UNIVERSITY OF CALIFORNIA-IRVINE
RESEARCH REACTOR
LICENSE NO. R-116
DOCKET NO. 50-326

TECHNICAL RAI RESPONSES (DATED 10/3/2011)

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUT OR DENOTED BY BRACKETS

UNIVERSITY OF CALIFORNIA, IRVINE

BERKELEY • DAVIS • IRVINE • LOS ANGELES • RIVERSIDE • SAN DIEGO • SAN FRANCISCO   SANTA BARBARA • SANTA CRUZ

George E. Miller
Senior Lecturer Emeritus
Department of Chemistry and
Director, Nuclear Reactor Facility
Faculty Advisor for Science, UCI Center for Education Partnerships

IRVINE, CA 92697-2025
(949) 824-6649 or 824-6082
FAX: (949) 824-8571
email : gemiller@uci.edu

October 3, 2011

US Nuclear Regulatory Commission
Document Control Desk
Washington DC 20555
Attention: Linh Tran, Senior Project Manager
Walter Meyer, Senior Project Manager

Dear Ms Tran and Mr Meyer:
Please find attached the response to requests for additional information by you and other staff. Several items are included:

1. Revised MHA analysis based on additional work by general atomic.
2. Copy of General Atomic report 39364 as released.
3. Proposed Technical Specifications, revision 6, dated September 2011
4. UCI ALARA policy statement.

Please let me know if there are additional items needed.

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

Executed on October 3, 2011

G. E. Miller.

Dr. George E. Miller
Director

A020
NRR

UCI Radiation Safety Program
As Low as Reasonably Achievable (ALARA) Program

The University of California campus Radiation Safety Division of Environmental Health and Safety maintains an aggressive ALARA policy. This policy and practice is enforced at the UCI Nuclear Reactor facility and is delineated in Section G of the UCI Radiation Safety Manual available on-line at http://www.ehs.uci.edu/programs/radiation/RadiationSafetyManual.pdf

G. ALARA REVIEW OF SIGNIFICANT EXPOSURES

1. Routine ALARA Exposure Investigations
   a. A routine ALARA exposure investigation will be performed by an EH&S health physicist, with review by the RSO, to determine the conditions of exposure and to suggest preventative measures to maintain future exposures ALARA, whenever:
(1) Any single dosimeter reading or bioassay measurement is:
   (a) For adults -- 5 percent or more of any applicable annual regulatory limit.
   (b) For minors under age 18 -- 10 percent or more of any applicable annual regulatory limit.
   (c) For the embryo/fetus of a declared pregnant woman – 10 percent or more of the regulatory limit for the embryo/fetus during pregnancy.
(2) Any accumulated annual combined external and internal dose reaches:
   (a) For adults -- 10 percent or more of any applicable annual regulatory limit during the year.
   (b) For minors under age 18 -- 20 percent or more of any applicable annual regulatory limit during the year.
   (c) For the embryo/fetus of a declared pregnant woman – 20 percent or more of the regulatory limit for the embryo/fetus during pregnancy.
   b. A lower exposure may be investigated for any reason.
   c. The Radiation Safety Committee (RSC) may authorize higher investigation levels for individuals or groups of individuals, as appropriate.

2. Special ALARA Exposure Investigations
A special ALARA exposure investigation will be performed by the RSO, with review by the RSC, whenever any accumulated annual combined external and internal dose reaches:
   (1) For adults -- 30 percent or more of any applicable annual regulatory limit during the year.
   (2) For minors under age 18 -- 50 percent or more of any applicable annual regulatory limit during the year.
   (3) For the embryo/fetus of a declared pregnant woman -- 50 percent or more of the regulatory limit for the embryo/fetus during pregnancy.

H. INVESTIGATION OF OVER-EXPOSURES
1. The Environmental Health & Safety Office is responsible for notification of the US NRC and/or the California DPH, as appropriate, in cases of known or suspected exposure above the regulatory dose equivalent limits.
2. Medical evaluation and/or treatment by a qualified physician for any overexposed personnel may be required by the RSO, the RSC, or the appropriate regulatory agency.
3. Whenever the annual regulatory dose limits have been reached or exceeded, personnel will usually be required to avoid work with radioisotopes and/or radiation-producing machines for the remainder of the year.

13.2 Maximum Hypothetical Accident (MHA) - Rupture of Single Fuel Element in Air.

13.2.1 Summary

The MHA for reactors with TRIGA® hydride fuel has been adopted (NUREG-1537) to be the release of fission products from a single fuel element whose cladding has been stripped. This is analyzed both under water and in air. The assumption for the MHA is made that maximum fission product inventory has been achieved in the most active fuel element, i.e., that the reactor has been operating for an infinite time period prior to the release. This is obviously a highly conservative assumption for the UCI reactor which has operated on a very intermittent schedule since it was commissioned. The actual likelihood for such a complete rupture event is very small, but is not further considered here. This section examines the consequences of the postulated MHA. The most significant activity released from the perspective of personal exposure risk is that of iodine isotopes since this delivers exposure to the thyroid gland. Direct radiation is also considered for risk to any occupants of adjacent areas not subject to risk from the volatile materials.

The released radioactivity is the product of the fission product inventory and the release fraction. Consideration is then given to the exposure risk to personnel within the facility, those in surrounding spaces including laboratories or offices, and release beyond the building via the building ventilation system. Factors included are the facility volume, the ventilation purge rate and an estimated plating factor for iodine radio-nuclides on walls and facility structures which will reduce inhalation exposures.

The total number of fission product nuclei present from the fuel was estimated below[1] and the release fraction was used from NUREG 1282. The modest levels predicted in the MHA come about as a result of the very low value of fission product release fraction established for TRIGA® fuel elements that have stripped their cladding.

13.2.2 Fission Product Inventory

The source term for the reactor inventory was calculated by General Atomic using SCALE 6.1 code assuming 1000 effective full power days (EFPD) and separated into halogen and noble gas groups. The first group (Group I) comprises bromine and iodine isotopes that will dissolve if water is present, and be airborne if it is not. The second group (Group II) comprises the insoluble volatiles: krypton and xenon isotopes. The second are the major source of radioactivity in the room (and outside) if the unclad element were to be under water. SCALE 6.1 uses the ORIGEN-S code for isotope generation and depletion calculations. Details of the calculation methodologies and validation references are contained in the original GA report cited. The fuel composition used was based on the number densities calculated [2] earlier for this core. The calculation showed the reactor turning subcritical within about 100 EFPD's mostly due to build-up of ^{135}Xe, but this was ignored to determine the "saturation" fission product inventory, which occurred at about 125 EFPD. Exceptions are the very long-lived nuclides such as ^{85}Kr and ^{129}I, and short-lived activation daughter products which are separately identified (by *) in Table 13-1.

[1] GA 39364R00001, "ORIGEN and Dose Calculations for the UCI TRIGA Reactor", Revision A., September 2011.
[2] GA 911196, "Nuclear Analysis of the University of California TRIGA Reactor", Rev 1 July 2011.

TABLE 13-1

GASEOUS FISSION PRODUCTS IN SINGLE UCI TRIGA MAXIMUM POWER FUEL ELEMENT

	Nuclide	Half-life	Inventory (curies)
Group I			
	Br 82	35.28 h	[redacted]
	Br 82m*	6.13 m	[redacted]
	Br 83	2.4 h	[redacted]
	Br 84	31.8 m	[redacted]
	Br 84m	6. m	[redacted]
	Br 85	2.9 m	[redacted]
	Br 86	55.1 s	[redacted]
	Br 87	55.65 s	[redacted]
	I-128*	24.99 m	[redacted]
	I 129*	1.57×10^7 y	[redacted]
	I-130*	12.36 h	[redacted]
	I-130m*	8.84 m	[redacted]
	I 131	8.02 d	[redacted]
	I-132	2.295 h	[redacted]
	I 133	20.8 h	[redacted]
	I 134	52.5 m	[redacted]
	I-134m	3.52 m	[redacted]
	I 135	6.57 h	[redacted]
	I 136	83.4 s	[redacted]
	I-136m	46.9 s	[redacted]
	I-137	24.13 s	[redacted]
		Total Iodines	[redacted]
		Total Group I	[redacted]
Group II			
	Kr 83m	1.83 h	[redacted]
	Kr 85*	10.776 y	[redacted]
	Kr 85m	4.48 h	[redacted]
	Kr 87	76.3 m	[redacted]
	Kr 88	2.84 h	[redacted]
	Kr 89	3.15 m	[redacted]
	Kr 90	32.32 s	[redacted]
	Kr 91	8.57 s	[redacted]
	Xe 131m	11.93 d	[redacted]
	Xe 133	5.25 d	[redacted]
	Xe 133m	2.19 d	[redacted]
	Xe 135	9.14 h	[redacted]
	Xe 135m	15.29 m	[redacted]
	Xe 137	3.818 m	[redacted]
	Xe 138	14.08 m	[redacted]
	Xe 139	39.68 s	[redacted]
	Xe 140	13.6 s	[redacted]
		Total Group II	[redacted]

13.2.3 Fission Product Release Fraction

NUREG-1282 quotes a release fraction of 1×10^{-5} that has been measured for 8.5% uranium TRIGA® fuel. Some indications in experiments are that the release slowly increases with fuel temperature. However, facilities have adopted a *release fraction* = 1.5×10^{-5} as an appropriate value for hypothetical use for fuel temperatures below 400°C.

Multiplying this release fraction by the inventory of gaseous fission products produced in the maximum power fuel element, as given in Table 13-1, gives the total activity that would be released should the integrity of a fuel element cladding be compromised. These values are shown in Table 13-2

Table 13-2 Releases from Maximum Power Fuel Element Unclad in Air

	Total Curies	Total Released (Curies)
Noble Gases	[redacted]	[redacted]
Iodine	[redacted]	[redacted]
Halogens	[redacted]	[redacted]

13.2.4 Pool Water Activity if Rupture is Under Water.

We review briefly the course of events if the rupture did occur in water, in which case the soluble (Group I) halogen products will remain in the water, totaling [redacted] or [redacted]

Since the volume of water in the reactor pool is 8.7×10^7 cm^3, the activity concentration is [redacted] In 24 hours the activity would decrease by more than a factor of 10 to [redacted] as all but Br-82, I-131 and I-133 nuclides have half-lives shorter than 10 hours, and the activity contribution of I-129 is negligible. The demineralizer can be used to safely concentrate these for subsequent disposal. [redacted] of activity on the resin is a readily handled quantity by trained personnel. Thus the personnel exposure consequences of such an incident are relatively minor.

13.2.5 Predicted Exposures to Personnel Inside the Reactor Room if Rupture in Air.

The cited General Atomic report[1] computed the external dose rates and thyroid TEDE rates to personnel in the reactor room as a result of the described MHA: fuel element unclad in air, using a fission product release fraction of 1.5×10^{-5}. The concentrations in Tables 13-3 and 13-4 are based on the fission product release uniformly dispersed within the 6.6×10^8 cm^3 volume of the reactor room. External dose conversion factors for submersion in air were obtained from Federal Guidance Report No. 12[3] except for Br-85, I-136, Kr-89, Kr-90, and Xe-137. Scaling factors for these radionuclides were derived from a DOE tabulation of external dose conversion factors[4].The Group I external dose rate to an individual in the reactor room is 31 mrem/hr and the Group II external dose rate is almost 19 mrem/hr for a total external dose rate of almost 51 mrem/hr. Some of the radionuclides from the ORIGEN calculation do not have external dose conversion factors. These radionuclides either have very weak gamma emissions or very short half-lives so their contribution to the total dose is negligible.

---

[3] "Federal Guidance Report No. 12: External Exposure to Radionuclides in Air, Water, and Soil," US Environmental Protection Agency, EPA-402-R-93-081, September 1993.

[4] "External Dose-Rate Conversion Factors for Calculation of Dose to the Public," US Dept. of Energy, DOE/EH-0070, July 1988.

Table 13-3 Group I Fission Product Release and External Dose

Nuclide	Inventory (Ci)	Release (Ci)	Concentration (Ci/m^3)	DCF (Sv per Bq-s/m^3)	External Dose (mrem/hr)
Br-82	[redacted]	[redacted]	[redacted]	1.30E-13	6.97E-04
Br-82m	[redacted]	[redacted]	[redacted]		
Br-83	[redacted]	[redacted]	[redacted]	3.82E-16	2.39E-03
Br-84	[redacted]	[redacted]	[redacted]	9.41E-14	1.04E+00
Br-84m	[redacted]	[redacted]	[redacted]		
Br-85	[redacted]	[redacted]	[redacted]	3.56E-15	5.70E-02
Br-86	[redacted]	[redacted]	[redacted]		
Br-87	[redacted]	[redacted]	[redacted]		
I-128	[redacted]	[redacted]	[redacted]	4.16E-15	4.77E-05
I-129	[redacted]	[redacted]	[redacted]	3.80E-16	2.92E-10
I-130	[redacted]	[redacted]	[redacted]	1.04E-13	5.23E-03
I-130m	[redacted]	[redacted]	[redacted]		
I-131	[redacted]	[redacted]	[redacted]	1.82E-14	6.06E-01
I-132	[redacted]	[redacted]	[redacted]	1.12E-13	5.66E+00
I-133	[redacted]	[redacted]	[redacted]	2.94E-14	2.27E+00
I-134	[redacted]	[redacted]	[redacted]	1.30E-13	1.18E+01
I-134m	[redacted]	[redacted]	[redacted]		
I-135	[redacted]	[redacted]	[redacted]	7.98E-14	5.80E+00
I-136	[redacted]	[redacted]	[redacted]	1.35E-13	4.17E+00
I-136m	[redacted]	[redacted]	[redacted]		
I-137	[redacted]	[redacted]	[redacted]		
Total Group I	[redacted]				31.4

Table 13-4 Group II Fission Product Release and External Dose

Nuclide	Inventory (Ci)	Release (Ci)	Concentration (Ci/m^3)	DCF (Sv per Bq-s/m^3)	External Dose (mrem/hr)
Kr-83m	[redacted]	[redacted]	[redacted]	1.50E-18	9.26E-06
Kr-85	[redacted]	[redacted]	[redacted]	1.19E-16	7.89E-05
Kr-85m	[redacted]	[redacted]	[redacted]	7.48E-15	1.15E-01
Kr-87	[redacted]	[redacted]	[redacted]	4.12E-14	1.23E+00
Kr-88	[redacted]	[redacted]	[redacted]	1.02E-13	4.11E+00
Kr-89	[redacted]	[redacted]	[redacted]	9.21E-14	4.77E+00
Kr-90	[redacted]	[redacted]	[redacted]	6.07E-14	3.38E+00
Kr-91	[redacted]	[redacted]	[redacted]		
Xe-131m	[redacted]	[redacted]	[redacted]	3.89E-16	1.52E-04
Xe-133	[redacted]	[redacted]	[redacted]	1.56E-15	1.21E-01
Xe-133m	[redacted]	[redacted]	[redacted]	1.37E-15	1.14E-03
Xe-135	[redacted]	[redacted]	[redacted]	1.19E-14	6.48E-01
Xe-135m	[redacted]	[redacted]	[redacted]	2.04E-14	1.90E-01
Xe-137	[redacted]	[redacted]	[redacted]	9.06E-15	6.42E-01
Xe-138	[redacted]	[redacted]	[redacted]	5.77E-14	4.19E+00
Xe-139	[redacted]	[redacted]	[redacted]		
Xe-140	[redacted]	[redacted]	[redacted]		
Total Group II	[redacted]				19.4

The thyroid committed dose rate and whole body committed effective dose rate from inhalation to an individual within the reactor room are presented in Table 13-5 based on the inventory and releases presented in Table 13-4. The concentration in the room is multiplied by the breathing rate of 3.47 x 10^{-4} m^3/s to give the inhaled amount of iodine. Committed effective dose equivalent (CEDE) and thyroid dose conversion factors (DCFs) for inhalation were obtained from Federal Guidance Report No. 11[5]. Some of the radionuclides from the ORIGEN calculation did not have inhalation dose conversion factors. These radionuclides have very short half-lifes such that their contribution to the total dose is negligible.

The total thyroid dose rate is 1.37 mrem/sec or 4.9 rem/hr. The CEDE whole body dose rate is 0.046 mrem/sec or 0.17 rem/hr. Adding the external dose rate of 51 mrem/hr (from previous page) gives a Total Effective Dose Equivalent (TEDE) of 0.22 rem/hr.

Table 13-5 Thyroid and CEDE Dose Rates from Reactor Room Inhalation

Nuclide	Reactor Room Concentration (Ci/m^3)	Thryoid DCF (Sv per Bq)	Thyroid Dose (mrem/sec)	CEDE DCF (Sv per Bq)	CEDE Dose (mrem/sec)
Br-82	[redacted]	2.38E-10	1.23E-07	4.13E-10	2.13E-07
Br-82m	[redacted]				
Br-83	[redacted]	3.29E-12	1.99E-06	2.41E-11	1.46E-05
Br-84	[redacted]	3.12E-12	3.33E-06	2.61E-11	2.79E-05
Br-84m	[redacted]				
Br-85	[redacted]				
Br-86	[redacted]				
Br-87	[redacted]				
I-128	[redacted]	5.34E-11	5.91E-08	1.28E-11	1.42E-08
I-129	[redacted]	1.56E-06	1.16E-07	4.69E-08	3.48E-09
I-130	[redacted]	1.99E-08	9.64E-05	7.14E-10	3.46E-06
I-130m	[redacted]				
I-131	[redacted]	2.92E-07	9.37E-01	8.89E-09	2.85E-02
I-132	[redacted]	1.74E-09	8.48E-03	1.03E-10	5.02E-04
I-133	[redacted]	4.86E-08	3.62E-01	1.58E-09	1.18E-02
I-134	[redacted]	2.88E-10	2.51E-03	3.55E-10	3.10E-03
I-134m	[redacted]				
I-135	[redacted]	8.46E-09	5.92E-02	3.32E-10	2.33E-03
I-136	[redacted]				
I-136m	[redacted]				
I-137	[redacted]				
Total			1.37		0.046

[5] "Federal Guidance Report No. 11: Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," US Environmental Protection Agency, EPA-520/1-88-020, September 1988.

On detection of release of radioactive fission products into the reactor room, the continuous air monitor will close the normal exhaust and start the emergency purge. The room air will then be exhausted through a filter at 240 cfm (0.113 m^3/sec) which will not remove the noble gases and may not remove all the iodine isotopes released. However, credit may also be taken for plate-out on surfaces of the iodine isotopes. In Oregon State University's SAR, they adopted a method which suggests halogen release plates out to reduce the airborne concentrations to less than 25% of the initial release.
With this assumption, the revised dose rate expected to the thyroid in the unlikely event of a fuel element cladding rupture in air, is 0.25 x 4.9 rem/hr = 1.2 rem/hr and a revised TEDE dose rate of 0.25 x 0.17 rem/hr + 0.051 rem/hr = 0.093 rem/hr. If the water in the pool is assumed to be present there is a high likelihood that this will rapidly absorb the majority of halogens from the air and reduce the source for thyroid exposure to an individual in the facility to even less than 25% of that released.

Given a 5 minute evacuation time for personnel from the facility, the revised maximum estimate is 0.1 rem to the thyroid and 0.008 rem TEDE. Without plate-out credit, the value is 0.4 rem thyroid and 0.018 rem TEDE.

Given that 50 rem to the thyroid is the limiting annual exposure[6] (or 30 rem recommended in NUREG 1537) a person will have a much longer time than 5 minutes to evacuate. The TEDE compared to 5 rem (NURGE 1537) is not limiting. These are very conservative estimate since in addition to the comments made above that would reduce exposure, 1000 hour operation at full power prior to the accident was assumed as was instant cladding rupture. Also no credit was taken for operation of the room emergency purge exhaust, which will also act to reduce the dose received.

13.2.6 Predicted Exposure to Persons Outside the Building

The ventilation system discharges the emergency purge exhaust into the main building exhaust which has three exhaust fans designed to create a flow of 3 x 80,000 cfm, further diluted to 3 x 128,000 cfm at a height of 120 feet above the building roof by means of Axijet® structures. From that point, natural dispersion, aided by the plume jet, will further dilute the effluents.

From the above, it is reasonable to assume a conservative dilution of the exhaust effluent by a factor of 384,000/240 = 1600. Using this, and assuming no climatic or other dilution outside the building exhaust flow, the maximum dose to an individual may be calculated.

The calculation assumes:

(1) Complete mixing in the reactor room at all times

(2) The person is immersed in the effluent from the building exhaust flow in which the concentration of radioactivity X_t at any time t is equal to $C_t/1600$ where C_t is the concentration in the exhaust and in the room and C_o is the initial concentration in the room.

(3) Inhaled activity $A_t = X_t$ multiplied by the breathing rate, R, of 3.47 x 10^{-4} m^3/s.

(4) Activity is removed from the room by leakage ($\lambda_L = 0.113/660$ sec $^{-1}$) and radioactive decay, λ_D.

The integrated concentration of radionuclides that an individual is exposed to up to time t is

$$X_t = \frac{C_o}{1600(\lambda_L + \lambda_D)}\left[1 - \exp(-(\lambda_L + \lambda_D)t)\right]$$

6 10CFR 20.1201(1)(i)

Similarly, the inhaled activity up to time t is

$$A_t = \frac{RC_o}{1600(\lambda_L + \lambda_D)}[1 - \exp(-(\lambda_L + \lambda_D)t)]$$

For infinite exposure, the quantity in the square brackets equals one. The Group I and Group II external doses to an individual exposed to the rooftop exhaust for infinite exposure are presented in Tables 13-6 and 13-7. The thyroid and CEDE whole body inhalation doses for infinite exposure are presented in Table 13-8. The TEDE dose is the sum of the external and CEDE dose which equals 0.18 mrem. No credit is taken for plate-out, filtering or downwind dispersion.

Atmospheric dispersion is commonly expressed in terms of the χ/Q ratio which is the ratio of concentration (Ci/m^3) over source rate (Ci/sec). At the rooftop exhaust, the concentration is

$$\chi = \frac{C_o}{1600}$$

The source rate is

$$Q = 0.113\left[m^3/_{sec}\right] \times C_o$$

The effective dispersion at the rooftop exhaust is 5.531×10^{-3} sec/m^3. Atmospheric dispersion is calculated using the following equation

$$\frac{X}{Q} = \frac{1}{\pi \sigma_y \sigma_z u}$$

where u is the wind speed in m/sec, and σ_y and σ_z are the standard deviations of the Gaussian plume in the y and z directions. The Gaussian plume dispersion parameters as a function of downwind distance x in meters are calculated using the following relationships from the MACCS computer code description in NUREG/CR-4691, Vol. 2[7].

$$\sigma_y = a\,x^b$$

$$\sigma_z = c\,x^d$$

Using conservative weather conditions of Stability Class F, the Gaussian plume parameters are: a=0.0722, b=0.9031, c=0.2, and d=0.6020. Assuming a conservative wind speed of 1 m/sec (2.2 mph), a virtual source distance of 1129 m is calculated using the methodology of the MACCS computer code. The associated atmospheric dispersion parameters are σ_y=41.3 m and σ_z=13.8 m. This virtual source distance is the distance from a point source which reproduces the effective atmospheric dispersion at the rooftop exhaust. The nearest residence to Rowland Hall is Campus Village Housing which is at least 200 m to the west. The atmospheric dispersion parameters at a distance of 1329 m are σ_y=47.8 m and σ_z=15.2 m which gives an atmospheric dispersion of 4.381×10^{-4} sec/m^3. This atmospheric dispersion results in a further dilution of the rooftop exhaust by a factor of 12.6. The resulting TEDE dose for infinite time exposure is 0.014 mrem.

[7] "MELCOR Accident Consequence Code System (MACCS) – Model Description," U.S., Nuclear Regulatory Commission, NUREG/CR-4691, Vol. 2, February 1990.

Table 13-6 Group I External Dose From Rooftop Exhaust

Nuclide	Integrated Concentration ($Ci\text{-}s/m^3$)	DCF (Sv per $Bq\text{-}s/m^3$)	External Dose (mrem)
Br-82	[redacted]	1.30E-13	6.85E-07
Br-82m	[redacted]		
Br-83	[redacted]	3.82E-16	1.65E-06
Br-84	[redacted]	9.41E-14	3.39E-04
Br-84m	[redacted]		
Br-85	[redacted]	3.56E-15	2.38E-06
Br-86	[redacted]		
Br-87	[redacted]		
I-128	[redacted]	4.16E-15	1.31E-08
I-129	[redacted]	3.80E-16	2.96E-13
I-130	[redacted]	1.04E-13	4.86E-06
I-130m	[redacted]		
I-131	[redacted]	1.82E-14	6.11E-04
I-132	[redacted]	1.12E-13	3.85E-03
I-133	[redacted]	2.94E-14	2.18E-03
I-134	[redacted]	1.30E-13	5.22E-03
I-134m	[redacted]		
I-135	[redacted]	7.98E-14	5.02E-03
I-136	[redacted]	1.35E-13	8.53E-05
I-136m	[redacted]		
I-137	[redacted]		
Total Group I			0.017

Table 13-7 Group II External Dose From Rooftop Exhaust

Nuclide	Integrated Concentration ($Ci\text{-}s/m^3$)	DCF (Sv per $Bq\text{-}s/m^3$)	External Dose (mrem)
Kr-83m	[redacted]	1.50E-18	5.82E-09
Kr-85	[redacted]	1.19E-16	8.00E-08
Kr-85m	[redacted]	7.48E-15	9.36E-05
Kr-87	[redacted]	4.12E-14	6.62E-04
Kr-88	[redacted]	1.02E-13	2.98E-03
Kr-89	[redacted]	9.21E-14	2.16E-04
Kr-90	[redacted]	6.07E-14	2.72E-05
Kr-91	[redacted]		
Xe-131m	[redacted]	3.89E-16	1.53E-07
Xe-133	[redacted]	1.56E-15	1.22E-04
Xe-133m	[redacted]	1.37E-15	1.13E-06
Xe-135	[redacted]	1.19E-14	5.85E-04
Xe-135m	[redacted]	2.04E-14	3.55E-05
Xe-137	[redacted]	9.06E-15	3.49E-05
Xe-138	[redacted]	5.77E-14	7.34E-04
Xe-139	[redacted]		
Xe-140	[redacted]		
Total Group II			0.005

Table 13-8 Thyroid and CEDE Inhalation Doses from Rooftop Exhaust

Nuclide	Total Activity Inhaled (Ci)	Thyroid DCF (Sv per Bq)	Thyroid Dose (mrem)	CEDE DCF (Sv per Bq)	CEDE Dose (mrem)
Br-82	[redacted]	2.38E-10	4.35E-07	4.13E-10	7.55E-07
Br-82m					
Br-83		3.29E-12	4.94E-06	2.41E-11	3.62E-05
Br-84		3.12E-12	3.90E-06	2.61E-11	3.26E-05
Br-84m					
Br-85					
Br-86					
Br-87					
I-128		5.34E-11	5.83E-08	1.28E-11	1.40E-08
I-129		1.56E-06	4.22E-07	4.69E-08	1.27E-08
I-130		1.99E-08	3.23E-04	7.14E-10	1.16E-05
I-130m					
I-131		2.92E-07	3.40E+00	8.89E-09	1.04E-01
I-132		1.74E-09	2.08E-02	1.03E-10	1.23E-03
I-133		4.86E-08	1.25E+00	1.58E-09	4.07E-02
I-134		2.88E-10	4.01E-03	3.55E-10	4.95E-03
I-134m					
I-135		8.46E-09	1.85E-01	3.32E-10	7.25E-03
I-136					
I-136m					
I-137					
Total			4.86		0.158

The high variability of local climatic conditions actually prevalent when an accident occurs make it relatively ineffective to try to predict accident exposures with any greater degree of certainty. It seems very likely that the present predictions will be conservative, and that accidents of this hypothetically very serious nature at this facility will not result in unacceptably hazardous exposures either to personnel in the facility, or to the general public. Since the probability of such an event could increase during fuel handling operations, in order to further guard against the likelihood of public exposures, operational regulations at the reactor preclude the presence of any inessential or untrained personnel within the facility whenever operations such as fuel handling are in progress. Events involving handling of irradiated fuel elements in air are subject to even greater restrictions and personnel training, and include procedures to exclude the public from all areas adjacent to the facility. This was the practice followed during the transfer of one cask load of irradiated fuel elements into the facility in 1974. The protocol for that procedure is on file for further use and inspection.

13.2.7 Predicted Exposure to Rowland Hall Occupants Adjacent to the Facility

The calculations presented in section 13.2.5 can be applied to assessment of potential exposure in an MHA to individuals located in adjacent areas outside the reactor facility.

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

Section 13.2.5 established a maximum dose rate to an individual within the facility of about 50 mrem/hr. Since this is an immersion calculation, it is reasonable to assume a distance factor before estimating the dose rate through a wall. The closest a person can stand outside the facility is in the adjacent hallway at the "window" wall. Assuming a whole body distance on each side of the wall, and including the wall thickness, it is reasonable to credit a distance factor of 5 feet. Making a further assumption that the source is an effective point source inside the facility, this could provide a distance factor of 5^2 or 25. Thus the external dose rate estimate is reduced by 50/25 or to 2 mrem/hr. making no allowances for attenuation by air or wall materials, This is at the acceptable rate[8] for public individuals over short time periods. In such an accident, the building will be cleared within less than 1 hour of the accident so that extensive exposure at this rate (which would need to exceed 25 hours to attain the 50 mrem annual limit[9]) would not be realized. Since there is no internal exposure, this exposure is the TEDE estimate. As it is so low, even if there are large uncertainties in the estimate, this appears to be entirely acceptable. Anyone at a further distance, such as in nearby laboratories or offices, would receive less as a result of additional distance factors and absorption reductions. It is important to emphasize that the building ventilation design is such that no mechanism exists for mixing of air between the reactor area and adjacent rooms, so no internal exposure can result from the MHA except to persons in the facility.

[8] 10CFR20.1302(2)(ii)
[9] 10CFR20.1302(2)(ii)

Proposed
Technical Specifications
for the
U. C. Irvine
TRIGA Mark I Nuclear Reactor

Revision 2011-06
To conform to NRC requirements

Final: September 2011

1. DEFINITIONS

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

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

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

1.3 CHANNEL CALIBRATION 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 include equipment actuation, alarm or trip, and shall be deemed to include a CHANNEL TEST.

1.4 CHANNEL CHECK 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.

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

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

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

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

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

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

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

d. Fast Transient (FTR): a rod with scram capabilities that can be rapidly ejected from the reactor core to produce a pulse. It shall have a void follower.

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

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

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

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

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

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

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

1. FUEL ELEMENT is a single TRIGA® fuel rod.

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

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

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

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

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

1.19 OPERATIONAL CORE means a CORE CONFIGURATION that meets all license requirements, including Technical Specifications.

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

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

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

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

1.24 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) The reactor is shutdown and 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) No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods;
(d) No experiments are being moved or serviced that have a reactivity worth exceeding the maximum value allowed for a single experiment, or $1.00.

1.25 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 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 the maximum value allowed for a single experiment, or $1.00.

1.26 REFERENCE CORE CONDITION is when the core is at ambient temperature (cold) and the reactivity worth of xenon is negligible (less than $0.30).

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

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

1.29 SAFETY CHANNEL means a measuring channel in the reactor safety system.

1.30 SCRAM TIME is the elapsed time between the initiation of a scram signal and a specified movement of a control or safety device.

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

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

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

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

1.35 SUBSTANTIVE CHANGES are changes in the original intent or safety significance of an action or event.

1.36 SURVEILLANCE INTERVALS that are permitted are established as follows:

a. quinquennial – interval not to exceed 6 years
b. biennial – interval not to exceed 2-1/2 years
c. annual – interval not to exceed 15 months
d. semi-annual – interval not to exceed 7-1/2 months
e. quarterly – interval not to exceed 4 months
f. monthly - interval not to exceed 6 weeks
g.daily – refers to each day when the reactor is to be operated or before any operation extending more than one day

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

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limit - Fuel Element Temperature

Applicability. This specification applies to the fuel element temperature.

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

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

Basis. The important parameter for a TRIGA® reactor is the fuel element temperature, since it can be measured. The loss in the integrity of the fuel element cladding could arise from an excessive build-up of pressure in the fuel element. The safety limit for high hydride TRIGA® fuel is based on data including the experimental evidence obtained during high performanc reactor tests of this fuel. These data indicate that the stress will remain below the ultimate stress provided the fuel temperature does not exceed 1150°C and the fuel cladding is water cooled.
The safety limit for the stainless steel clad, high hydride ($Zr/H_{1.7}$) fuel element is based on analysis (McClellan Nuclear Research Center reactor SAR 4.5.4.1.3 and Oregon State University SAR 4.5.3.1) which also indicates that the stress in the cladding due to the hydrogen pressure from the dissociation of the zirconium hydride will remain below the yield stress provided the temperature of the fuel does not exceed 1150°C and the fuel cladding is water cooled.

2.2 Limiting Safety System Settings

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

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

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

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

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

3. LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Core Parameters

3.1.1 Steady-state Operation

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

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

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

Basis. Calculations have been performed which show that for operation at 250 kW, the maximum fuel temperature is 253 °C and the minimum DNB ratio is greater than 7.27. In addition, experience at other TRIGA® reactors and thermal and hydraulic calculations (OSU SAR 4.5.3) indicates that power levels up to 1.9 Mw can be safely used with natural convection cooling of the fuel elements, in a similar circular grid plate core configuration.

3.1.2 Shutdown Margin

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

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

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

The shutdown margin provided by the control rods shall be greater than $0.55 with:

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

b. the most reactive control rod fully withdrawn; and

c. The reactor in the reference core condition.

Basis. The value of the shutdown margin and limits on experiments assure that the reactor can be shut down from any operating condition even if the most reactive control rod should remain in the fully-withdrawn position. Assuming a SHIM rod worth of $3.40, and a core excess reactivity value of $3.00, the maximum permitted, this means that the three remaining control rods must provide a negative worth of at least $3.55, for a minimum total control rod worth permitted of $6.95 ($3.40 + $3.55). The total control rod worth is projected to be above $8.00 at this facility for some considerable time.

3.1.3 Core Excess Reactivity

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

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

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

Basis. An excess reactivity limit of \$3.00 allows for flexibility in operating the reactor in steady state mode while limiting the reactivity addition for pulse operation. Computations presented in the SAR (Chapter 13.3) establish that a sudden insertion of \$3.00 results in a fuel temperature of approximately 350°C, well below the established safety limit for this fuel (TS 2.1). Such calculations are conservative, being based on a purely adiabatic model. The specifications assure that no insertion of reactivity above this value shall be possible, even under non-normal operating conditions. The only possible activity that could result in requiring fuel movement to attain this condition would be the unlikely addition of an experiment with a high positive reactivity worth.

3.1.4 Pulse Operation

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

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

Specifications. The reactor shall not be operated in the pulse mode unless, in addition to the other requirements of Section 3.1,

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

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

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

3.1.5 This section intentionally left blank.

3.1.6 Fuel Element Inspection Parameters

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

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

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

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

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

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

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

Basis. These criteria have been successfully used for hundreds of fuel inspections over many years in similar reactors to successfully identify elements that have cladding issues prior to serious failure.

3.1.7 Core Configuration

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

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

Specifications.

a. The core shall be an arrangement of TRIGA 8.5/20 LEU fuel.

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

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

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

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

Bases.

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

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

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

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

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

3.2 Reactor Control and Safety Systems

3.2.1 Control Rods

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

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

Specifications. The reactor shall not be operated unless the control rods are operable. Control rods shall not be considered operable if:

a. damage is apparent to rods or drive assemblies that could affect operation; or

b. the scram time for the control rods is greater than 1 second for 90% reactivity insertion.

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

Basis. Experience has shown that rod movement is assured in the absence of damage and that scram times of less than 1 second are more than adequate to reduce reactivity and fuel temperatures rapidly to assure safety in view of known transient behavior of TRIGA® reactors. The total worth of the two transient rods is limited so as to restrict the pulse size.

3.2.2 Reactor Measuring Channels

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

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

Specifications. The reactor shall not be operated in the specified mode unless the measuring channels described in Table 1 are operable.

Table 1. Minimum Measuring Channels

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

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

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

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

3.2.3 Reactor Safety System

Applicability This specification applies to the reactor safety system channels.

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

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

Table 2. Minimum Reactor Safety Channels

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

Table 3. Minimum Interlocks

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

Bases

Scrams. The fuel temperature scram provides the protection to assure that if a condition results in which the LSSS is approached, an immediate shutdown will occur to keep the fuel temperature well below the safety limit. The power level scrams are provided as added protection against abnormally high fuel temperature and to assure that reactor operation stays within the licensed limits. The manual scram allows the operator to shut down the system if an unsafe or abnormal condition occurs. A high voltage scram on each channel assures that detector response is operating at all times. The seismic switch will scram the reactor if earth movement in any dimension exceeds 3%g (0.03g) in case the operator is prevented from operating the manual scram at the time. This level corresponds to movement noticeable by most persons, but (by MM scale) results in no damage to structures. The preset timer scram provides pulse "clipping" to reduce energy production at the tail of a pulse.

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

3.3 Coolant Systems

3.3.1 Pool Water Level

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

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

Specifications.

a. The reactor shall not be operated unless the pool water level is at no more than 1 foot below the tank edge (at least 15 feet above the upper grid plate of the core and at least 24 feet above the tank floor).
b. An audible alarm, with reporting to the UCIPD dispatch desk if not locally silenced by an operator, shall operate 24/7 to alert personnel if the water level in the reactor pool falls below the above limit. Visual checking of water level may be substituted during periods when the alarm is found to be inoperable.
c. Records shall be maintained of the date, time and quantity of all make up water added to the pool.

Basis. Facility design calculations and subsequent measurements show that these water levels are sufficient to reduce full power operational radiation levels to acceptable levels within the facility and in any occupied areas above or surrounding the reactor. This is also true for shut down levels. The alarm will notify appropriate responders well before any increase in radiation levels to the surroundings occurs. The alarm and the operational make up water records will, if it occurs unusually frequently, alert operators to the possibility that pool leakage might be occurring, . The pool level is normally maintained at approximately 10 inches below the tank edge. Thus the alarm level is at 2 inches below and results from evaporation or leakage of only 160 gallons (640 liters, or 0.6% of total pool water).

3.3.2 Pool Water Temperature

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

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

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

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

3.3.3 Pool Water Conductivity

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

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

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

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

3.3.4 Pool Water pH

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

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

Specification. The pool water pH level shall be maintained between 5.5 and 7.5.

Basis. While no credible mechanism exists in the pool for pH to be out of this range, ANSI 15.1 and NUREG 1537 recommend such limits.

3.3.5 Pool Water Radioactivity

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

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

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

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

3.4 This section intentionally left blank

3.5 Ventilation Systems

3.5.1 Ventilation System.

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

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

Specification.

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

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

2. a minimum total exhaust flow rate from the reactor area of 4000 cfm is present.

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

b. The reactor shall not be operated unless it is verified that the ventilation system goes into the emergency mode upon manual actuation or a signal of high radiation activity from a continuous particulate air monitor (CAM) measuring air from above the pool as described in TS 3.5.2. Verification shall be by observing the emergency flow rate is at least 240 cfm, the absence of regular exhaust flow, and the pressure differential reading between the the reactor area and the outside is negative.

Basis. Through a combination of inflow dampers and outflow exhaust, facility design establishes and exceeds these pressure differentials and flows. The differentisal pressure assists in confinement of radioactive materials. The SAR establishes that normal operation effectively dilutes ^{41}Ar levels below 10 CFR20 limits and as detailed in facility annual reports. An automatic emergency mode with a small filtered purge exhaust is provided to limit release of radioactivity to the environment. Operation of the normal system adequately dilutes the argon 41 released even under unusual experimental operations. The two hour exemption should not dimish the effectiveness of the CAM in detecting any release of radioactivity. The requirement not to pulse while the ventilation system is undergoing repair reduces the likelihood of fuel element failure during such times.

3.5.2 Ventilation During Emergency Situations

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

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

Specification. A signal of high radiation activity alarm from a continuous particulate air monitor (CAM) measuring air from above the pool or manual operation from the control room shall carry out the following functions:

a. close off inflow air by closing dampers; and

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

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

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

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

3.6 Emergency Power

Applicability. This specification applies to the availability of emergency power.

Objective. To assure certain information related to personnel safety is available in the event of main electrical power failure.

Specification. Emergency electrical power, activated rapidly upon main electrical power failure, shall be provided to facility lighting, radiation monitoring and security monitoring systems.

Basis. Provision of power to these systems will assure that personnel present at the time, or responding to an event, will have information to assist in monitoring their safety and the safety and security of the facility.

3.7 Radiation Monitoring Systems and Effluents

3.7.1 Radiation Monitoring Systems

Applicability. This specification applies to monitoring of radiation levels.

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

Specifications.

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

Radiation Area Monitors (RAM):	2
Continuous Particulate Radiation Monitor (CAM):	1

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

Basis. These instruments and dosimeters will provide adequate notification of abnormal levels that could result in exposures or uncontrolled releases. The environmental dosimeters provide information that can be used to track long term trends that might need attention.

3.7.2 Effluents

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

Objective. To assure that concentration of ^{41}Ar in accessible unrestricted areas shall be below the applicable limits of 10 CFR Part 20.

Specification.

a. The annual average concentration of ^{41}Ar released to the environment shall not exceed 1×10^{-8} µCi/mL.
b. The quantity of radioactivity in liquid effluents released from the facility to the environment shall not exceed the limits of 10CFR Appendix B. Table 3.

Basis.
a. The building exhaust and room ventilation system provides a dilution factor of 100 reducing the facility room concentration predicted from calculations and measurements to be created as a result of an indefinite period of operation to well below 10CFR Part 20 Appendix B, Table 2 requirements (1×10^{-8} µCi/mL) when released. The exposure risk to the public is reduced since the discharge plume is at a high level above the roof so that no person would be submersed in the cloud before further atmosphereic dilution occurred. Annual reports from this facility have shown that levels released have been well below this value.
b. This specification establishes assurance that any release of radioactive materials contained in liquids released to the environment does not exceed acceptable limits proscribed in regulations. Past operations at this facility have not been curtailed or otherwise affected by the need to stay within such limits.

3.8. Limitations on Experiments

3.8.1 Reactivity Limits

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

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

Specifications.

a. The absolute value of any unsecured experiment shall not exceed $1.00, and
b. The reactivity worth of an individual experiment shall not exceed $3.00, and
c. The sum of absolute values of all experiments shall not exceed $3.00.

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

3.8.2 Materials

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

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

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

a. Fueled experiments shall be limited such that the total inventory of iodine isotopes 131 through 135 in the experiment is not greater than 0.3 curies and the Strontium 90 inventory is not greater than 1 microcurie; and

b. Explosive materials shall not be irradiated in quantities greater than 25 milligrams of TNT equivalent. Explosive materials in lesser quantities may be irradiated provided that the pressure produced upon accidental detonation of the explosive has been calculated and/or experimentally determined to be less than half the design pressure of the container; and

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

Basis. It is shown in the SAR, Chapter 13, that a release of 0.016 curies of iodine activity will result in a maximum dose to the thyroid of a person in the facility who evacuates in 5 minutes is 0.4

rem, of less than 1/75th of the recommended (NUREG 1537) TEDE limit of of 30 rem to the thyroid. An individual exposed to exhaust in an unrestricted area would receive less than 0.021mrem.. The resulting projected whole body exposure to personnel will be 0.018 rem, 1/275th of the 5 rem limit in NUREG 1537. The limit on iodine inventory in an experiment, assuming 100% release, is thus conservatively set at 20 times the value (of 0.016 Curies). The limit for Strontium 90 is that which corresponds to the iodine yield of 0.3 curies for a given number of fission events and would be no significant hazard. Specifications b., and c. reduce the likelihood of damage to reactor components resulting from experiment failure, and use information from NRC Reg. Guide 2.2.

3.8.3 Failures or Malfunctions

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

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

Specifications. Where the possibility exists that the failure of an experiment under normal operating conditions of the experiment or reactor, credible accident conditions in the reactor, or possible accident conditions in the experiment could release radioactive gases or aerosols to the reactor facility or any unrestricted area, the quantity and type of material in the experiment shall be limited such that the airborne radioactivity in the reactor facility or the unrestricted area will not exceed the applicable dose limits in 10CFR 20. In calculating such a limit, it shall be assumed that 100% of the gases or aerosols escape from the experiment, unless a specific effective experiment design is in place for trapping such effluents, in which case at least 10% of the gases or aerosols shall be assumed to escape.

Basis. This specification is intended to assist experiment review and design in meeting the goals of 10CFR20 by reducing the likelihood of excessive facility personnel or public exposure by gases or aerosols as a result of experiment failure.

3.9. This section intentionally left blank.

4. SURVEILLANCE REQUIREMENTS

4.0 General

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

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

Specifications.

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

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

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

4.1 Reactor Core Parameters

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

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

Specifications.

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

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

c. The shutdown margin shall be determined at each day's shutdown, or at the end of any operation exceeding one day, or following any significant change (>$0.25) in core configuration.

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

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

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

4.2 Reactor Control and Safety Systems

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

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

Specifications.

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

b. Control rod drop (scram) times for all four control rods shall be determined annually.

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

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

e. On each day that pulse mode operation of the reactor is planned, a functional performance check of the transient (pulse) rod system shall be performed.

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

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

h. A channel check of the pool water temperature measuring channel shall be performed prior to each day's operation or prior to each operation lasting more than one day.

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

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

4.3 Reactor Pool Water

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

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

Specifications.

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

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

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

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

e. The pool water pH value shall be measured quarterly.

f. The pool water radioactivity shall be measured quarterly.

g. The pool water loss rate shall be evaluated on each occasion when make-up water is added to the pool. Any unusually increased loss rate shall be investigated as a possible pool leak before any further reactor operation,

h. If there is any indication of fuel element leakage of fission products, of pool water leakage from the tank, or leakage from the secondary cooling system into pool water, all pool water quality and quantity measurements shall be re-measured immediately and repeated at least weekly until the absence of any problem is confirmed. When the reactor has not been operated for periods greater than 1 month, all pool water measurements listed above shall be made or checked before reactor operation is resumed.

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

4.4 This section intentionally left blank.

4.5 Ventilation Systems

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

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

Specifications.

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

b. A check of the exhaust flow rates from the reactor area in both normal and emergency modes shall be performed daily.

c. It shall be verified biennually that the ventilation system is operable in both normal and emergency modes by verification of flow rates, valve/damper action and filter.

d. A channel test of the function of the particulate high radiation (CAM) alarm and the control room manual switch to properly set the ventilation system into emergency mode shall be performed daily.

Basis. Based on experience, these surveillances will assure that the ventilation system is functioning as specified. (Section 3. 5).

4.6 Emergency Power

Applicability. This specification applies to the provision of emergency electrical power to room lighting, radiological safety, and security instrumentation.

Objective. To assure proper connection and function of the emergency electrical power so that personnel are provided lighting and information relating to radiological safety in the event of main electrical power failure.

Specification. It shall be determined annually that the radiological safety instruments required by Section 3.7.1.a. are attached to the correct circuit for emergency electric power provision. It shall be determined annually that the emergency power generator has been successfully tested for operation and automatic load transfer.

Basis. It is important for safety that verification of emergency power functions be carried out. Past experience has shown that this frequency is adequate to assure continuity of this service.

4.7 Radiation Monitoring System and Effluents

Applicability. This specification applies to the surveillance requirements for the radiation monitoring instrumentation required by Section 3.7.1.a of these specifications and the effluent releases specified by section 3.7.2.

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

Specifications.

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

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

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

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

e. Any liquid effluents to be released to the environment from the facility shall be analyzed for radioactive content prior to release.

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

4.8 Experiment Limits

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

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

Specifications.

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

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

Basis. Past experience has shown that adherence to requirementsdescribed in Sections 3.8 and 6.5 are adequate to assure safe experimentation at this facility.

5.0 DESIGN FEATURES

5.1 Site and Facility Description

Specifications

The site shall be the reactor facility as described below.

The reactor facility shall be a restricted access area consisting of a main area two associated laboraty areas, and a control room on a single level in the basement of Rowland Hall, on the University of California Irvine campus. The minimum free air volume of the reactor room and the two associated laboratories shall be 10,000 cubic feet. Normal entry to these areas shall be restricted to a single doorway from the control room. Large doors in the reactor room shall be provided to the adjacent loading dock to provide emergency egress and/or access for incoming or outgoing large items. Full visibility shall be provided between the control room and the reactor area.
The reactor shall be housed in a closed room designed to restrict leakage.

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

5.2. Reactor Coolant System

Specifications.

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

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

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

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

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

f. A method for pH water measurement shall be available in the reactor room.

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

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

5.3. Reactor Core and Fuel

5.3.1 Reactor Core.

Specifications.

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

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

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

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

Basis. Standard TRIGA® fuel and reactor core design has a long and successful history of use.Model calculations indicate acceptable neutronic and thermionic conditions for the core design under extended use and burn-up.The Am-Be source is in a sealed capsule and has a long useful life.

5.3.2. Control Rods.

Specifications.

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

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

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

5.3.3. Reactor Fuel.

Specifications. Standard TRIGA® fuel elements shall have the following characteristics:

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

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

hydrogen atoms to 1.0 zirconium atom.

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

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

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

5.4. Fuel Storage

Specifications.

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

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

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

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

5.5. Ventilation System

Specifications.

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

b. During normal operations, the ventilation system shall be capable of exhausting air or other gases from the reactor area at a rate of 4000 cfm.

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

d. During emergency situations involving release of radioactive materials into the air, an emergency exhaust with a HEPA filter shall be provided to exhaust a minimum of 230 ft^3/minute from the reactor area.

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

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

Basis

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

6 0 ADMINISTRATIVE CONTROLS

6.1 Organization and Structure.

6.1.1 Structure.

The reactor facility is housed in the School of Physical Sciences of the University of California, Irvine. The reactor is related to the University structure of positions shown in the organization chart, Figure 1.

6.1.2 Responsibilities.

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

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

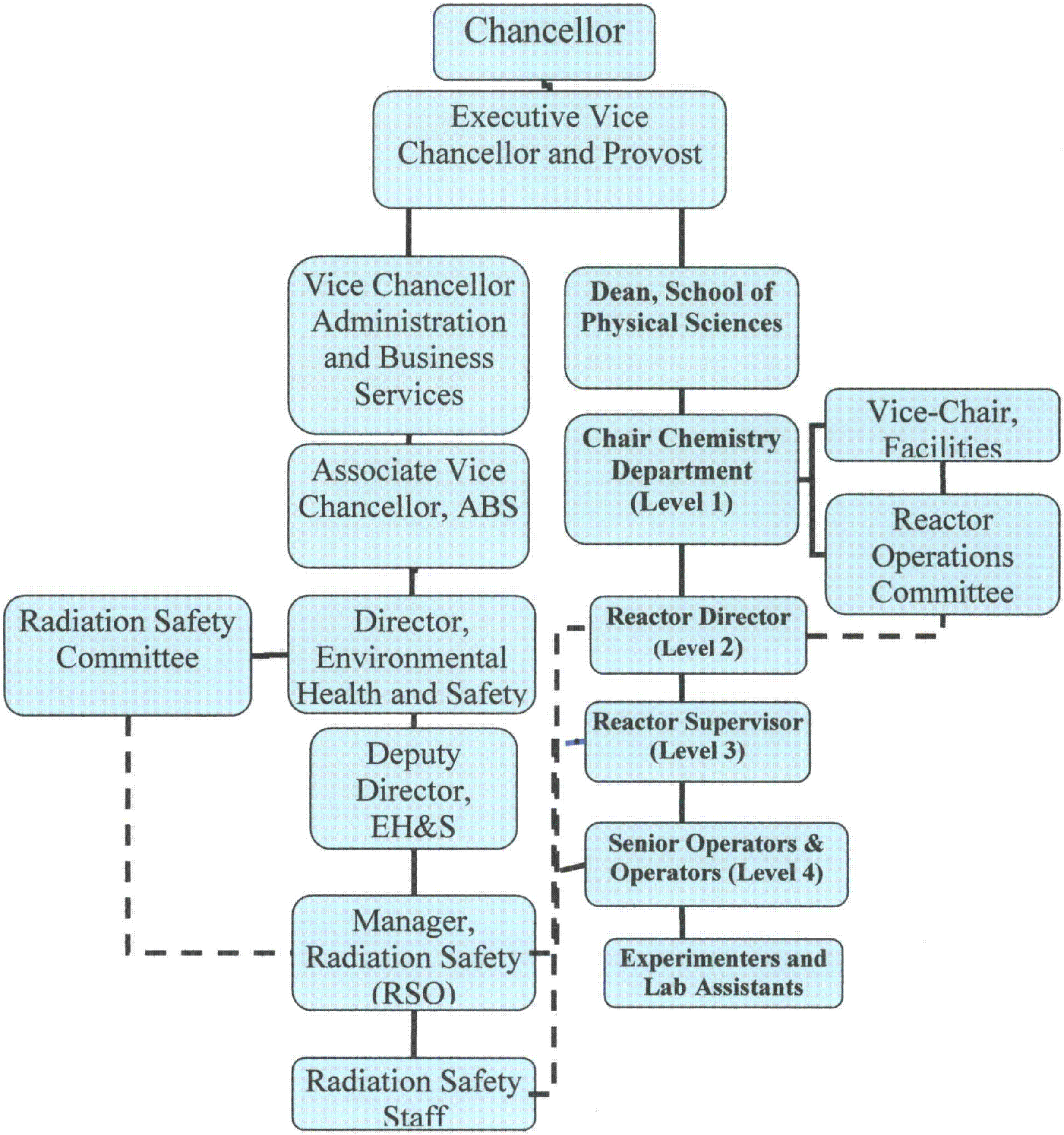
c. Operations shall be supervised by the Reactor Supervisor who shall hold a valid senior operator's license for the facility. This position shall be responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, the provisions of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

d. Reactor operators shall be responsible for operation of the reactor and performing needed maintenance and surveillance, including radiological safety and necessary supervision of experimenters. Senior reactor operators shall assume duties for supervision of operators as required by the US Nuclear Regulatory Commission in Part 55 of 10 CFR, and Section 6.1.3. of these technical specifications.

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

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

Figure 1. UCI Reactor Organization Chart



6.1.3 Staffing.

a. The minimum staffing when the reactor is not secure shall include:
1. A licensed operator with direct access to the reactor controls;
2. A second designated individual present within Rowland Hall able to carry out prescribed instructions and with the ability to check on the safety of the licensed operator and to act in the event of emergency; and
3. A licensed Senior Operator (SRO) readily available on call. Readily available on call means -
i. has been specifically designated and the designation known to the operator on duty,
ii. can be rapidly contacted by phone,by the operator on duty,
iii. is capable of getting to the reactor facility within 30 minutes under normal conditions.

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

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

d. The following events require the presence in the facility of a licensed Senior Reactor Operator:
1. initial start-up,and approach to power and final daily shutdown;
2. fuel or control-rod relocations within the core region;
3. insertion, removal, or relocation of any experiment worth more than $1.00; and
4. restart following any unplanned or unscheduled shutdown, or significant power reduction.

6.1.4 Selection and Training of Personnel.

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

6.2 Review and audit

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

6.2.1 ROC Composition and Qualifications

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

6.2.2 ROC Charter and rules

The following responsibilities constitute the charter of the ROC.

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

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

3. Review and audit of facility staff and operations as indicated in sections 6.2.3 and 6.2.4.

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

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

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

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

6.2.3 ROC Review function

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

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

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

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

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

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

6. Review of operating abnormalities that have safety significance;

7. Review of reportable occurrences listed in Sections 6.6.1, 6.6.2, or 6.7.2;

8. Review of audit reports, incuding reports from the Manager, Radiation Safety, regarding the radiation protection program.

6.2.4 ROC Audit function

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

1. Facility operations for conformance to the technical specifications and applicable license or other conditions;

2. Retraining and requalification of operators according to the Requalification Plan;

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

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

6.3 Radiation Safety

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

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

6.4 Operating Procedures

Written procedures, reviewed and approved by the ROC, shall be in effect and implemented for the following items. The procedures shall be adequate to assure the safety of the reactor but not preclude the use of independent judgment and action should the situation require such.

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

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

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

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

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

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

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

8. Shipping and/or transfer of radioactive materials.

Substantive changes to procedures shall be made only with the approval of the ROC. Temporary changes procedures that do not change their original intent may be made by the Reactor Supervisor. All such temporary changes to procedures shall be documented and subsequently reviewed by the Reactor Director and the ROC. Substantive changes affecting radiological safety shall be made only with the approval of the RSO. Temporary, minor, changes in radiological safety procedures may be made by the Reactor Supervisor, but shall be reported to the RSO as soon as possible.

6.5 Experiment Review and Approval

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

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

2. Substantive changes to existing experiments or classes shall be made only after review by the ROC and RSO or their designees. Minor changes that do not significantly alter the experiment may be approved by a senior reactor operator (SRO),and shall be submitted to the ROC for review at its next scheduled meeting.

6.6 Required Actions

6.6.1 Actions To Be Taken In Case of a Safety Limit Violation.

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

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

2. The event shall be reported immediately to the Reactor Director, the ROC chairperson, and the RSO.

3. The event shall be reported to the NRC Operations Center within 24 hours and followed by a written report sent within 14 days to the NRC Document Control Desk;

4. A report, and any applicable follow-up report, shall be prepared and reviewed by the ROC, for submission to NRC, describing:

   a. applicable circumstances leading to the violation including, where known, the cause and contributing factors;
   b. effects of the violation upon reactor facility components, systems, or structures, and on the health and safety of personnel and the public; and
   c. corrective action to prevent occurrence.

6.6.2 Actions to be taken in the event of an occurrence of the type identified in Section 6.7.2, other than a safety limit violation.

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

2. Operation shall not be resumed until authorized by the Reactor Director..

3. The occurrence shall be reported to NRC as required in Section 6.7.2 of these specifications, and reviewed by the ROC at their next meeting.

6.7 Reports

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

6.7.1. Annual Operating Report.

A routine annual report shall be submitted by the Reactor Director to NRC at the end of each 12-month period for operations for the preceding year's activities between July 1st through June 30th. The report shall include:

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

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

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

4. a tabulation of major changes in the reactor facility and procedures, and tabulations of new experiments that are significantly different from those performed previously, including a summary of safety evaluations performed to assess that they do not require prior NRC approval and are authorized by 10CFR 50.59;

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

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

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

6.7.2 Special Reports.

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

a. violation of a safety limit (fuel temperature);

b. release of radioactivity from the site above allowed limits;

c. operation with actual safety system settings for required systems less conservative than the limiting safety system settings in these specifications;

d. operation in violation of limiting conditions for operation unless prompt remedial action is taken as permitted in section 3;

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

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

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

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

2. A report within 30 days (in writing) of:

   a. permanent significant changes in facility organization; and

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

6.8 Records

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

6.8.1 Records to be retained for a period of at least 5 years or for the life of the component involved if less than 5 years.

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

2. principal maintenance activities;

3. reportable occurrences;

4. surveillance activities required by the Technical Specifications;

5. reactor facility radiation and contamination surveys;

6. experiments performed with the reactor;

7. fuel inventories, receipts and shipments;

8. approved changes in operating procedures; and

9. ROC records of meetings and audit reports.

6.8.2 Records to be retained for at least one certification cycle.

Records of retraining and requalification of licensed operators (and SRO's) shall be retained at all times the individual has duties as an operator or his or her license is renewed.

6.8.3 Records to be retained for the lifetime of the reactor facility.

Applicable annual reports containing this information may also be used as records for the following:

1. gaseous and liquid radioactive effluents released to the environs;
2. off-site environmental monitoring surveys;
3. radiation exposures for all personnel monitored; and
4. drawings of the reactor facility and safety related components.