TEXAS A&M UNIVERSITY SYSTEM
TEXAS ENGINEERING EXPERIMENT STATION
RESEARCH REACTOR
LICENSE NO. R-83
DOCKET NO. 50-128

TECHNICAL RAI RESPONSES (DATED 01/12/2012)

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUT OR DENOTED BY BRACKETS

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January 12, 2012

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

2012-0001

Subject: Response to NRC Requests for Additional Information Questions 1 and 3 for the 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 NRC. In July 2009, the NSC submitted an updated SAR, dated June 2009, to the Nuclear Regulatory Commission (NRC). This updated 2009 version of our SAR incorporated the information from the conversion SAR and the startup of the new LEU reactor core. On January 12, 2012 the NRC submitted a Request for Additional Information as a part of the review process. This request included 1 question about the technical aspects of the NSC's SAR submittal. Attached to this letter are the NSC's response to Questions 1 of the NRC's RAI and an updated version the NSC's response to Question 3 of the RAIs submitted September 29th, 2011.

If you have any questions, please contact Jim Remlinger, Jerry Newhouse, or W. Dan Reece at 979-845-7551.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 12, 2012.

W. D. Reece
Director, Nuclear Science Center

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

A020
NRR

RAI 1

NUREG-1537, Part 1, Section 4.5.3, Operating Limits, states that the applicant should present information about reactor operating limits relating to reactivity, specifically control rod worth, excess reactivity and shutdown margin. The Reactor Startup Report dated April 30, 2007, and submitted to the NRC by letter dated August 30, 2010, provides calculated and measured control rod worth, excess reactivity and shutdown margin for the initial startup of the low-enriched uranium 30/20 core, which may not apply to the current reactor operating limits. The following information is needed to complete our review:

a. Please provide the most recent control rod worth data and excess reactivity calculation with the reactor against the thermal column and discuss the changes in excess reactivity due to fuel burn-up and the build-up of samarium-149 since the initial startup.

b. Please provide a discussion of the current specification of shutdown margin in Technical Specification 3.1.3, Shutdown Margin, and consider increasing the shutdown margin to be consistent with the reactivity worth of xenon referenced in the definition of reference core condition or provide an explanation why a change to the shutdown margin specification is not necessary.

Response A

Control rod worth and excess reactivity were most recently measured and calculated on July 9th, 2011, during our last annual maintenance period. Control rod worth was measured as $14.62. Core excess against the thermal column was calculated to be $6.50. Core excess away from the thermal column was calculated to be $5.55. Shutdown margin against the thermal column was calculated to be $2.20. Shutdown margin away from the thermal column was calculated to be $3.15. The excess reactivity in 30/20 fuel monotonically decreases during fuel burn-up and samarium-149 build-up. While the most rapid decrease was during the earliest days of operation, it will continue to decrease during the entire lifetime of the core.

Response B

After reviewing the reactivity worth of xenon referenced in the definition of reference core condition, we will increase the shutdown margin described in TS 3.1.3 from $0.25 to $0.50.

[redacted]

RAI 3

NUREG-1537 Chapter 13, Accident Analysis, recommends MHA doses analysis to the public. The MHA analysis presented in the TAMU TRIGA SAR, as supplemented, is incomplete in that; (1) while providing considerable information, it does not provide a clear presentation of the analyzed scenario, and (2) it does not discuss the dose to on site, non-occupational individuals in the Laboratory Building such as students, faculty, visitors, etc.

a. Please provide a dose assessment for the maximum exposed individual member of the public in the Laboratory Building. Please describe the assumptions used and any systems, plans, procedures, or stay times for which credit is taken in the analysis.

c. Please discuss the assumptions used in the dose assessment for the members of the public such as ground release rate, exposure pathways (inhalation, immersion), exposure time for members of public, and dose conversion factors.

Response A and C

Note: In addition to the below please see the attached Excel workbook which accounts for dose contribution on a nuclide basis.

Using the data in Table 13-1, the saturated activities of the significant fission products for a single LEU 30/20 fuel element at 1.0 MW are:

1. Total iodine fission products – [redacted]
2. Total halogen (Br and I) fission products – [redacted]
3. Total gaseous (Kr and Xe) fission product – [redacted]

Applying the release fraction of 2.6 x 10^{-5} to the total inventory in a single element operating at 1.0 MW yields the following activities that would be released in a cladding failure:

1. Total iodine activity - [redacted]
2. Total halogen activity - [redacted]
3. Total gaseous activity - [redacted]

If the release accident occurred with water in the pool, the halogens will remain in the water. The resulting concentration would be [redacted] [redacted]. Within 24 hours, this value would decay to [redacted] [redacted] The demineralizer system (Section 5.4.1) would remove these soluble fission products and they would eventually be transferred to the liquid waste system.

NSC calculated the results of the release of fission products from a single fuel element without water in the reactor pool, and with the ventilation system shut down[i]. The only case where significant exposure occurs requires the simultaneous failure of the fuel element clad,

[redacted]

catastrophic failure of the pool and liner, and a failure of the ventilation system with personnel remaining within the reactor facility for a period of five minutes after release.

Calculations were performed to determine the immersion deep dose equivalent (DDE) and thyroid committed effective dose equivalent (CEDE) to an individual due to gamma emitters uniformly dispersed throughout the volume of the reactor building. The maximum stay time in the reactor building is assumed to be 5 minutes following the release. We assumed that 100 % of volatiles escape into the reactor building volume when the pool water is lost. If the volatiles were dispersed uniformly in the reactor building, the resulting concentration is calculated to be $1.1x10^{-4}$ μCi/cc. The maximum external dose from submersion in these photon emitters was calculated to be 6.2 mR.

The thyroid CEDE to an individual in the reactor building is calculated from the concentrations of various iodine isotopes released from the initial inventory and dispersed throughout the volume of the reactor building. Assuming standard man breathing rate the resulting dose is calculated from using the flux to dose conversion factor from Federal Guidance Report 11 for each iodine isotope. For an individual remaining in the reactor building for 5 minutes following release, the thyroid CEDE is calculated to be 560 mrem.

The dose assessment for the members of the public outside the fence line was calculated assuming that all the fission products are released to the atmosphere outside the reactor building. Worst case conditions were assumed: a wind speed of 2 m/s and stability class F and with ventilation system shutdown, all releases at ground level, leaking from the reactor building, and the unrestricted area was occupied during the entire release. No credit was given for iodine plate out or decay of isotopes during hold up. Using these assumptions, the dose rate to an individual just outside the site boundary at 100 m from the reactor building was calculated to be 0.4 mrem/hr and the thyroid CEDE calculated to be 2.8 mrem. Conservative calculations were also performed to determine the immersion DDE and the thyroid CEDE to personnel in the permanently occupied area about 800 m away from the building. The immersion dose rate was calculated to be 0.1 mR/hr and thyroid CEDE was calculated to be 0.8 mrem.

Members of the general public can also 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 20 minutes. Because of the shelter provided by the building, the dose rates inside the building are assumed to be no worse than dose rates at the fenceline directly under the plume where we assumed occupation during the entire event. We are taking these unrestricted access doses as an upper bound on personnel in the laboratory building.

Table 13-2 shows the calculated exposure to population outside the building and exposure to operating personnel inside the facility. The exposure to personnel in unrestricted areas during the accident is calculated to be minimal. Thus, no realistic hazard of consequence will result from the DBA.

[redacted]

Table 13-2: Summary of Radiation Exposures Following the Release of Fission Products from a Single Fuel Element

Exposure Criteria	Immersion DDE	Thyroid CEDE	TEDE
	mrem	mrem	mrem
Maximum exposure to operating personnel at 5 minutes after release	6.2	560	566.2
Maximum exposure to personnel in unrestricted area at 1 hour after release	0.4	2.8	3.2
Exposure to personnel in nearest permanently occupied area (nearest resident) at 1 hour after release	0.1	0.8	0.9
Exposure to personnel inside the Laboratory Building	0.4	2.8	3.2

[i] Vasudevan, L., "Radiation Exposure Following the release of Fission Products During a Design Basis Accident Condition," Nuclear Science Center Internal Report, August 2006.

[redacted]

Nearest Resident Exposure Thyroid									
Nuclide	**Inventory (Ci)**	**Release (Ci)**	**Concentration (µCi/cc)**	**Activity Uptake (µCi)**	**Activity Uptake (Bq)**	**Dose Conversion Factor (Sv/Bq)**	**Dose (SV)**	**Dose (Rem)**	**Dose (mRem)**
I-131	[redacted]	[redacted]	[redacted]	5.09E-04	18.82771429	2.92E-07	5.50E-06	5.50E-04	5.50E-01
I-132	[redacted]	[redacted]	[redacted]	7.64E-04	28.2828	1.74E-09	4.92E-08	4.92E-06	4.92E-03
I-133	[redacted]	[redacted]	[redacted]	1.18E-03	43.70228571	4.86E-08	2.12E-06	2.12E-04	2.12E-01
I-134	[redacted]	[redacted]	[redacted]	1.36E-03	50.27137143	2.88E-10	1.45E-08	1.45E-06	1.45E-03
I-135	[redacted]	[redacted]	[redacted]	1.11E-03	40.89874286	8.46E-09	3.46E-07	3.46E-05	3.46E-02

Sum:	**8.03E-04**	**0.80**

[redacted]

Maximum Exposed Member of the Public Thyroid									
Nuclide	Inventory (Ci)	Release (Ci)	Concentration (µCi/cc)	Activity Uptake (µCi)	Activity Uptake (Bq)	Dose Conversion Factor (Sv/Bq)	Dose (SV)	Dose (Rem)	Dose (mRem)
I-131	[redacted]	[redacted]	[redacted]	1.78E-03	65.897	2.92E-07	1.92E-05	1.92E-03	1.92E+00
I-132	[redacted]	[redacted]	[redacted]	2.68E-03	98.9898	1.74E-09	1.72E-07	1.72E-05	1.72E-02
I-133	[redacted]	[redacted]	[redacted]	4.13E-03	152.958	4.86E-08	7.43E-06	7.43E-04	7.43E-01
I-134	[redacted]	[redacted]	[redacted]	4.76E-03	175.9498	2.88E-10	5.07E-08	5.07E-06	5.07E-03
I-135	[redacted]	[redacted]	[redacted]	3.87E-03	143.1456	8.46E-09	1.21E-06	1.21E-04	1.21E-01

Sum:	**2.81E-03**	**2.81**

[redacted]

Maxiumum Exposed Person Thyroid								
Nuclide	**Inventory (Ci)**	**Release (Ci)**	**Concentration (µCi/cc)**	**Activity Uptake (µCi)**	**Activity Uptake (Bq)**	**Dose Conversion Factor (Sv/Bq)**	**Dose (SV)**	**Dose (Rem)**
I-131	[redacted]	[redacted]	[redacted]	3.56E-01	1.32E+04	2.92E-07	3.85E-03	3.85E-01
I-132	[redacted]	[redacted]	[redacted]	5.35E-01	19797.96	1.74E-09	3.44E-05	3.44E-03
I-133	[redacted]	[redacted]	[redacted]	8.27E-01	30591.6	4.86E-08	1.49E-03	1.49E-01
I-134	[redacted]	[redacted]	[redacted]	9.51E-01	35189.96	2.88E-10	1.01E-05	1.01E-03
I-135	[redacted]	[redacted]	[redacted]	7.74E-01	28629.12	8.46E-09	2.42E-04	2.42E-02

Sum:	**0.56**

[redacted]

Nearest Resident Exposure Whole Body							
Nuclide	Inventory (Ci)	Release (Ci)	Concentration (μCi/cc)	Concentration (Bq/m^3)	DCF (Sv/s per Bq/m^3)	External Dose (Sv/s)	External Dose (mR/hr)
Br-83	[redacted]	[redacted]	[redacted]	[redacted]	3.82E-16	1.34E-14	4.81E-06
Br-84	[redacted]	[redacted]	[redacted]	[redacted]	9.41E-14	6.40E-12	2.30E-03
Br-85	[redacted]	[redacted]	[redacted]	[redacted]	3.56E-15	2.85E-13	1.02E-04
I-131	[redacted]	[redacted]	[redacted]	[redacted]	1.82E-14	3.46E-12	1.25E-03
I-132	[redacted]	[redacted]	[redacted]	[redacted]	1.12E-13	3.20E-11	1.15E-02
I-133	[redacted]	[redacted]	[redacted]	[redacted]	2.94E-14	1.30E-11	4.67E-03
I-134	[redacted]	[redacted]	[redacted]	[redacted]	1.30E-13	6.60E-11	2.38E-02
I-135	[redacted]	[redacted]	[redacted]	[redacted]	7.98E-14	3.30E-11	1.19E-02
I-136	[redacted]	[redacted]	[redacted]	[redacted]	1.35E-13	2.86E-11	1.03E-02
Kr-83m	[redacted]	[redacted]	[redacted]	[redacted]	1.50E-18	5.25E-17	1.89E-08
Kr-85m	[redacted]	[redacted]	[redacted]	[redacted]	7.48E-15	5.98E-13	2.15E-04
Kr-85	[redacted]	[redacted]	[redacted]	[redacted]	1.19E-16	5.94E-16	2.14E-07
Kr-87	[redacted]	[redacted]	[redacted]	[redacted]	4.12E-14	6.73E-12	2.42E-03
Kr-88	[redacted]	[redacted]	[redacted]	[redacted]	1.02E-13	2.36E-11	8.50E-03
Kr-89	[redacted]	[redacted]	[redacted]	[redacted]	9.21E-14	2.77E-11	9.97E-03
Kr-90	[redacted]	[redacted]	[redacted]	[redacted]	6.07E-14	1.98E-11	7.14E-03
Kr-91	[redacted]	[redacted]	[redacted]	[redacted]			
Xe-131m	[redacted]	[redacted]	[redacted]	[redacted]	3.89E-16	8.10E-16	2.91E-07
Xe-133m	[redacted]	[redacted]	[redacted]	[redacted]	1.37E-15	1.75E-14	6.30E-06
Xe-133	[redacted]	[redacted]	[redacted]	[redacted]	1.56E-15	6.71E-13	2.42E-04
Xe-135m	[redacted]	[redacted]	[redacted]	[redacted]	2.04E-14	1.56E-12	5.63E-04
Xe-135	[redacted]	[redacted]	[redacted]	[redacted]	1.19E-14	3.49E-12	1.26E-03
Xe-137	[redacted]	[redacted]	[redacted]	[redacted]	9.06E-15	3.64E-12	1.31E-03
Xe-138	[redacted]	[redacted]	[redacted]	[redacted]	5.77E-14	2.41E-11	8.67E-03
Xe-139	[redacted]	[redacted]	[redacted]	[redacted]			
Xe-140	[redacted]	[redacted]	[redacted]	[redacted]			

Sum:	[redacted]	[redacted]	[redacted]				**0.11**

[redacted]

Maximum Exposed Member of the Public Whole Body							
Nuclide	Inventory (Ci)	Release (Ci)	Concentration (μCi/cc)	Concentration (Bq/m^3)	DCF (Sv/s per Bq/m^3)	External Dose (Sv/s)	External Dose (mR/hr)
Br-83	[redacted]	[redacted]	[redacted]	[redacted]	3.82E-16	4.68E-14	1.68E-05
Br-84	[redacted]	[redacted]	[redacted]	[redacted]	9.41E-14	2.24E-11	8.06E-03
Br-85	[redacted]	[redacted]	[redacted]	[redacted]	3.56E-15	9.96E-13	3.59E-04
I-131	[redacted]	[redacted]	[redacted]	[redacted]	1.82E-14	1.21E-11	4.36E-03
I-132	[redacted]	[redacted]	[redacted]	[redacted]	1.12E-13	1.12E-10	4.03E-02
I-133	[redacted]	[redacted]	[redacted]	[redacted]	2.94E-14	4.54E-11	1.63E-02
I-134	[redacted]	[redacted]	[redacted]	[redacted]	1.30E-13	2.31E-10	8.31E-02
I-135	[redacted]	[redacted]	[redacted]	[redacted]	7.98E-14	1.15E-10	4.15E-02
I-136	[redacted]	[redacted]	[redacted]	[redacted]	1.35E-13	1.00E-10	3.60E-02
Kr-83m	[redacted]	[redacted]	[redacted]	[redacted]	1.50E-18	1.84E-16	6.61E-08
Kr-85m	[redacted]	[redacted]	[redacted]	[redacted]	7.48E-15	2.09E-12	7.53E-04
Kr-85	[redacted]	[redacted]	[redacted]	[redacted]	1.19E-16	2.08E-15	7.49E-07
Kr-87	[redacted]	[redacted]	[redacted]	[redacted]	4.12E-14	2.36E-11	8.49E-03
Kr-88	[redacted]	[redacted]	[redacted]	[redacted]	1.02E-13	8.26E-11	2.97E-02
Kr-89	[redacted]	[redacted]	[redacted]	[redacted]	9.21E-14	9.70E-11	3.49E-02
Kr-90	[redacted]	[redacted]	[redacted]	[redacted]	6.07E-14	6.94E-11	2.50E-02
Kr-91	[redacted]	[redacted]	[redacted]	[redacted]			
Xe-131m	[redacted]	[redacted]	[redacted]	[redacted]	3.89E-16	2.83E-15	1.02E-06
Xe-133m	[redacted]	[redacted]	[redacted]	[redacted]	1.37E-15	6.12E-14	2.20E-05
Xe-133	[redacted]	[redacted]	[redacted]	[redacted]	1.56E-15	2.35E-12	8.46E-04
Xe-135m	[redacted]	[redacted]	[redacted]	[redacted]	2.04E-14	5.47E-12	1.97E-03
Xe-135	[redacted]	[redacted]	[redacted]	[redacted]	1.19E-14	1.22E-11	4.40E-03
Xe-137	[redacted]	[redacted]	[redacted]	[redacted]	9.06E-15	1.28E-11	4.59E-03
Xe-138	[redacted]	[redacted]	[redacted]	[redacted]	5.77E-14	8.43E-11	3.03E-02
Xe-139	[redacted]	[redacted]	[redacted]	[redacted]			
Xe-140	[redacted]	[redacted]	[redacted]	[redacted]			
Sum:	[redacted]	[redacted]	[redacted]				**0.37**

[redacted]

Maxium Exposed Person Whole Body							
Nuclide	**Inventory (Ci)**	**Release (Ci)**	**Concentration (µCi/cc)**	**Concentration (Bq/m^3)**	**DCF (Sv/s per Bq/m^3)**	**External Dose (Sv/s)**	**External Dose (mR/hr)**
Br-83	[redacted]	[redacted]	[redacted]	[redacted]	3.82E-16	9.35E-12	3.37E-03
Br-84	[redacted]	[redacted]	[redacted]	[redacted]	9.41E-14	4.48E-09	1.61E+00
Br-85	[redacted]	[redacted]	[redacted]	[redacted]	3.56E-15	1.99E-10	7.17E-02
I-131	[redacted]	[redacted]	[redacted]	[redacted]	1.82E-14	2.42E-09	8.72E-01
I-132	[redacted]	[redacted]	[redacted]	[redacted]	1.12E-13	2.24E-08	8.06E+00
I-133	[redacted]	[redacted]	[redacted]	[redacted]	2.94E-14	9.08E-09	3.27E+00
I-134	[redacted]	[redacted]	[redacted]	[redacted]	1.30E-13	4.62E-08	1.66E+01
I-135	[redacted]	[redacted]	[redacted]	[redacted]	7.98E-14	2.31E-08	8.30E+00
I-136	[redacted]	[redacted]	[redacted]	[redacted]	1.35E-13	2.00E-08	7.20E+00
Kr-83m	[redacted]	[redacted]	[redacted]	[redacted]	1.50E-18	3.67E-14	1.32E-05
Kr-85m	[redacted]	[redacted]	[redacted]	[redacted]	7.48E-15	4.18E-10	1.51E-01
Kr-85	[redacted]	[redacted]	[redacted]	[redacted]	1.19E-16	4.16E-13	1.50E-04
Kr-87	[redacted]	[redacted]	[redacted]	[redacted]	4.12E-14	4.71E-09	1.70E+00
Kr-88	[redacted]	[redacted]	[redacted]	[redacted]	1.02E-13	1.65E-08	5.95E+00
Kr-89	[redacted]	[redacted]	[redacted]	[redacted]	9.21E-14	1.94E-08	6.98E+00
Kr-90	[redacted]	[redacted]	[redacted]	[redacted]	6.07E-14	1.39E-08	5.00E+00
Kr-91	[redacted]	[redacted]	[redacted]	[redacted]			
Xe-131m	[redacted]	[redacted]	[redacted]	[redacted]	3.89E-16	5.67E-13	2.04E-04
Xe-133m	[redacted]	[redacted]	[redacted]	[redacted]	1.37E-15	1.22E-11	4.41E-03
Xe-133	[redacted]	[redacted]	[redacted]	[redacted]	1.56E-15	4.70E-10	1.69E-01
Xe-135m	[redacted]	[redacted]	[redacted]	[redacted]	2.04E-14	1.09E-09	3.94E-01
Xe-135	[redacted]	[redacted]	[redacted]	[redacted]	1.19E-14	2.44E-09	8.80E-01
Xe-137	[redacted]	[redacted]	[redacted]	[redacted]	9.06E-15	2.55E-09	9.18E-01
Xe-138	[redacted]	[redacted]	[redacted]	[redacted]	5.77E-14	1.69E-08	6.07E+00
Xe-139	[redacted]	[redacted]	[redacted]	[redacted]			
Xe-140	[redacted]	[redacted]	[redacted]	[redacted]			
Sum:	[redacted]	[redacted]	[redacted]				**74.24**

Dose in 5 minutes: 6.19

TECHNICAL SPECIFICATIONS

Included in this section are the technical specifications and the bases for the technical specifications. These bases, which provide the technical support for the individual technical specifications, are included for informational purposes only. They are not part of technical specifications and they do not constitute limitations or requirements to which the licensee must adhere.

1 Definitions

1.1 30/20 LEU 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 ZrH_x): 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%.

4. Cladding: 304 stainless steel.

1.2 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.

1.3 Annually

Items which are designated by these specifications or by standard operating procedures to have an annual surveillance requirement shall allow, for operational flexibility only, an interval not to exceed 15 months.

1.4 Audit

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

1.5 Biennially

Items which are designated by these specifications or by standard operating procedures to have a biennial surveillance requirement shall allow, for operational flexibility only, an interval not to exceed 30 months.

1.6 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.

1.6.1 Channel Test

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

1.6.2 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.

1.6.3 Channel Check

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

1.7 Confinement

Confinement means a closure of the overall facility that results in the control of the movement of air into and out of the facility through a controlled or defined path.

1.8 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.

1.8.1 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.

1.8.2 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.

1.8.3 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.

1.9 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.

1.10 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.

1.11 Excess Reactivity

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

1.12 Experiment

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

1.12.1 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.

1.12.2 Unsecured Experiment

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

1.12.3 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.

1.13 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.

1.14 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.

1.15 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.

1.16 Fuel Element

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

1.17 Instrumented Fuel Element (IFE)

An instrumented fuel element is a special fuel element in which one or more sheathed chromel-alumel or equivalent thermocouples are embedded in the fuel near the horizontal center plane of the fuel element at a point approximately 0.3 inch from the center of the fuel body.

1.18 License

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

1.19 Licensee

A licensee is an individual or organization holding a license.

1.20 LEU Core

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

1.21 Limiting Safety System Setting (LSSS)

The limiting safety system setting is a setting for automatic protective action related to those variables having significant safety functions.

1.22 Measuring Channel

A measuring channel is the combination of sensor, interconnecting cables or lines, amplifiers, and output device which are connected for the purpose of measuring the value of a variable.

1.23 Measured Value

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

1.24 Operable

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

1.25 Operating

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

1.26 Operational Core – Steady State

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

1.27 Operational Core – Pulse

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

1.28 Operator

See Reactor Operator.

1.29 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.

1.30 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.

1.31 Pulse Mode

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

1.32 Quarterly

Items that are designated by these specifications or by standard operating procedures to have a quarterly surveillance requirement shall allow, for operational flexibility only, an interval not to exceed 4 months.

1.33 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.

1.34 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.

1.35 Reactor Operating

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

1.36 Reactor Operator

A Reactor Operator is an individual who is licensed to manipulate the controls of a reactor.

1.37 Reactor Safety Systems

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

1.38 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) The minimum number of neutron-absorbing control devices is fully inserted or other safety devices are in shutdown position, as required by technical specifications;

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

(c) The reactor is shutdown

1.39 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 and all of the following conditions exist:

(a) 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;

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

1.40 Reference Core Condition

The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible (<0.30 dollar).

1.41 Reportable Occurrence

Any of the following events is a reportable occurrence:

(1) Violation of a safety limit.

(2) Release of fission products from a leaking fuel element..

(3) Release radioactivity from the site above limits established by 10 CFR 20 or specified within these Technical Specifications.

(4) Operation with actual safety system settings for required systems less conservative than specified in the Technical Specifications.

(5) Operation in violation of a Limiting Condition of Operation listed in Section 3.

(6) 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.

(7) An unanticipated or uncontrolled change in reactivity greater than $1.00.

(8) Abnormal and significant degradation in reactor fuel or cladding, or both, coolant

boundary, or confinement boundary where appropriate.

(9) 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.

1.42 Review

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

1.43 Safety Channel

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

1.44 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.

1.45 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.

1.46 Semiannually

Items which are designated by these specifications or by standard operating procedures to have a semiannual surveillance requirement shall allow, for operational flexibility only, an interval not to exceed 7.5 months.

1.47 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.

1.48 Shall, Should and May

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

1.49 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.

1.50 Steady State Mode

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

1.51 True Value

The true value is the actual value of a parameter.

1.52 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.

1.53 Weekly

Items which are designated by these specifications or by standard operating procedures to have a weekly surveillance requirement shall allow, for operational flexibility only, ant interval not to exceed 10 days.

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 limits 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. One category of error between the true temperature value and the measured temperature value is due to the accuracy of the fuel element channel and any overshoot in reactor power resulting from a reactor transient during steady state mode of operation. Although a lesser contributor to error, a minimum safety margin of 10% was applied on an absolute temperature basis. Adjusting the fuel temperature safety limit to degrees Kelvin, K, and applying a 10% safety margin results in a safety limit reduction of 150°C. Applying this first margin of safety, the safety setting would be 1000°C for LEU. However, to arrive at the final LSSS it is necessary to allow for the difference between the measured temperature value and the peak core temperature, which is a function of the location of the thermocouple within the core. For example, if the thermocouple element were located in the hottest position in the core, the difference between the true and measured temperatures would be only a few degrees since the thermocouple junction is at the mid-plane of the element and close to the anticipated hot spot. However, at the NSC, this core position is not available due to the location of the transient rod. For the NSC, the location of the instrumented elements is therefore restricted to the positions closest to the central element. Calculations indicate that, for this case, the true temperature at the hottest location in the core will differ from the measured temperature by no more than 40%. When applying this 40% worst case measurement scenario and considering the previously mentioned sources of error between the true and measured values, a final LSSS temperature of 975°F (525°C) is imposed on operation. Viewed on an absolute temperature scale, K, this represents a 37% safety margin in the LEU safety limit.

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.

Specifications

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.

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 assure 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

TRIGA fuel is fabricated with a nominal hydrogen-to-zirconium ratio of 1.6 for LEU fuel. This yields delta phase zirconium hydride that has high creep strength and undergoes no phase changes at temperatures over 1000°C. However, after extensive steady state operation at 1.0 MW the hydrogen will redistribute due to migration from the central high temperature regions of the fuel to the cooler outer regions. When the fuel is pulsed, the instantaneous temperature distribution is such that the highest values occur at the surface of the element and the lowest values occur at the center. The higher temperatures in the outer regions occur in fuel with a hydrogen-to-zirconium ratio that has now substantially increased above the nominal value. This produces hydrogen gas pressures considerably in excess of the expected for $ZrH_{1.6}$. If the pulse insertion is such that the temperature of the fuel exceeds 874°C, then the pressure will be sufficient to cause expansion of microscopic holes in the fuel that grows with each pulse. The pulsing limit of 830°C is obtained by examining the equilibrium hydrogen pressure of zirconium hydride as a function of temperature. The decrease in temperature from 874°C to 830°C reduces hydrogen pressure by a factor of two, which is an acceptable safety factor. This phenomenon does not alter the safety limit since the total hydrogen in a fuel element does not change. Thus, the pressure exerted on the clad will not be significantly affected by the distribution of hydrogen within the element.

In practice 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

These specifications apply 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.

Specifications

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 Core Configuration Limitation

Applicability

This specification applies to a LEU core.

Objective

The objective is to ensure that the fuel temperature safety limit will not be exceeded due to power peaking effects in full LEU cores and with various experimental facilities installed.

Specifications

1. The TRIGA core assembly shall be LEU.

2. The reactor shall not be taken critical with a core lattice position vacant except for positions on the periphery of the core assembly. Water holes in the inner fuel region shall be limited to single rod positions. Vacant core positions 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. Safety and accident analysis were only performed for a TRIGA core using LEU fuel.

2. Vacant core positions containing experiments or an experimental facility will prevent accident 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. Reference: Basis of TS 2.2

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.

Specifications

1. The reactor shall not be operated knowingly with damaged fuel.

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.

   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.

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

Basis

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

3.1.6 Maximum Excess Reactivity

Applicability

This specification applies to the maximum excess reactivity, above reference core condition, which may be loaded into the reactor core at any time.

Objective

The objective is to ensure that the core analyzed in the safety analysis report approximates the operational core within reasonable limits.

Specifications

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

Basis

Although maintaining a minimum shutdown margin at all times ensures that the reactor can be shut down, that specification does not address the total reactivity available within the core. 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.

Note: General Atomics' (GA) modeling of the NSC LEU core predicted an excess reactivity of $6.22. The initial data for the NSC reactor start up with a cold, xenon-free core in 2006 yielded a measured excess reactivity of $7.483. GA's modeling of the PRNC FLIP core had predicted $6.26, while the experimentally determined measured value was $7.12.

3.2 Reactor Control and Safety Systems

3.2.1 Reactor Control Systems

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 assure safe operation of the reactor.

Specifications

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

Table 1: Measuring Channels Required for Operation

Measuring 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 specifications 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 circuits 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.

Specifications

The reactor shall not be operated unless the safety circuits and interlocks described in Table 14-2 are operable. However during periods of maintenance, surveillance, calibration and repair, the reactor may be operated as necessary to calibrate and ensure channel operability provided that all other conditions of the TS are met during the operation.

Table 2: Safety Circuits and Interlocks Required for Operation

Safety Channel	Number Operable	Function	Operating Mode	
			S.S.	Pulse
Fuel Element Temperature	1	Scram ≤ the LSSS	X	X
High Power Level	2	Scram ≤ 125%	X	-
Console Scram Button	1	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
Log Power	1	Prevent withdrawal of Shim Safeties at <$4x10^{-3}$ W	X	-
Log Power	1	Prevent pulsing above 1 kW	-	X
Transient Rod Position	1	Prevent application of air unless fully inserted	X	-
Shim Safety and Regulating Rod Position	1	Prevent withdrawal	-	X
Pulse Stop Electro-Mechanical Interlock	1	Prevent application of air unless pulse stop is installed	-	X
Pool Water Temperature	1	Manual scram if temperature reaches 60 C	X	X

Basis

1. The fuel temperature and high power level scrams provide protection to assure 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. The interlock to prevent startup of the reactor at power levels less than 4 x 10-3 W, which corresponds to approximately 2 cps, assures that sufficient neutrons are available for proper startup.

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

7. 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.

8. 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.

9. 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.

10. 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.

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 scammable control rods shall be operable. 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 scrammable 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 assure the safety of the reactor.

3.3 Confinement

3.3.1 Operations that Require Confinement

Applicability

This specification applies to confinement requirements during operation of the reactor and the handling of radioactive materials.

Objective

To maintain normal or emergency airflow into and out of the reactor building during operations that produce or could potentially produce airborne radioactivity.

Specifications

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

1. Reactor operating

Note: For periods of maintenance to the central exhaust system during reactor operation, 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.

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

Basis

1. This basis applies during the conduct of those activities defined as reactor operations. ^{41}Ar is produced during operation of the reactor in experimental facilities and in the

reactor pool; thus, air control within the building and the exhaust system in necessary to maintain proper airborne radiation levels in the reactor building and release levels in the exhaust stack. Other radioactivity releases to the reactor building must be considered during reactor operation, such as fission product release from a leaking fuel element or a release from fixed experiments in or near the core. 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. The handling of radioactive materials can result in the accidental or controlled release of airborne radioactivity to the reactor building environment or direct release to the building exhaust system. In these cases, the control of air into and out of the reactor building is necessary.

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.

Specifications

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 exhaust fan 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.

Note: 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.

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.

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.

Objective

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

Specification

The reactor shall not be operated unless the radiation monitoring channels listed in Table 14-3 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

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

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

Note: When required monitors are inoperable, or for periods of maintenance, the intent of this specification will be satisfied if it is replaced for a period of no more than 1 week only if a gamma sensitive instrument may be substituted for the inoperable channel and has its own alarm which shall be kept under visual observation. The suitability of the substitute instrument shall be determined by the SRO on duty. If two of the above monitors are not operating, the reactor shall be shutdown.

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 concentration of ^{41}Ar in the effluent gas from the facility as diluted by atmospheric air in the lee of the facility due to the turbulent wake effect shall not exceed 1.0×10^{-8} μCi/ml averaged over one year.

Basis

The maximum allowable concentration of ^{41}Ar in air in unrestricted areas as specified in Appendix B, Table II of 10CFR20 is 1.0×10^{-8} μCi/ml. Section 11 of the SAR for the NSC substantiates a 5.0×10^{-3} atmospheric dilution factor for a 2.0 mph wind speed.

3.5.3 Xenon and Iodine Monitoring

Applicability

This specification applies to the radiation monitoring systems necessary to monitor and control the concentration of any effluent releases during the production of ^{125}I from the radioactive decay of ^{125}Xe.

Objective

The objective is to ensure sufficient radiation monitoring information is available to the operator, and the central exhaust system is automatically isolated to ensure that the health and safety of the general public is not endangered during the production of ^{125}I.

Specification

No experiment that involves active handling of ^{125}Xe and ^{125}I may be performed unless the following radiation monitoring system is operable (Table 14-4). No experiment may be performed, except decay of ^{125}Xe, unless the ^{125}Xe FAM channel is operable.

Table 4: Radiation Monitoring Required for Xenon-125 and Iodine-125 Experiments

Radiation Monitoring Channels	Number
^{125}Xe Effluent Monitoring Channel (FAM Ch. 5)	1
^{125}I Air Monitor	1

Upon receipt of an alarm signal from the ^{125}Xe Effluent Monitoring Channel the central exhaust system shall be automatically isolated.

Note: When required monitors are inoperable, or for periods of maintenance, the intent of this specification will be satisfied if this monitor is replaced for a period of no more than 1 week only if a gamma sensitive instrument may be substituted for the inoperable channel and has its own alarm which shall be kept under visual observation. The suitability of the substitute instrument shall be determined by the SRO on duty.

Basis

The required facility air monitors (3.5.1) are not calibrated for ^{125}I. The ^{125}Xe FAM channel provides information to operators in the event there is a significant release of ^{125}Xe during the production of ^{125}I. The ^{125}I air monitor measures ^{125}I radiation levels in the immediate area where the iodine handling is being performed.

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.

Specifications

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

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

2. The reactivity worth of any secured experiment shall be less than two dollars.

3. The sum of the absolute reactivity of all experiments shall be less than five dollars.

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.

Specifications

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 25 milligrams (TNT-equivalent) shall not be irradiated in the reactor pool. Explosive materials in quantities less than 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 25 milligrams (TNT-equivalent) shall be restricted from the reactor pool, the upper research level, the demineralizer room, cooling equipment room and the interior of the pool containment structure.

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

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

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

2. Each fueled experiment shall be controlled such that its inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1 Ci.

3. Significant amounts of 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. (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. ; Amendment No. 7 to Facility License No. R-83)

2. The 1 Ci limitation on Iodine 131 through 135 ensures that in the event of failure of a fueled experiment leading to total release of the iodine, the exposure dose at the exclusion area boundary will be less than that allowed by 10 CFR 20 for an unrestricted area.

3. 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.

Specifications

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 a) 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.

    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 assure safe operation of the reactor.

3.6.4 Xenon Irradiation for Iodine Production

Applicability

This specification applies to the experiments that produce ^{125}I from the activation of enriched ^{124}Xe and the decay of ^{125}Xe.

Objective

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

Specifications

1. ^{124}Xe activation experiments shall be controlled such that the total single experiment activity produced is limited to no more than 2000 Ci of ^{125}Xe.

2. The total facility ^{125}Xe inventory of all experiments shall not exceed 3500 Ci.

Basis

1. The 2000 Ci limitation on Xenon-125 produced in any one experiment assures that in the event of a failure of an experiment leading to the accidental release of xenon, the exposure to the general public would be less than 0.05 rem per year (10 CFR 20.1301).

2. In the NRC's safety evaluation supporting amendment 15 to our facility license the commission concluded that the NSC's request for a total inventory limit of Xe-125 of 3500 Ci was reasonable based on the facility experimental design and operating plan for Iodine Production.

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.

Specifications

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

4. In the event of a significant fission product leak from a fuel rod or a significant 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.

5. Before discharge, the facility liquid effluents collected in the holdup tanks shall be analyzed for the nature and concentration of radioactive effluents. The total quantity of radioactive material that is released into the sanitary sewerage system in a year shall not exceed 1 curie (37 GBq) of all radioactive materials combined. 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.

Calculations show that a total annual discharge of 30 Ci ^{41}Ar into the environment will result in a dose to a member of the general public of 6.3 millirem and this demonstrates compliance with the annual dose constraint of 10 millirem set by 10 CFR 20.1101.

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.

Specifications

1. Conductivity of the bulk pool water shall be no higher than 5 x 10^{-6} mhos/cm (5 μSiemens/cm).

2. The pH of the bulk pool water shall be between 5.5 and 7.5.

3. 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

1) The Director, or his designee, shall be notified when pool temperature reaches 50 C, with further operation dependent on his approval.

2) 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. (Evaluation of Pool Temperature on Fuel Temperature and MDNBR for the NSC TRIGA Reactor, August 2010). 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.

Specifications

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 assure 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.

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 significant core configuration and/or control rod changes.

Basis

The reactivity worth of the control rods is measured to assure 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 insure no significant changes in the shutdown margin.

4.1.4 Core Configuration Limitation

The LCO 3.1.4 Core Configuration Limitation does not require a Surveillance Requirement.

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.

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 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 biennially and following significant core configuration and/or control rod changes.

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 Control System

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.

Specifications

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.

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.

4.2.2 Reactor Safety Systems and Interlocks

Applicability

These specifications apply 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.

Specifications

1. A channel check of each of the reactor safety system channels and interlocks for the intended mode of operation shall be performed before each day's operation or before each operation extending more than one day, except for the pool level alarm channel, which shall be tested weekly.

2. Whenever a reactor scram caused by high power level or high fuel element temperature occurs, an evaluation shall be conducted to determine whether the fuel element temperature safety limit was exceeded.

3. A calibration of the fuel element temperature measuring channels shall be performed semiannually.

4. A calibration of the pool water temperature measuring channel shall be performed semiannually.

5. A channel check of the fuel element temperature measuring channel shall be made daily whenever the reactor is operated.

Basis

Channel tests will assure that the safety system channels are operable on a daily basis or prior to an extended run, except for the pool level alarm channel, which is verified weekly. If the period between operations extends beyond a year, then the annual channel test requirement will assure operability.

4.2.3 Scram Time

Applicability

This specification applies to the surveillance of control rod scram times.

Objective

The objective is to verify that all scrammable control rods meet the scram time requirement.

Specification

The scram time shall be measured annually.

Basis

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

During periods of operation, or radioactive material handling, the central exhaust system shall be verified operable weekly. 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 assure the proper operation of the system and control of the release of radioactive material.

4.4 Ventilation Systems

The LCO 4 Ventilation Systems does not require a Surveillance Requirement.

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 facility air monitoring system (FAM) shall be calibrated annually and shall be tested weekly.
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.

TS 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.

Specifications

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 6.4 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 an experiment shall be estimated or measured, as appropriate, before reactor operation with said experiment.

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 As Low As Reasonably Achievable (ALARA) Radioactive Effluents Released

Surveillance for the LCO 3.7 As Low As Reasonably Achievable (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 to know the purity of the pool water.

Specifications

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.

3. pH of the bulk pool water shall be measured and recorded quarterly.

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

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.

Specifications

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

2. The pool water level alarm shall be verified operable weekly.

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.

Specifications

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.

4.9 Surveillance Deferment

Applicability

This specification describes which surveillance requirements may be deferred during extended reactor shutdown, and which shall be performed prior to resuming reactor operations.

Objective

The objective is to give facility personnel the ability to avoid conducting unnecessary surveillances during periods of reactor shutdown.

Specifications

	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 Excess Reactivity	Yes	Yes
5.	4.1.5 Reactor Fuel Elements	No	N/A
6.	4.2.1 Reactor Control Systems	No	N/A
7.	4.2.2 Reactor Safety Systems and Interlocks	Yes, except pool level	Yes
8.	4.2.3 Scram Time	Yes	Yes
9.	4.3 Confinement	Yes	Yes
10.	4.5 Radiation Monitoring Systems and Effluents	No	N/A
11.	4.6 Experiments	Yes	Yes
12.	4.8.1 Primary Coolant Purity	Yes - Quarterly	Yes
13.	4.8.2 Primary Coolant Level and Leak Detection	No	Yes
14.	4.8.3 Primary Coolant Temperature	Yes	Yes

Basis

1. It is impossible to perform a calorimetric while the reactor is shutdown; a calorimetric shall be performed prior to resuming operations.

2. It is impossible to perform pulse surveillance while the reactor is shutdown; the surveillance shall be performed prior to resuming operations.

3. It is impossible to determine control rod worth while the reactor is shutdown; the surveillance shall be performed prior to resuming operations.

4. It is impossible to determine control rod worth while the reactor is shutdown; the surveillance shall be performed prior to resuming operations.

5. The possibility for cladding degradation or damage still exists while the reactor is shutdown, therefore this surveillance shall not be deferred.

6. The possibility for control rod degradation or damage still exists while the reactor is shutdown, therefore this surveillance shall not be deferred.

7. With the exception of the pool level alarm, whose surveillance shall not be deferred, it is not necessary to conduct surveillance on reactor safety systems, which are effectively automatic scram systems, while the reactor is shutdown. The surveillance shall be performed prior to resuming operations.

8. It is impossible to determine scram time while the reactor is shutdown; the surveillance shall be performed prior to resuming operations.

9. The central exhaust system is only required during reactor operations or radioactive materials handling. The surveillance shall be performed prior to resuming operations or handling.

10. The possibility for cladding degradation or damage still exists while the reactor is shutdown, therefore this surveillance shall not be deferred.

11. No experiments can be installed in the reactor while it is shutdown; the surveillance shall be performed prior to resuming operations.

12. The possibility for cladding degradation or damage still exists while the reactor is shutdown; however it is much less likely. Therefore the periodicity of this surveillance may be changed to quarterly. The surveillance shall be performed prior to resuming operations.

13. Pool level is not affected by reactor operation. The surveillance must be maintained regardless of reactor operation.

14. As the reactor is the source of heat for the pool, it is impossible for the pool to reach 50 C while the reactor is shutdown; the surveillance shall be performed prior to resuming operations.

5 Design Features

5.1 Site and Facility Description

Applicability

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

Objective

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

Specifications

1. The restricted area is that area inside the fence surrounding the site.

2. The reactor confinement building houses the TRIGA reactor and has a minimum free volume of 180,000 cubic feet.

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

4. Emergency controls for the shutdown of the central exhaust systems and isolation of the reactor confinement building shall be located in the reactor control room and emergency support center.

5. The licensed area of the facility is the area inside the site boundary.

Basis

The restricted area is described in SAR 2.2.1.1. The facility is designed such that the central exhaust system will maintain a negative pressure in the reactor confinement building with respect to the outside atmosphere so that there will be no uncontrolled discharge to the environment. Controls for startup and operation of the central exhaust system are located in the reactor control room and emergency support center. 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.

Specifications

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%.

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.

Specifications

1. The core shall be an arrangement of TRIGA 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. Core lattice position 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

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. Water in combination with graphite reflectors can be used for neutron economy and the enhancement of experimental facility radiation requirements.

3. Vacant core positions containing experiments or an experimental facility will prevent accident 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.

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.

Specifications

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.

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 for continuously monitoring airborne radioactivity.

Objective

The objective is to describe the radiation monitoring equipment available to the operator to assure safe operation of the reactor.

Specification

The radiation monitoring equipment listed in Table 14-5 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	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	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.

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.

Specifications

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 5.5.1 and 5.5.2 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.

Specifications

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 a 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 fission product facility air monitor.

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.

Specifications

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 and outlet pipes to the heat exchanger shall have emergency covers 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 readout in the control room and 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. Covers for these lines will be stored in the reactor pool. The time required to uncover the reactor core due to failure of a single pool coolant pipe system is calculated to be 17 minutes. Extensive operator trials have shown it is reasonable to expect the covers to be installed in less than 5 minutes, which is much shorter than 17 minutes.

4. This alarm is observed in the reactor control room, in the emergency support center, and at a continuously manned 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 licensee for the administration and the proper and safe operation of the facility. Figure 12-1 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.

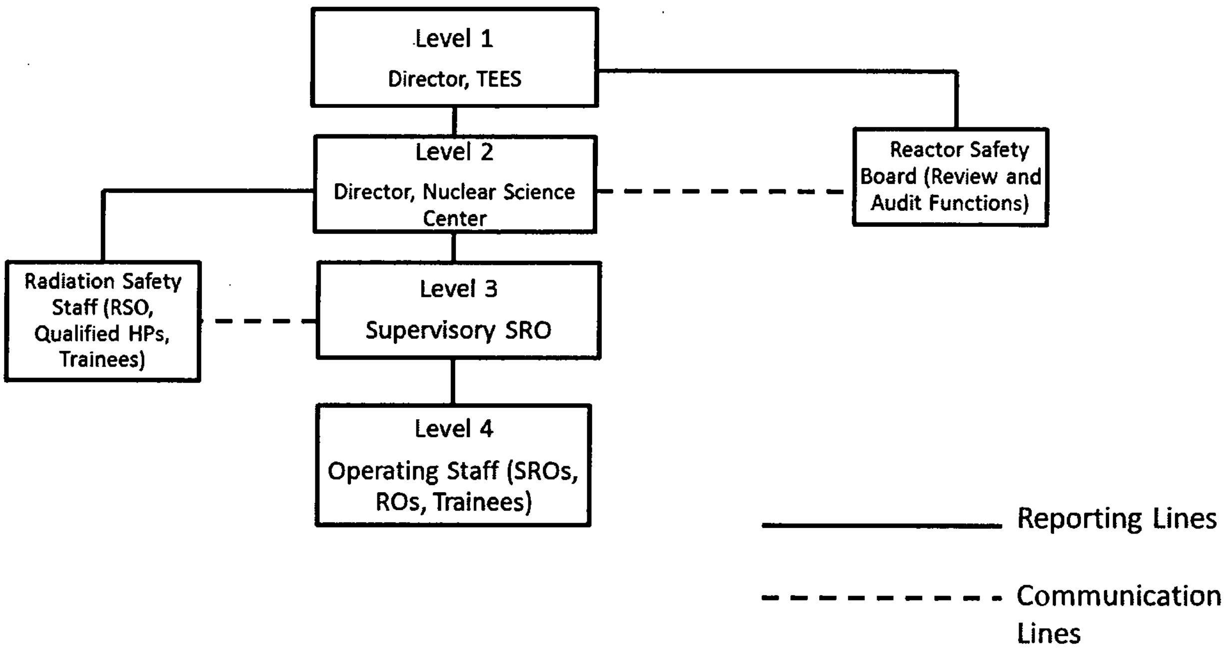

Figure 1: Organization Chart for Reactor Administration

6.1.1 Structure

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

2. The TEES licensee 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. Radiation Safety Staff:

The Radiation Safety Officer has the responsibility for implementation of the radiation protection program at the NSC. The Radiation Safety Officer shall direct the actions of the Radiation Safety Staff, including qualified Health Physicists and Trainees. The individual reports to Level 2 management.

5. 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.

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.

Note: During periods of reactor maintenance the reactor operator or the operator trainee may be replaced by maintenance personnel. The licensed senior reactor operator may be permitted as the only operations person present at the facility to perform a pre-startup check of the reactor or perform general reactor maintenance. The unsecured reactor shall always be under the direct control of the Senior Reactor Operator.

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

c. 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).

d. 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. Administrative personnel

   b. Radiation safety personnel

   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 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 of an in-core experiment with a reactivity worth greater than one dollar.

   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.

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, normally the Deputy Director of TEES, 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 normally appointed by the licensee 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 3 voting members of the RSB excluding ex-officio members.

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.

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. 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 whether a proposed change, test, experiment, or changes in equipment and systems would constitute an unevaluated 10CFR50.59 safety question or a change in technical specification;

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, license or charter;

5. Review of the NSC radiation protection program;

6. Review of violations of technical specifications, license, or charter, 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 6.6.2 of these TS;

9. Review of audit reports.

6.2.4 RSB Audit Function

The RSB or a subcommittee thereof shall audit reactor operations and radiation protection programs annually. 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;

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;

3. The retraining and requalification program for the operating staff;

4. The facility security plan and records;

5. The reactor facility emergency plan and implementing procedures.

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 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 in effect 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. Civil disturbances on or near the facility site;

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

9. Actions to be taken to correct specific and foreseen potential malfunctions of systems, including responses to alarms and abnormal reactivity changes;

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

11. Surveillance procedures for shipping radioactive materials.

6.4 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. The NSC Director or his designated alternate may approve minor changes that do not significantly alter the experiment.

6.5 Required Actions

6.5.1 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 NRC.

2. An immediate report of the occurrence shall be made to the Chairman, RSB, NSC Director, and reports shall be made to the NRC in accordance with Section 6.6.2 of these specifications, and

3. A report shall be prepared which shall include an analysis of the cause and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to the RSB for review and then submitted to the NRC when authorization is sought to resume operation of the reactor.

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

In the event of a reportable occurrence as stated in TS 1.41, the following action shall be taken:

1. NSC staff shall return the reactor to normal operating 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 Director or his designated alternate shall notify the Chairman of the RSB 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.

5. A report shall be made to the NRC in accordance with Section 6.6.2 of these specifications.

6.6 Reporting Requirements

6.6.1 Annual 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 brief narrative summary of (1) 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 emergency shutdowns and inadvertent scrams, including reasons thereof;

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. 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.

        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

        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).

    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.

8. A description and summary of any environmental surveys performed outside the facility.

6.6.2 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 be followed by a written report that describes the circumstances of the event and includes a discussion of the effect of the violation upon the health and safety of the public within 14 days of any of the following:

   a. Violation of safety limits (6.5.1).

   b. Any accidental release of radioactivity above permissible limits in unrestricted areas whether or not the release resulted in property damage, personal injury, or exposure;

   c. Any reportable occurrences as defined in TS 1.41. The written report (and, to the extent possible, the preliminary telephone or faxed report) shall describe, analyze, and evaluate safety implications, and outline the corrective measures taken or planned to prevent reoccurrence of the event;

2. A written report within 30 days of:

   a. Personnel changes in the facility organization involving Level 1 and Level 2.
   b. Significant changes in the transient or accident analysis as described in the Safety Analysis Report.

6.7 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:

6.7.1 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.

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.

9. Records of meeting and audit reports of the RSB.

6.7.2 Records to be Retained for at Least One Training Cycle

1. Retraining and Requalification of certified operations personnel.

2. Records of the most recent complete cycle shall be maintained for individuals employed.

6.7.3 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.

6.7.4 Records to be Retained until the Commission terminates the license of the reactor

1. Exceeding a safety limit.

2. Failure of an automatic safety function to function as required.

3. Failure to meet a Limiting Condition for Operation.