition for the application of the term "significant" or justify why no change is necessary.

NSC Response: The use of the word "significant" in the TEES/TAMUS proposed TS 3.7, "As Low As Reasonable Achievable (ALARA) Radioactive Effluents Released," Specification 3 does not seem to add any value to the intent of the Specification. Said another way, removing the word "significant" will not change the meaning of the Specification, and, in fact, makes it more clear. Additionally, removing the word "significant" eliminates the need to define it. The Specification will be revised to remove the word "significant."

> Revision: In the event of a fission product leak from a fuel rod or an airborne radioactive release from a sample being irradiated, as detected by the facility air monitor (FAM), the reactor shall be shut down until the source of the leak is located and eliminated. However, the reactor may continue to be operated on a short-term basis, as needed, to assist in determining the source of the leakage.

14. *The TEES/TAMUS proposed TS 3.5.3, "Xenon and Iodine Monitoring," Basis, does not appear to have a corresponding SAR description of the purpose or basis for the Xenon-125 (^{125}Xe) and Iodine-125 (^{125}I) monitors, or description of set points based on radiological consequences.*

    a. *NUREG-1537, Part 1, Section 11.1.1.1, "Airborne Radiation Sources," provides guidance for describing the radiological consequences of a release to both workers and members of the public, including how the monitoring systems help to ensure that the limits of 10 CFR Part 20, "Standards for Protection Against Radiation," are maintained. Provide a description, including the potential radiological doses (including calculations), to both workers and members of the public, for an accidental release of xenon and iodine. Demonstrate how the monitoring set points in TS 3.5.3 ensure that the potential doses from an accidental release would remain within the limits of 10 CFR Part 20.*

    b. *NUREG-1537, Part 1, Section 7.7, "Radiation Monitoring System," provides guidance that the licensee should provide a design basis, a system description, performance analysis and a conclusion about the suitability of the system to perform its function. Provide a Basis in TS 3.5.3, based on a SAR description of the ^{125}Xe and ^{125}I systems in TS 3.5.3, or justify why this information is not necessary.*

    c. *Table 4 lists the ^{125}I Air Monitor, which is not described in the SAR. Is this monitor part of the facility air monitoring (FAM) system similar to the ^{125}Xe Effluent Monitoring Channel (FAM Ch. 5)? Provide a description of the ^{125}I Air Monitor, or justify why this information is not necessary.*

<u>NSC Response:</u> In performing the dose calculations (see below) to respond to 14a, we found that a 2000 Ci single experiment limit was unreasonably unsafe. A failure of such an experiment would, in 5 minutes, expose a person in the confinement building to a whole body dose of over 924 rem and a CEDE of 33,600 rem. To bring doses following an experiment failure to within the bound describe in the MHA, a maximum single experiment activity of 200 uCi would need to be established. We have two responses to this information.

First, we immediately issued an administrative control suspending Xenon/Iodine experiments. Second, we would like to remove proposed TSs that allow such experiments. Specifically, proposed TSs 3.5.3, "Xenon and Iodine Monitoring," and 3.6.4, "Xenon Irradiation for Iodine Production."

Calculations:

Below I have calculated the Total Effective Dose Equivalent, Effective Dose Equivalent, and Committed Effective Dose Equivalent (TEDE, EDE and CEDE respectively) of an occupational worker 5 min after an accidental release of the maximum permissible amount of ^{125}Xe (2,000 Ci) and the EDE, CEDE, and TEDE from the ^{125}I produced from the decay of ^{125}Xe. We would like the maximum dose to be below the NSC's Maximum Hypothetical Accident (MHA), which is a TEDE of 24 mrem (DDE of 6.2 mrem and a CEDE dose of 17.8 mrem) for occupational workers. I calculated the EDE, CEDE, & TEDE for the maximum amount of ^{125}Xe permissible and the resulting amount of ^{125}I. I found the results to be too high of a CEDE dose, and therefore back calculated how much ^{125}Xe would result in an activity of ^{125}I that would give us a CEDE of 17.8 mrem or less. The end result was that we could have a maximum of 200μCi of ^{125}Xe in order to restrict the CEDE to 17.8 mrem or less and a TEDE of 24 mrem or less.

Building Volume 8000ft^3

Conversion from ft^3 to cc

$$(8000ft^3 * 2.5x10^4) = 5.1x10^9 cc$$

^{125}Xe Effective Dose Equivalent

$$\left(\frac{\left(\frac{\left(\frac{Ci}{concentration}\right) * conversion\ factor}{DAC\ \left(\mu Ci/_{cc}\right)}\right) * time\ in\ minutes}{2000\ hrs\ in\ minutes}\right) * annual\ limit\ (5\ rem) = exposure\ in\ rem$$

$$\left(\frac{2000Ci}{5.1x10^9}\right) * 1x10^6 = 0.392\frac{\mu Ci}{cc}\ Xe^{125} concentration$$

$$(1x10^6 = conversion\ factor\ to\ go\ from\ Ci\ to\ \mu Ci)$$

$$\left(\frac{0.392\ \frac{\mu Ci}{cc}}{2.0x10^{-5}\frac{\mu Ci}{cc}}\right) = 1.96x10^4\ \frac{\mu Ci/cc}{DAC}$$

$$(2.0x10^{-5}\frac{\mu Ci}{cc} = DAC\ as\ per\ 10CFR20)$$

$$1.96x10^4 * 5min = 9.80x10^4 DACmin$$

$$\left(\frac{9.80x10^4}{1.2x10^5\ min}\right) * 5\ rem = 817\ mrem\ whole\ body\ dose$$

$$(1.2x10^5 = 2000\ hrs\ in\ \text{minutes})$$

$$817\ mrem\ whole\ body\ dose\ from\ Xe^{125}$$

^{125}I Activity

^{125}Xe half-life 17hrs or 1020 minutes

$$\lambda_{xe} = \frac{ln2}{1020\ min} = 6.8x10^{-4}\ Min^{-1}$$

^{125}I activity after 5 min of ^{125}Xe decay

$$A = 2000\ Ci\left(1 - e^{-6.8x10^{-4}}\right) = 1.36\ Ci\ of\ I^{125} after\ 5\ minutes$$

^{125}I Effective Dose Equivalent:

$$\left(\frac{1.36\ Ci}{5.1x10^{9}cc}\right) * 1x10^{6} = 1.33x10^{-3}\ \mu Ci\ /cc$$

$$(I^{125} Concentration)$$

$$\left(\frac{1.33x10^{-3}\ \mu Ci/cc}{3.0x10^{-8}}\right) = 4.43x10^{4}\ \frac{\mu Ci/cc}{DAC}$$

$$(3.0x10^{-8} = I^{125} DAC\ as\ per\ 10CFR20)$$

$$(8.88x10^{3}\ \frac{\mu Ci/cc}{DAC} * 5\ minutes = 2.22x10^{5} DACmin$$

$$\left(\frac{9.8x10^{4}\ DACmin}{1.2x10^{5} min}\right) * 5\ rem = \ 924\ rem\ whole\ body$$

^{125}I Committed Effective Dose Equivalent for 1.4 Ci

$$\sum_{T} W_T * H_{T,50} = CEDE\ dose$$

CEDE: As per Federal Guidance Report No. 11 page 136 the I^{125}CDE $(H_{T,50})$is equvalent to $2.16x10^{-7}\ {}^{Sv}/_{Bq}$ for the thyroid. The thyroid is the maximally exposed tissue and all other tissues are below 10% and therefore, will not be taken into consideration for the CEDE calculation. The thyroid $W_T$is equal to 0.03.

$$CEDE = 0.03 * 2.16x10^{-7}\ {}^{Sv}/_{Bq} = 6.48x10^{-9}\ {}^{Sv}/_{Bq}$$

$$6.48x10^{-9}\ {}^{Sv}/_{Bq} * 3.7x10^{6} = 2.4x10^{-2}\ {}^{rem}/_{\mu Ci}$$

$$\left(3.7x10^{6} = conversion\ factor\ to\ {}^{rem}/_{\mu Ci}\right)$$

$$2.4x10^{-2}\ {}^{rem}/_{\mu Ci} * 1.4x10^{6} \mu Ci = 3.36x10^{4} rem\ Thyroid\ CEDE$$

Combine ^{125}Xe and ^{125}I Total Effective Dose Equivalent

$$TEDE = (0.817 + 924 + 3.36x10^4)rem = 3.45x10^4\ rem$$

As you can see the above calculations give us a TEDE above 24mrem. Below I have calculated a reduced amount of ^{125}Xe to result in a CEDE of less than 17.8 mrem using the same methods as above.

^{125}Xe Effective Dose Equivalent

$$\left(\frac{200\mu Ci}{5.1x10^9}\right) * 1x10^6 = 3.92x10^8 \frac{\mu Ci}{cc}\ Xe^{125} concentration$$

$$\left(\frac{3.92x10^8\ \frac{\mu Ci}{cc}}{2.0x10^{-5}\frac{\mu Ci}{cc}}\right) = 1.96x10^{-3}\ \frac{\mu Ci/cc}{DAC}$$

$$1.96x10^{-3} * 5\ min = 9.80x10^{-3} DACmin$$

$$\left(\frac{1.96x10^{-6}}{1.2x10^5\ min}\right) * 5\ rem = 81.7\ \mu rem\ whole\ body\ dose$$

^{125}I Effective Dose Equivalent

$$\left(\frac{0.678\ \mu Ci}{5.1x10^9 cc}\right) * 1x10^6 = 1.33x10^{-10}\ \mu\ Ci\ /cc$$

$$(I^{125} Concentration)$$

$$\left(\frac{1.33x10^{-10}\ \mu Ci/cc}{3.0x10^{-8}}\right) = 4.43x10^{-3}\frac{\mu Ci/cc}{DAC}$$

$$(3.0x10^{-8} = I^{125} DAC\ as\ per\ 10CFR20)$$

$$(4.43x10^{-3}\frac{\mu Ci/cc}{DAC} * 5\ minutes = 2.22x10^{-2} DACmin$$

$$\left(\frac{1.78x10^3}{1.2x10^5 min}\right) * 5\ rem = \ 924\ \mu rem\ whole\ body$$

^{125}I Activity after 5 min

$$\lambda_{xe} = \frac{ln2}{1020\ min} = 6.8x10^{-4}\ Min^{-1}$$

$$A = 200\ \mu Ci\left(1 - e^{-6.8x10^{-4}}\right) = 0.678\ \mu Ci\ of\ I^{125} after\ 5\ minutes$$

^{125}I Committed Effective Dose Equivalent for 0.678 µCi

$$CEDE = 0.03 * 2.16x10^{-7}\ {}^{Sv}/_{Bq} = 6.48x10^{-9}\ {}^{Sv}/_{Bq}$$

$$6.48x10^{-9}\ {}^{Sv}/_{Bq} * 3.7x10^{6} = 2.4x10^{-2}\ {}^{rem}/_{\mu Ci}$$

$$2.4x10^{-2}\ {}^{rem}/_{\mu Ci} * 0.678\ \mu Ci = 16.3\ mrem\ Thyroid\ CEDE$$

Combine ^{125}Xe and ^{125}I Total Effective Dose Equivalent

$$TEDE = EDE + CEDE$$

$$TEDE = (\ 0.817 + 0.924 + 16.3)mrem = 18.0\ mrem\ TEDE$$

15. *The TEES/TAMUS proposed TS 4.5, "Radiation Monitoring Systems and Effluents," Specification 2, states, in part, requirements for the emergency xenon monitor. The following items were identified:*

    a. *The emergency xenon monitor is not defined in the TSs or SAR. Is this the same monitor described in proposed TS 3.5.3, Table 4, as the ^{125}Xe Effluent Monitoring Channel (FAM Ch. 5)? Provide a description of the emergency xenon monitor, revise the name, or justify why no change is needed.*

    b. *The emergency xenon monitor is not channel tested consistent with the requirements of the FAM system. Revise TS 4.5, Specification 2, to require a channel test of the emergency xenon monitor or justify why no change is needed.*

NSC Response: The TEES/TAMUS proposed TS 4.5, "Radiation Monitoring Systems and Effluents," Specification 2 appears to be a vestige of the past. As described in TS 3.5.3, "Xenon and Iodine Monitoring," FAM Ch. 5 is the Xenon Monitoring Channel. Since FAM Ch. 5 is channel tested consistently with the requirements of the FAM system.

Since this Surveillance Requirement specification seems to be a vestigial reference (the phrase "emergency xenon monitor" does not occur elsewhere) and does not correspond to anything in Chapter 3, Limiting Conditions for Operation, the proposed revision is to delete the specification.

16. The TEES/TAMUS proposed TS 1.3, "Definitions," Excess Reactivity, states in part, control devices, which are not defined. Provide a definition for control devices, revise to control rods, or justify why no change is necessary.

NSC Response: The TEES/TAMUS proposed TS 1.3, "Definitions," Excess Reactivity will be revised to change "control devices" to "control rods."

Revision: Excess reactivity is the 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 (*keff* = 1) at reference core conditions.

17. The TEES/TAMUS proposed TS 1.3, "Definitions," License, states, in part, "the responsible authority," which is not defined. Provide a definition for responsible authority, change to U.S. NRC, or justify why no change is necessary.

NSC Response: The TEES/TAMUS proposed TS 1.3, "Definitions," License, will be revised to change "responsible authority" to "U.S. NRC."

> Revision: The written authorization, by the U.S. NRC, for an individual or organization to carry out the duties and responsibilities associated with a personnel position, material, or facility requiring licensing.

18. The TEES/TAMUS proposed TS 1.3, "Definitions," Limiting Safety System Settings (LSSS), states, in part, "[t]he limiting safety system setting is a temperature," but does not indicate if it refers to the fuel element temperature. Provide a revision of the definition of the limiting safety system setting that indicates that it is the temperature of the fuel element or justify why no change is necessary.

NSC Response: The TEES/TAMUS proposed TS 1.3, "Definitions," Limiting Safety System Setting (LSSS) will be revised to indicate that it is the fuel element temperature.

Revision: The limiting safety system setting is the fuel element temperature, which if exceeded, shall cause a reactor scram to be initiated, preventing the safety limit from being exceeded.

19. The TEES/TAMUS proposed TS 1.3, "Definitions," Reactor Secured, Item (2)(a), states, in part, the number of neutron-absorbing control devices, without providing a definition or number. Provide a definition of neutron-absorbing control devices, including a number, change to state "control rods," or justify why no change is necessary.

NSC Response: The TEES/TAMUS proposed TS 1.3, "Definitions," Reactor Secured, Item (2)(a) will be revised to state "control rods," and to include a number.

Revision: All control rods are fully inserted;

20. *The TEES/TAMUS proposed TS 1.3, "Definitions," Reportable Occurrence, Item (4), does not include the guidance provided in ANSI/ANS-15.1-2007, Section 6.7.2, Special Reports, Item (1)(iv), that states, in part, "Reactor trips resulting from a known cause are excluded." Revise the definition of Reportable Occurrence to match the guidance in ANSI/ANS-15.1-2007, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 1.3, "Definitions," Reportable Occurrence, Item (4), will be revised to follow the guidance provided in ANSI/ANS-15.1-2007, Section 6.7.2, Special Reports, Item (1)(iv).

> Revision: An unanticipated or uncontrolled change in reactivity greater than $1.00. Reactor trips resulting from a known cause are excluded;

21. *The TEES/TAMUS proposed TS 1.3, "Definitions," Reportable Occurrence, Item (5), states, in part, "where appropriate," which is not clearly defined. Provide a definition for examples that would constitute exceptions to the reporting requirements of proposed TS 1.3 "Definitions," Reportable Occurrence, Item (5), eliminate the exception "where appropriate," or justify why no change is necessary.*

NSC Response: It is unclear what "where appropriate" is supposed to mean in this definition as written. It appears to be a corrupted version of ANSI/ANS-15.1-2007, Section 6.7.2, Special Reports, Item (1)(v), which states, in part, "where applicable." In any case, it does not appear that the phrase adds any value to the intent of the definition. It will be revised to eliminate "where appropriate."

> Revision: Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or confinement boundary; and

22. *The TEES/TAMUS proposed TS 3.1.5, "Reactor Fuel Parameters," does not have a specification for burnup. NUREG-1537, Part 1, Appendix 14.1, Section 3.1, "Reactor Core Parameters," Item (6)(b), "TRIGA Fuel," provides guidance that the fuel matrix should not exceed 50 percent of its initial concentration. Provide a burnup limit consistent with the guidance in NUREG-1537, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS3.1.5, "Reactor Fuel Parameters," will be revised to include a specification that follows the guidance of NUREG-1537, Part 1, Appendix 14.1, Section 3.1, "Reactor Core Parameters," Item (6)(b), "TRIGA Fuel."

Revision: 3. The burnup of the uranium-235 in the UZrH fuel matrix shall not exceed 50 percent of the initial concentration.

23. The TEES/TAMUS proposed TS 3.2.2, "Reactor Safety Systems and Interlocks," has the following items:

a. *The specification provides requirements for operation with safety channels out of service for maintenance, surveillance, calibration or repair. However, no other limitations are provided, e.g., time or number of channels which may be out of service concurrently. Provide a revised specification, including the supporting basis, for allowing safety channels to be out of service, or justify why no change is necessary.*

b. *Table 2a, Safety Channel, Fuel Element Temperature, Function, states, "Scram <= the LSSS," but does not provide a set point (numerical value). NUREG-1537, Part 1, Appendix 14.1, Section 3.2, "Reactor Control and Safety Systems," Item (4), "Scram Channels," provides guidance that the set points should be provided in a table. Provide a set point (numerical value) for TS 3.2.2 LSSS, or justify why no change is necessary.*

c. *Table 2a, Safety Channel, High Power Level, states, "Scram <= 125%," which is not defined. Revise 125% to relate to the licensed power level, or define in terms of megawatts, and provide a basis for the setting being 25% above the licensed power level, as opposed to 10% or 15%, or justify why no change is necessary.*

NSC Response A: The TEES/TAMUS proposed TS 3.2.2, "Reactor Safety Systems and Interlocks," will be revised to specify limitations on operation with safety channels out of service for maintenance, surveillance, calibration, or repair.

Revision A: Specification: The reactor shall not be operated unless the safety circuits and interlocks described in Tables 2a and 2b are operable. However, any single safety channel or interlock may be inoperable with the reactor operating for the purpose of maintenance, surveillance, calibration, or repair. If any required safety channel or interlock becomes inoperable while the reactor is operating, for reasons other than identified in this TS, then the channel shall be restored to operation within 5 minutes or the reactor shall be immediately shutdown.

Basis: During period of maintenance, surveillance, calibration, and repair, true signals generated by the reactor may be required. This operation is allowable provided all other TS conditions are met. If multiple channels have failed simultaneously, then involvement of the U.S. NRC in recovery planning will be necessary.

NSC Response B: Table 2a, Safety Channel, Fuel Element Temperature, Function will be revised to provide a numerical value for the scram set point.

Revision B: Scram ≤ 975°F

NSC Response C: Table 2a, Safety Channel, High Power Level will be revised to relate the percentage value to the licensed power level. During most operational configurations, a setting of 15% above the licensed power level would provide sufficient room for error in the measuring channels. However, during operation against our beam port/thermal column coupler box the reactor operates near a large volume of graphite. The reflection from this experiment raises affects the signal generated in the high power level safety channels by approximately a positive 10%. A setting of 25% above the licensed power level is necessary to avoid nuisance scrams during these operations.

Revision C: Scram ≤ 1.25MW

24. *The TEES/TAMUS proposed TS 3.2.2, "Reactor Safety Systems and Interlocks," Table 2b, Function, provides descriptions of the Safety Channels' functions that are not clearly worded. NUREG-1537, Part 1, Appendix 14.1, Section 3.2, "Reactor Control and Safety Systems," Item (4), "Scram Channels," provides guidance that the scram channels should be described in a table. Provide a more comprehensive description of the Function for the Safety Channels in Table 2b, or justify why no changes are necessary.*

NSC Response: The TEES/TAMUS proposed TS 3.2.2, "Reactor Safety Systems and Interlocks," Table 2b, Interlocks Required for Operation, Function will be revised to provide a more comprehensive description of the Function for the Safety Channels in Table 2b.

Revision: Log Power: Prevents withdrawal of the Shim Safety Control Rods at an indicated log power of less than $4x10^{-3}$ W.

Log Power: Prevents pulsing of the Transient Rod when log power is above 1 kW.

Transient Rod Position: Prevents application of air to the Transient Rod unless the Transient Rod is fully inserted.

Shim Safety and Regulating Rod Position: Prevents Shim Safety and Regulating Control Rod withdrawal during a pulse.

Pulse Stop Electro-Mechanical Interlock: Prevents application of air to the Transient Rod unless the mechanical pulse stop is installed.

25. *The TEES/TAMUS proposed TS 3.6.1, "Reactivity Limits," Specification 1, states, in part, requirements for the reactivity worth, but does not describe if the reactivity worth involves positive or negative reactivity. Revise TS 3.6.1, Specification 1, to include the absolute reactivity worth, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 3.6.1, "Reactivity Limits," Specification 1 will be revised to include the absolute reactivity worth.

> Revision: The absolute reactivity worth of any single, movable or unsecured experiment shall be less than $1.

26. The TEES/TAMUS proposed TS 3.6.2, "Material Limitations," has the following items:

a. *Specification 1.a, describes explosive materials in quantities greater than 25 milligrams and less than 25 milligrams, but does not include quantities equal to 25 milligrams. Revise TS 3.6.2 to include quantities up to 25 milligrams, or justify why no change is necessary.*

b. *Specification 1.b, describes interior of the pool containment structure, which does not appear to be defined in the TSs or SAR. Provide a description or clearly define the area or location meant by the interior of the pool containment structure, or justify why no change is necessary.*

c. *Specification 2, states, in part, significant amounts of corrosive materials, which is not clearly defined. Revise TS 3.6.2, Specification 2, to provide a qualitative definition for significant amounts of justify why no change is necessary.*

NSC Response A: The TEES/TAMUS proposed TS 3.6.2, "Material Limitations," Specification 1.a will be revised to include quantities up to 25 milligrams. Additionally, similar revisions to Specifications 1.b, 1.c, 1.d, and 1.e will be made to improve clarity.

> Revision A: Explosive materials in quantities greater than or equal to 25 milligrams (TNT-equivalent) shall not be irradiated in the reactor pool. Explosive materials in quantities up 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;

NSC Response B: It is unclear what we were trying to describe with the phrase "pool containment structure." The specification lists "..the reactor pool, the upper research level, the demineralizer room, cooling equipment room…" all of which make sense. "Pool containment structure" is not a thing that exists. The TEES/TAMUS proposed TS 3.6.2, "Material Limitations," Specification 1.b will be revised to delete this reference.

> Revision B: Explosive materials in quantities greater than or equal to 25 milligrams (TNT-equivalent) shall only be allowed in the lower research level and laboratory building, excluding the heat exchanger room and demineralizer room;

NSC Response C: The TEES/TAMUS proposed TS 3.6.2, "Material Limitations," Specification 2 will be revised to more concretely describe the requirement for double encapsulation.

> Revision C: Corrosive materials used in a reactor experiment shall be double encapsulated. Exceptions may only be made if a detailed analysis and/or prototype testing

with small amounts of materials demonstrates that the experiment presents negligible risk.

27. The TEES/TAMUS proposed TS 3.2.1, "Reactor Measuring Channels," does not appear to have a corresponding TS 4.0 surveillance requirement. Provide a surveillance requirement for TS 3.2.1, or justify why one is not necessary.

NSC Response: The 4.0 TS that should correspond to the TEES/TAMUS proposed TS 3.2.1, "Reactor Measuring Channels," 4.2.1, specifies surveillance of the "Reactor Control System," i.e. the control rods and their associated equipment. These specifications seem to have a more appropriate location in 4.2.3, "Scram Time" as 3.2.3 is "Minimum Number of Operable Scrammable Control Rods and Scram Time." The 4.0 TS will be revised to move the specifications in the existing 4.2.1, "Reactor Control System," to 4.2.3, "Scram Time." The name of TS 4.2.3 will be revised to "Minimum Number of Operable Scrammable Control Rods and Scram Time." TS 4.2.1 will be revised to "Reactor Measuring Channels," and contain specifications that correspond to TS 3.2.1, "Reactor Measuring Channels."

While the TEES/TAMUS proposed TS 4.2.1 Reactor Measuring Channels will provide surveillance requirements for TS 3.2.1, calibration requirements for these channels already exist in TSs 4.1.1, 4.2.2, and 4.8.3 and will not be repeated in TS 4.2.1.

Revisions:

4.2.1 Reactor Channels

Applicability

These specifications apply to the surveillance requirements for reactor channels.

Objective

The objective is to verify the condition and operability of system components directly related to channels that measure key reactor parameters.

Specifications

A channel test of each of the reactor channels for the intended mode of operation, as identified in Table 1, shall be performed before each day's operation or before each operation extending more than one day.

Basis

Channel tests will ensure that the safety system channels are operable on a daily basis or prior to an extended run.

4.2.3 Minimum Number of Operable Scrammable Control Rods and Scram Time

Applicability

These specifications apply to the surveillance requirements for reactor control systems.

Objective

The objective is to verify the condition and operability of system components affecting safe and proper control of the reactor.

Specification

1. The control rods shall be visually inspected for deterioration biennially.

2. Operability tests of the control rod mechanism shall follow modification or repairs.

3. The Transient Rod drive cylinder and associated air supply system shall be inspected, cleaned and lubricated semiannually.

4. The scram time shall be measured annually or whenever any work is done on the rods or the rod drive system.

Basis

1. The visual inspection of the control rods is made to evaluate corrosion and wear characteristics caused by operation of the reactor.

2. These tests provide verification that the control rod has full travel and that the rod drop time is within specification.

3. Inspection and maintenance of the transient rod drive assembly reduces the probability of failure of the system due to moisture-induced corrosion of the pulse cylinder and piston rod assembly.

4. Measurement of the scram time on an annual basis is a check not only of the scram system electronics, but also is an indication of the capability of the control rods to perform properly.

28. The TEES/TAMUS proposed TS 4.2.2, "Reactor Safety Systems and Interlocks," Specification 1, states, in part, "[a] channel check," as identified in Table 2, except for the pool level alarm. However, the pool level alarm is not listed in TS 3.2.2, Table 2. Provide a revised TS 4.2.2, or TS 3.2.2, Table 2, or both, or justify why no change is necessary.

NSC Response: Before addressing the specific issue of this question, another revision in the TEES/TAMUS proposed TS 4.2.2, "Reactor Safety Systems and Interlocks," Specification 1 needs to be made. "Channel check..." is clearly an error that should read "Channel test..."

The reference to the pool level alarm channel in TS 4.2.2 is in error. The pool level alarm used to be a part of TS 3.2.2, and when it was moved to the pairing of TSs 3.8.2 and 4.8.2, the reference was not deleted in 4.2.2. The TEES/TAMUS proposed TS 4.2.2, "Reactor Safety Systems and Interlocks," Specification 1 and Basis will be revised to remove the extraneous reference to the pool level alarm.

> Revision: Specification: A channel test of each of the reactor safety system channels and interlocks for the intended mode of operation, as identified in Table 2, shall be performed before each day's operation or before each operation extending more than one day.
>
> Basis: Channel tests will ensure that the safety system channels are operable on a daily basis or prior to an extended run. If the period between operations extends beyond a year, then the annual channel test requirement will ensure operability.

29. The TEES/TAMUS proposed TS 4.5, "Radiation Monitoring Systems and Effluents," Specification 1, states, in part, "the FAM," but does not identify it as a "system" as described in the SAR. Revise the FAM to include system, or justify why no change is necessary.

NSC Response: The TEES/TAMUS proposed TS 4.5, "Radiation Monitoring Systems and Effluents," Specification 1 will be revised to include system.

Revision: The area radiation monitoring system (ARM) and the FAM system shall be calibrated annually, shall be channel tested weekly, and shall be channel checked prior to reactor operation.

30. *The TEES/TAMUS proposed TS 4.8.1, "Primary Coolant Purity," Specifications 2 and 3, states, in part, that conductivity and pH measurements of the bulk pool water shall be performed quarterly during extended shutdowns, and quarterly, respectively. NUREG-1537, Part 1, Appendix 14.1, Section 4.3, "Coolant Systems," Item (6) "Conductivity and pH," provides guidance that the conductivity and pH should be measured weekly, or monthly, if the reactor is shutdown for long periods of time, if justification is provided in the SAR. A review of the SAR did not find justification for the proposed quarterly periodicity. Revise TS 4.8.1, Specifications 2 and 3, to weekly, or monthly with an accompanying SAR justification, respectively, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 4.8.1, "Primary Coolant Purity," Specifications 2 will be revised to follow the guidance of NUREG-1537, Part 1, Appendix 14.1, Section 4.3, "Coolant Systems," Item (6) "Conductivity and pH."

Revision:

2. Conductivity of the bulk pool water shall be measured and recorded weekly.

Based on discussions with NRC staff during a site visit on February 4th, 2015, we request to have the pH limits removed. These discussions focused on our proposed conductivity limit (no higher than 5×10^{-6}mhos/cm for an averaged period of two weeks), the periodicity of conductivity measurement (procedurally required every day the reactor operates), and the close relationship between conductivity and pH in high purity systems.

31. *The TEES/TAMUS proposed TS 5.3, "Reactor Core," Specification 3, states, in part, "[c]ore lattice position." Should this state "positions"? Revise TS 5.3, Specification 3, to state core lattice positions or justify why no change is needed. Additionally, the basis to TS 5.3, Specification 3, states, in part, "accident." Should this be "accidental"? Revise TS 5.3, Basis 3, to state accidental, or justify why no change is necessary.*

NSC Response: This question is linked with question 52. Question 52 rearranges the contents of TEES/TAMUS proposed Technical Specifications 3.1.4, "Reactor Core Configuration," and 5.3, "Reactor Core." The specification referenced in this question has been moved to Specification 3.1.4 and will be referred to in that way.

The answer to both questions is "yes." These typographical errors in the TEES/TAMUS proposed TS 3.1.4 "Reactor Core Configurations," will be revised.

> Revision: Specification: Core lattice positions shall not be vacant except for positions on the periphery of the core assembly while the reactor is operating. Water holes in the inner fuel region shall be limited to single rod positions. Vacant core positions not on the periphery shall contain experiments or an experimental facility to prevent accidental fuel additions to the core.
>
> Basis: Vacant core positions containing experiments or an experimental facility will prevent accidental fuel additions to the reactor core. They will be permitted only on the periphery of the core or a single rod position to prevent power peaking in regions of high power density.

32. *The TEES/TAMUS proposed TS 5.4, "Control Rods," Specification 2, provides a description of the regulating rod, but does not describe any control rod follower characteristics (i.e., is it a water follower, consistent with the follower descriptions provided in TS 5.4, Specifications 1 and 3). Revise TS 5.4, Specification 2, to include a description of the control rod follower characteristics, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 5.4, "Control Rods," Specification 2 will be revised to include a description of the control rod follower characteristics.

> Revision: The regulating control rod may not have scram capability and shall be a stainless rod or contain borated graphite, B4C powder or boron and its compounds in solid form as poison in aluminum or stainless steel cladding. This rod is water followed in that pool water takes the place of the rod as it is withdrawn. It has no physical follower attachment.

33. The TEES/TAMUS proposed TS 5.5, "Radiation Monitoring System," has the following items:

a. *The Applicability statement indicates that the area radiation monitoring (ARM) and FAM systems are required for continuous monitoring, whereas, proposed TS 3.5.1, "Radiation Monitoring," indicates that the ARM and FAM Channels 1, 3, and 4, are required during reactor operation. Reconcile the applicability statements for TS 5.5 and TS 3.5.1, or justify why no change is necessary.*

b. *Proposed TS 5.5, Table 5, Radiation Monitoring Channel column, provides ARM and FAM system information, but does not state the applicable channel(s). Revise Table 5 to provide radiation monitoring channel information, or justify why no change is necessary.*

c. *The Basis does not indicate how the alarm setpoints are established in order to protect the workers or public. NUREG-1537, Part 1, Appendix 14.1, Section 3.7.1, "Monitoring System," Item (3) "Area Monitors," provides guidance that the alarm and automatic setpoints should be specified to ensure that personnel exposures and potential doses remain below the limits of 10 CFR Part 20. Provide the basis for the ARM and FAM setpoints, or justify why no change is necessary.*

NSC Response A: The TEES/TAMUS proposed TS 5.5, "Radiation Monitoring System," Applicability statement will be revised to agree with proposed TS 3.5.1"Radiation Monitoring." Also, the applicability statement of proposed TS 3.5.1 "Radiation Monitoring," and thereby proposed TS 5.5, "Radiation Monitoring System," will be expanded beyond "reactor operation" to include "movement of irradiated fuels or fueled experiments, conduct of core or control rod work that could cause a change in reactivity of more than one dollar, or handling of radioactive materials with the potential for airborne release."

Revision A: This specification describes the functions and essential components of the area radiation monitoring (ARM) equipment and the facility air monitoring (FAM) system equipment for continuously monitoring airborne radioactivity as described in TS 3.5.1 "Radiation Monitoring."

NSC Response B: The TEES/TAMUS proposed TS 5.5, Table 5, Radiation Monitoring Channel column will be revised to include channel numbers that correspond to where the listed equipment is used.

NSC Response C: The reactor bridge Area Radiation Monitor (ARM) alarm set point is normally set at 14 mR for an alert and 16 mR alarm. The background for this ARM is normally no greater than 5 mR and this set point alerts the Reactor Operator that the ARM is approaching

or has exceeded three times the background. The alarm set point can be adjusted if operational activities warrant the necessity of the adjustment. If the alarm set point is raised above the normal alarm set point additional controls are put into place in order to ensure ALARA.

The Facility Air Monitor (FAM) alarm set points are calculated during the annual calibration of the specific FAM channel. Each channel has an individual calibration and alarm set point. The following list is the basis of each alarm set point for each individual FAM channel.

Channel 1	The alarm set point is calculated by $\left(\frac{Cs^{137} Effluent\ Concentration * dilution\ factor}{conversion\ factor}\right) * air\ system\ efficiency$ $Dilution\ Factor = 200$ $Air\ system\ efficiency = 0.33$ $Conversion\ factor = \frac{1}{YRTKQ} = \mu Ci/cc/net\ cpm$ $Y = \frac{gross\ background}{activity}$ $R = flowrate$ $T = Transit\ time\ of\ sample, derived\ from\ d/v$ $d =$ detector diameter(inches) $v =$ advance rate of the filter paper $\left(\frac{\text{inches}}{\text{hr}}\right)$ $K = \left(\frac{dpm}{\mu Ci}\right) * \left(\frac{cm^3}{ft^3}\right)$ $Q = \frac{average\ particulate\ counting\ time}{max\ particulate\ counting\ time}$
Channel 3	$\frac{Effluent\ concentration\ Ar^{41} * dilution\ factor}{conversion\ factor_{Ar^{41}}}$ $Conversion\ factor_{Ar^{41}} = \frac{concentration}{counts}\ slope\ in\ \mu Ci/cc/net\ cpm$
Channel 5	$\frac{Effluent\ Concentration\ Xe^{125} * dilution\ factor}{Conversion\ Factor_{Xe^{125}}}$ $Conversion\ factor_{Xe^{125}=0.77*CF_{Ar^{41}}}$ 0.77 is based on the differences of the photon energies and photon yields of the two isotopes.

The methods of determining alarm set point described above will ensure that doses remain ALARA by notifying the Control Room in the event radiation levels exceed those limits specified in 10 CFR part 20 and the facility's ALARA program.

34. The TEES/TAMUS proposed TS 5.7, "Reactor Building and Central Exhaust System," has the following items:

a. *Specifications 1, 2, and 3, provide requirements associated with the reactor building volume, exhaust system, and emergency controls, that are also stated in TS 5.1, "Site and Facility Description," Specifications 2, 3, and 4. In order to reduce unnecessary redundancy and avoid potential confusion, consider consolidating the requirements of TS 5.1, Specifications 2, 3, and 4, into TS 5.7, or justify why no change is necessary.*

b. *Specification 3, states, in part, the fission product air monitor, However, this system is defined in the TSs.* (**Note:** I believe the NRC author meant to write "...is not defined...) *Provide a definition or description of the fission product facility air monitor, or justify why no change is necessary.*

NSC Response A: The TEES/TAMUS proposed TS 5.1, "Site and Facility Description," Specifications 2, 3, and 4, will be consolidated into proposed TS 5.7, "Reactor Building and Central Exhaust System." The remaining Specifications 1 and 5 will be consolidated into a single specification. The name, applicability, and objective of proposed TS 5.1 "Site and Facility Description" will be revised to reflect these moves and consolidations.

NSC Response B: The TEES/TAMUS proposed TS 5.7, "Reactor Building and Central Exhaust System," Specification 3 incorrectly references a Radiation Monitoring Channel described in proposed TS 3.5.1, "Radiation Monitoring" and in proposed TS 5.5, "Radiation Monitoring System." It will be revised to reference this defined channel correctly.

Revision B: Emergency isolation controls for the central exhaust system shall be located in the emergency support center and the system shall be designed to shut down in the event of an alarm on the stack particulate monitor (FAM Ch.1) radiation monitoring channel.

35. *The TEES/TAMUS proposed TS 5.8, "Reactor Pool Water Systems," has the following issues:*

    a. *Specification 3, states, in part, that emergency covers are required in case of pool water loss due to external pipe system failure. The SAR does not provide a description of the use of the emergency covers. The proposed TS 5.8 Basis provides a description of the storage and assumed installation times for the use the covers, but no supporting information is provided in the SAR. Prove a description of the covers, including a basis for the assumptions provided in the TS Basis, or justify why no change is necessary.*

    b. *Specification 4 appears to be missing "at" prior to "a continuously monitored remote location." Provide a revised Specification 4, or justify why no change is necessary.*

NSC Response A: The two primary discharges (heat exchanger outlets) to the pool are equipped with flapper valves. When normal flow is stopped, these valves close over the discharge pipes. A leak in the pipe would result in additional pressure against the flapper from the pool, which helps to seal it. The one primary suction (heat exchanger inlet) from the pool is covered with a manual plug. The "less than 5 minutes" projected installation time is based on extensive testing and practice with employees. Experienced employees can install it in approximately 10 seconds, and even totally green employees installing it for the first time can install it in approximately 1 minute. "Less than 5 minutes," was chosen as a reasonable safety margin to account for a green employee being relied upon to install it in a stressful emergency situation. Two sets of leakage rate tests conducted several years apart show a rate of approximately 5gpm when the covers are installed.
The TEES/TAMUS proposed TS 5.8, "Reactor Pool Water Systems," Specification 3 will be revised to reflect that only one cover exists.
The basis of the TEES/TAMUS proposed TS 5.8, "Reactor Pool Water Systems," Specification 3 will be revised to include more descriptive information of the pipes, similar to what is written above. This information will also be added to the next revision of the SAR.
NSC Response B: Specification 4 will be modified to include "at" prior to "a continuously monitored remote location", as well as to make "readout" plural.

> Revision B: A pool level alarm with readouts in the control room and at a continuously monitored remote location shall indicate a pool level less than 3 feet below the reference operating level.

36. *The TEES/TAMUS proposed TS 6.1.2, "Responsibility," provides requirements for responsibility for the safe operation of the reactor facility. However, no information is provided on the use of alternates. ANSI/ANS-15.1-2007, Section 6.1.2, "Responsibility," provides guidance for the use of alternates. Provide a revision to TS 6.1.2 that adds clarification on the use of alternates, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 6.1.2, "Responsibility," will be revised to follow the guidance of ANSI/ANS-15.1-2007, Section 6.1.2, "Responsibility."

Revision: Responsibility for the safe operation of the reactor facility shall be in accordance with the line organization established in Section 6.1.1.. In all instances, responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon appropriate qualifications.

37. The TEES/TAMUS proposed TS 6.1.3, "Staffing," has the following items:

a. *Specification 1.b, states, in part, "period of reactor maintenance" and "periods when the reactor is unsecured..." which appear to be typographical errors. Should "period" be "periods" and "unsecured" be "not secured"? Provide a revised TS 6.1.3, Specification 1.b, or justify why no change is necessary.*

b. *Specification 3, describes activities requiring designated individuals, including loading fuel and recovery from an unplanned or unscheduled shutdown. However, unloading fuel and power reductions are not included. Provide a revised TS 6.1.3, Specification 3 that includes unloading fuel and power reductions, as appropriate, or justify why no changes are necessary.*

NSC Response A: Yes, the noted words are indeed typographical errors. The TEES/TAMUS proposed TS 6.1.3, "Staffing," Specification 1.b, will be revised to correct these errors.

NSC Response B: The revisions to the TEES/TAMUS proposed TS 6.1.3, "Staffing" Specification 3.a, 3.b and 3.e, will address the required items.

Revision B:

3. The following designated individuals shall direct the events listed:

    a. The NSC Director, or his designated alternate who shall be an SRO, shall direct any loading or unloading of fuel or control rods within the reactor core region,

    b. The NSC Director or his designated alternate who shall be SROs shall direct any loading or unloading of an in-core experiment with a reactivity worth greater than $1,

    .
    .
    .

    e. The senior reactor operator on duty shall be present at the facility and shall direct all significant reactor power changes after initial startup. A significant reactor power change is defined as one that would disable the automatic servo control, i.e. equal to or great than 5% of reactor power.

38. *The TEES/TAMUS proposed TS 6.2.1, "Reactor Safety Board (RSB)," has the following items:*

    a. *The word "normally" is used in two instances.*

    b. *The title "Deputy Director of TEES" is used without a corresponding definition in TS 6.1.1.*

    c. *"Level 1" is used without a corresponding title.*

    These items may crease a potential for misinterpretation of the requirements contained in TS 6.2.1. ANSI/ANS-15.1-2007, Section 6.2, "Review and audit," provides guidance for the review and audit function. Provide a revision of TS 6.2.1 consistent with the guidance in ANSI/ANS-15.1-2007, or justify why no change is necessary.

NSC Response A-C: The TEES/TAMUS proposed TS 6.2.1, "Reactor Safety Board (RSB) will be revised to address the above items. The two uses of the word "normally" will be deleted. "Deputy of Director of TEES" was used in error. It will be revised to "Director of TEES (Level 1 Management)." "Level 1" will be revised to "Director of TEES (Level 1 Management)."

39. The TEES/TAMUS proposed TS 6.2.2, "RSB Charter and Rules," has the following items:

a. *Specification 2, "Quorum," states, in part, that "[a] quorum is comprised of 3 voting members." This limits the quorum to three, and doesn't match the guidance in ANSI/ANS-15.1-2007, which states that a quorum is not less than half of the voting membership. Revise TS 6.2.2, Specification 2, to follow the guidance in ANSI/ANS-15.1-2007, or justify why no change is necessary.*

b. *Specification 5, "Meeting Minutes," does not follow the guidance in ANSI/ANS-15.1-2007, which provides that the meeting minutes will be provided to the Level 1 in a timely manner. Revise TS 6.2.2, Specification 5, to follow the guidance in ANSI/ANS-15.1-2007, or justify why no change is necessary.*

NSC Response A: The TEES/TAMUS proposed TS 6.2.2, "RSB Charter and Rules," Specification 2, "Quorum," will be revised to follow the guidance of ANSI/ANS-15.2-2007, Section 6.2.2, "Charter and rules."

Revision A: Quorum: A quorum is comprised of not less than one-half of the voting membership where the operating staff does not constitute a majority;

NSC Response B: We have no problem with revising the TEES/TAMUS proposed TS 6.2.2, "RSB Charter and Rules," Specification 5, "Meeting Minutes," to explicitly list the Director of TEES (Level 1 Management) along with the rest of the RSB, and to include the phrase "timely manner."

Revision B: Meeting Minutes: The Chairman will designate one individual to act as recording secretary. It will be the responsibility of the secretary to prepare the minutes which will be distributed to the RSB, including the Director of TEES (Level 1 Management), within three months. The RSB will review and approve the minutes of the previous meetings. A complete file of the meeting minutes will be maintained by the Chairman of the RSB and by the Director of the NSC.

40. The TEES/TAMUS proposed TS 6.2.3, "RSB Review Function," has the following items:

a. *Specification 1, states, in part, "unreviewed safety question," which is not consistent with 10 CFR 50.59, "Changes, tests, and experiments," or the guidance in ANSI/ANS-15.1-2007. Revise TS 6.2.3, Specification 1, to comply with 10 CFR 50.59 and the guidance in ANSI/ANS-15.1-2007, or justify why no change is necessary.*

b. *Specifications 4 and 6, state, in part, "the charter" which is not applicable to NRC licensees. Revise TS 6.2.3, Specifications 4 and 6, consistent with NRC licensees, or justify why no change is necessary.*

NSC Response A: The TEES/TAMUS proposed TS 6.2.3, "RSB Review Function," Specification 1 will be revised to comply with 10 CFR 50.59 and the guidance in ANSI/ANS-15.1-2007.

Revision A: Review and evaluation of determinations of whether proposed changes to equipment, systems, tests, experiments, or procedures can be made under 10 CFR 50.59 or would require a change in technical specifications or license conditions;

NSC Response B: The TEES/TAMUS proposed TS 6.2.3, "RSB Review Function," Specifications 4 and 6, will be revised to be consistent to NRC licenses.

Revision B: Review of proposed changes to the technical specifications and U.S. NRC issued license;

Review of violations of technical specifications, U.S. NRC issued license, and violations of internal procedures or instructions having safety significance;

41. *The TEES/TAMUS proposed TS 6.2.4, "RSB Audit Function," contains components of the audit function as provided in ANSI/ANS-15.1-2007, but lacks information provided in the introductory paragraph concerning discussions with cognizant personnel, audit responsibilities, timeliness, and resolution of deficiencies. Revise TS 6.2.4 to follow the guidance in ANSI-15.1-2007, or justify why no changes are necessary.*

NSC Response: The TEES/TAMUS proposed TS 6.2.4, "RSB Audit Function," will be revised to follow the guidance of ANSI/ANS-15.1-2007, Section 6.2.4, "Audit Function."

Revision:

RSB Audit Function

The audit function shall include selective (but comprehensive) examination of operating records, logs, and other documents. Discussions with cognizant personnel and observation of operations should be used also as appropriate. In no case shall the individual immediately responsible for an area perform an audit in that area. Audits shall include but are not limited to the following:

1. Facility operations, including radiation safety, for conformance to the technical specifications, applicable license conditions, and standard operating procedures: at least once per calendar year (interval between audits not to exceed 15 months);

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

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

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

5. The reactor facility security plan and implementing procedures: 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 Director of TEES (Level 1 Management). A written report of the findings of the audit shall be submitted to the Director of TEES (Level 1 Management) and the review and audit group members within 3 months after the audit has been completed.

42. *The TEES/TAMUS proposed TS 6.3, "Radiation Safety," provides requirements for implementing the radiation protection program at TEES/TAMUS NSC. However, TS 6.3 does not follow the guidance in ANSI/ANS-15.1-2007, which refers to the guidance in ANSI/ANS-15.11-2004, "Radiation Protection at Research Reactor Facilities." Revise TS 6.3 to follow the guidance in ANSI/ANS-15.1-2007, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 6.3, "Radiation Safety," will be revised to follow the guidance in ANSI/ANS-15.1-2007.

> Revision: The Radiation Safety Officer shall be responsible for implementing the radiation safety program for the TEES/TAMUS NSC TRIGA Research Reactor. The requirements of the radiation safety program are established in 10 CFR 20. The Program should use the guidelines of the ANSI/ANS-15.11-1993; R2004, "Radiation Protection at Research Reactor Facilities."

43. The TEES/TAMUS proposed TS 6.4, "Procedures," states, in part, that operating procedures "shall be in effect," which may not be clearly defined in all situations. Consider modifying the statement, such as, "shall be used" to ensure effective procedure adherence. Revise TS 6.4 to provide a clear definition for the use of procedures, or justify why no change is necessary.

NSC Response: The TEES/TAMUS proposed TS 6.4, "Procedures," will be revised to provide a clear definition for the use of procedures.

Revision: Operating procedures shall be used for the following items:

44. *The TEES/TAMUS proposed TS 4.6, "Experiments," Specification 1, states, in part, that experiments shall be reviewed for compliance with Section 6.5, "Experiment Review and Approval," of the TSs. TS 6.5 require reviews and approvals by the RSB and TEES Director* (**note:** I believe the NRC author meant to write "NSC Director."). *However, there is no requirement to review the proposed experiments in accordance with TS 3.6, "Limitations on Experiments." Consider adding a review for compliance with TS 3.6 to Specification 1 of TS 4.6. Revise TS 4.6, Specification 1 to add TS 3.6 to the review requirement, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 4.6, "Experiments," Specification 1 will be revised to include a review for compliance with TS 3.6 along with a review for compliance with TS 6.5.

> Revision: A new experiment shall not be installed in the reactor or its experimental facilities until a hazard analysis has been performed and reviewed for compliance with Section 3.6 and Section 6.5 of the Technical Specifications.

45. *The TEES/TAMUS proposed TS 6.5, "Experiment Review and Approval," Specification 2, provides requirements for changes to previously approved experiments. However, the requirements do not include a review in accordance with the requirements of 10 CFR 50.59. Revise TS 6.5 to add the review requirements associated with 10 CFR 50.59, or justify why no changes are necessary.*

NSC Response: The TEES/TAMUS proposed TS 6.5, "Experiment Review and Approval," Specification 2 requires a review by the RSB. The RSB review, as described in TS 6.2.3, "RSB Review Function," Specification 1 requires a review in accordance with the requirements of 10 CFR 50.59. The statement "The NSC Director or his designated alternate may approve minor changes that do not significantly alter the experiment," will be removed to ensure all changes to previously approved experiments undergo a 10 CFR 50.59 review.

46. *The TEES/TAMUS proposed TS 6.6.1, "Action to be Taken in the Event of a Safety Limit Violation," Specification 2, states, in part, that an immediate notification is made to NRC, and a report in accordance with TS 6.7.2, "Special Reports" (within the next day). The immediate notification is not provided in the guidance in ANSI/ANS-15.1-2007, and may result in duplicative reporting. Revise TS 6.6.1, Specification 2, to follow the guidance in ANSI/ANS-15.1-2007, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 6.6.1, "Action to be Taken in the Event of a Safety Limit Violation," Specification 2 will be revised to follow the guidance in ANSI/ANS-15.1-2007, "Action to be taken in case of safety limit violation."

> Revision: An immediate notification of the occurrence shall be made to the RSB Chairman and the NSC Director, and reports shall be made to the U.S. NRC in accordance with Section 6.7.2 of these specifications;

47. *The TEES/TAMUS proposed TS 6.7.1, "Annual Operating Report," provides requirements for information for the annual report. However, several of the Specification 1 through 8 are not consistent with the guidance in ANSI/ANS-15.1-2007, Section 6.7.1, "Operating Reports." For example, Specification 3 provides the number of emergency shutdowns; whereas the guidance in ANSI/ANS-15.1-2007 requests unscheduled shutdowns and, where applicable, corrective actions to preclude recurrence. Specification 6 does not include any estimate of the individual radionuclides, as practicable. Revise TS 6.7.1 to include the guidance provided in ANSI/ANS-15.2-2007, Section 6.7.1, or justify why no changes are necessary.*

NSC Response: The TEES/TAMUS proposed TS 6.7.1, "Annual Operating Report," will be revised to include the guidance provided in ANSI/ANS-15.1-2007, Section 6.7.1, "Operating Reports."

Revisions:

1. A narrative summary of (1) reactor operating experience...

3. The number of unscheduled shutdowns and inadvertent scrams, including, where applicable corrective action to preclude recurrence;

6. 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 before the point of such release or discharge. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed or recommended, a statement to this effect is sufficient:

48. The TEES/TAMUS proposed TS 6.7.2, "Special Reports," Specification 1.b, states, in part, "accidental release." The guidance in ANSI/ANS-15.1-2007 does not include accidental, but refers to any release. Revise TS 6.7.2, Specification 1.b to follow the guidance in ANSI/ANS-15.1-2007, or justify why no change is necessary.

NSC Response: The TEES/TAMUS proposed TS 6.7.2, "Special Reports," Specification 1.b will be revised to follow the guidance in ANSI/ANS-15.1-2007, Section 6.7.2, "Special Reports."

Revision: Any release of radioactivity from the site above allowed limits…

49. The TEES/TAMUS updated SAR, dated May 2011, does not appear to provide any information on accidents associated with the mishandling or malfunction of equipment. NUREG-1537, Part 1, Section 13.1.9, "Mishandling or Malfunction of Equipment" provides guidance for types of accidents that should be evaluated. Provide an analysis of the accidents that may be associated with mishandling or malfunction of equipment, or justify why no information is necessary.

NSC Response: NUREG-1537, Part 1, Section 13.1.9, "Mishandling or Malfunction of Equipment" identifies six examples of initiators in this category. Namely, (1) operator error at the controls; (2) other operator errors; (3) malfunction or loss of safety-related instruments or controls, such as amplifiers or power supplies; (4) electrical fault in control/safety rod systems; (5) malfunction of confinement or containment system; (6) rapid leak of contaminated liquid, such as waste or primary coolant. Each of these will be considered. While some conditions considered may result in a license violation, the focus here will be on accidents.

(1) Operator error at the controls. As there are no systems that are greatly affected by the operational status of the reactor, e.g. pressure loops, etc., an error resulting in a reactor scram creates no potential of an accident. In this case, we would proceed with our standard recovery from an unscheduled shutdown procedure. There are no possible operator errors at the controls in any reactor condition that could result in a fuel temperature high enough to damage the fuel without an automatic scram being actuated.

(2) Other operator errors. The most credible operator error outside of those possible at the controls would be to open a sequence of valves that would begin draining the pool. This accident would, in the worst case, be realized when the low pool level alarm actuated at a pool level 3 feet lower than the reference operating level. This accident would be an isolable pool leak (NOUE), and our emergency response procedure would be activated. The source of the leak would be isolated (valves closed) and pool level restored to an acceptable level.

Outside of this, other errors are no more unique than in any laboratory environment. Regular TAMUS laboratory inspections, as well as our own internal safety plans, keep these risks to a minimum.

(3) Malfunction or loss of safety-related instruments or controls, such as amplifiers or power supplies. A malfunction or loss of safety-related instruments or controls result in a failure of the channel, which results in an automatic reactor scram. As there are no systems that are greatly affected by the operational status of the reactor, e.g. pressure loops, etc., an error resulting in a reactor scram creates no potential of an accident.

(4) Electrical fault in control/safety rod systems. All of our scrammable rods will scram on electrical fault. The shim safety control rods rely on electromagnets to couple the rod to the drive system, and the transient rod relies on a solenoid valve whose deenergized position is to vent air from the system. An electrical fault in this system will result in a reactor scram, since all of the scrammable rods will drop. As there are no systems that are greatly affected by the operational status of the reactor, e.g. pressure loops, etc., an error resulting in a reactor scram creates no potential of an accident.

(5) Malfunction of confinement or containment system. Confinement is only required while the reactor is operating, irradiated fuels or fueled experiments are being moved, core or control rod work that could cause a change in reactivity of more than one dollar is being performed, or during handling of radioactive materials with the potential for airborne release. During these conditions, the control room is staffed. A failure of the confinement system results in an alarm in the control room. While it is permissible to continue these operations for one hour during confinement system maintenance, they would be stopped while we decided whether there is a need to use that permission or not. At that point, the operations that make confinement necessary would not be happening, and so no accident is possible.

(6) Rapid leak of contaminated liquid, such as waste or primary coolant. We can exclude primary coolant as a source of contaminated liquid. TS 3.8.1, "Primary Coolant Purity," Specification 3 requires the primary coolant to be releasable under 10 CFR 20, Appendix B, Table 2. Our only source of contaminated liquid is our liquid effluent. A rapid leak from the liquid waste tanks could not result in unacceptable dose to NSC personnel, and the area would not be accessible to members of the public. Contamination response would proceed in accordance with existing procedures.

50. *The TEES/TAMUS proposed TS 1.3, "Definitions," "Experiment," provides an exclusion for experimental samples, which is not consistent with the guidance in ANSI/ANS-15.1-2007, Reactivity* (**note:** I believe the NRC author meant to write "Experiment."). *Revise the definition of Experiments to follow the guidance in ANSI/ANS-15.1-2007, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 1.3, "Definitions," "Experiment," will be revised to follow the wording of ANSI/ANS-15.1-2007.

Revision: An operation, hardware, or target (excluding devices such as detectors, foils, etc.)...

51. *The TEES/TAMUS proposed TS 1.3, "Definitions," provides definitions for "Channel" and "Measuring Channel," which appear identical. Redundant definitions could unnecessarily complicate TS compliance. Revise the definitions of Channel and Measuring Channel for consistency, or justify why no change is necessary.*

NSC Response: The term "measuring channel" occurs 12 times outside of the TEES/TAMUS proposed TS 1.3, "Definitions." However, it could be replaced by "channel" in every instance and the definition of "measuring channel" be deleted without affecting the meaning of the document. The term "measuring channel," and its definition, will be replaced by "channel" throughout the document.

52. The TEES/TAMUS proposed TS 5.3, "Reactor Core," Specifications 1, 3 and 4 appear identical to proposed TS 3.1.4, "Core Configuration Limitation," Specifications 1, 2, and 3. Redundant specifications could unnecessarily complicate TS compliance. Revise TS 5.3, or TS 3.1.4, for consistency, or justify why no change is necessary.

NSC Response: The TEES/TAMUS proposed TS 3.1.4, "Reactor Core Configuration," will be revised to remove specifications with no possible normal surveillance associated with them to proposed TS 5.3, "Reactor Core."

Revision:

3.1.4 Reactor Core Configuration

Specifications

1. Control rods shall not be manually removed from the core unless the core has been shown to be subcritical, and shutdown margin requirements met, with those control rods removed.

2. Core lattice positions shall not be vacant except for positions on the periphery of the core assembly while the reactor is operating. Water holes in the inner fuel region shall be limited to single rod positions. Vacant core positions not on the periphery shall contain experiments or an experimental facility to prevent accidental fuel additions to the core.

3. The instrumented fuel element, if serving as the Limiting Safety System sensor, shall be located adjacent to the central bundle with the exception of the corner positions.

5.3 Reactor Core

Specifications

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

2. The reflector, excluding experiments and experimental facilities, shall be any combination of graphite, water, and heavy water.

3. Fuel shall not be inserted or removed from the core unless the reactor is subcritical by at least 50 cents more than the calculated worth of the most reactive fuel assembly.

53. *The TEES/TAMUS proposed TS 3.2.3, "Minimum Number of Operable Scrammable Control Rods and Scram Time," provides operability requirements for the scrammable control rods. However, the regulating rod is not included, and, as such, it is not clear if the regulating rod is required to be operable for reactor operation or reactor secured. Provide TS requirements on the operability of the regulating rod, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 3.2.3, "Minimum Number of Operable Scrammable Control Rods and Scram Time," will be revised to clearly include the operability of the regulating rod.

> Revision: During operation all control rods shall be operable. For scrammable control rods, the scram time measured from the instant a SCRAM signal is initiated to the instant that the slowest scrammable rod reaches its fully inserted position shall not exceed 1.2 seconds. During core manipulations, i.e. core loading and unloading, all installed control rods shall be operable.

54. *The TEES/TAMUS updated SAR, Section 5.3, "Secondary Coolant System," describes cooling tower water chemistry control through make-up and blow down, but does not explain the system's operation (i.e., if it's automatic or manually operated, or how the blow down material is handled). Provide a more comprehensive description of the cooling tower water chemistry control system, including the handling of any cooling tower blow down, or justify why no additional information is necessary.*

NSC Response: The cooling tower water is regularly monitored for radioactive material that may be present from failures in the boundary of the heat exchanger. Radioactive material in this water would require further investigation. An automatic control system, provided by ChemCal, monitors water chemistry and initiates chemical additions (see response to RAI 1 for chemical descriptions) or blow downs as necessary. A ChemCal technician visits the site monthly to perform maintenance on the system and make adjustments based on the previous month's data, and calibrations performed at the time. This automatic control system can be overridden and any of the components be controlled manually. However, manual control of the system is extremely rare. The system blows down through the site's sanitary sewer system and is in compliance with all state and local regulations.

55. *The TEES/TAMUS proposed TS 3.5.1, "Radiation Monitoring," Table 3, Footnote, and TS 3.5.3, "Xenon and Iodine Monitoring," Table 4, Footnote, provide requirements for out of service radiation monitors. However, the requirements are not clearly delineated. Provide revised TS 3.5.1, and TS 3.5.3, Table 3, and 4, Footnotes, or justify why no changes are necessary.*

NSC Response: The TEES/TAMUS proposed TS 3.5.1, "Radiation Monitoring," Table 3, Footnote, and TS 3.5.3, "Xenon and Iodine Monitoring," Table 4, Footnote will be revised to improve clarity.

> Revision: When a required channel becomes inoperable, operations may continue only if a portable gamma-sensitive ion chamber is utilized as a temporary substitute, provided that the substitute can be observed by the reactor operator, can be installed within 1 hour of discovery, and is not used longer than one week. If two of the above monitors are not operating, operations shall cease.

56. *The TEES/TAMUS response to RAI no. 3, by letter dated January 12, 2012 (a redacted version is available in ADAMS Accession No. ML120260016), provided an assumption that the doses to any members of the public in the laboratory areas are no worse than the doses present at the fence line under the plume. The basis for this assumption was not provided. Provide the basis to substantiate this assumption provided in the response to RAI no. 3, or justify why no additional information is necessary.*

NSC Response: Members of the general public can be present in the laboratory building adjacent to the reactor building. After evacuation of the confinement building, our Emergency Plan calls for evacuation of the laboratory building if the dose rate in the laboratory building exceeds 5 mrem. We assume under these conditions that evacuation would take at most 5 minutes. Doses to unrestricted personnel inside the laboratory building are assumed to be no worse than the dose received by the maximally exposed operating personnel in confinement for the same amount of time after the release. The basis for this assumption is that the confinement building would contain most of the radioactivity release and the doses for unrestricted personnel would be less than those in restricted areas. Additionally, the doses for personnel in restricted areas are within regulations for doses to the general public.

Exposure Criteria	Immersion DDE	Inhalation CEDE	TEDE
	mrem	mrem	mrem
Maximum exposure to operating personnel at 5 minutes after release	6.2	1.78E+1	24
Maximum exposure to personnel in unrestricted area at 1 hour after release	0.4	8.91E-2	.4891
Exposure to personnel in nearest permanently occupied area (nearest resident) at 1 hour after release	0.1	2.55E-2	.1255
Exposure to personnel inside the Laboratory Building	6.2	1.78E+1	24

57. The TEES/TAMUS proposed TS 3.3.1, "Operations that Require Confinement," provides requirements for confinement. However, some the guidance provided in ANSI/ANS-15.1-2007 is not contained in the proposed TS 3.3.1. Revise TS 3.3.1 to include the guidance provided in ANSI/ANS-15.1-2007, or justify why no change is necessary.

NSC Response: The TEES/TAMUS proposed TS 3.3.1, "Operations that Require Confinement," will be revised to include the guidance provided in ANSI/ANS-15.1-2007 3.4.1, "Operations that require containment or confinement."

Revision: Confinement of the reactor building shall be required during the following operations:

1. Reactor operating;

2. Movement of irradiated fuels or fueled experiments;

3. Core or control rod work that could cause a change in reactivity of more than one dollar; or

4. Handling of radioactive materials with the potential for airborne release.

Appendix A

TO

FACILITY LICENSE NO. R-83
DOCKET NO. 50-128

TECHNICAL SPECIFICATIONS AND BASES

TEXAS ENGINEERING EXPERIMENT STATION
NUCLEAR SCIENCE CENTER (NSC)

MARCH 2015

TECHNICAL SPECIFICATIONS

1 Introduction

1.1 Scope

This document constitutes the Technical Specifications for the Facility License No. R-83 as required by 10 CFR 50.36 and supersedes all prior Technical Specifications. This document includes the "bases" 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.

1.2 Format

These specifications are formatted to NUREG-1537 and ANSI/ANS 15.1-2007.

1.3 Definitions

ALARA

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

Audit

An audit is a quantitative examination of records, procedures, or other documents after implementation from which appropriate recommendations are made.

Channel

A channel is the combination of sensors, lines, amplifiers, and output devices that are connected for the purpose of measuring the value of a parameter.

Channel Test

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

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 that the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include a channel test.

Channel Check

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.

Confinement

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

Control Rod

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

Regulating Control Rod

The regulating rod is a low-worth control rod used primarily to maintain an intended power level that need not have scram capability. Its axial position may be varied manually or by the servo controller.

Shim Safety Control Rod

A shim safety rod is a control rod having an electric motor drive and scram capabilities. It shall have a fueled follower section.

Transient Control Rod

The transient rod is a pneumatically driven control rod with scram capabilities that is capable of providing rapid reactivity insertion to produce a pulse.

Core Configuration

The core configuration includes the number, type, or arrangement of fuel elements, reflector elements, and regulating/shim-safety/transient rods occupying the core grid.

Core Lattice Position

The core lattice position is that region in the core (approximately 3" x 3") over a grid-plug hole. A fuel bundle, an experiment, or a reflector element may occupy the position.

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 (*keff* = 1) at reference core conditions.

Experiment

An operation, hardware, or target (excluding devices such as detectors, foils, etc.) that is designed to investigate non-routine reactor characteristics, or that is intended for irradiation within the pool, or in a beam port or irradiation facility. Hardware rigidly secured to a core or shield structure so as to be a part of its design to carry out experiments is not normally considered an experiment.

Secured Experiment

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

Unsecured Experiment

An unsecured experiment is any experiment or component of an experiment that does not meet the definition of a secured experiment.

Movable Experiment

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

Experimental Facilities

Experimental facilities shall mean beam ports, including extension tubes with shields, thermal columns with shields, vertical tubes, through tubes, in-core irradiation baskets, irradiation cell, pneumatic transfer systems, and in-pool irradiation facilities.

Experiment Safety Systems

Experiment safety systems are those systems, including their associated input circuits, which are designed to initiate a scram for the primary purpose of protecting an experiment or to provide information for operator intervention.

Fuel Bundle

A fuel bundle is a cluster of two, three, or four fuel elements and/or non-fueled elements secured in a square array by a top handle and a bottom grid plate adapter. Non-fueled elements shall be fabricated from stainless steel, aluminum, boron, or graphite materials.

Fuel Element

A fuel element is a single TRIGA fuel rod of LEU 30/20 type.

Instrumented Fuel Element (IFE)

An instrumented fuel element is a special fuel element in which one or more thermocouples are embedded for the purpose of measuring the fuel temperatures during operation.

License

The written authorization, by the U.S. NRC, for an individual or organization to carry out the duties and responsibilities associated with a personnel position, material, or facility requiring licensing.

Licensee

A licensee is an individual or organization holding a license.

LEU Core

An LEU core is an arrangement of TRIGA-LEU fuel in a reactor grid plate.

Limiting Safety System Setting (LSSS)

The limiting safety system setting is the fuel element temperature, which if exceeded, shall cause a reactor scram to be initiated, preventing the safety limit from being exceeded.

Measured Value

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

Operable

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

Operating

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

Operational Core – Steady State

A steady state operational core shall be an LEU core which meets the requirements of the Technical Specifications.

Operational Core – Pulse

A pulse operational core is a steady state operational core for which the maximum allowable pulse reactivity insertion has been determined.

Pool Water Reference Operating Level

The pool water reference operating level is 10 inches below the top of the pool wall. This level is designed to prevent pool water from rising above the top of the liner and from submerging the fission product facility air monitor intake.

Protective Action

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

Pulse Mode

Pulse mode operation shall mean any operation of the reactor with the mode selector switch 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 Console Secured

The reactor console is secured whenever all scrammable rods have been verified to be fully inserted and the console key has been removed from the console.

Reactor Operating

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

Reactor Operator

A Reactor Operator is 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 automatic reactor protection or to provide information for initiation of manual protective action.

Reactor Secured

The reactor is secured when:

Either

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

Or

(2) All of the following conditions exist:

(a) All control rods are fully inserted;

(b) The console key switch is in the "off" position and the key is removed from the console lock;

(c) The reactor is shutdown;

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

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

Reactor Shutdown

The reactor is shut down if it is subcritical by at least $1.00 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) and the reactivity worth of xenon is negligible.

Reportable Occurrence

Any of the following events is a reportable occurrence:

(1) Operation with actual safety system settings for required systems less conservative than the LSSS specified in the Technical Specifications;

(2) Operation in violation of a Limiting Condition of Operation listed in Section 3 unless prompt remedial action is taken as permitted in Section 3;

(3) Operation with a required reactor or experiment safety system component in an inoperative or failed condition which renders or could render the system incapable of performing its intended safety function. If the malfunction or condition is caused during maintenance, then no report is required;

(4) An unanticipated or uncontrolled change in reactivity greater than $1.00. Reactor trips resulting from a known cause are excluded;

(5) Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or confinement boundary; and

(6) 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 an unsafe condition with regard to reactor operations.

Review

A review is a qualitative examination of records, procedures, or other documents prior to implementation from which appropriate recommendations are made.

Safety Channel

A safety channel is a channel in the reactor safety system.

Safety Limit

Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. For the Texas A&M NSC TRIGA reactor the safety limit is the maximum fuel element temperature that can be permitted with confidence that no damage to any fuel element cladding will result.

Scram Time

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

Senior Reactor Operator

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

Shall, Should and May

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

Shutdown Margin

Shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems, starting from any permissible operating condition. This margin is determined assuming that the most reactive scrammable rod and any non-scrammable rods are fully withdrawn, and that the reactor will remain subcritical by this calculated margin without any further operator action.

Steady State Mode

Steady state mode of operation shall mean operation of the reactor with the mode selector switch in the steady state position.

Surveillance Intervals

The maximum surveillance intervals are provided for operational flexibility and the average surveillance intervals should be maintained over the long term.

Annually - an interval not to exceed 15 months.
Biennially - an interval not to exceed 30 months.
Monthly - an interval not to exceed 6 weeks.
Quarterly - an interval not to exceed 4 months.
Semiannually - an interval not to exceed 7.5 months.
Weekly - an interval not to exceed 10 days.

True Value

The true value is the actual value of a parameter.

Unscheduled Shutdown

An unscheduled shutdown is 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. It does not include shutdowns that occur during testing or check out operations.

2 Safety Limit and Limiting Safety System Setting

2.1 Safety Limit-Fuel Element Temperature

Applicability

This specification applies to the temperature of 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 will result.

Specification

The temperature in a stainless steel-clad TRIGA LEU fuel element shall not exceed 2100 °F (1150°C) under any conditions of operation.

Basis

The most important safety limit for a TRIGA reactor is fuel element temperature. This parameter is well suited as a single specification because it can be measured directly with a thermocouple. A loss in the integrity of the fuel element cladding could arise from a buildup of excessive pressure if the fuel element temperature exceeds the temperature safety limit. The fuel element temperature and the ratio of hydrogen to zirconium in the fuel-moderator material determine the magnitude of the pressure buildup. The mechanism for the pressure buildup is the dissociation of hydrogen from the zirconium hydride moderator that has been blended with uranium to form the fuel mixture encased within the fuel element cladding.

The temperature safety limit for the LEU fuel element is based on data which indicates that the internal stresses within the fuel element, due to hydrogen pressure from the dissociation of the zirconium hydride, will not result in compromise of the stainless steel cladding if the fuel temperature is not allowed to exceed 2100°F (1150°C) and the fuel element cladding is water cooled.

2.2 Limiting Safety System Setting

Applicability

This specification applies to the scram setting that prevents the safety limit from being reached.

Objective

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

Specification

The limiting safety system setting shall be 975°F (525°C) as measured in an instrumented fuel element (IFE). The IFE shall be located adjacent to the central bundle with the exception of the corner positions.

Basis

The limiting safety system setting (LSSS) is a temperature that, if exceeded, will cause a reactor scram to be initiated preventing the safety limit from being exceeded.

The temperature safety limit for LEU fuel is 2100°F (1150°C). Due to various errors in measuring temperature in the core, it is necessary to arrive at a LSSS for the fuel element safety limit that takes into account these measurement errors. The results of analysis provided in SAR 13.3 Evaluation of LSSS for NSC LEU 30/20 Fuel indicate that a LSSS temperature of 975°F (525°C) is appropriate.

In the pulse mode of operation, the above temperature limiting safety system setting will apply. However, the temperature channel will have no effect on limiting peak powers generated because of its relatively long time constant (seconds) as compared with the width of the pulse (milliseconds). In this mode, however, a temperature trip will act to reduce the amount of energy generated in the entire pulse transient by cutting the "tail" off the energy transient in the event the pulse rod remains stuck in the fully withdrawn position.

3 Limiting Conditions for Operation

3.1 Reactor Core Parameters

3.1.1 Steady State Operation

Applicability

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

Objective

The objective is to ensure that the fuel temperature safety limit will not be exceeded during steady state operation.

Specification

The reactor power level shall not exceed 1.0 megawatt (MW) during steady state operation.

Basis

Calculations show that reactor operations with a pool temperature of 60°C will not risk reaching a fuel temperature greater than the LSSS, or a DNBR of unity for 1MW steady state. (Response to U.S. NRC Request for Additional Information, Review of The Fuel Pool Temperature on Fuel Temperature, submitted November 13, 2014).

3.1.2 Pulse Mode Operation

Applicability

This specification applies to the peak temperature generated in the fuel as the result of a pulse insertion of reactivity.

Objective

The objective is to ensure that respective pulsing will not induce damage to the reactor fuel.

Specification

The reactivity to be inserted for pulse operation shall not exceed that amount which will produce a peak fuel temperature of 1526°F (830°C). In the pulse mode the pulse rod shall be limited by mechanical means or the rod extension physically shortened so that the reactivity insertion will not inadvertently exceed the maximum value.

Basis

The pulsing limit of 830°C will be translated to a reactivity insertion limit for each specific core. The peaking factors from the thermocouple element to the hottest spot in the core must be calculated for each core configuration that is to be used. Temperature would then be measured for small pulse insertions.

The initial core calibration in 2006 established the maximum allowable pulse insertion to be $1.91. Any subsequent pulse insertion change shall only be made after core recalibration and following approval by the NSC staff.

3.1.3 Shutdown Margin

Applicability

This specification applies to the reactivity condition of the reactor and the reactivity worth of control rods and experiments. They apply for all modes of operation.

Objective

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

Specification

The reactor shall not be operated unless the shutdown margin provided by control rods is greater than $0.50 with:

1. Irradiation facilities and experiments in place and the total worth of all non-secured experiments in their most reactive state,

2. The highest worth control rod and the regulating rod fully withdrawn, and

3. The reactor in reference core condition.

Basis

The value of the shutdown margin ensures that the reactor can be shut down from any operating condition even if the highest worth control rod should remain in the fully withdrawn position. Since the regulating rod is not scrammable, its worth is not used in determining the shutdown reactivity.

3.1.4 Reactor Core Configuration

Applicability

This specification applies to the configuration of the 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 will not be produced.

Specification

1. Control rods shall not be manually removed from the core unless the core has been shown to be subcritical, and shutdown margin requirements met, with those control rods removed.

2. Core lattice positions shall not be vacant except for positions on the periphery of the core assembly while the reactor is operating. Water holes in the inner fuel region shall be limited to single rod positions. Vacant core positions not on the periphery shall contain experiments or an experimental facility to prevent accidental fuel additions to the core.

3. The instrumented fuel element, if serving as the Limiting Safety System sensor, shall be located adjacent to the central bundle with the exception of the corner positions.

Basis

1. Manipulation of core components will be allowed only when a single manipulation cannot result in inadvertent criticality.

2. Vacant core positions containing experiments or an experimental facility will prevent accidental fuel additions to the reactor core. They will be permitted only on the periphery of the core or a single rod position to prevent power peaking in regions of high power density.

3. SAR 13.3.1 Steady State Mode provides an evaluation of the LSSS in steady state mode. It states in part, "The location of the fuel cluster containing the instrumented fuel element shall be chosen to be as close as possible to the hottest fuel element in the core." The location(s) as close as possible to the hottest fuel element in the core are those adjacent to the central bundle with the exception of the corner positions. These adjacent positions are: C5 east, D4 north, E5 west, D6 south.

3.1.5 Reactor Fuel Parameters

Applicability

This specification applies to all fuel elements.

Objective

The objective is to maintain the integrity of the fuel elements' cladding.

Specification

1. The reactor shall not be operated knowingly with damaged fuel, 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. In measuring the transverse bend, the bend exceeds 0.125 inch over the length of the cladding, or

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

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

    d. A visual inspection reveals bulges, gross pitting or corrosion.

3. The burnup of the uranium-235 in the UZrH fuel matrix shall not exceed 50 percent of the initial concentration.

Basis

Gross failure or obvious visual deterioration of the fuel is sufficient to warrant declaration of the fuel as damaged. The elongation, bend, and burn-up limits are values that have been found acceptable to the U.S. NRC (NUREG-1537).

3.1.6 Maximum Excess Reactivity

Applicability

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

Objective

The objective is to ensure that the reactor can be shutdown at all times.

Specification

The maximum reactivity in excess of reference core condition shall not exceed 5.5% Δk/k ($7.85).

Basis

This specification, although over-constraining the reactor system, helps ensure that the licensee's operational power densities, fuel temperatures, and temperature peaks are maintained within the evaluated safety limits. The specified excess reactivity allows for power coefficients of reactivity, xenon poisoning, most experiments, and operational flexibility.

3.2 Reactor Control and Safety Systems

3.2.1 Reactor Channels

Applicability

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

Objective

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

Specification

The reactor shall not be operated in the specific mode of operation unless the channels listed in Table 1 are operable.

Table 1: Channels Required for Operation

Channel	Minimum Number Operable	Operating Mode	
		S.S.	Pulse
Fuel Element Temperature	1	X	X
Linear Power Level	1	X	-
Log Power Level	1	X	-
Integrated Pulse Power	1	-	X
Pool Water Temperature	1	X	X

Basis

Fuel temperature displayed at the control console gives continuous information on this parameter, which has a specified safety limit. The power level monitors ensure that the reactor

power level is adequately monitored for both steady state and pulsing modes of operation. The specification on reactor power level indications are included in this section, since the power level is related to fuel temperature. The specification on pool water temperature indication is included in this section to allow monitoring in support of TS 3.8.3 and 4.8.3.

3.2.2 Reactor Safety Systems and Interlocks

Applicability

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

Objective

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

Specification

The reactor shall not be operated unless the safety circuits and interlocks described in Tables 2a and 2b are operable. However, any single safety channel or interlock may be inoperable with the reactor is operating for the purpose of maintenance, surveillance, calibration, or repair. If any required safety channel or interlock becomes inoperable while the reactor is operating, for reasons other than identified in this TS, then the channel shall be restored to operation within 5 minutes or the reactor shall be immediately shutdown.

Table 2a: Safety Channels Required for Operation

Safety Channel	Number Operable	Function	Operating Mode	
			S.S.	Pulse
Fuel Element Temperature	1	Scram ≤ 975°F	X	X
High Power Level	2	Scram ≤ 1.25MW	X	-
Console Scram Button	1	Manual Scram	X	X
High Power Level Detector Power Supply	2	Scram on loss of supply voltage	X	-
Preset Timer	1	Transient Rod Scram 15 seconds or less after pulse	-	X
Pool Water Temperature	1	Manual scram if temperature reaches 60 C	X	X

Table 2b: Interlocks Required for Operation

Safety Channel	Number Operable	Function	Operating Mode	
			S.S.	Pulse
Log Power	1	Prevents withdrawal of the Shim Safety Control Rods at an indicated log power of less than $4x10^{-3}$ W.	X	-
Log Power	1	Prevents pulsing of the Transient Rod when log power is above 1 kW.	-	X
Transient Rod Position	1	Prevents application of air to the Transient Rod unless the Transient Rod is fully inserted.	X	-
Shim Safety and Regulating Rod Position	1	Prevents Shim Safety and Regulating Control Rod withdrawal during a pulse.	-	X
Pulse Stop Electro-Mechanical Interlock	1	Prevents application of air to the Transient Rod unless the mechanical pulse stop is installed.	-	X

Basis

During period of maintenance, surveillance, calibration, and repair, true signals generated by the reactor may be required. This operation is allowable provided all other TS conditions are met. If multiple channels have failed simultaneously, then involvement of the U.S. NRC in recovery planning will be necessary.

Safety Channels Required for Operation

1. The fuel temperature and high power level scrams provide protection to ensure that the reactor can be shutdown before the safety limit on fuel element temperature will be exceeded.

2. The manual console scram allows the operator to shut down the system if an unsafe or abnormal condition occurs.

3. In the event of failure of the power supply for a high power level safety detector, operation of the reactor without adequate instrumentation is prevented.

4. The preset timer ensures that the reactor power level will reduce to a low level after pulsing.

5. A manual scram is a sufficient response because pool water temperature is a slow-changing parameter. It is surveyed frequently enough to give time for an operator to respond.

Interlocks Required for Operation

1. The interlock to prevent startup of the reactor at power levels less than 4 x 10-3 W, which corresponds to approximately 2 cps, ensures that sufficient neutrons are available for proper startup.

2. The interlock to prevent pulsing at powers above 1 kW ensures that the magnitude of the pulse will not cause the fuel element temperature safety limits to be exceeded.

3. The interlock to prevent application of air to the transient rod unless the cylinder is fully inserted is to prevent pulsing of the reactor in steady state mode.

4. The interlock to prevent the withdrawal of the shim safeties or regulating rod in the pulse mode is to prevent the reactor from being pulsed while on a positive period.

5. The interlock to prevent application of air to the transient rod unless the cylinder is fully inserted prevents a reactor pulse of sufficient worth to exceed the temperature safety limit.

3.2.3 Minimum Number of Operable Scrammable Control Rods and Scram Time

Applicability

This specification applies to the minimum number of operable scrammable control rods in the core, where operable is specified in terms of maximum scram time from the instant that any SCRAM signal is initiated.

Objective

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

Specification

During operation, all control rods shall be operable. For scrammable control rods, the scram time measured from the instant a SCRAM signal is initiated to the instant that the slowest scrammable rod reaches its fully inserted position shall not exceed 1.2 seconds. During core manipulations, i.e. core loading and unloading, all installed control rods shall be operable.

Basis

This specification ensures that the reactor will be promptly shutdown 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.

3.3 Confinement

3.3.1 Operations that Require Confinement

Applicability

This specification applies to the area housing the reactor and the ventilation system controlling that area.

Objective

To provide restrictions on radioactive airborne material releases into the environment.

Specification[1]

Confinement of the reactor building shall be required during the following operations:

1. Reactor operating;
2. Movement of irradiated fuel elements or fuel bundles;
3. Core or control rod work that could a change in reactivity of more than one dollar; or
4. Handling of radioactive materials with the potential for airborne release.

[1] For periods of maintenance to the central exhaust system, entry doors to the reactor building shall remain closed except for momentary opening for personnel entry or exit. The central exhaust system shall not remain inoperable during periods of maintenance for more than one hour.

Basis

This specification describe when the central exhaust system shall operate to control any release of radioactive material in the confinement building.

3.3.2 Equipment to Achieve Confinement

Applicability

This specification applies to the equipment and controls needed to provide confinement of the reactor building.

Objective

The objective is to ensure that a minimum of equipment is in operation to achieve confinement as specified in Section 3.3.1 and that the control panel for this equipment is available for normal and emergency situations.

Specification[2]

1. The minimum equipment required to be in operation to achieve confinement of the reactor building shall be the central exhaust system, which consists of the central exhaust fan, isolation louvers, and associated duct work.

2. The central exhaust system shall be considered operating when it creates a minimum of 0.1 inch of water negative pressure at the sample point in the central exhaust system duct work.

3. Controls for establishing the operation of the central exhaust system during normal and emergency conditions shall be available in the Emergency Support Center.

4. The central exhaust system shall be isolated automatically by alarm level signals from the stack particulate, fission product, or stack gas (xenon) facility air monitor.

[2] During periods of maintenance to the central exhaust system, entry doors to the reactor building shall remain closed except for momentary opening for personnel entry or exit. The central exhaust fan system shall not remain inoperative during periods of maintenance for more than one hour.

Basis

1. Operation of the central exhaust fan will achieve confinement of the reactor building during normal and emergency conditions when the controls for air input are set such that the central exhaust fan capacity remains greater than the amount of air being delivered to the reactor building. The exhaust fan has sufficient capacity to handle extra air intake to the building during momentary opening of doors. Isolation of the central exhaust fan for periods of less than one hour is for operability verification during weekly ventilation checks. This limit provides enough time to complete these checks.

2. Negative pressure in the confinement building mitigates leakage of unmonitored airborne material to the environment.

3. The control panel for the central exhaust system provides for manual selection of air input to the reactor building and the automatic or manual selection of air removal. The air supply and exhaust systems work together to maintain a small negative pressure in the reactor building. These controls are available in the emergency support center for accessibility during emergency conditions.

4. An automatic isolation of the central exhaust system will mitigate leakage of unmonitored airborne material to the environment.

3.4 Ventilation System

The LCO for Ventilation System is covered by TS 3.3.2 Equipment to Achieve Confinement

3.5 Radiation Monitoring Systems and Effluents

3.5.1 Radiation Monitoring

Applicability

This specification applies to the radiation monitoring information that must be available to the reactor operator during reactor operation, movement of irradiated fuel elements or fuel bundles, conduct of core or control rod work that could cause a change in reactivity of more than one dollar, or handling of radioactive materials with the potential for airborne release.

Objective

The objective is to ensure that sufficient radiation monitoring information is available to the operator to ensure safe operation of the facility.

Specification

The above operations shall not be conducted unless the radiation monitoring channels listed in Table 3 are operable, displays and alarms are operable in the control room, and displays are operable in the Emergency Support Center.

Table 3: Radiation Monitoring Channels Required for Operation[3]

Radiation Monitoring Channels	Function	Number
Reactor Bridge ARM	Monitor radiation levels within the reactor bay	1
Stack Particulate Monitor (FAM Ch. 1)	Monitor radiation levels in the exhaust air stack	1
Stack Gas Monitor (FAM Ch. 3)	Monitor radiation levels in the exhaust air stack	1
Building Particulate Monitor (FAM Ch. 4)	Monitor radiation levels within the reactor bay	1

[3] When a required channel becomes inoperable, operations may continue only if a portable gamma-sensitive ion chamber is utilized as a temporary substitute, provided that the substitute can be observed by the reactor operator, can be installed within 1 hour of discovery, and is not used longer than one week. If two of the above monitors are not operating, operations shall cease.

Basis

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

3.5.2 Argon-41 Discharge Limit

Applicability

This specification applies to the concentration of Argon-41 (^{41}Ar) that may be discharged from the TRIGA reactor facility.

Objective

The objective is to ensure that the health and safety of the public is not endangered by the discharge of ^{41}Ar from the TRIGA reactor facility.

Specification

The total annual discharge of ^{41}Ar into the environment shall not exceed 30 Ci per year.

Basis

If the 30 Ci is assumed to be released continuously over one year, then the Ar-41 concentration at the point of discharge, which is the top of the stack, is 2.5×10^{-7} μCi/ml. This concentration is diluted by a factor of 200 to get the Ar-41 concentration at the site boundary 1.0×10^{-9} μCi/ml. 1.0×10^{-9} μCi/ml corresponds to a dose of 12.6 mrem.

3.6 Limitations on Experiments

3.6.1 Reactivity Limits

Applicability

This specification applies to the reactivity limits on experiments installed in the reactor and its experimental facilities.

Objective

The objective is to ensure control of the reactor during the handling of experiments adjacent to or in the reactor core.

Specification

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

1. The absolute reactivity worth of any single, movable or unsecured experiment shall be less than $1,

2. The reactivity worth of any secured experiment shall be less than $2, and

3. The sum of the absolute reactivity of all experiments shall be less than $5.

Basis

1. This specification is intended to ensure that the worth of a single unsecured experiment will be limited to a value such that the safety limit will not be exceeded if the positive worth of the experiment were suddenly inserted. This does not restrict the number of unsecured experiments adjacent to or in the reactor core except by reactivity worth and the requirements of these TS.

2. The maximum worth of a single experiment is limited so that its removal from the reactor in reference core condition will not result in the reactor achieving a power level high enough to exceed the fuel element temperature safety limit. Since experiments of such worth must be secured, its removal from the reactor operating at full power would result in a relatively slow power increase such that the reactor protective systems would act to prevent high power levels from being attained.

3. This limit poses a restriction on the number of experiments being run at any given time to prevent excessive positive and negative reactivity effects from experiments.

3.6.2 Material Limitations

Applicability

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

Objective

The objective is to prevent damage to the reactor or excessive release of radioactivity by limiting materials quantity and radioactive material inventory of the experiment.

Specification

1. Explosive materials in quantities inclusively between 25 milligrams and 5 pounds (TNT-equivalent) shall not be allowed within the reactor building except as noted below in TS. Explosive materials in quantities greater than 5 pounds (TNT-equivalent) shall not be allowed within the reactor building. Irradiation of explosive materials shall be restricted as follows:

    a. Explosive materials in quantities greater than or equal to 25 milligrams (TNT-equivalent) shall not be irradiated in the reactor pool. Explosive materials in quantities up 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. Explosive materials in quantities greater than or equal to 25 milligrams (TNT-equivalent) shall only be allowed in the lower research level and laboratory building, excluding the heat exchanger room and demineralizer room;

    c. Irradiation of explosive materials in quantities greater than or equal to 25 milligrams (TNT-equivalent) shall be permitted only in the neutron radiograph facility;

    d. Explosive materials in quantities greater than or equal to 5 pounds (TNT-equivalent) shall not be irradiated in experimental facilities; and

    e. Cumulative exposures for explosive materials in quantities greater than or equal to 25 milligrams (TNT-equivalent) shall not exceed 10^{12} n/cm^2 for neutron or 25 Roentgen for gamma exposures.

2. Corrosive materials used in a reactor experiment shall be double encapsulated. Exceptions may only be made if a detailed analysis and/or prototype testing with small amounts of materials demonstrates that the experiment presents negligible risk.

Basis

1. This specification is intended to prevent damage to the reactor or reactor safety systems resulting from failure of an experiment involving explosive materials.

    a. This specification is intended to prevent damage to the reactor core and safety related reactor components located within the reactor pool in the event of failure of an experiment involving the irradiation of explosive materials. Limited quantities of less than 25 milligrams (TNT-equivalent) and proper containment of such experiment provide the required safety for in-pool irradiation provided that 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. (Regulatory Guide 2.2)

b. This specification is intended to prevent damage to vital equipment by restricting the quantity and location of explosive materials within the reactor building. Explosives in quantities exceeding 25 milligrams (TNT-equivalent) are restricted from areas containing the reactor bridge, reactor console, pool water coolant and purification systems, and reactor safety related equipment. (Amendment No. 7 to Facility License No. R-83)

c. This specification supports the same goal as the previous specification. The neutron radiograph facility was analyzed and shown to be able to withstand an explosion of the described quantity. (Amendment No. 7 to Facility License No. R-83)

d. The failure of an experiment involving the irradiation of up to 5 pounds (TNT-equivalent) of explosive material in an experimental facility located external to the reactor pool structure will not result in damage to the reactor or the reactor pool containment structure.

e. This specification is intended to prevent any increase in the sensitivity of explosive materials due to radiation damage during exposures.

   See: Kaufman J.V.R. (Jul. 29, 1958). The Effect of Nuclear Radiation on Explosives. *Proceedings of the Royal Society of London. Series A, Mathematical and Physical Science, Vol. 246* (No. 1245, A Discussion on the Initiation and Growth of Explosion in Solids);

   Urizar M.J., Loughran E.D., Smith L.C. (Jan. 01, 1960). A Study of the Effects of Nuclear Radiation on Organic Explosives; and

   Amendment No. 7 to Facility License No. R-83.

2. This specification is intended to prevent damage to the reactor or reactor safety systems resulting from failure of an experiment involving corrosive materials.

3.6.3 Failures and Malfunctions

Applicability

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

Objective

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

Specification

1. Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under (a) normal operating conditions of the experiment or reactor, (b) credible accident conditions in the reactor, or (c) possible accident conditions in the experiment shall be limited in activity such that if 100% of the gaseous activity or radioactive aerosols produced escaped to the reactor building or the atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the limit of Appendix B of 10CFR20.

2. In calculations pursuant to 1) above, the following assumptions shall be used:

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

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

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

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

Basis

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

2. These assumptions are used to evaluate the potential airborne radioactivity release due to an experiment failure.

3. Operation of the reactor with reactor fuel or structure damage is prohibited to avoid release of fission products. Potential damage to reactor fuel or structure must be brought to the attention of the NSC Director or his designated alternate for review to ensure safe operation of the reactor.

3.7 As Low As Reasonable Achievable (ALARA) Radioactive Effluents Released

Applicability

This specification applies to the measures required to ensure that the radioactive effluents released from the facility are in accordance with ALARA criteria.

Objective

The objective is to constrain the annual radiation exposure to the general public resulting from operation of the reactor to a level as low are reasonably achievable below the constraints listed in 10 CFR 20.1101.

Specification

1. In addition to the radiation monitoring specified in Section 5.5, an environmental radiation monitoring program shall be conducted to measure the integrated radiation exposure in and around the environs of the facility on a quarterly basis.

2. The annual radiation exposure (dose) to the public due to reactor operation shall not exceed the limits defined in 10 CFR 20.1301. The facility perimeter shall be monitored to ensure this specification is being met.

3. In the event of a fission product leak from a fuel rod or an airborne radioactive release from a sample being irradiated, as detected by the facility air monitor (FAM), the reactor shall be shut down until the source of the leak is located and eliminated. However, the reactor may continue to be operated on a short-term basis, as needed, to assist in determining the source of the leakage.

4. The facility liquid effluents collected in the holdup tanks shall be discharged in accordance with 10 CFR 20.2003 "Disposal by release into sanitary sewerage." The liquid effluent shall also meet local sanitary sewer discharge requirements.

Basis

The simplest and most reliable method of ensuring that ALARA release limits are accomplishing their objective of minimal facility-caused radiation exposure to the general public is to actually measure the integrated radiation exposure in the environment on and off the site.

3.8 Primary Coolant Conditions

3.8.1 Primary Coolant Purity

Applicability

This specification applies to the quality of the primary coolant in contact with the fuel cladding.

Objective

The objectives are to minimize the possibility for corrosion of the cladding on the fuel elements and to minimize neutron activation of dissolved materials.

Specification

1. The reactor shall not be operated for a period exceeding two weeks if the two week averaged conductivity of the bulk pool water is higher than 5×10^{-6} mhos/cm.

2. The concentrations of radionuclides in the bulk pool water shall be no higher than the values presented for water in 10 CFR Appendix B to Part 20 Table 2.

Basis

A small rate of corrosion continuously occurs in a water-metal system. In order to limit this rate, and thereby extend the longevity and integrity of the fuel cladding, a water cleanup system is required. Experience with water quality control at many reactor facilities has shown that maintenance within the specified limits provides acceptable control.

By limiting the concentrations of dissolved materials in the water, the radioactivity of neutron activation products is limited. This is consistent with the ALARA principle, and tends to decrease the inventory of radionuclides in the entire coolant system, which will decrease personnel exposure during maintenance and operations.

3.8.2 Primary Coolant Level and Leak Detection

Applicability

This specification applies to the water level that must be in the pool and requirements for leak detection for reactor operation.

Objective

The objective is to ensure proper shielding and cooling of the reactor and the ability to detect leaks.

Specification

1. The reactor shall not be operated if the pool level is below 3 feet from the reference operating level.

2. The reactor shall not be operated if the pool level unexpectedly drops one foot from its operating level.

3. The pool level alarm shall initiate an alarm signal in the control room and at a continuously monitored off-site facility if the pool level is lower than 3 feet from its reference operating level.

Basis

1. The intake to the diffuser system is approximately 18 feet above the core and 8.5 feet below the reference operating level. Setting this level as the specification will both ensure the availability of the diffuser system, and provide more than adequate shielding and cooling for the reactor.

2. An unexpected one foot drop from the operating pool level, whatever level that may be, indicates leakage.

3. An operable pool level alarm that provides an off-site alarm will ensure proper notification if a low pool level or significant unexpected change occurs.

3.8.3 Primary Coolant Temperature

Applicability

This specification applies to the maximum allowable primary coolant temperature.

Objective

The objective is to maintain fuel temperature less than the LSSS, and to maintain the departure of nucleate boiling ratio (DNBR) greater than unity, and to limit any degradation of the reactor systems.

Specification

The reactor shall not be operated when pool temperature exceeds 60° C.

Basis

Calculations show that reactor operations with a pool temperature of 60 C will not risk reaching a fuel temperature greater than the LSSS, or a DNBR of unity. In fact, a conservative calculation predicts a DNBR of 1.54. (Response to U.S. NRC Request for Additional Information, Review of The Fuel Pool Temperature on Fuel Temperature… dated November 13, 2014). For reactor

pulses, the NSC already accounts for pool temperature. As described in the SAR (4.5.13 Pulse Operation – NSC – BOL, Measured), the NSC calculates peak core temperature. The temperature in this calculation is in °C. Ambient fuel temperature is already included in pulse calculations.

4 Surveillance Requirements

Applicability

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

Objective

The objective is to verify the proper operation of any system related to reactor safety.

Specification

Surveillance requirements may be deferred during reactor shutdown (except TS 4.1.5, 4.2.3, 4.5, 4.8.1, and 4.8.2); 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.

	TS	Possible to Defer During Shutdowns?	Required prior to operations?
1.	4.1.1 Steady State Operation	Yes	Yes
2.	4.1.2 Pulse Mode Operation	Yes	Yes
3.	4.1.3 Shutdown Margin	Yes	Yes
4.	4.1.4 Core Configuration Limitation	Yes	Yes
5.	4.1.5 Reactor Fuel Elements	No	N/A
6.	4.1.6 Maximum Excess Reactivity	Yes	Yes
7.	4.2.1 Reactor Channels	Yes	Yes
8.	4.2.2 Reactor Safety Systems and Interlocks	Yes	Yes
9.	4.2.3 Minimum Number of Operable Scrammable Control Rods and Scram Time	No	N/A
10.	4.3 Confinement	Yes	Yes
11.	4.5 Radiation Monitoring Systems and Effluents	No	N/A
12.	4.6 Experiments	Yes	Yes
13.	4.8.1 Primary Coolant Purity	No	N/A
14.	4.8.2 Primary Coolant Level and Leak Detection	No	N/A
15.	4.8.3 Primary Coolant Temperature	Yes	Yes

Any additions, modifications, or maintenance to the central exhaust system, the core and its associated support structure, the pool or its penetrations, the pool coolant system, the rod drive mechanism, or the reactor safety system shall be made and tested in accordance with the specifications to which the systems were originally designed and fabricated or to specifications

approved by the Reactor Safety Board. A system shall not be considered operable until it is successfully tested.

Basis

This specification relates to changes in reactor systems, which could directly affect the safety of the reactor. As long as changes or replacements to these systems continue to meet the original design specifications, then it can be assumed that they meet the presently accepted operating criteria.

4.1 Reactor Core Parameters

4.1.1 Steady State Operation

Applicability

This specification applies to the surveillance requirement of the power level monitoring channels.

Objective

The objective is to verify that the maximum power level of the reactor meets the license requirements.

Specification

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

Basis

The power level channel calibration will ensure that the reactor will be operated at the proper power level.

4.1.2 Pulse Mode Operation

Applicability

This specification applies to the surveillance requirements for operation of the reactor in the pulse mode.

Objective

The objective is to verify that operation of the reactor in the pulse mode is proper and safe and to determine if any significant changes in fuel characteristics have occurred.

Specification

The reactor shall be pulsed semiannually to compare fuel temperature measurements and core pulse energy with those of previous pulses of the same reactivity value. The reactor shall not be declared operational for pulsing until such pulse measurements are performed and are determined to be acceptable.

Basis

The reactor is pulsed at suitable intervals to make a comparison with previous similar pulses and to determine if changes in fuel or core characteristics are taking place.

4.1.3 Shutdown Margin

Applicability

This specification applies to the surveillance requirement of control rod calibrations and shutdown margin.

Objective

The objective is to verify that the requirements for shutdown margins are met for operational cores.

Specification

The reactivity worth of each control rod and the shutdown margin shall be determined annually and following changes in the core, in-core experiments, or control rods.

Basis

The reactivity worth of the control rods is measured to ensure that the required shutdown margin is available and to provide an accurate means for determining the reactivity worth of experiments inserted in the core. Experience with TRIGA reactors gives assurance that measurement of the reactivity worth on an annual basis is adequate to ensure no significant changes in the shutdown margin.

4.1.4 Core Configuration Limitation

Applicability

This specification applies to the surveillance requirements for core configuration.

Objective

The objective is to verify the core is in a safe, reviewed, and approved configuration.

Specification

Each core configuration change shall be determined to meet the requirements of TS 3.1.4 prior to the core loading.

Basis

The requirements of TS 3.1.4 ensure acceptable safety analysis is complete for a core configuration, as well as prevent accidental fuel damage, fuel addition, or criticality events.

4.1.5 Reactor Fuel Elements

Applicability

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

Objective

The objective is to verify the continuing integrity of the fuel element cladding and to ensure that no fuel damage has occurred.

Specification

1. The following fuel elements shall be inspected visually for damage or deterioration and measured for length and bend annually:

    a. At least four elements which occupy the highest pulse temperature positions in the core,

    b. At least one-fifth of the fuel elements used in operation of the reactor over the previous inspection year,

    c. The four elements (a) above may be included in the inspection of the fuel elements of (b) above, and

    d. Over a 5 year period every fuel element used in operation of the reactor shall be inspected.

2. If any element is found to be damaged, the entire core will be inspected.

Basis

The frequency of inspection is based on the parameters most likely to affect the fuel cladding of a pulsing reactor operated at moderate pulsing levels and utilizing fuel elements whose characteristics are well known. Experience has shown that temperature is the major contributor to fuel damage. Inspection of four fuel elements which occupy the highest pulse temperature positions in the core provides surveillance for detection of the most probable fuel element

damage should it occur. Inspection of one-fifth of the elements used in operation of the reactor provides surveillance of the lower temperature elements and over a five year period provides for inspection of all elements.

4.1.6 Maximum Excess Reactivity

Applicability

This specification applies to the surveillance requirements of reactor excess reactivity.

Objective

The objective is to verify that requirements on excess reactivity are met for operational cores.

Specification

The excess reactivity shall be determined annually and following changes in the core, in-core experiments, or control rods for which the predicted change in reactivity exceeds the absolute value of the specified shutdown margin.

Basis

The excess reactivity of the core is measured to ensure that during all states of operation criticality can be maintained for licensed operational limits. With the accumulation of fission product poison buildup and fissile material burnup, excess reactivity must be available for power transients and maintaining criticality.

4.2 Reactor Control and Safety Systems

4.2.1 Reactor Channels

Applicability

This specification applies to the surveillance requirements for reactor channels.

Objective

The objective is to verify the condition and operability of system components directly related to channels that measure key reactor parameters.

Specification

A channel test of each of the reactor channels for the intended mode of operation, as identified in Table 1, shall be performed before each day's operation or before each operation extending more than one day.

Basis

Channel tests will ensure that the safety system channels are operable on a daily basis or prior to an extended run.

4.2.2 Reactor Safety Systems and Interlocks

Applicability

This specification applies to the surveillance requirements for measurements, tests, and calibrations of the control and safety systems.

Objective

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

Specification

1. A channel test of each of the reactor safety system channels and interlocks for the intended mode of operation, as identified in Table 2, shall be performed before each day's operation or before each operation extending more than one day.

2. A channel calibration of the fuel element temperature channels shall be performed semiannually.

Basis

Channel tests will ensure that the safety system channels are operable on a daily basis or prior to an extended run. If the period between operations extends beyond a year, then the annual channel test requirement will ensure operability.

4.2.3 Minimum Number of Operable Scrammable Control Rods and Scram Time

Applicability

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

Objective

The objective is to verify the condition and operability of system components affecting safe and proper control of the reactor.

Specification

1. The control rods shall be visually inspected for deterioration biennially.

2. Operability tests of the control rod mechanism shall follow modification or repairs.

3. The Transient Rod drive cylinder and associated air supply system shall be inspected, cleaned and lubricated semiannually.

4. The scram time shall be measured annually or whenever any work is done on the control rods or the control rod drive system.

Basis

1. The visual inspection of the control rods is made to evaluate corrosion and wear characteristics caused by operation of the reactor.

2. These tests provide verification that the control rod has full travel and that the rod drop time is within specification.

3. Inspection and maintenance of the transient rod drive assembly reduces the probability of failure of the system due to moisture-induced corrosion of the pulse cylinder and piston rod assembly.

4. Measurement of the scram time on an annual basis is a check not only of the scram system electronics, but also is an indication of the capability of the control rods to perform properly.

4.3 Confinement

Applicability

This specification applies to the central exhaust system.

Objective

The objective is to ensure the proper operation of the central exhaust system to prevent uncontrolled releases of radioactive material to the environment.

Specification

1. The central exhaust system shall be channel checked prior to reactor operation or radioactive material handling.

2. During periods of operation, or radioactive material handling, the central exhaust system shall be verified operable weekly including automatic isolation on receipt of a high radiation signal. This specification is not required during periods of non-operation, e.g., holidays, extended maintenance outages.

Basis

Experience accumulated over several years of operation has demonstrated that the tests of the central exhaust system on a weekly basis are sufficient to ensure the proper operation of the system and control of the release of radioactive material.

4.4 Ventilation Systems

The Ventilation System surveillance requirements are specified in TS 4.3 above.

4.5 Radiation Monitoring Systems and Effluents

Applicability

This specification applies to the surveillance requirements for the area radiation monitoring equipment and the Facility Air Monitoring (FAM) system and to effluents.

Objective

The objective is to ensure that the radiation monitoring equipment is operating with appropriate alarm settings and to ensure that gaseous and liquid effluents are in accordance with 10 CFR 20.

Specification

1. The area radiation monitoring system (ARM) and the FAM system shall be calibrated annually, shall be channel tested weekly, and shall be channel checked prior to reactor operation.

2. The level of ^{41}Ar in the effluent gas shall be continuously monitored during operation of the reactor.

3. The environmental monitoring program required by TS 3.7 shall measure the integrated radiation exposure on a quarterly basis.

4. The annual discharge of ^{41}Ar shall be calculated for each annual report.

5. Before discharge, the facility liquid effluents shall be analyzed for radioactive content.

Basis

Experience has shown that weekly verification of area radiation and air monitoring system operations in conjunction with annual calibration is adequate to correct for any variation in the system due to a change of operating characteristics over a long time span.
Monitoring and calculating the amount of gaseous and liquid effluents will allow assurance that they are in accordance with 10 CFR 20.

4.6 Experiments

Applicability

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

Objective

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

Specification

1. A new experiment shall not be installed in the reactor or its experimental facilities until a hazard analysis has been performed and reviewed for compliance with Section 3.6 and Section 6.5 of the Technical Specifications. Minor modifications to a reviewed and approved experiment may be made at the discretion of the Director, or his designee, with concurrence from the Radiation Safety Officer, or his designee. The Director, or his designee, and the Radiation Safety Officer, or his designee, shall review the hazards associated with the modifications and determine that the modifications do not create a significantly different, a new, or a greater safety risk than the original approved experiment, and does not require a review under 10CFR50.59.

2. The performance of an experiment classified as an approved experiment shall not be performed until a licensed senior operator and the Radiation Safety Officer, or his designee has reviewed it for compliance with these TS.

3. The reactivity worth of the experiment shall be estimated or measured, as appropriate, before reactor operation.

Basis

It has been demonstrated over a number of years of experience that experiments and irradiations reviewed by the Reactor Staff and the Reactor Safety Board as appropriate can be conducted without endangering the safety of the reactor or exceeding the limits in the technical specifications.

4.7 ALARA Radioactive Effluents Released

Surveillance for the LCO 3.7 ALARA Radioactive Effluents Released is incorporated into TS 4.5.

4.8 Primary Coolant Conditions

4.8.1 Primary Coolant Purity

Applicability

This specification applies to the surveillance requirements for coolant purity.

Objective

The objective is ensure the water quality and radioactivity of the reactor coolant remains within defined limits.

Specification

1. A sample of the coolant shall be collected and analyzed for radioactive material content at least weekly during periods of reactor operation and at least quarterly during extended shutdowns.

2. Conductivity of the bulk pool water shall be measured and recorded weekly.

Basis

1. Weekly sampling during operation is sufficient to predict trends of radioactive material content from fuel or other sources.

2. A small rate of corrosion continuously occurs in any water-metal system. In order to limit this rate, and thereby extend the longevity and integrity of the fuel cladding, a water cleanup system is required. Experience with water quality control at many reactor facilities has shown that maintenance within the specified limits provides acceptable control.

4.8.2 Primary Coolant Level and Leak Detection

Applicability

This specification applies to the surveillance requirements for primary coolant level and leak detection.

Objective

The objective is to verify the operability of the pool level alarm and monitor for pool leakage.

Specification

1. The reactor pool water level shall be recorded at least weekly.

2. The pool water level alarm shall be channel tested weekly.

3. The pool water level alarm shall be channel checked prior to reactor operation.

Basis

1. A weekly record of pool level provides a large set of comparable data over time. This data can be used to determine if changes in pool level are due to leakage.

2. Experience has shown that a weekly verification of operability is sufficient to ensure reliability of the alarm.

4.8.3 Primary Coolant Temperature

Applicability

This specification applies to the surveillance requirements for primary coolant temperature channel.

Objective

The objective is to verify the operability of the primary coolant temperature channel.

Specification

1. Primary coolant temperature shall be recorded every 30 minutes while the reactor is operating, or immediately following reactor startup if the reactor is to be operated for less than 30 minutes.

2. The primary coolant temperature channel shall be calibrated semiannually.

Basis

1. Changes in primary coolant temperature occur slowly due to the large volume of the pool. 30 minute intervals are sufficient to track and predict trends in temperature.

2. Experience with semiannual calibration has shown very high reliability of the temperature channels with need for adjustment being very rare.

5 Design Features

5.1 Site Description

Applicability

This specification applies to the NSC site location.

Objective

The objective is to specify the bounds of the site.

Specification

The licensed area of the facility is the area inside the site boundary. The boundary is defined by the fence surrounding the site. This description coincides with that of the restricted area.

Basis

The restricted area is described in SAR 2.2.1.1, and the site boundary is shown in Figure 2-2 in the SAR.

5.2 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.

Specification

TRIGA LEU 30/20 Fuel: The individual unirradiated LEU fuel elements shall have the following characteristics:

1. Uranium content: maximum of 30 wt% enriched to maximum 19.95% Uranium-235 with nominal enrichment of 19.75% Uranium-235,

2. Hydrogen-to-zirconium atom ratio (in the ZrHx): nominal 1.6 H atoms to 1.0 Zr atoms with a maximum H to Zr ratio of 1.65,

3. Natural erbium content (homogeneously distributed): nominal 0.90 wt%, and

4. Cladding: 304 stainless steel.

Basis

The fuel specification permits a maximum uranium enrichment of 19.95%. This is about 1% greater than the design value for 19.75% enrichment. Such an increase in loading would result in an increase in power density of less than 1%. An increase in local power density of 1% reduces the safety margin by less than 2%. (TAMU LEU Conversion SAR, December 2005)

The fuel specification for a single fuel element permits a minimum erbium content of about 5.6% less than the design value of 0.90 wt%. (However, the quantity of erbium in the full core must not deviate from the design value by more than –3.3%). This variation for a single fuel element would result in an increase in fuel element power density of about 1-2%. Such a small increase in local power density would reduce the safety margin by less than 2%. (TAMU LEU Conversion SAR, December 2005)

The maximum hydrogen-to-zirconium ratio of 1.65 could result in a maximum stress under accident conditions in the fuel element clad about a factor of 2 greater than for a hydrogen-to-zirconium ratio of 1.60. This increase in the clad stress during an accident would not exceed the rupture strength of the clad. (GA Report E-117-883, February 1980)

Stainless steel clad has been shown through decades of operation to provide a sufficient barrier against fission product release.

5.3 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 to provide assurance that excessive power densities will not be produced.

Specification

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

2. The reflector, excluding experiments and experimental facilities, shall be any combination of graphite, water, and heavy water.

3. Fuel shall not be inserted or removed from the core unless the reactor is subcritical by at least 50 cents more than the calculated worth of the most reactive fuel assembly.

Basis

1. Standard TRIGA cores have been in use for years and their characteristics are well documented. LEU cores including 30/20 fuel have also been operated at General Atomics and their successful operational characteristics are available. General Atomics and Texas A&M have conducted a series of studies documenting the viability of using LEU fuel in TRIGA reactors.

2. The core will be assembled in the reactor grid plate that is located in a pool of light water. Light water in combination with graphite or heavy water reflectors can be used for neutron economy and the enhancement of experimental facility radiation requirements.

3. Manipulation of core components will be allowed only when a single manipulation cannot result in inadvertent criticality.

5.4 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.

Specification

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

2. The regulating control rod may not have scram capability and shall be a stainless rod or contain borated graphite, B4C powder or boron and its compounds in solid form as poison in aluminum or stainless steel cladding. This rod is water followed in that pool water takes the place of the rod as it is withdrawn. It has no physical follower attachment.

3. The transient control rod shall have scram capability and contain borated graphite or boron and its compounds in solid form as a poison in an aluminum or stainless steel clad. The transient rod shall have an adjustable upper limit to allow a variation of reactivity insertions. This rod shall incorporate an air follower.

Basis

Using neutron absorbing borated graphite, B_4C powder, or boron and its compounds, satisfies the poison requirements for the control rods. Since the regulating rod normally is a low worth rod, using a solid stainless steel rod could satisfy its function. These materials must be contained in a suitable clad material, such as aluminum or stainless steel, to ensure mechanical stability during movement and to isolate the poison from the pool water environment. 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 in the LEU fueled region 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 a reactor pulse. The nuclear behavior of the air follower that is incorporated into the transient rod is similar to a void.

5.5 Radiation Monitoring System

Applicability

This specification describes the functions and essential components of the area radiation monitoring (ARM) equipment and the facility air monitoring (FAM) system equipment for monitoring airborne radioactivity as described in TS 3.5.1 "Radiation Monitoring."

Objective

The objective is to describe the radiation monitoring equipment available to the operator to ensure safe operation of the facility.

Specification

The radiation monitoring equipment listed in Table 5 shall have the following characteristics:

Table 5: NSC Radiation Monitoring Equipment

Radiation Monitoring Channel	Detector Type	Function
Area Radiation Monitor (ARM)	Gamma sensitive detector	Monitor radiation fields in key locations. Alarm and readout in the control room and readout in the emergency support center.
Facility Air Monitor (FAM) – Particulates (FAM Ch. 1, 4)	Beta-Gamma sensitive detector	Monitors concentration of airborne radioactive particulates. Alarm and readout in the control room and readout in the emergency support center.
Facility Air Monitor (FAM) – Gases (FAM Ch. 3, 5)	Gamma sensitive detector	Monitors concentration of radioactive gases. Alarm and readout in the control room and readout in the emergency support center.

An alarm signal from the stack particulate, fission product, or stack gas (xenon) facility air monitor shall automatically isolate the central exhaust system.

Basis

The radiation monitoring system is intended to provide information to operating personnel of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the environment. Automatic isolation capability of the central exhaust system will mitigate the spread of radioactivity to the environment.

The reactor bridge Area Radiation Monitor (ARM) alarm set point is normally set at 14 mR for an alert and 16 mR alarm. The background for this ARM is normally no greater than 5 mR and this set point alerts the Reactor Operator that the ARM is approaching or has exceeded three times the background. The alarm set point can be adjusted if operational activities warrant the necessity of the adjustment. If the alarm set point is raised above the normal alarm set point additional controls are put into place in order to ensure ALARA.

The Facility Air Monitor (FAM) alarm set points are calculated during the annual calibration of the specific FAM channel. Each channel has an individual calibration and alarm set point. The following list is the basis of each alarm set point for each individual FAM channel.

5.6 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 that is being stored will not become critical and will not reach an unsafe temperature.

Specification

1. All fuel elements and fueled devices shall be stored in a geometrical array for which the k-effective is less than 0.8 for all conditions of moderation and reflection.

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

Basis

The limits imposed by Specifications 5.6.1 and 5.6.2 are conservative and ensure safe storage.

5.7 Reactor Building and Central Exhaust System

Applicability

This specification applies to the building that houses the reactor.

Objective

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

Specification

1. The reactor shall be housed in a facility designed to restrict leakage. The minimum free volume in the facility shall be 180,000 cubic feet.

2. The reactor building shall be equipped with a central exhaust system designed to exhaust air or other gases from the reactor building and release them from a stack at minimum of 85 feet from ground level.

3. Emergency isolation controls for the central exhaust system shall be located in the emergency support center and the system shall be designed to shut down in the event of an alarm on the stack particulate monitor (FAM Ch.1) radiation monitoring channel.

Basis

The facility is designed such that the central exhaust system will normally maintain a negative pressure with respect to the atmosphere so that there will be no significant uncontrolled leakage to the environment. The free air volume within the reactor building is confined when there is an emergency isolation of the central exhaust system. Controls for startup and operation of the central exhaust system are located in the emergency support center. Proper handling of airborne radioactive materials (in emergency situations) can be conducted from the emergency support center minimizing exposure to operating personnel.

5.8 Reactor Pool Water Systems

Applicability

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

Objective

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

Specification

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

2. The pool water inlet and outlet pipe for the demineralizer, diffuser, and skimmer systems shall not extend more than 15 feet below the top of the reactor pool when fuel is in the core.

3. Pool water inlet to the heat exchanger shall have an emergency cover within the reactor pool for manual shut off in case of pool water loss due to external pipe system failure.

4. A pool level alarm with readouts in the control room and at a continuously monitored remote location shall indicate a pool level less than 3 feet below the reference operating level.

Basis

1. This specification is based on thermal and hydraulic calculations, which show that the TRIGA-LEU core can operate continuously in a safe manner at power levels up to 2,420 kW, with natural convection flow and sufficient bulk pool cooling.

2. In the event of accidental siphoning of pool water through inlet and outlet pipes of the demineralizer, skimmer, or diffuser systems, the pool water level will drop to no more than 15 feet from the top of the pool, providing 17 feet of water above the core.

3. Inlet and outlet coolant lines to the pool heat exchanger terminate at the bottom of the pool. In the event of pipe failure, these lines must be manually sealed from within the reactor pool. The primary outlet pipes from the heat exchanger (inlet pipes to the pool) are equipped with flapper valves. During no-flow or reverse-flow conditions, these flapper valves close and severely restrict the flow of water through the pipe. The primary inlet pipe to the heat exchanger (outlet pipe from the pool) must have a cover manually installed. The cover for this line will be stored in the reactor pool.

4. This alarm is observed in the reactor control room, in the emergency support center, and at a continuously staffed remote location.

6 Administrative Controls

6.1 Organization

The Nuclear Science Center is operated by the Texas A&M University System's Texas Engineering Experiment Station (TEES), with responsibility within TEES resting with the Director or his designee. The Director of the Nuclear Science Center is responsible to the TEES director, or his designee, for the administration and the proper and safe operation of the facility. Figure 1 shows the administration chart for the Nuclear Science Center. The Reactor Safety Board advises the director of the NSC on all matters or policy pertaining to safety. The NSC Radiological Safety Officer provides onsite advice concerning personnel and radiological safety and provides technical assistance and review in the area of radiation protection.

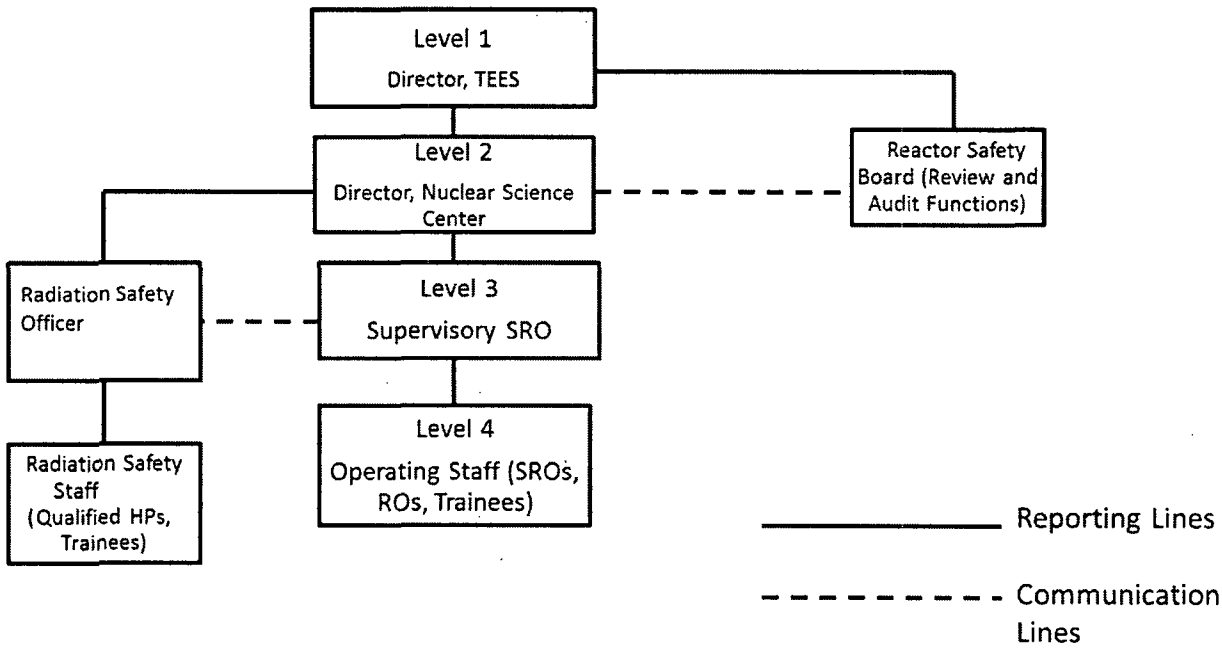

Figure 1: Organization Chart for Reactor Administration

6.1.1 Structure

1. A line management organizational structure provides for personnel who will administrate and operate the reactor facility.

2. The Director of TEES and the Director of the NSC have line management responsibility for adhering to the terms and conditions of the Nuclear Science Center Reactor (NSC) license and technical specifications and for safeguarding the public and facility personnel from undue radiation exposure. The facility shall be under the direct control of the NSC Director or a licensed senior reactor operator.

3. Management Levels:

    a. Level 1: TEES Licensee (Director of TEES): Responsible for the NSC facility license.

    b. Level 2: NSC Director: Responsible for reactor facility operation and shall report to Level 1.

    c. Level 3: Supervisory SRO (One of the following - Associate Director, Reactor Manager, or Reactor Supervisor): Responsible for the day-to-day operation of the NSC including shift operation and shall report to Level 2.

    d. Level 4: Reactor Operating Staff: Licensed reactor operators and senior reactor operators and trainees. These individuals shall report to Level 3.

4. Reactor Safety Board (RSB):

    The RSB is responsible to the licensee for providing an independent review and audit of the safety aspects of the NSC.

6.1.2 Responsibility

Responsibility for the safe operation of the reactor facility shall be in accordance with the line organization established in Section 6.1.1. In all instances, responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon appropriate qualifications.

The reactor facility shall be under the direct control of the Supervisory SRO. The Supervisory SRO shall be responsible for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, procedures and requirements of the Radiation Safety Officer and the Reactor Safety Board.

6.1.3 Staffing

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

    a. At least two individuals shall be present at the facility complex and shall consist of at least a licensed senior reactor operator and either a licensed reactor operator or operator trainee;

    b. During periods of reactor maintenance the two individuals who shall be present at the facility complex may consist of a licensed senior reactor operator and a member of the maintenance staff who is able to carry out prescribed written instructions. During periods when the reactor is not secured, it shall be under the direct control the of the senior reactor operator;

    c. A licensed reactor operator or senior reactor operator shall be in the control room;

d. The NSC Director or his designated management alternate is readily available for emergencies or on call (the individual can be rapidly reached by phone or radio and is within 30 minutes or 15 miles of the reactor facility); and

e. At least one member of the Radiation Safety Staff shall be readily available at the facility or on call (the individual can be rapidly reached by phone or radio and is within 30 minutes or 15 miles of the reactor facility), to provide advice and technical assistance in the area of radiation protection.

2. A list of reactor facility personnel by name and telephone number shall be readily available for use in the control room. The list shall include:

   a. Management personnel,

   b. Radiation safety personnel, and

   c. Other operations personnel

3. The following designated individuals shall direct the events listed:

   a. The NSC Director or his designated alternate who shall be SROs shall direct any loading or unloading of fuel or control rods within the reactor core region,

   b. The NSC Director or his designated alternate who shall be SROs shall direct any loading or unloading of an in-core experiment with a reactivity worth greater than $1,

   c. The senior reactor operator on duty shall be present at the facility and shall direct the recovery from an unplanned or unscheduled shutdown,

   d. The senior reactor operator on duty shall be present at the facility and shall direct each reactor startup and approach to power, and

   e. The senior reactor operator on duty shall be present at the facility and shall direct all significant reactor power changes after initial startup. A significant reactor power change is defined as one that would disable the automatic servo control, i.e. equal to or great than 5% of reactor power.

6.1.4 Selection and Training of Personnel

The selection and training of operations personnel shall be in accordance with the following:

1. Responsibility: The NSC Director or his designated alternate is responsible for the selection, training, and requalification of the facility reactor operators and senior reactor operators.

2. Selection: The selection of operations personnel shall be consistent with the standards related to selection in ANSI/ANS-15.4-2007

3. Training Program: The Training Program shall be consistent with the standards related to training in ANSI/ANS-15.4-2007.

4. Requalification Program: The Requalification Program shall be consistent with the standards related to requalification in ANSI/ANS-15.4-2007.

6.2 Review and Audit

6.2.1 Reactor Safety Board (RSB)

The Reactor Safety Board shall be comprised of at least 3 voting members knowledgeable in fields which relate to Nuclear Safety. One of these members, the Director of TEES (Level 1 Management), will serve as the Chairman. If the Chairman is unable to attend one or a number of committee meetings he may designate a committee member as Chairman Pro-tem. The members are appointed by the Director of TEES (Level 1 Management) to serve one year terms. It is expected that the members will be reappointed each year as long as they are willing to serve so that their experience and familiarity with the past history of the NSC will not be lost to the committee. The Director of the NSC, TAMU Radiological Safety Officer, Head of the Department of Nuclear Engineering, and a senior member of the NSC Radiation Safety Staff shall be ex-officio members of the RSB.

6.2.2 RSB Charter and Rules

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

1. Meeting Frequency: The RSB shall meet annually at intervals not to exceed 15 months. (Note: The facility license requires a meeting at least once per year and as frequently as circumstances warrant consistent with effective monitoring of facility activities);
2. Quorum: A quorum is comprised of not less than one-half of the voting membership where the operating staff does not constitute a majority;
3. Voting Rules: On matters requiring a vote, if only a quorum is present a unanimous vote of the quorum is required; otherwise a majority vote is required;
4. Subcommittees: The Chairman may appoint subcommittees comprised of members of the RSB including ex-officio members to perform certain tasks. Subcommittees or members of the RSB may be authorized to act for the board; and
5. Meeting Minutes: The Chairman will designate one individual to act as recording secretary. It will be the responsibility of the secretary to prepare the minutes which will be distributed to the RSB, including the Director of TEES (Level 1 Management), within three months. The RSB will review and approve the minutes of the previous meetings. A complete file of the meeting minutes will be maintained by the Chairman of the RSB and by the Director of the NSC.

6.2.3 RSB Review Function

The review responsibilities of the Reactor Safety Board or a designated subcommittee shall include, but are not limited to the following:

1. Review and evaluation of determinations of whether proposed changes to equipment, systems, tests, experiments, or procedures can be made under 10 CFR 50.59 or would require a change in technical specifications or license conditions;

2. Review of new procedures, major revisions of procedures, and proposed changes in reactor facility equipment or systems which have significant safety impact to reactor operations;

3. Review of new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity;

4. Review of proposed changes to the technical specifications and U.S. NRC issued license;

5. Review of the NSC radiation protection program;

6. Review of violations of technical specifications, U.S. NRC issued license, and violations of internal procedures or instructions having safety significance;

7. Review of operating abnormalities having safety significance;

8. Review of reportable occurrences listed in Section 6.6.1 and 6.6.2 of these TS; and

9. Review of audit reports.

6.2.4 RSB Audit Function

The audit function shall include selective (but comprehensive) examination of operating records, logs, and other documents. Discussions with cognizant personnel and observation of operations should be used also as appropriate. In no case shall the individual immediately responsible for an area perform an audit in that area. Audits shall include but are not limited to the following:

1. Facility operations, including radiation protection, for conformance to the technical specifications, applicable license conditions, and standard operating procedures: at least once per calendar year (interval between audits not to exceed 15 months);

2. 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);

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

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

5. The reactor facility security plan and implementing procedures: 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 Director of TEES (Level 1 Management). A written report of the findings of the audit shall be submitted to the Director of TEES (Level 1 Management) and the review and audit group members within 3 months after the audit has been completed.

6.2.5 Audit of ALARA Program

The Chairman of the RSB or his designated alternate (excluding anyone whose normal job function is within the NSC) shall conduct an audit of the reactor facility ALARA program annually. The auditor shall transmit the results of the audit to the RSB at the next scheduled meeting for its review and approval.

6.3 Radiation Safety

The Radiation Safety Officer shall be responsible for implementing the radiation safety program for the TEES/TAMUS NSC TRIGA Research Reactor. The requirements of the radiation safety program are established in 10 CFR 20. The Program should use the guidelines of the ANSI/ANS-15.11-1993; R2004, "Radiation Protection at Research Reactor Facilities."

6.4 Procedures

Written operating procedures shall be prepared, reviewed, and approved before initiating any of the activities listed in this section. The procedures shall be reviewed and approved by the NSC Director or his designated alternate, the Reactor Safety Board, and shall be documented in a timely manner. Procedures shall be adequate to ensure the safe operation of the reactor but shall not preclude the use of independent judgment and action should the situation require such. Operating procedures shall be used for the following items:

1. Startup, operation, and shutdown of the reactor;

2. Fuel loading, unloading, and movement within the reactor;

3. Control rod removal or replacement;

4. Routine maintenance of the control rod, drives and reactor safety and interlock systems or other routine maintenance of major components of systems that could have an effect on reactor safety;

5. Surveillance checks, calibrations, and inspections of reactor instrumentation and controls, control rod drives, area radiation monitors, facility air monitors, the central exhaust system and other systems as required by the Technical Specifications;

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

7. Implementation of required plans such as emergency or security plans;

8. Radiation protection program to maintain exposures and releases as low as reasonably achievable (ALARA);

9. Use, receipt, and transfer of by-product material, if appropriate; and

10. Surveillance procedures for shipping radioactive materials.

6.5 Experiment Review and Approval

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

1. All new experiments or class of experiments shall be reviewed by the RSB as required by TS 6.2.3 and implementation approved in writing by the NSC Director or his designated alternate.

2. Substantive changes to previously approved experiments shall be made only after review by the RSB and implementation approved in writing by the NSC Director or his designated alternate.

6.6 Required Actions

6.6.1 Action to be Taken in the Event of a Safety Limit Violation

In the event a safety limit is violated:

1. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the U.S. NRC;

2. An immediate notification of the occurrence shall be made to the RSB Chairman and the NSC Director, and reports shall be made to the U.S. NRC in accordance with Section 6.7.2 of these specifications; and

3. A report shall be prepared which shall include:

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

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

    c. Corrective action to be taken to prevent recurrence.

This report shall be submitted to the RSB for review and then submitted to the U.S. NRC when authorization is sought to resume operation of the reactor.

6.6.2 Action to be Taken in the Event of a Reportable Occurrence Other Than A Safety Limit Violation

Action to be taken in the event of a reportable occurrence other than a safety limit violation:

1. NSC staff shall return the reactor to normal operating via the approved NSC procedure or shut down conditions. If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed unless authorized by the NSC Director or his designated alternate;

2. The NSC Director or his designated alternate shall be notified and corrective action taken with respect to the operations involved;

3. The NSC Director or his designated alternate shall notify the RSB Chairman who shall arrange for a review by the RSB;

4. A report shall be made to the RSB which shall include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence; and

5. A report shall be made to the U.S. NRC in accordance with Section 6.7.2 of these specifications.

6.7 Reports

6.7.1 Annual Operating Report

An annual report covering the operation of the reactor facility during the previous calendar year shall be submitted to the NRC before March 31 of each year providing the following information:

1. A narrative summary of (l) reactor operating experience (including experiments performed), (2) changes in facility design, performance characteristics, and operating procedures related to reactor safety and occurring during the reporting period, and (3) results of surveillance tests and inspections;

2. Tabulation of the energy output (in megawatt days) of the reactor, hours reactor was critical, and the cumulative total energy output since initial criticality;

3. The number of unscheduled shutdowns and inadvertent scrams, including, where applicable corrective action to preclude recurrence;

4. Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safety of the operation of the reactor and the reasons for any corrective maintenance required;

5. 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 Section 50.59 of 10 CFR Part 50;

6. 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 before the point of such release or discharge. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed or recommended, a statement to this effect is sufficient:

    a. Liquid Waste (summarized on a monthly basis)

        i. Radioactivity discharged during the reporting period.

            1. Total radioactivity released (in Curies),

            2. The effluent concentration used and the isotopic composition if greater than 1×10^{-7} μCi/cc for fission and activation products,

            3. Total radioactivity (in curies), released by nuclide during the reporting period based on representative isotopic analysis, and

            4. Average concentration at point of release (in μCi/cc) during the reporting period.

        ii. Total volume (in gallons) of effluent water (including dilution) during periods of release.

    b. Airborne Waste (summarized on a monthly basis)

        i. Radioactivity discharged during the reporting period (in Curies) for:

            1. ^{41}Ar, and

            2. Particulates with half-lives greater than eight days.

    c. Solid Waste

        i. The total amount of solid waste transferred (in cubic feet),

        ii. The total activity involved (in Curies), and

iii. The dates of shipment and disposition (if shipped off site).

7. A summary of radiation exposures received by facility personnel and visitors, including dates and time where such exposures are greater than 25% of that allowed or recommended; and

8. A description and summary of any environmental surveys performed outside the facility.

6.7.2 Special Reports

In addition to the requirements of applicable regulations, reports shall be made to the NRC Document Control Desk and special telephone reports of events should be made to the Operations Center as follows:

1. There shall be a report not later than the following working day by telephone and confirmed in writing by fax or similar conveyance to the NRC Headquarters Operation Center, and followed by a written report that describes the circumstances of the event and sent within 14 days to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, DC 20555, of any of the following:

   a. Violation of safety limit (see TS 6.6.1);

   b. Any release of radioactivity from the site above allowed limits; and

   c. Any reportable occurrences as defined in TS 1.3.

2. A written report within 30 days in writing to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, DC, 20555, of:

   a. Permanent changes in the facility organization involving Level 1 and Level 2; and

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

6.8 Records

Records of facility operations in the form of logs, data sheets, or other suitable forms shall be retained for the period indicated as follows:

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

1. Normal reactor facility operation (but not including supporting documents such as checklists, log sheets, etc. which shall be maintained for a period of at least one year),

2. Principal maintenance operations,

3. Reportable occurrences,

4. Surveillance activities required by the technical specifications,

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

6. Experiments performed with the reactor,

7. Fuel inventories, receipts, and shipments,

8. Approved changes in operating procedures, and

9. Records of meeting and audit reports of the RSB.

6.8.2 Records to be Retained for at Least One Certification Cycle

Records of retraining and requalification of certified operations personnel shall be maintained at all times the individual is employed or until the certification is renewed. For the purposes of this technical specification, a certification is an NRC issued operator license.

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. Off-site environmental monitoring surveys required by the technical specifications,

3. Radiation exposure for all personnel monitored,

4. Drawings of the reactor facility, and

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