KANSAS STATE UNIVERSITY | TRIGA Mk II Nuclear Reactor Laboratory

US Nuclear Regulatory Commission

Washington, DC 20555-0001

2 May 2017

Subject: Facility Response to 3/28/17 Request for Additional Information (Acc. # ML17038A272)

To Whom It May Concern,

On March 28th, 2017, the NRC sent a third Request for Additional Information (RAI) to the Kansas State University nuclear reactor facility (license R-88, docket 50-188) regarding a license amendment request (LAR) to add up to four 12%-loaded fuel elements to the core. The original LAR was submitted on April 9, 2012 (Acc. # ML1219A063).

The resolution of the question asked in the most recent RAI required several amendments to Chapter 4 of the facility Safety Analysis Report and to the facility Technical Specifications. These changes were reviewed and approved by the Reactor Safeguards Committee on 4/28/17 by unanimous vote, pending several minor editorial changes that have been made since the vote.

The following constitutes the facility's response to the RAI, and is organized as a numbered list of abbreviated questions in order of appearance in the RAI, followed by the facility response.

1. ... *Please provide information that validates the pool boiling model used in the SAR or alternatively provide a traditional hot channel analysis using the Bernath correlation...with core inlet conditions at the TS limit, or another correlation that has been validated against acceptable data.*

A RELAP-5 single-channel model was used to update the SAR analysis for a fuel element heated to 24 kWth in water at a pressure of 1.43 kPa, corresponding to the depth of the fuel in the reactor pool. 24 kWth is the power in a fuel element with an 85-element core operating at 1.25 MW and an element-to-average power peaking value of 1.63. The bulk water temperature used in the analysis was 49°C. This value is slightly lower than the maximum of 54°C specified in the Technical Specifications. However, based on this analysis, the TS will be revised to permit a maximum bulk water temperature of 44°C in order to avoid bulk boiling of the coolant.

The departure from nucleate boiling ratio (DNBR) reached a minimum value of slightly above 2.0. This is significantly lower than the DNBR calculated in the current version of the SAR. The reactor core grid plates contain an array of 8 mm interstitial holes meant to accommodate flux wires or other experiments. The presence of an experiment in these holes may reduce the temperature at which bulk boiling will occur; therefore an additional experimental design constraint is being proposed for the TS that forbids the insertion of tubes or flux wires in the interstitial holes at bulk water temperatures

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greater than 37°C. A drawing of the upper grid plate showing the interstitial holes is attached for reference.

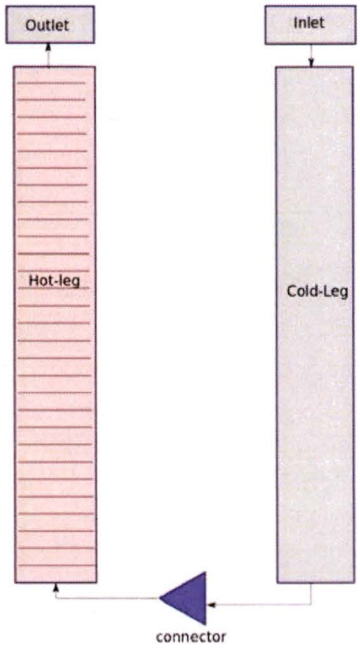

Figure 1 - RELAP single channel model

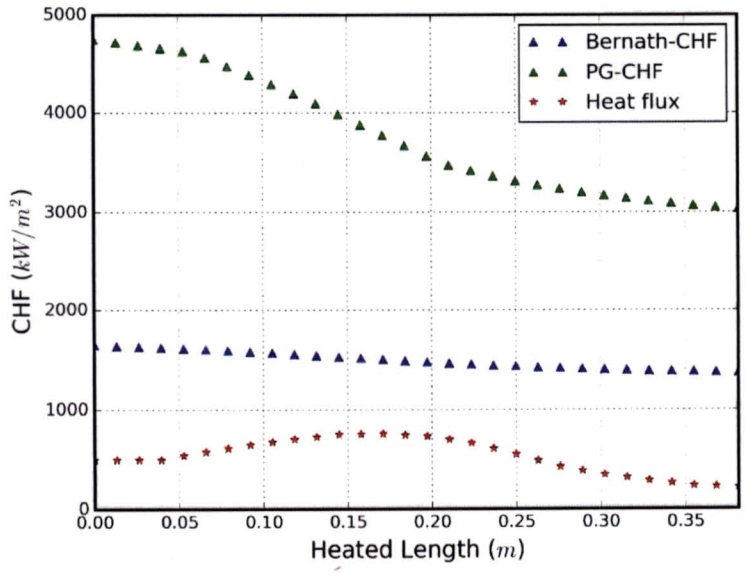

Figure 2 - Critical heat flux (CHF) versus heated length

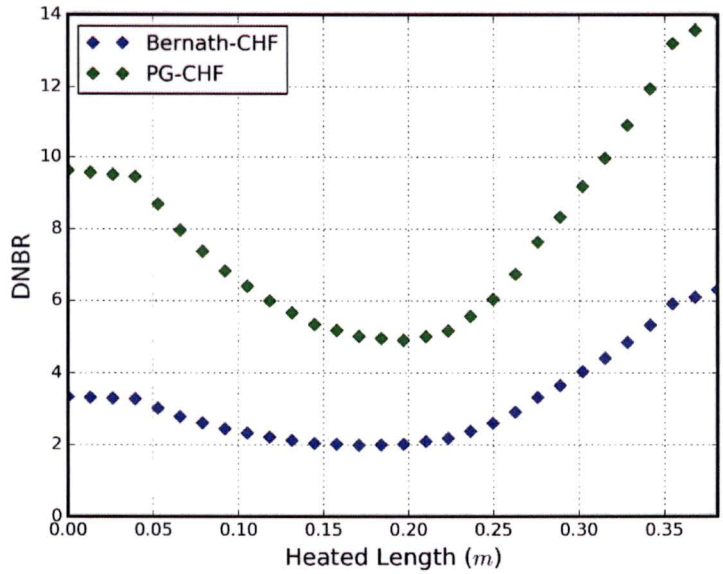

Figure 3 - Departure from nucleate boiling ratio (DNBR) versus heated length

2. *...Propose a licensed thermal power limit that is within the range of the currently installed nuclear instrumentation or describe how the nuclear instrumentation system is capable of measurement of the full range of reactor power levels anticipated as described in the safety analyses including instrument uncertainties based on the current licensed thermal power limit.*

The three nuclear instruments at KSU have maximum readings of: 1.0 MW (NLW-1000, log channel), 1.20 MW (NMP-1000, multi-range linear channel) and 1.10 MW (NPP-1000, percent power / pulse channel). Two of these are required by the Limiting Conditions of Operation during steady state mode, one is required during pulse mode (TS 3.3, attached). Therefore readings at or slightly above 1.0 MW would be readable on the console instruments, but the console instruments are incapable of reading up to the license limit of 1.25 MW. In order to address this issue, the revised Technical Specifications have been amended as follows. The Limiting Safety System Setting has been left unchanged at 1.25 MW. However, an additional Limiting Condition of Operation has been added which sets the maximum operating power to 1.00 MW, with a requirement to reduce power if it exceeds 1.05 MW. In this way the facility will still have some margin to the LSSS if, for example, it is determined that the power channels were reading low due to detector uncertainty or flux shifting following power calibration. However, the maximum power will not be allowed to exceed the useful range of the control panel instrumentation. The new LCO is found in the attached Technical Specifications, section 3.10. Note that the upgraded control console instrumentation planned for installation in January 2018 is capable of reading 1.25 MW of power.

3. *...Provide an explanation resolving the significant deviation between the reactor power observed on June 27, 2016 (735 kWt) and the maximum reactor power predicted in support of this license amendment (600 kWt) and provide a revised analysis for the maximum core operating power after the addition of the four 12.5 U wt% fuel elements.*

The reason for the deviation between the power observed on 6/27/2016 (735 kWth) and the predicted maximum power of 600 kWth in the original license amendment proposal is primarily due to the length of time involved in the LAR process. In 2012, when the LAR was originally submitted, the most recent operation at near full power had been 3/28/12, when the reactor operated at 550 kW with the three most reactive rods fully-withdrawn, and the regulating rod withdrawn to a position corresponding to only 0.04$ remaining. The reactor power is limited by core reactivity, so during the annual fuel inspections in the following years, several old fuel elements were replaced with fresh elements, resulting in an increase in the available excess reactivity. Based on the increase in power level versus when the original LAR was submitted, the maximum power level would be expected to be slightly greater than 1 MW (i.e., 1.035 MW).

4. *...Provide clarification on how the MCNP calculations are integrated with the facility measurements to determine the maximum excess reactivity and the minimum shutdown margin.*

In order to reduce the error on the estimates for maximum excess reactivity and minimum shutdown margin, MCNP was used to determine only the change in reactivity due to loading 12% fuel, not the total reactivity. Measured values of excess reactivity were taken from the daily reactivity balance, in which the calibrated integral control rod worth curves are used to determine how much excess reactivity remains above the critical configuration at zero power with no xenon. The estimated values were therefore calculated as follows:

$$\text{max. reactivity (estimated)} = \text{max. reactivity (measured)} + \Delta\rho\ (\text{MCNP})$$

$$\text{min. shutdown margin (estimated)} = \text{min. shutdown margin (measured)} - \Delta\rho\ (\text{MCNP})$$

$$\Delta\rho\ (\text{MCNP}) = (k_f - k_i) / k_i$$

5. ... *If operation with one or more control rods inoperable but fully inserted is acceptable, please provide the supporting analyses and evaluations from which operational acceptability is derived.*

The operation of the reactor core with one or more control rods inoperable but fully inserted was judged to be safe due to the understanding that the safety function of control rods is to provide a means to control reactivity, i.e., if a control rod was inoperable but inserted, then it was fulfilling its safety function. The response to the previous RAI did not consider the changes in power peaking due to the insertion of a control rod. In order to address this additional consideration, a reactor model was created in MCNP6 in which the fuel elements were all 9.0 wt%-loaded elements. The fuel elements were all modeled as having the highest allowable density of uranium, i.e, fresh fuel at 9.0 wt% U. The maximum element-to-core-average power peaking was calculated using fission heating cell tallies, and the results were compared between cases where all rods were fully withdrawn, versus having individual rods failed but fully inserted. Table 1 shows a summary of the results. The power peaking in most cases is B4, which is the fuel element located between the pulse rod channel and central thimble, such that this location is heavily moderated when the pulse rod is fully withdrawn. When the pulse (i.e., transient) rod is inoperable and fully inserted, the maximum power peaking would occur in fuel element C11. In some cases the power peaking is greater than in the all-rods-out case, but no case approached the SAR-assumed 2.00 power peaking factor used for PCT analysis.

Rod Inserted	Peak Power Factor	Element
None	1.558	B4
Regulating Rod	1.617	B4
Shim Rod	1.630	B4
Safety Rod	1.609	B4
Pulse Rod	1.438	C11
SAR	2.00	N/A

Table 1: Peak Power Factors

Using recent measurements of integral control rod worth, the peak element power was calculated for an all-rods-out case at the current license power limit of 1250 kW for a fully-loaded core, and then for a case with each of the four control rods fully inserted. For these cases, the power of the core was reduced based on the current set of integral control rod worth curves and an assumed power coefficient of -0.0045 $ / kW, which is similar to the value of -0.0038$ / kw calculated by dividing the recent peak power of 735 kW by the present-day 2.77$ of excess reactivity. The maximum power per element for this set of models is given in Table 2.

Rod Inserted	Approximate Reactor Power (kW)	Average Power Per Element (kW)	Peak Element Power (kW)
Regulating Rod (max, IRW = 0.87$)	882	10.38	16.78
Shim Rod (max, IRW = 2.13$)	738	8.68	14.15
Safety Rod (max, IRW = 1.68$)	758	8.91	14.35

Rod Inserted	Approximate Reactor Power (kW)	Average Power Per Element (kW)	Peak Element Power (kW)
Pulse Rod (max, IRW = 2.77$)	685	7.74	11.13
1250 kW, all rods out	**1250**	**14.71**	**22.91**

Table 2: Power comparison between different core configurations, no 12% fuel installed.

A similar study was conducted using a peak power of 1250 kWth with all rods withdrawn, but with a single 12.3%-weighted fuel element in location E-10. This study confirms that the maximum power in a single fuel element remains well below the SAR-assumed value of 2.0 element-to-average, even with a high-load fuel element on the opposite side of the core relative to the inserted control rod. The results are depicted in Figure 4.

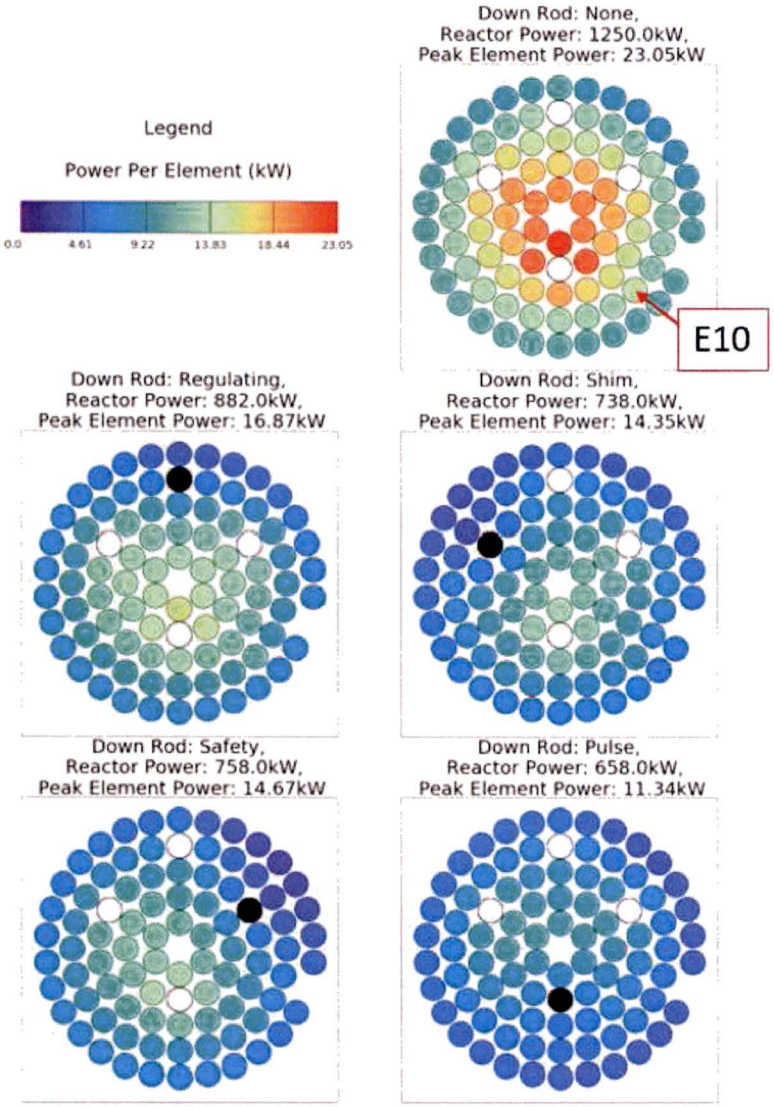

Figure 4 -Power peaking map with different control rods stuck in, with a 12% element in location E-10 and a maximum core power of 1.25 MW.

Therefore the case of an inoperable control rod being fully inserted may slightly increase maximum power peaking but will greatly reduce the maximum power per element. The slight increase in power peaking will also be more than offset by the reduction in core reactivity due to the insertion of the rod poison, resulting in a massive decrease to the maximum fuel element power.

The proposed revision to the TS includes a specification of minimum of 3 operable control rods, and states that inoperable control rods must be fully inserted. (See attached, TS 3.4).

6. ... *"Provide a revision to the proposed TS describing the geometric limitation and the controls that will help ensure compliance and include information on the acceptability of the specific location where the 12.5% fuel elements will be installed. Otherwise, describe how the current or previously proposed TS adequately address the control of this geometric limitation on the location [of] 12.5 wt% fuel elements in the proposed reactor core."*

MCNP results transmitted in the initial LAR and in previous RAI responses indicate that 12.5 wt% (max.) fuel in the E- and F-rings will be kept below the fission heating density (i.e., power) of the 9.0 wt% (max.) elements in the B-ring, even if placed adjacent to another 12.5%-loaded element. The additional restriction to avoided placing the 12.5% elements adjacent to control rod channels was intended to avoid conflict with the NRC over local power peaking during pulsing, although results from MCNP calculations performed at KSU have not shown reason for concern. The proposed TS have been revised to include a specific list of locations in which 12.5% fuel cannot be located. (See attached). In order to ensure that the 12% fuel is only added to locations where it permitted, the fuel handling procedures will be revised to include guidance to check the Technical Specifications list of approved locations prior to loading 12% fuel.

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

Regards,

Jeffrey A. Geuther

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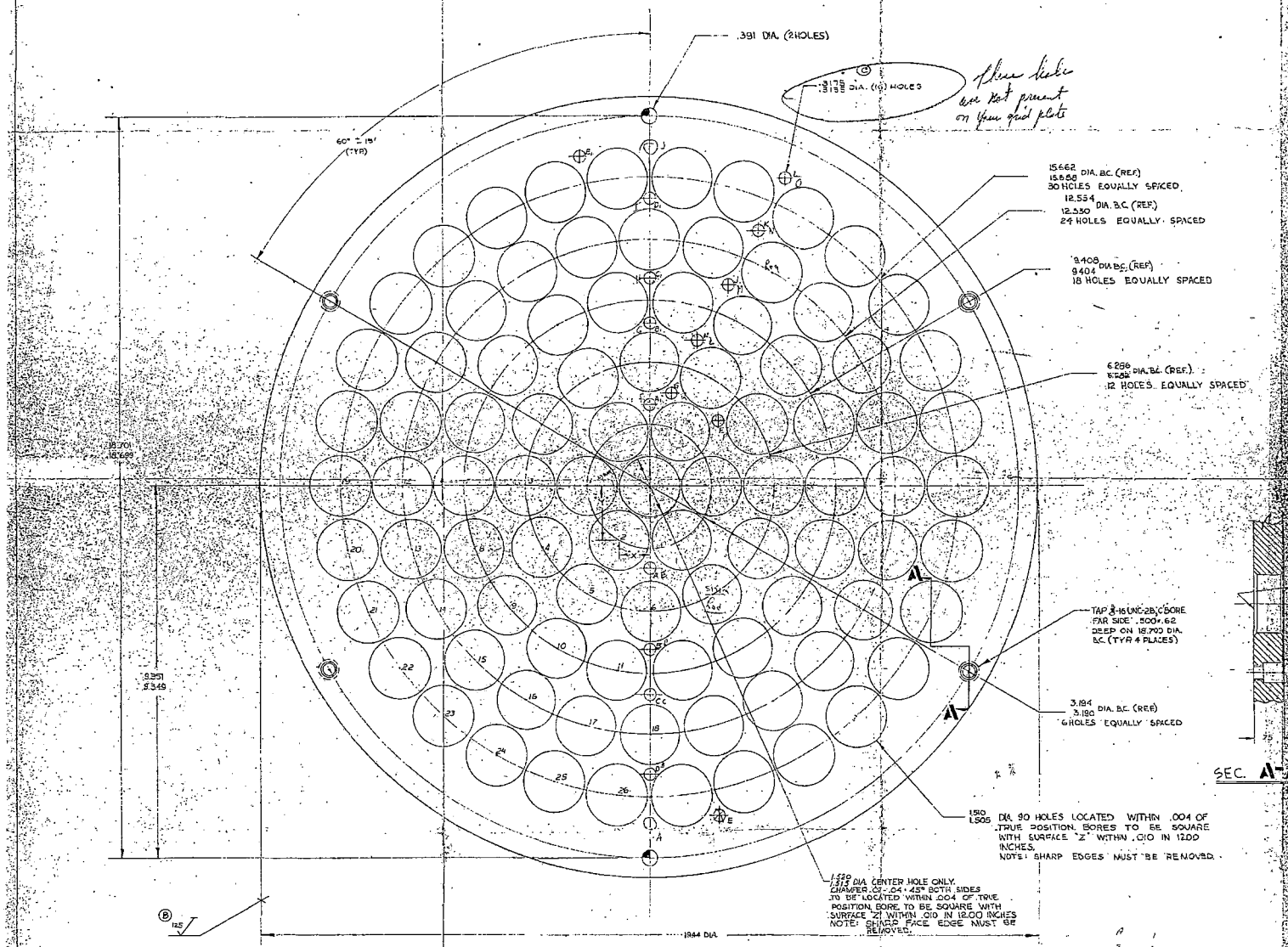
Attachments (5): GA Drawing TOS21E106 – Top Grid Plate

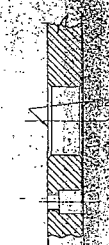
SAR Ch. 4 markup copy

SAR Ch. 4 clean copy

TS markup copy

TS clean copy





SYM	REVISIONS	DR	CHK	APPD
A	REMOVED [illegible] TAPPED HOLES	H3-58		[illegible]
	& DIMENS[illegible]			
B	ADDED $\sqrt{125}$ FINISH	H225[illegible]		[illegible]
C	ADDED $\frac{.3175}{.3135}$ DIA. (16) HOLES	[illegible] 3/28/62	[illegible]	[illegible]

these holes are not present on your grid plate

CODE	X DIM.	Y DIM.
1	1.5960	0
2	.7980	1.3621
3	3.1420	0
4	2.7210	1.5710
5	1.5710	2.7210
6	0	3.1420
7	4.7030	0
8	4.4194	1.6084
9	3.6026	[illegible]
10	2.3515	4.0729
11	.8166	4.6315
12	[illegible]	0
13	[illegible]	[illegible]
14	5.4163	[illegible]
15	[illegible]	[illegible]
16	3.3330	[illegible]
17	1.0216	[illegible]
18	[illegible]	[illegible]
19	7.6200	0
20	7.6586	1.6279
21	[illegible]	[illegible]
22	[illegible]	4.6024
23	5.2391	5.8185
24	[illegible]	7.2408
25	2.4185	[illegible]
26	.8182	7.7869
A-A	0	2.023
B-B	0	4.119
C-C	0	5.259
D-D	0	7.242
E-E1	1.764	6.295
F	1.750	1.656
G	.582	2.358
H	1.208	3.690
J	2.000	5.082
K	2.275	6.223
L	3.452	7.149

AUG 2 1963

T0S210E106

A 1
B 6
C 12
D 18
E 24
30
91

2. ANODIZE PER MIL SPEC A-8625A TYPE II
MODIFY TO 2 HOUR ANODIZE - SEALED
1. REMOVE ALL BURRS & SHARP EDGES.
NOTES:

			19.44 DIA × 3/4 THK	AL ALLOY	6061-T6
REQUIRED	ITEM	PART NO.	DESCRIPTION	MATL	MATL SPEC

LIST OF MATERIAL

UNLESS OTHERWISE SPECIFIED
DIMENSIONS ARE IN INCHES
TOLERANCES:
DECIMALS FRACTIONS ANGLES
.XX ± .01 ± ±1°
.XXX ± .005
ALL MACHINED SURFACES $\sqrt{125}$

DR	KINNEY	[illegible]
CHK	R. WASSERMAN	[illegible]
APPD	K. ANDERSON	[illegible]
APPD	ARKELYAN	[illegible]
APPD	R. E. [illegible]	

NEXT ASSEMBLY
T0S210E100

TOP GRID - CORE

GENERAL DYNAMICS
GENERAL ATOMIC DIVISION
[illegible] SAN DIEGO [illegible], CALIFORNIA

T0S210E106

SCALE 1/1 WT. SHEET OF REV C

4. Reactor Description

4.1 Summary Description

The Kansas State University (KSU) Nuclear Reactor Facility, operated by the Department of Mechanical and Nuclear Engineering, is located in Ward Hall on the campus in Manhattan, Kansas. The Department is also the home of the Tate Neutron Activation Analysis Laboratory. The TRIGA reactor was obtained through a 1958 grant from the United States Atomic Energy Commission and is operated under Nuclear Regulatory Commission License R-88 and the regulations of Chapter 1, Title 10, Code of Federal Regulations. Chartered functions of the Nuclear Reactor Facility are to serve as: 1) an educational facility for all students at KSU and nearby universities and colleges, 2) an irradiation facility for researchers at KSU and for others in the central United States, 3) a facility for training nuclear reactor operators, and 4) a demonstration facility to increase public understanding of nuclear energy and nuclear reactor systems.

The KSU TRIGA reactor is a water-moderated, water-cooled thermal reactor operated in an open pool and fueled with heterogeneous elements consisting of nominally 20 percent enriched uranium in a zirconium hydride matrix and clad with stainless steel. Principal experimental features of the KSU TRIGA Reactor Facility are:

- Central thimble
- Rotary specimen rack
- Thermalizing column with bulk shielding tank
- Thermal column with removable door
- Beam ports
  - Radial (2)
  - Piercing (fast neutron) (1)
  - Tangential (thermal neutron) (1)

The reactor was licensed in 1962 to operate at a steady-state thermal power of 100 kilowatts (kW). The reactor has been licensed since 1968 to operate at a steady-state thermal power of 250 kW and a pulsing maximum thermal power of 250 MW. Application is made concurrently with license renewal to operate at a maximum of 1,250 kW, with fuel loading to support 500 kW steady state thermal power and with pulsing to $3.00 reactivity insertion. All cooling is by natural convection. The 250-kW core consists of 81 fuel elements typically (at least 83 planned for the 1,250-kW core), each containing as much as 39 grams of ^{235}U. The reactor core is in the form of a right circular cylinder about 23 cm (approximately 9 in.) radius and 38 cm (14.96 in.) depth, positioned with axis vertical near the base of a cylindrical water tank 1.98 m (6.5 ft.) diameter and 6.25 m (20.5 ft.) depth. Criticality is controlled and shutdown margin assured by three control rods in the form of aluminum or stainless-steel clad boron carbide or borated graphite. A fourth control rod would be used for 1,250-kW operation. A biological shield of reinforced concrete at least 2.5 m (8.2 ft) thick provides radiation shielding at the side and at the base the reactor tank. The tank and shield are in a 4078-m^3 (144,000 $ft.^3$) confinement building

made of reinforced concrete and structural steel, with composite sheathing and aluminum siding. Sectional views of the reactor are shown in Figures 4.1 and 4.2.

Criticality was first achieved on October 16, 1962 at 8:25 p.m. In 1968 pulsing capability was added and the maximum steady-state operating power was increased from 100 kW to 250 kW. The original aluminum-clad fuel elements were replaced with stainless-steel clad elements in 1973. Coolant system was replaced (and upgraded in 2000), the reactor operating console was replaced, and the control room was enlarged and modernized in 1993, with support from the U.S. Department of Energy. All neutronic instrumentation was replaced in 1994.

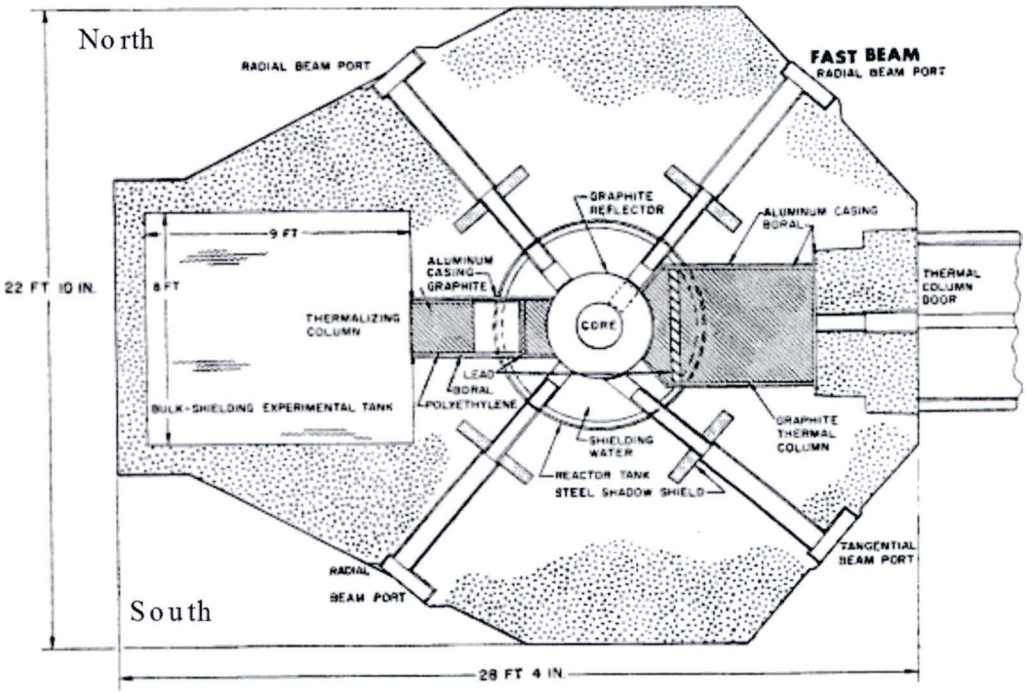


Figure 4. 1, Vertical Section Through the KSU TRIGA Reactor.

4.2 Reactor Core

The General Atomics TRIGA reactor design began in 1956. The original design goal was a completely and inherently safe reactor. Complete safety means that all the available excess reactivity of the reactor can be instantaneously introduced without causing an accident. Inherent safety means that an increase in the temperature of the fuel immediately and automatically results in decreased reactivity through a prompt negative temperature coefficient. These features were accomplished by using enriched uranium fuel in a zirconium hydride matrix.

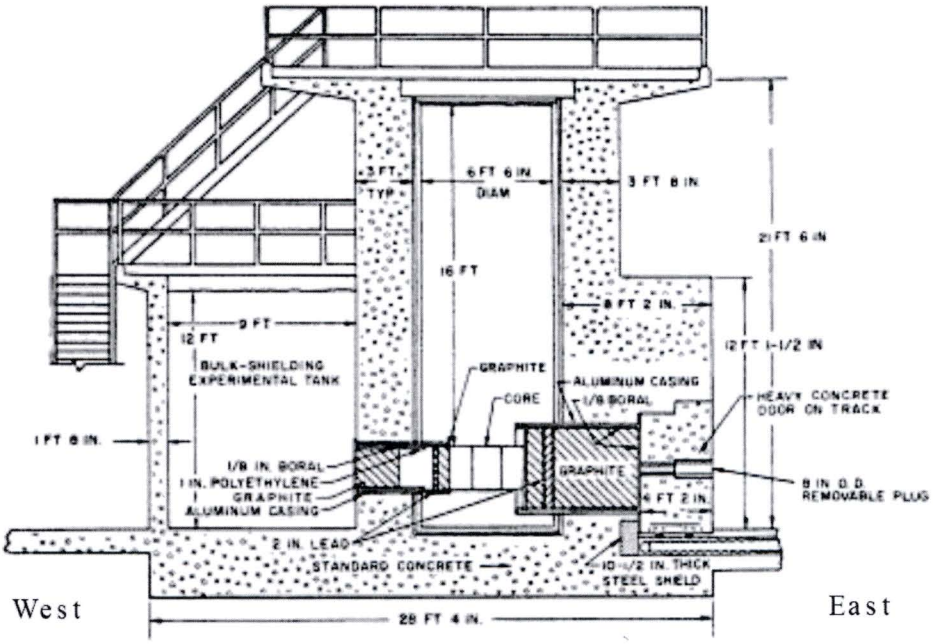


Figure 4. 2, Horizontal Section Through the KSU TRIGA Reactor.

The basic parameter providing the TRIGA system with a large safety factor in steady state and transient operations is a prompt negative temperature coefficient, relatively constant with temperature (-0.01% $\Delta k/k°C$). This coefficient is a function of the fuel composition and core geometry. As power and temperature increase, matrix changes cause a shift in the neutron energy spectrum in the fuel to higher energies. The uranium exhibits lower fission cross sections for the higher energy neutrons, thus countering the power increase. Therefore, fuel and clad temperature automatically limit operation of the reactor.

It is more convenient to set a power level limit that is based on temperature. The design bases analysis indicates that operation at up to 1900 kW (with an 83 element core and 120°F inlet water temperature) with natural convective flow will not allow film boiling, and therefore high fuel and clad temperatures which could cause loss of clad integrity could not occur. An 85-element core distributes the power over a larger volume of heat generating elements, and therefore results in a more favorable, more conservative, thermal hydraulic response.

4.2.1 Reactor Fuel[1]

TRIGA fuel was developed around the concept of inherent safety. A core composition was sought which had a large prompt negative temperature coefficient of reactivity such that if all the available excess reactivity were suddenly inserted into the core, the resulting fuel temperature would automatically cause the power excursion to terminate before any core damage resulted. Zirconium hydride was found to possess a basic mechanism to produce the desired characteristic. Additional advantages were that ZrH has a high heat capacity, results in relatively small core sizes and high flux values due to the high hydrogen content, and could be used effectively in a rugged fuel element size.

TRIGA fuel is designed to assure that fuel and cladding can withstand all credible environmental and radiation conditions during its lifetime at the reactor site. As described in 3.5.1 (Fuel System) and NUREG 1282, fuel temperature limits both steady-state and pulse-mode operation. The fuel temperature limit stems from potential hydrogen outgassing from the fuel and the subsequent stress produced in the fuel element clad material. The maximum temperature limits of 1150°C (with clad < 500°C) and 950°C (with clad > 500°C) for U-ZrH ($H/Zr_{1.65}$) have been set to limit internal fuel cladding stresses that might challenge clad integrity (NUREG 1282). These limits are the principal design bases for the safety analysis.

a. Dimensions and Physical Properties.

The KSU TRIGA reactor is fueled by stainless steel clad Mark III fuel-elements. Three instrumented aluminum-clad Mark II elements are still available for use in the core. General properties of TRIGA fuel are listed in Table 4.1. The Mark III elements are illustrated in Figure 4.3. To facilitate hydriding in the Mk III elements, a zirconium rod is inserted through a 0.635 cm. (1/4-in.) hole drilled through the center of the active fuel section.

Instrumented elements have three chromel-alumel thermocouples embedded to about 0.762 cm (0.3 in.) from the centerline of the fuel, one at the axial center plane, and one each at 2.54 cm. (1 in.) above and below the center plane. Thermocouple leadout wires pass through a seal in the upper end fixture, and a leadout tube provides a watertight conduit carrying the leadout wires above the water surface in the reactor tank.

[1]Unless otherwise indicated, fuel properties are taken from the General Atomics report of Simnad [1980] and from authorities cited by Simnad.

Graphite dummy elements may be used to fill grid positions in the core. The dummy elements are of the same general dimensions and construction as the fuel-moderator elements. They are clad in aluminum and have a graphite length of 55.88 cm (22 in.).

Table 4.1, Nominal Properties of Mark II and Mark III TRIGA Fuel Elements in use at the KSU Nuclear Reactor Facility.

Property	Mark II	Mark III
Dimensions		
Outside diameter, $D_o = 2r_o$	1.47 in. (3.7338 cm)	1.47 in. (3.7338 cm)
Inside diameter, $D_i = 2r_i$	1.41 in (3.6322 cm)	1.43 in. (3.6322 cm)
Overall length	28.4 in. (72.136 cm)	28.4 in. (72.136 cm)
Length of fuel zone, L	14 in. (35.56 cm)	15 in. (38.10 cm)
Length of graphite axial reflectors	4 in. (10.16 cm)	3.44 in (8.738 cm)
End fixtures and cladding	aluminum	304 stainless steel
Cladding thickness	0.030 in. (0.0762 cm)	0.020 in. (0.0508 cm)
Burnable poisons	Sm wafers	None
Uranium content		
Weight percent U	8.0	8.5
^{235}U enrichment percent	20	20
^{235}U content	36 g	38 g
Physical properties of fuel excluding cladding		
H/Zr atomic ratio	1.0	1.6
Thermal conductivity (W cm^{-1} K^{-1})	0.18	0.18
Heat capacity [T ≥0°C] (J cm^{-3} K^{-1})		2.04 + 0.00417T
Mechanical properties of delta phase U-ZrH[a]		
Elastic modulus at 20°C		9.1×10^6 psi
Elastic modulus at 650°C		6.0×10^6 psi
Ultimate tensile strength (to 650°C)		24,000 psi
Compressive strength (20°C)		60,000 psi
Compressive yield (20°C)		35,000 psi

[a]Source: Texas SAR [1991].

b. Composition and Phase Properties

The Mark III TRIGA fuel element in use at Kansas State University contains nominally 8.5% by weight of uranium, enriched to 20% ^{235}U, as a fine metallic dispersion in a zirconium hydride matrix. The H/Zr ratio is nominally 1.6 (in the face-centered cubic delta phase). The equilibrium hydrogen dissociation pressure is governed by the composition and temperature. For $ZrH_{1.6}$, the equilibrium hydrogen pressure is one atmosphere at about 760°C. The single-phase, high-hydride composition eliminates the problems of density changes associated with phase changes and with thermal diffusion of the hydrogen. Over 25,000 pulses have been performed with the TRIGA fuel elements at General Atomic, with fuel temperatures reaching peaks of about 1150°C.

The zirconium-hydrogen system, whose phase diagram is illustrated in Chapter 3, is essentially a simple eutectoid, with at least four separate hydride phases. The delta and epsilon phases are respectively face-centered cubic and face-centered tetragonal hydride phases. The two phase delta + epsilon region exists between $ZrH_{1.64}$ and $ZrH_{1.74}$ at room temperature, and closes at $ZrH_{1.7}$ at 455°C. From 455°C to about 1050°C, the delta phase is supported by a broadening range of H/Zr ratios.

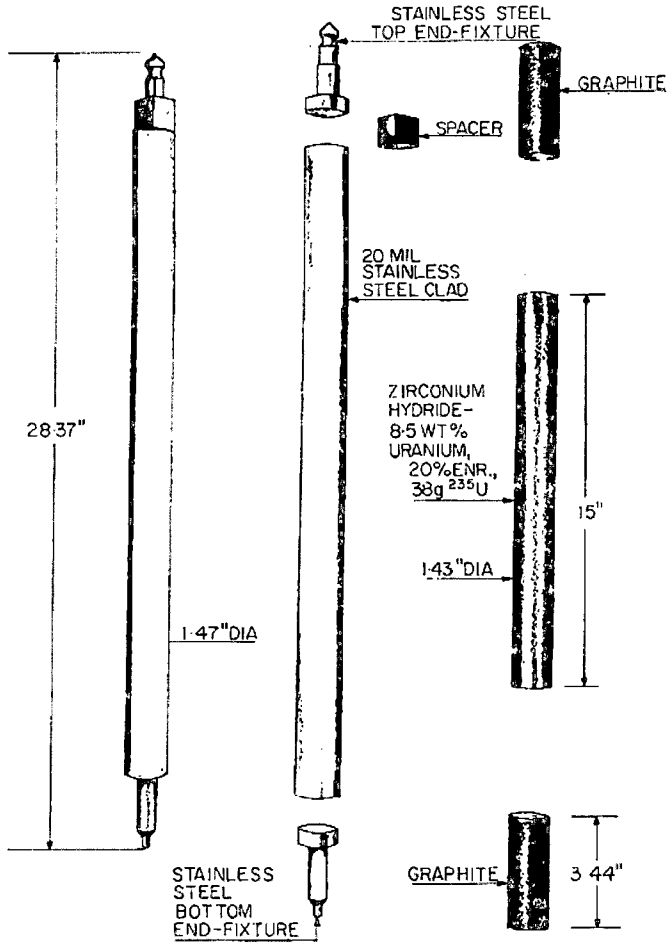


Figure 4.3, TRIGA Fuel Element.

c. Core Layout

A typical layout for a KSU TRIGA II 250-kW core (Core II-18) is illustrated in Figure 4.4. The layout for the 1,250-kW core is expected to be similar, except that the graphite elements will be replaced by fuel elements, one additional control rod will be added, and control rod positions will be adjusted..

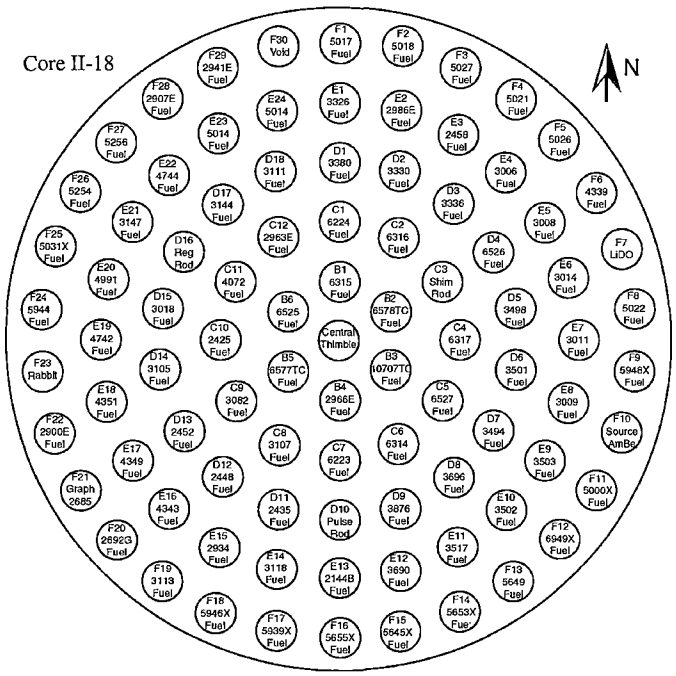


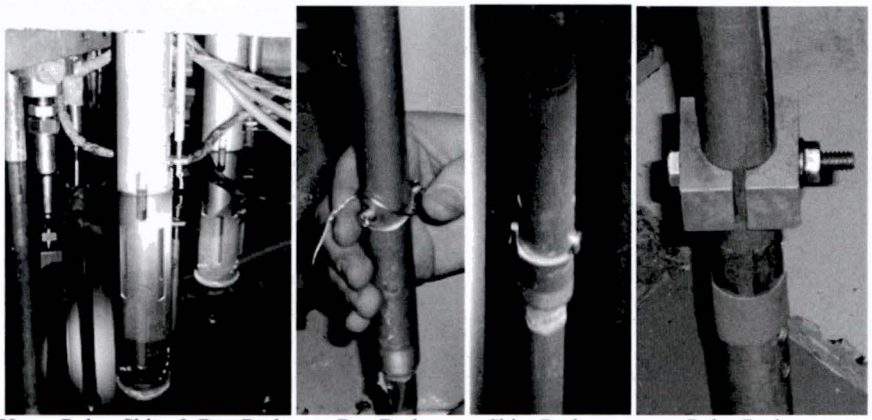
Figure 4.4, Core Layout (250 kW).

The additional fuel elements are required to compensate for higher operating temperatures from the higher maximum steady state power level. The additional control rod is required to meet reactivity control requirements at higher core reactivity associated with the additional fuel. The control rod positions will be different to allow a higher worth pulse rod (the 250 kW pulse rod reactivity worth is $2.00, the 1,250 kW core pulse rod reactivity worth is $3.00), balancing the remaining control rod's worth to meet minimum shutdown margin requirements, and meeting physical constraints imposed by the dimensions of the pool bridge

4.2.2 Control Rods

The pulse rod is 3.175 cm. (1.25 in.) diameter. Other rods are 2.225 cm (7/8 in.) diameter. Control rods are 50.8 cm. (20 in.) long boron carbide or borated graphite, clad with a 0.0762 cm. (30-mil) aluminum sheath.

The control rod drives are connected to the control rod clutches through three extension shafts. The clutch and upper extension shaft for standard rods extend through an assembly designed with slots that provides a hydraulic cushion (or buffer) for the rod during a scram, and also limits the bottom position of the control rods so that they do not impact the bottom of the control rod guide tube (in the core). The buffers for two standard rods are shown in the left hand picture below (slotted tubes on the right hand side) along with the top section of the pulse/transient rod extension. The pulse rod drive clutch connects to a solid extension shaft through a pneumatic cylinder; the dimensions of the cylinder limits bottom travel.



Upper Pulse, Shim & Reg Rods Reg Rod Shim Rod Pulse Rod

Figure 4.5, Control Rod Upper Extension Assemblies

The bottom of the pulse rod is shown on the left hand side of Figure 4.5. The upper extension shaft is a hollow tube, the middle extension is solid. The upper extension shaft is connected to the middle extension shaft with lock wire or a pin and lock wire for standard rods, with a bolted collar for the pulse rod (the mechanical shock during a pulse requires a more sturdy fastener). Securing the upper control rod extension to the middle extension at one of several holes drilled in the upper part of the middle extension (Figure 4.6) provides adjustment for the control rods necessary to ensure the control rod full in position is above the bottom of the guide tube.

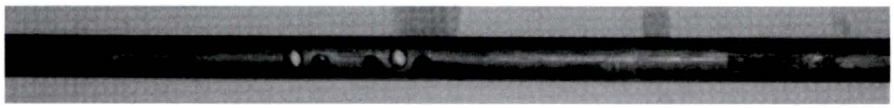


Figure 4.6, Middle Extension Rod Alignment Holes

The middle solid extension is similarly connected to the lower extension. The lower extension is hollow, the middle extension fits into the lower extension and a hole drilled in the overlap secures the lower extension to the middle extension. Typically the lower extension has a tighter fit than the upper extension because the lower and middle extension are not separated for inspections and because the interface with upper extension is used to set the bottom position of the control rod. Pictures of the lower connector for the pulse rod and one standard rod are shown at the left in Figure 4.7..

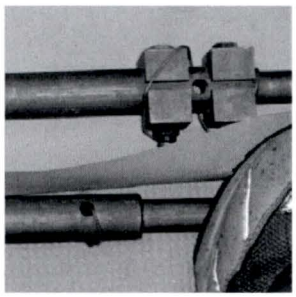
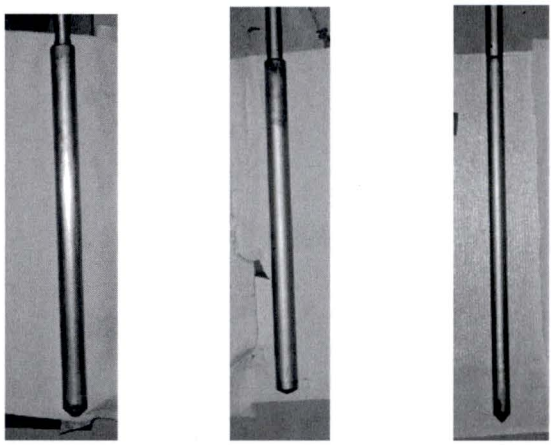


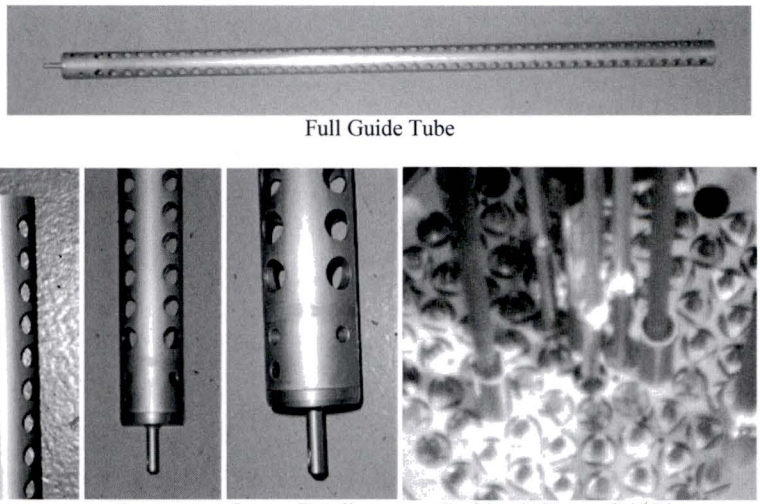
Figure 4.7, Standard & Pulse Rod Lower Coupling

The bottom of the lower extension attaches directly to the control rod. Pictures of the control rods taken during the 2003 control rod inspection are in Figure 4.8. The rods move within control rod guide tubes, shown in Figure 4.9. The guide tubes have perforated walls. The guide tubes have a small metal wire in the tip that fits into the lower grid plate; a setscrew inside the bottom of the guide tube pushes the wire against the lower grid plate to secure the guide tube.



Pulse Rod Shim Rod Reg Rod

Figure 4.8, Control Rods During 2003 Inspection



Full Guide Tube

Upper Lower Lower Detail Position in Upper Grid Plate

Figure 4.9, Control Rod Guide Tubes

a. Control Function

While three control rods were adequate to meet Technical Specification requirements for reactivity control with the 100 kW and 250 kW cores, reactivity limits for operation at a maximum power level of 1,250 kW requires four control rods (three standard and one transient/pulsing control rod). The control-rod drives are mounted on a bridge at the top of the reactor tank. The control rod drives are coupled to the control rod through a connecting rod assembly that includes a clutch. The standard rod clutch is an electromagnet; the transient rod clutch is an air-operated shuttle. Scrams cause the clutch to release by de-energizing the magnetic clutch and venting air from the transient rod clutch; gravity causes the rod to fall back into the core. Interlocks ensure operation of the control rods remains within analyzed conditions for reactivity control, while scrams operation at limiting safety system settings. A detailed description of the control-rod system is provided in Chapter 7; a summary of interlocks and scrams is provided below in Table 4.2 and 4.3. Note that (1) the high fuel temperature and period scrams are not required, (2) the fuel temperature scram limiting setpoint depends on core location for the sensor, and (3) the period scram can be prevented by an installed bypass switch.

Table 4.2, Summary of Control Rod Interlocks

INTERLOCK	SETPOINT	FUNCTION/PURPOSE
Source Interlock	2 cps	Inhibit standard rod motion if nuclear instrument startup channel reading is less than instrument sensitivity/ensure nuclear instrument startup channel is operating
Pulse Rod Interlock	Pulse rod inserted	Prevent applying power to pulse rod unless rod inserted/prevent inadvertent pulse
Multiple Rod Withdrawal	Withdraw signal, more than 1 rod	Prevent withdrawal of more than 1 rod/Limit maximum reactivity addition rate
Pulse Mode Interlock	Mode switch in Hi Pulse	Prevent withdrawing standard control rods in pulse mode
Pulse-Power Interlock	10 kW	Prevent pulsing if power level is greater than 10 kW

NOTE: (1) Pulse-Power Interlock normally set at 1 kW, (2) only Pulse Mode Interlock required by Technical Specifications

b. Evaluation of Control Rod System

The reactivity worth and speed of travel for the control rods are adequate to allow complete control of the reactor system during operation from a shutdown condition to full power. The TRIGA system does not rely on speed of control as significant for safety of the reactor; scram times for the rods are measured periodically to monitor potential degradation of the control rod system. The inherent shutdown mechanism (temperature feedback) of the TRIGA prevents unsafe excursions and the control system is used only for the planned shutdown of the reactor and to control the power level in steady state operation.

Table 4.3, Summary of Reactor SCRAMs

Measuring Channel	Limiting Trip Setpoint		Actual Setpoint
	Steady State	Pulse	
Linear Channel High Power	110%	N/A	104%
Power Channel High power	110%	N/A	104%
Detector High Voltage	90%	90%	90%
High Fuel Temperature[1]	600°C B Ring element		450°C
	555°C C Ring element		
	480°C D Ring element		
	380°C E Ring element		350°C
Period [1]	N/A	N/A	3 sec

NOTE [1]: Period trip and temperature trip are not required by Technical Specifications

The reactivity worth of the control system can be varied by the placement of the control rods in the core. The control system may be configured to provide for the excess reactivity needed for 1,250 kW operations for eight hours per day (including xenon override) and will assure a shutdown margin of at least $0.50.

Nominal speed of the standard control rods is about 12 in. (30.5 cm) per minute (with the stepper motor specifically adjusted to this value), of the transient rod is about 24 in. (61 cm) per minute, with a total travel about 15 in. (38.1 cm). Maximum rate of reactivity change for standard control rods is specified in Technical Specifications.

4.2.3 Neutron Moderator and Reflector

Hydrogen in the Zr-H fuel serves as a neutron moderator. Demineralized light water in the reactor pool also provides neutron moderation (serving also to remove heat from operation of the reactor and as a radiation shield). Water occupies approximately 35% of the core volume. A graphite reflector surrounds the core, except for a cutout containing the rotary specimen rack (described in Chapter 10). Each fuel element contains graphite plugs above and below fuel approximately 3.4 in. in length, acting as top and bottom reflectors.

The radial reflector is a ring-shaped, aluminum-clad, block of graphite surrounding the core radially. The reflector is 0.457-m (18.7 in.) inside diameter, 1.066-m (42 in.) outside diameter, and 0.559-m (20 in.) height. Embedded as a circular well in the reflector is an aluminum housing for a rotary specimen rack, with 40 evenly spaced tubular containers, 3.18-cm (1.25 in.) inside diameter and 27.4-cm (10.8 in.) height. The rotary specimen rack housing is a watertight assembly located in a re-entrant well in the reflector.

The radial reflector assembly rests on an aluminum platform at the bottom of the reactor tank. Four lugs are provided for lifting the assembly. A radial void about 6 inches (15.24 cm) in diameter is located in the reflector such that it aligns with the radial piercing beam port (NE beam port). The reflector supports the core grid plates, with grid plate positions set by alignment fixtures. Graphite inserts within the fuel cladding provide additional reflection. Inserts are placed at both ends of the fuel meat, providing top and bottom reflection.

4.2.4 Neutron Startup Source

A 2-curie americium-beryllium startup source (approximately 2×10^6 n s^{-1}) is used for reactor startup. The source material is encapsulated in stainless steel and is housed in an aluminum-cylinder source holder of approximately the same dimensions as a fuel element. The source holder may be positioned in any one of the fuel positions defined by the upper and lower grid plates. A stainless-steel wire may be threaded through the upper end fixture of the holder for use in relocating the source manually from the 22-ft level (bridge level) of the reactor.

4.2.5 Core Support Structure

The fuel elements are spaced and supported by two 0.75-in. (1.9 cm) thick aluminum grid plates. The grid plates have a total of 91 spaces, up to 85 of which are filled with fuel-moderator elements and dummy elements, and the remaining spaces with control rods, the central thimble, the pneumatic transfer tube, the neutron source holder, and one or more voids. The bottom grid plate, which supports the weight of the fuel elements, has holes for receiving the lower end fixtures. Space is provided for the passage of cooling water around the sides of the bottom grid plate and through 36 special holes in it. The 1.5-in. (3.8 cm) diameter holes in the upper grid plate serve to space the fuel elements and to allow withdrawal of the elements from the core. Triangular-shaped spacers on the upper end fixtures allow the cooling water to pass through the upper grid plate when the fuel elements are in position. The reflector assembly supports both grid plates.

4.3 Reactor Tank

The KSU TRIGA reactor core support structure rests on the base of a continuous, cylindrical aluminum tank surrounded by a reinforced, standard concrete structure (with a minimum thickness of approximately 249 cm. or 8 ft 2 in), as illustrated in Figures 4.1 and 4.2. The tank is a welded aluminum structure with 0.635 cm. (1/4-in.) thick walls. The tank is approximately 198 cm (6.5-ft) in diameter and approximately 625 cm (20.5-ft) in depth. The exterior of the tank was coated with bituminous material prior to pouring concrete to retard corrosion. Each experiment facility penetration in the tank wall (described below) has a water collection plenum at the penetration. All collection plenums are connected to a leak-off volume through individual lines with isolation valves, with the leak-off volumes monitored by a pressure gauge. The bulk shield tank wall is known to have a small leak into the concrete at the thermalizing column plenum, therefore a separate individual leak-off volume (and pressure gauge) is installed for the bulk shield tank; all other plenums drain to a common volume. In the event of a leak from the pool

through an experiment facility, pressure in the volume will increase; isolating individual lines allows identification of the specific facility with the leak.

A bridge of steel plates mounted on two rails of structural steel provides support for control rod drives, central thimble, the rotary specimen rack, and instrumentation. The bridge is mounted directly over the core area, and spans the tank. Aluminum grating with clear plastic attached to the bottom is installed that can be lowered over the pool. The grating can be lowered when activities could cause objects or material to fall into the reactor pool. The grating normally remains up to reduce humidity at electro-mechanical components of the control rod drive system and to prevent the buildup of radioactive gasses at the pool surface during operations.

Four beam tubes run from the reactor wall to the outside of the concrete biological shield in the outward direction. Tubes welded to the inside of the wall run toward the reactor core. Three of the tubes (NW, SW, and SE) end at the radial reflector. The NE beam tube penetrates the radial reflector, extending to the outside of the core. Two penetrations in the tank allow neutron extraction into a thermal column and a thermalizing column (described in Chapter 10).

4.4 Biological Shield

The reactor tank is surrounded on all sides by a monolithic reinforced concrete biological shield. The shielding configuration is similar to those at other TRIGA facilities operating at power levels up to 1 MW. Above ground level, the thickness varies from approximately 249 cm. (8 ft 2 in.) at core level to approximately 91 cm. (3 ft.) at the top of the tank.

The massive concrete bulk shield structure provides additional radiation shielding for personnel working in and around the reactor laboratory and provides protection to the reactor core from potentially damaging natural phenomena.

4.5 Nuclear Design

The strong negative temperature coefficient is the principal method for controlling the maximum power (and consequently the maximum fuel temperature) for TRIGA reactors. This coefficient is a function of the fuel composition, core geometry, and temperature. For fuels with 8.5% U, 20% enrichment, the value is nearly constant at 0.01% $\Delta k/k$ per °C, and varies only weakly dependent on geometry and temperature.

Fuel and clad temperature define the safety limit. A power level limit is calculated that ensures that the fuel and clad temperature limits will not be exceeded. The design bases analysis indicates that operation at 1,250 kW thermal power with an 83-element across a broad range of core and coolant inlet temperatures with natural convective flow will not allow film boiling that could lead to high fuel and clad temperatures that could cause loss of clad integrity.

Increase in maximum thermal power from 250 to 1,250 kW does not affect fundamental aspects of TRIGA fuel and core design, including reactivity feedback coefficients, temperature safety

limits, and fission-product release rates. Thermal hydraulic performance is addressed in Section 4.6.

4.5.1 Design Criteria - Reference Core

The limiting core configuration for this analysis is a compact core defined by the TRIGA Mk II grid plates (Section 4.2.5). The grid plates have a total of 91 spaces, up to 85 of which are filled with fuel-moderator elements and graphite dummy elements, and the remaining spaces with control rods, the central thimble, the pneumatic transfer tube, the neutron source holder, and one or more voids in the E or F (outermost two rings) as required to support experiment operations or limit excess reactivity. The bottom grid plate, which supports the weight of the fuel elements, has holes for receiving the lower end fixtures.

4.5.2 Reactor Core Physics Parameters

The limiting core configuration differs from the configuration prior to upgrade only in the addition of a fourth control rod, taking the place of a graphite dummy element or void experimental position. For this reason, core physics is not affected by the upgrade except for linear scaling with power of neutron fluxes and gamma-ray dose rates.

For comparison purposes, a tabulation of total rod worth for each control element from the K-State reactor from a recent rod worth measurement is provided with the values from the Cornell University TRIGA reactor as listed in NUREG 0984 (Safety Evaluation Report Related to the Renewal of the Operating license for the Cornell University TRIGA Research Reactor).

Table 4.4, 250 kW Core Parameters.

β (effective delayed neutron fraction)	0.007
ℓ (effective neutron lifetime)	43 :S
α_{Tf} (prompt temperature coefficient)	-\$0.017 EC^{-1} @ 250kW ~275EC
α_v (void coefficient)	-0.003 $1\%^{-1}$ void
α_p (power temperature coefficient – weighted ave)	-\$0.006 kW^{-1} to – \$0.01 kW^{-1}

Table 4.5, Comparison of Control Rod Worths.

	KSU TRIGA Mark II (250 kW)				Cornell University (500 kW)	
	Core II-19		Core III-1			
Pulse	D-10	\$1.96	C-4	\$2.12	D-10	\$1.88
Shim	C-3	\$2.88	D-4	\$1.85	D-16	\$2.20
Safety	NA	\$0.0	D-16	\$1.82	D-4	\$1.99
Regulating	D-16	\$1.58	E-1	\$0.79	E-1	\$0.58
TOTAL	NA	\$6.42	NA	\$6.58	NA	\$6.65

NOTE: Core III-1 has an experiment positioned to control the worth of the pulse rod

The pulse rod is similar to a standard control rod, and the worth of the pulse rod compares well with the comparable standard control rods in similar ring positions. A maximum pulse is analyzed for thermal hydraulic response and maximum fuel temperature.

4.5.3 Fuel and Clad Temperatures

This section analyzes expected fuel and cladding temperatures with realistic modeling of the fuel-cladding gap. Analysis of steady state conditions reveals maximum heat fluxes well below the critical heat flux associated with departure from nucleate boiling. Analysis of pulsed-mode behavior reveals that film boiling is not expected, even during or after pulsing leading to maximum adiabatic fuel temperatures.

Chapter 4, Appendix A of this chapter reproduces a commonly cited analysis of TRIGA fuel and cladding temperatures associated with pulsing operations. The analysis addresses the case of a fuel element at an average temperature of 1000°C immediately following a pulse and estimates the cladding temperature and surface heat flux as a function of time after the pulse. The analysis predicts that, if there is no gap resistance between cladding and fuel, film boiling can occur very shortly after a pulse, with cladding temperature reaching 470°C, but with stresses to the cladding well below the ultimate tensile strength of the stainless steel. However, through comparisons with experimental results, the analysis concludes that an effective gap resistance of 450 Btu hr^{-1} ft^{-2} $°F^{-1}$ (2550 W m^{-2} K^{-1}) is representative of standard TRIGA fuel and, with that gap resistance, film boiling is not expected. This section provides an independent assessment of the expected fuel and cladding thermal conditions associated with both steady-state and pulse-mode operations.

a. Spatial Power Distribution

The following conservative approximations are made in characterizing the spatial distribution of the power during steady-state operations.

- The hottest fuel element delivers twice the power of the average.

  Classically, the radial hot-channel factor for a cylindrical reactor (using R as the physical radius and R_e as the physical radius and the extrapolation distance) is given[2] by:

$$F_{R_e}^{N} = \frac{1.202 * \left(R/R_e \right)}{J_1\left[2.4048 * \left(R/R_e \right) \right]}$$

  with a radial peaking factor of 1.93 for the KSU TRIGA II geometry,. However, TRIGA fuel elements are on the order of a mean free path of thermal neutrons, and there is a significant change in thermal neutron flux across a fuel element.

[2] Elements of Nuclear Reactor Design, 2nd Edition (1983), J. Weisman, Section 6.3

Calculated thermal neutron flux data[3] indicates that the ratio of peak to average neutron flux (peaking factor) for TRIGA cores under a range of conditions (temperature, fuel type, water and graphite reflection) has a small range of 1.36 to 1.40.

Actual power produced in the most limiting actual case is 14% less than power calculated using the assumption; therefore using a peaking factor of 2.0 to determine calculated temperatures and will bound actual temperatures by a large margin, and is extremely conservative.

- The axial distribution of power in the hottest fuel element is sinusoidal, with the peak power a factor of $\pi/2$ times the average, and heat conduction radial only.

  The axial factor for power produced within a fuel element is given by:

$$g(z) = 1.514 * \cos\left(\frac{\pi}{2} * \frac{z}{2 * \ell + \ell_{ext}}\right), \tag{6}$$

  in which $\ell = L/2$ and ℓ_{ext} is the extrapolation length in graphite, namely, 0.0275 m. The value used to calculate power in the limiting location within the fuel element is therefore 4% higher a power calculated with the actual peaking factor. Actual power produced in the most limiting actual case is 4% less than power calculated using the assumption; therefore calculated temperatures will bound actual temperatures.

- The location on the fuel rod producing the most thermal power with thermal power distributed over 83 fuel rods is therefore:

$$q''_{max} = \frac{P}{83 \cdot \pi \cdot D_0 \cdot L} \cdot \frac{\pi}{2} \cdot 2 = \frac{P}{83 \cdot D_0 \cdot L} = P \cdot 0.8469 \tag{7}$$

- The radial and axial distribution of the power within a fuel element is given by

$$q'''(r,z) = q'''_{avg} f(r) g(z), \tag{5}$$

  in which r is measured from the vertical axis of the fuel element and z is measured along the axis, from the center of the fuel element. The axial peaking factor follows from the previous assumption of the core axial peaking factor, but (since there is a significant flux depression across a TRIGA fuel element) distribution of power produced across the radius of the fuel the radial peaking factor requires a different approach than the previous radial peaking factor for the core.

[3] GA-4361, Calculated Fluxes and Cross Sections for TRIGA Reactors (8/14/1963), G. B. West

- The radial factor is given by:

$$f(r) = \frac{a + cr + er^2}{1 + br + dr^2}, \tag{7}$$

in which the parameters of the rational polynomial approximation are derived from flux-depression calculations for the TRIGA fuel (Ahrens 1999a). Values are: a = 0.82446, b = -0.26315, c = -0.21869, d = -0.01726, and e = +0.04679. The fit is illustrated in Figure 4.11.

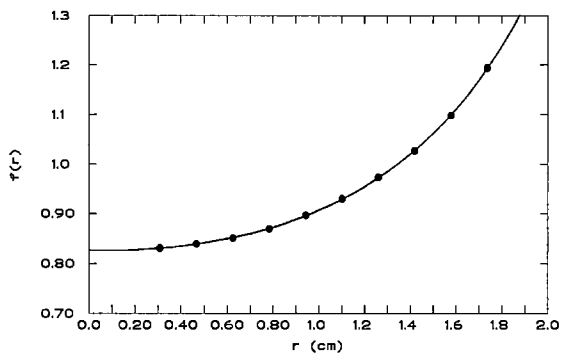


Figure 4.12, Radial Variation of Power Within a TRGIA Fuel Rod.
(Data Points from Monte Carlo Calculations [Ahrens 1999a])

a. Heat Transfer Models

The overall heat transfer coefficient relating heat flux at the surface of the cladding to the difference between the maximum fuel (centerline) temperature and the coolant temperature can be calculated as the sum of the temperature changes through each element from the centerline of the fuel rod to the water coolant, where the subscripts for each of the ΔT's represent changes between bulk water temperature and cladding outer surface, (br_0), changes between cladding outer surface and cladding inner surface (r_0r_i), cladding inner surface and fuel outer surface – gap (g), and the fuel outer surface to centerline (r_icl):

$$T_{cl} = T_b + \Delta T_{br_0} + \Delta T_{r_0r_i} + \Delta T_g + \Delta T_{r_icl} \qquad \text{Eq. 1}$$

A standard heat resistance model for this system is:

$$T_{cl} = T_b + q''\left[\frac{1}{h} + \frac{r_o \ln\left(r_o/r_i\right)}{k_c} + \frac{r_o}{r_i h_g} + \frac{r_o}{2k_f}\right] \qquad \text{Eq. 2}$$

and heat flux is calculated directly as:

$$q'' = U\Delta T = \frac{T_{\max} - T_b}{\frac{1}{h} + \frac{r_o \ln(r_o/r_i)}{k_c} + \frac{r_o}{r_i h_g} + \frac{r_o}{2k_f}}, \qquad (2)$$

in which r_o and r_i are cladding inner and outer radii, h_g is the gap conductivity, h is the convective heat transfer coefficient, and k_f is the fuel thermal conductivity. The gap conductivity of 2840 W m^{-2} K^{-1} (500 Btu h^{-1} ft^{-2} °F^{-1}) is taken from Appendix A. The convective heat transfer coefficient is mode dependent and is determined in context. Parameters are cross-referenced to source in Table 4.6.

Table 4.6: Thermodynamic Values

Parameter	Symbol	Value	Units	Reference
Fuel conductivity	k_f	18	W m^{-1} K^{-1}	Table 13.3
Clad conductivity	k_g	14.9	W m^{-1} K^{-1} (300 K)	Table 13.3
		16.6	W m^{-1} K^{-1} (400 K)	Table 13.3
		19.8	W m^{-1} K^{-1} (600 K)	Table 13.3
Gap resistance	h_g	2840	W m^{-2} K^{-1}	Appendix A
Clad outer radius	r_0	0.018161	M	Table 13.1
Fuel outer radius	r_i	0.018669	M	Table 13.1
Active fuel length	L_f	0.381	M	Table 13.1
No. fuel elements	N	83	N/A	Chap 13
Axial peaking factor	APF	$\pi/2$	N/A	Table 13.4

General Atomics reports that fuel conductivity over the range of interest has little temperature dependence, so that:

$$\frac{r_o}{2k_f} = 5.1858\text{E-}04\frac{\text{m}^2 K}{\text{W}}$$

Gap resistance has been experimentally determined as indicated, so that:

$$\frac{r_o}{r_i h_g} = 3.6196\text{E-}04\frac{\text{m}^2 K}{\text{W}}$$

Temperature change across the cladding is temperature dependent, with values quoted at 300 K, 400 K and 600 K. Under expected conditions, the value for 127°C applies so that:

$$\frac{r_o \ln \frac{r_o}{r_i}}{k_c} = 3.103e-5 \frac{m^2 K}{W}$$

Table 4.7, Cladding Heat Transfer Coefficient

Temp (°K)	Temp (°C)	m^2 K w^{-1}
300	27	3.457e-5
400	127	3.103e-5
600	327	2.601e-5

It should be noted that, since these values are less than 10% of the resistance to heat flow attributed to the other components, any errors attributed to calculating this factor are small.

The convection heat transfer coefficient was calculated at various steady state power levels. A graph of the calculated values results in a nearly linear response function.

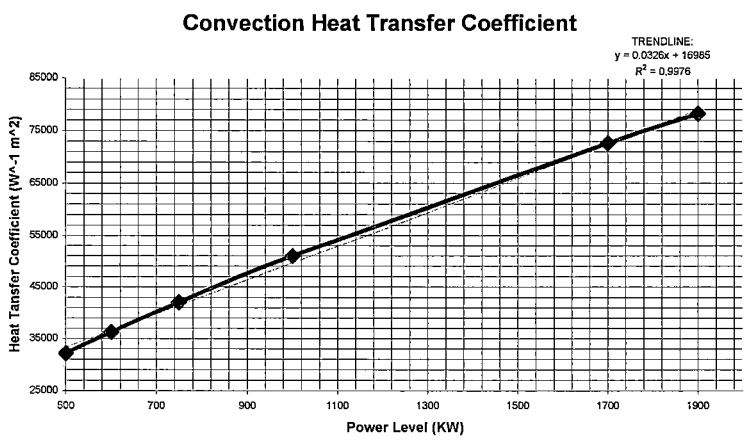


Figure 4.10, Convection Hear Transfer Coefficient versus Power Level

$$\frac{1}{h} = \frac{1}{0.0326P(watts) + 16985}$$

Core centerline temperature for the fuel rod producing the maximum heat as a function of power can be calculated as:

$$T_a = T_b + 0.423P\left[\frac{1}{0.0326P + 16985} + 3.103e-5 + 3.620e-4 + 5.186e-4\right] \quad (10)$$

c. Steady-State Mode of Operation

Centerline t~~T~~emperature calculations were performed on a "reference core" using the model as described above for the hottest location in the core were made. The reference core contains 83 fuel elements; temperature calculations using the reference core are conservative because at least 83 elements are required for steady state 500 kW operations, while analysis assumes 1.25 MW operation. A core with more than 83 elements will distribute heat production across a larger number of fuel elements, resulting in a lower heat flux per fuel rod than calculations based on the reference core. Since actual heat production will be less than heat calculated in analysis, actual temperatures will be lower. A power level of 1.25 MW steady state power at 20°C and 100°C was assumed with the following results:

Table 4.8, Calculated Temperature Data for 1,250 kW Operation

Fuel Centerline °C	Fuel/Gap Interface °C	Gap/Clad Interface °C	Clad/Water Interface °C	Bulk Water °C
503.2	229.0	37.7	21.2	20.0
582.0	307.8	116.4	100.0	100.0

For the purposes of calculation, the two extremes of cladding thermal conductivity were assumed (300 K value and 600 K value) to determine expected centerline temperature as a function of power level. Calculations show the effects of thermal conductivity changes are minimal. The graph also shows that fuel temperature remains below about 750 °C at power levels up to 1900 kW with pool temperature at 27 °C (300 K), and 1700 kW with pool temperatures at 100 °C.

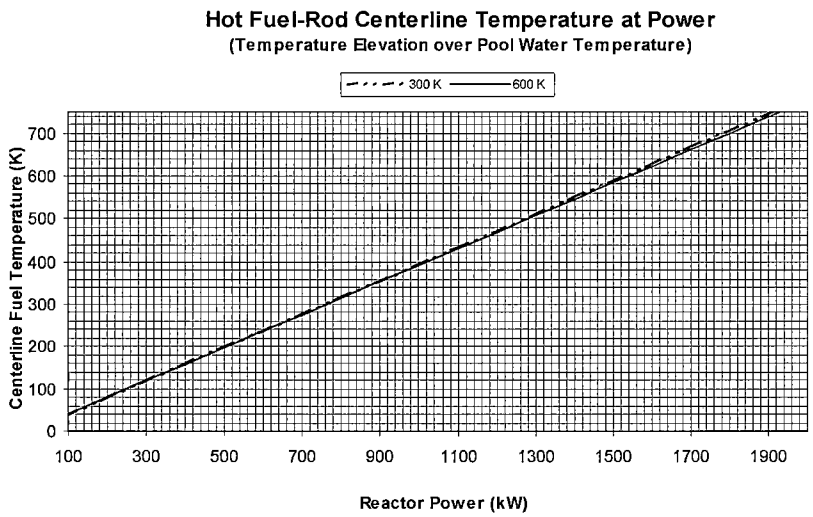


Figure 4.11, Hot Fuel-Rod Centerline Temperature

~~The margin to critical heat flux for the reference core was determined. Critical heat flux for saturated pool boiling is given by (*Heat Transfer*, A. Bejan, 1993, John Wiley & Sons):~~

$$\cancel{q''_{SAT} = 0.149 \cdot \rho_g^{0.5} \cdot \left(h_{g,sat} - h_{f,sat}\right) \cdot \left(g \cdot \sigma \cdot \left\{\rho_f - \rho_g\right\}\right)^{1/4}} \qquad \cancel{(3)}$$

~~where ρ_f is the density of the fluid, ρ_g is the density of the vapor, σ is the surface tension of the liquid phase in contact with vapor, $h_{f,sat}$ is the enthalpy of the saturated fluid, and $h_{g,sat}$ is the enthalpy of the vapor phase with all values at saturation conditions of temperature and pressure. Surface tension data provided by Bejan was fit to a polynomial (using temperature in °C) to generate data for the temperature range of interest, with an R^2 value of 0.999998:~~

—

$$\cancel{\sigma = 1.000E-11 \cdot T^4 + 7.370E-09 \cdot T^3 - 1.969E-06 \cdot T^2 + 4.709E-06 \cdot T + 7.1833E-02}$$

~~Pressure at the core is determined by barometric pressure at the facility elevation, vacuum maintained in the reactor bay and the weight of the water over the core. Barometric pressure associated with the Manhattan, Kansas airport is 29.92 in. Hg. The reactor bay is maintained at a slight vacuum, with the maximum gage pressure (6 in. of water) corresponding to -0.44 in Hg; nominal barometric pressure corrected for maximum reactor bay vacuum (a change of approximately 1.5%) corresponds to 99.83 kPa. Variations in local barometric pressure are on the order of the correction for reactor bay~~

~~pressure or less, so that the variations can be neglected without significant error. Normal pool level provides 16 feet of water over the core, contributing a nominal static pressure of 47.83 kPa. A pool level of 13 feet contributes a nominal static pressure of 38.58 kPa. (Vacuum breakers are installed in the cooling system piping about 3 feet below the surface of the pool water to limit potential siphoning of the pool.) Actual hydrostatic pressure of the pool water above the core is also determined by water density. With reactor bay atmospheric pressure at 99.827 kPa, water boils at 99.85°C. Therefore, values of thermodynamic properties as tabulated in Chapter 4 Appendix B were determined from the 1967 ASME (IFC) Steam Tables & IAPWS-IF97 with temperatures ranging from a maximum of 99°C to 15°C for pressures corresponding to 13 feet and 16 feet of water in the reactor pool.~~

~~For subcooled boiling, the critical heat flux is calculated by (Ivey and Morris 1978):~~

$$\cancel{q''_{sub} = q''_{SAT}\left(1 + 0.1\left(\frac{\rho_f}{\rho_g}\right)^{3/4} \frac{c_{p,f}\cdot\left(T_{SAT} - T_{sub}\right)}{h_{g,sat} - h_{f,sat}}\right)}, \quad \cancel{(4)}$$

~~where T_{sub} and $c_{p,f}$ correspond to the subcooled fluid. Specific heat capacities at the subcooled fluid temperatures and pressures evaluated above were determined from the 1967 ASME (IFC) Steam Tables & IAPWS-IF97 for the range of interest, as tabulated in Chapter 4 Appendix B. Critical heat flux calculations were compared to the actual heat flux previously calculated to determine the margin as a critical heat flux ratio (CHFR), provided below in Table 4.9.~~

~~Table 4.9, Critical Heat Flux Ratios (CHF versus Maximum Heat Flux) for 13 & 16 Feet of Water Over the Core~~

~~TEMP~~	~~CHFR (13 ft)~~	~~CHFR (16 ft)~~	~~AVE~~	~~Δ/AVE~~
~~15~~	~~6.58~~	~~6.84~~	~~6.71~~	~~3.86%~~
~~20~~	~~6.48~~	~~6.55~~	~~6.51~~	~~1.08%~~
~~25~~	~~6.18~~	~~6.26~~	~~6.22~~	~~1.20%~~
~~30~~	~~5.88~~	~~5.96~~	~~5.92~~	~~1.32%~~
~~35~~	~~5.59~~	~~5.67~~	~~5.63~~	~~1.45%~~
~~40~~	~~5.29~~	~~5.37~~	~~5.33~~	~~1.61%~~
~~45~~	~~4.99~~	~~5.08~~	~~5.03~~	~~1.77%~~
~~50~~	~~4.69~~	~~4.78~~	~~4.74~~	~~1.95%~~
~~55~~	~~4.39~~	~~4.49~~	~~4.44~~	~~2.16%~~
~~60~~	~~4.10~~	~~4.20~~	~~4.15~~	~~2.40%~~
~~65~~	~~3.80~~	~~3.91~~	~~3.86~~	~~2.67%~~
~~70~~	~~3.51~~	~~3.62~~	~~3.57~~	~~2.96%~~
~~75~~	~~3.22~~	~~3.33~~	~~3.28~~	~~3.31%~~
~~80~~	~~2.94~~	~~3.05~~	~~2.99~~	~~3.73%~~
~~85~~	~~2.66~~	~~2.76~~	~~2.71~~	~~3.98%~~
~~90~~	~~2.37~~	~~2.49~~	~~2.43~~	~~4.83%~~

~~**Table 4.9, Critical Heat Flux Ratios (CHF versus Maximum Heat Flux) for 13 & 16 Feet of Water Over the Core**~~

~~TEMP~~	~~CHFR (13 ft)~~	~~CHFR (16 ft)~~	~~AVE~~	~~Δ/AVE~~
~~95~~	~~2.09~~	~~2.21~~	~~2.15~~	~~5.60%~~
~~97~~	~~1.97~~	~~2.09~~	~~2.03~~	~~5.96%~~
~~99~~	~~1.86~~	~~1.98~~	~~1.92~~	~~6.35%~~

~~As indicated in Table 4.9, the actual heat flux is less than the critical heat flux for operating temperatures up to 55 °C by more than a factor of 4 considering both 13 feet and 16 feet of water above the core. The CHFR is greater than 2 for pool temperatures exceeding the maximum operating value up to 95 °C, and remains near 2 at values up to 99°C. The difference in the critical heat flux ratio for 13 and 16 feet of water is relatively small, with a minimum difference compared to the mean of the two values of 1.8% and a maximum of 3.86% below 60 °C, 6.4% across all pool temperatures considered.~~

~~It is clear from the table that there is a very wide margin between the operating heat flux and the critical heat flux even to unrealistically high pool water temperature, so that film boiling and excessive cladding temperature is not a consideration in steady-state operation.~~

For the analysis of critical heat flux, a single channel model was built in RELAP-5/MOD 3.3 patch 04 (Feldman 2008). A snapshot of the model is presented in Figure 4.12. It has two time-dependent volumes, enforcing the pressure boundary conditions, and two pipes, simulating the cold and hot channel connected via a single junction component of RELAP. Heat is added to the fluid by incorporating the heat structure component (simulating a fuel element) of RELAP with an appropriate axial power profile and power level. In this analysis, the power level for the B ring is at 24 kW (corresponding to an 85-element core with a ring-to-average peaking factor of 1.63). This power level is applied to the heat structure within the single channel. The model assumes an operating pressure of 143 kPa, and an operating temperature of 322.15 K (49.15°C).

The version of the RELAP code licensed to KSU uses PG-CHF correlation which is a state of the art best estimate CHF correlation developed by Nuclear Research institute of Rez in the Czech Republic. It is based on data in the Czech Republic data bank from 173 different sets of tube data, 23 sets of annular data, and 153 sets of rod bundle data. There are four forms of the PG-CHF correlation `Basic', `Flux', `Geometry', and `Power'. For the rod bundle it is applicable in the pressure range of 0.28 MPa to 18.73 MPa, for a mass flux of 34.1 to 7478 kg/s-m2, for 0.4-7.0 m length and for a diameter of 0.00241 to 0.07813 m. TRIGA has an operating pressure of 0.143 MPa and fuel rod length of 0.381 m, thus the operating conditions fall outside the range of the applicability of the PG-CHF correlation, and a different correlation is required to assess the departure from nucleate boiling ratio (DNBR ratio). One such correlation which is applicable for the low pressure range observed in TRIGA reactor facility is the Bernath correlation. The functional form of the Bernath correlation can be presented in the following equations.

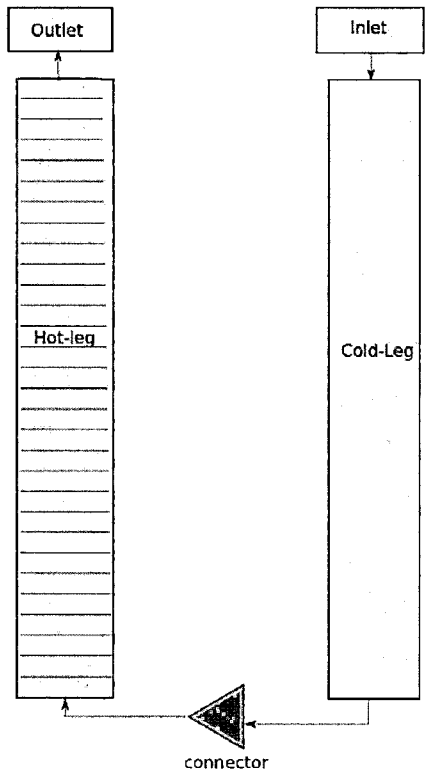


Figure 4.12 - RELAP single channel model used in CHF analysis

$$Q''_{BO} = h_{BO}\left(T_{wBO} - T_b\right) \quad (8)$$

$$h_{BO} = 10890\left(\frac{D_h}{D_h + D_H}\right) + \Delta v \quad (8)$$

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$$\Delta = \frac{48}{D_h^{0.6}}, \text{if } D_h \le 0.1 ft$$

$$\Delta = \frac{10}{D_h} + 90, \text{if } D_h \ge 0.1 ft$$

h_{BO} = film coefficient at CHF

D_h = hydraulic diameter (ft)

v = coolant velocity (ft / s)

T_{wBO} = wall temperature at burnout (°C)

D_H = heated diameter (ft)

The RELAP simulations were performed for the hot channel, i.e., a channel with a radial peaking factor of 1.63, assuming an 85-element core load and a power of 1.25 MWth, in order to obtain the pressure, temperature, and velocity distribution at different axial locations. With these calculations and the functional form of the Bernath correlation, the axial distribution of CHF was estimated in the hot channel. The methodology adopted for this analysis is described in literature (Feldman 2008). The hot channel model was based on the smallest hydraulic diameter in the core (between the A-ring and two B-ring elements) and the highest radial peaking factor. In the KSU TRIGA, the A-ring is occupied by the central thimble, not a fuel element. Since the actual hot channel would be between two B-ring elements and a C-ring element, the real hydraulic diameter will be slightly larger and the real heat flux into the channel will be slightly lower than the values assumed in the model. Therefore, this model is conservative in this regard.

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The axial CHF results from the PG and Bernath heat flux models are shown in Figure 4.13 and Figure 4.14. The DNBR ratio exceeds 2.0 for all locations along the heated length of the hot channel.

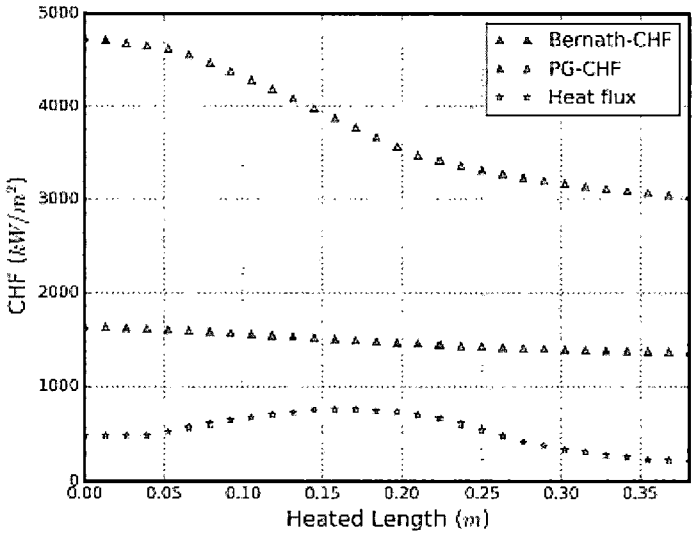

Figure 4.13 - CHF versus heated length

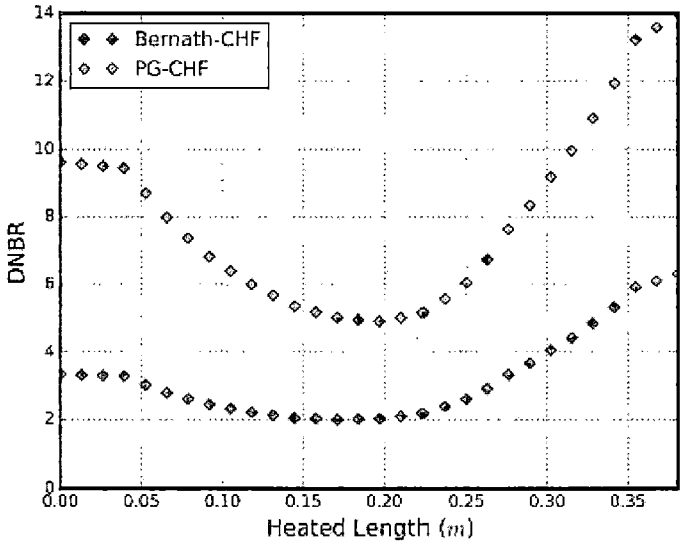

Figure 4.14 - DNBR versus heated length

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d. Pulsed Mode of Operation

Transient calculations have been performed using a custom computer code TASCOT for transient and steady state two-dimensional conduction calculations (Ahrens 1999). For these calculations, the initial axial and radial temperature distribution of fuel temperature was based on Eqs. (~~6~~9) and (~~7~~10), with the peak fuel temperature set to 746 °C, i.e., a temperature rise of 719 °C above 27 °C ambient temperature. The temperature rise is computed in Chapter 13, Section 13.2.3 for a 2.1% ($3.00) pulse from zero power and a 0.7% ($1.00) pulse from power operation. In the TASCOT calculations, thermal conductivity was set to 0.18 W cm^{-1} K^{-1} (Table 4.1) and the overall heat transfer coefficient U was set to 0.21 W cm^{-1} K^{-1}. The convective heat transfer coefficient was based on the boiling heat transfer coefficient computed using the formulation (Chen 1963, Collier and Thome 1994)

$$q'' = h_b(T_w - T_{sat}) = h(T_w - T_b). \qquad (\cancel{8}9)$$

The boiling heat transfer coefficient is given by the correlation (Forster & Zuber 1955)

$$h_b = 0.00122 * \left[\frac{k_f^{0.79} * c_{pf}^{0.45} * \lambda^{0.51}}{\sigma^{0.5} * \mu_f^{0.29} * \rho_g^{0.24} * \left(v_g - v_v\right)^{0.75} * T_{sat}^{0.75}} \right] * \left(T_w - T_{sat}\right)^{0.99}, \qquad (\cancel{9}10)$$

in which T_w is the cladding outside temperature, T_{sat} the saturation temperature (111.9 °C), and T_b the coolant ambient temperature (27°C). Fluid-property symbols and values are given in Appendix B. Subscripts f and g refer respectively to liquid and vapor phases. The overall heat transfer coefficient U varies negligibly for ambient temperatures from 20 to 60 °C, and has the value 0.21 W cm^{-1} K^{-1} at T_b = 27 °C.

Figure 4.14~~5~~ illustrates the radial variation of temperature within the fuel, at the midplane of the core, as a function of time after the pulse. Table 4.1~~0~~8 lists temperatures and heat fluxes as function of time after a 2.1% ($3.00) reactivity insertion in a reactor initially at zero power. The CHFR is based on the critical heat flux of 1.49 MW m^{-1} from Eqs. (3) and (4) and from Table 4.2 for saturated boiling. Figure 4A.3 of Appendix A, using the Ellion data, indicates a Leidenfrost temperature in excess of 500°C. Thus transition boiling, but not fully developed film boiling might be expected for a short time after the end of a pulse.

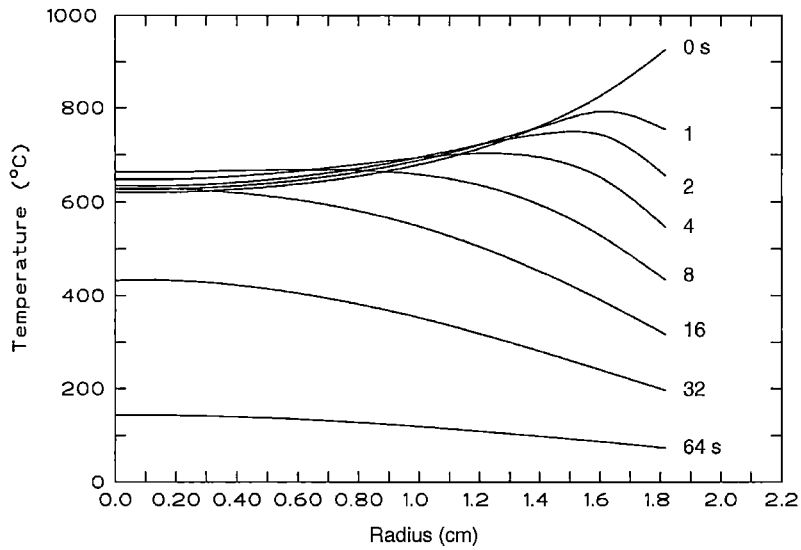

Figure 4. 145, Midplane Radial Variation of Temperature Within the Fuel Subsequent to a $3.00 Pulse.

Table 4.10, Heat Flux and Fuel Temperatures Following a $3.00 Pulse from Zero Power, with 27(°C) Coolant Ambient Temperature.

Time (s)	Q'' (W m^{-2})	CHFR	Fuel outside Temp. (°C)	Clad surface Temp. (°C)
0	-	-	953	-
1	3.57×10^5	4.2	781	224
2	7.34×10^5	2.0	683	432
4	8.52×10^5	1.7	574	498
8	7.54×10^5	2.0	461	443
16	5.71×10^5	2.6	344	342
32	3.46×10^5	4.3	224	218
64	1.04×10^5	14.4	100	84

4.6 Thermal Hydraulic Design and Analysis

A balance between the buoyancy driven pressure gain and the frictional and acceleration pressure losses accrued by the coolant in its passage through the core determines the coolant mass flow rate through the core, and the corresponding coolant temperature rise. The buoyancy pressure gain is given by

$$\Delta p_g = \rho_o \beta_o \Delta T g L, \qquad (\cancel{10}11)$$

in which ρ_o and β_o are the density and volumetric expansion coefficient at core inlet conditions (27°C, 0.15285 Mpa), g is the acceleration of gravity, 9.8 cm^2 s^{-1}, ΔT is the temperature rise through the core, and L is the height of the core (between gridplates), namely, 0.556 m. The frictional pressure loss is given by

$$\Delta p_f = \frac{\dot{m}^2 f L}{2 A^2 D_h \rho_o}, \qquad (\cancel{11}12)$$

in which $\dot{m}$ is the coolant mass flow rate (kg s^{-1}) in a unit cell approximated as the equivalent annulus surrounding a single fuel element, A is the flow area, namely, 0.00062 m^2, and D_h is the hydraulic diameter, namely, 0.02127 m. The friction factor f for laminar flow through the annular area is given by 100 Re^{-1} (Shah & London 1978), in which the Reynolds number is given by $D_h \dot{m} / A \mu_o$ in which μ_o is the dynamic viscosity at core inlet conditions.

Entrance of coolant into the core is from the side, above the lower grid plate (see Section 4.2.5), and the entrance pressure loss would be expected to be negligible. The exit contraction loss is given by

$$\Delta p_e = \frac{\dot{m}^2 K}{2 \rho_o A^2}. \qquad (\cancel{12}13)$$

The coefficient K is calculated from geometry of an equilateral-triangle spacer in a circular opening, for which

$$K \cong \left[\frac{A_t}{A_c}\right]^2 = \left[\frac{3 * R^2 \sin 60^o \cos 60^o}{\pi * R^2}\right] = 0.171, \quad (\cancel{13}14)$$

where R is the radius of the opening in the upper grid plate. Equations (~~11~~12) through (~~13~~14), solved simultaneously yield the mass flow rates per fuel element, and coolant temperature rises through the core listed in Table 4.~~9~~11.

Table 4.11, Coolant Flow Rate and Temperature Rise for Natural-Convection Cooling the TRIGA Reactor During Steady-State Operations.

P (kWt)	$\dot{m}$ (kg s^{-1})	ΔT (°C)
50	0.047	3.1
100	0.061	4.7
200	0.077	7.5
300	0.090	9.6
400	0.100	11.5
500	0.108	13.3
750	0.125	17.2
1000	0.139	20.6
1250	0.150	23.8

4.7 Safety Limit

As described in 3.5.1 (Fuel System) and NUREG 1282, fuel temperature limits both steady-state and pulse-mode operation. The fuel temperature limit stems from potential hydrogen outgassing from the fuel and the subsequent stress produced in the fuel element clad material by heated hydrogen gas. Yield strength of cladding material decreases at a temperature of 500°C; consequently, limits on fuel temperature change for cladding temperatures greater than 500°C. A maximum temperature of 1150°C (with clad < 500°C) and 950°C (with clad > 500°C) for U-ZrH ($H/Zr_{1.65}$) will limit internal fuel cladding stresses that might lead to clad integrity (NUREG 1282) challenges.

4.8 Operating Limits

4.8.1 Operating Parameters

The main safety consideration is to maintain the fuel temperature below the value that would result in fuel damage. Setting limits on other operating parameters, that is, limiting safety system settings, controls the fuel temperature. The operating parameters established for the KSU TRIGA reactor are:

- Steady-state power level
- Fuel temperature measured by thermocouple during pulsing operations
- Maximum step reactivity insertion of transient rod

4.8.2 Limiting Safety System Settings

Heat transfer characteristics (from the fuel to the pool) controls fuel temperature during normal operations. As long as thermal hydraulic conditions do not cause critical heat flux to be exceeded, fuel temperature remains well below any limiting value. Figure 4.13 illustrates that critical heat flux is not reached over a wide range of pool temperatures and power levels. As indicated in ~~Table 4.9~~Figure 4.14, the ratio of actual to critical heat flux is at least 2.0 for temperatures less than 100°C bulk pool water temperature for 1.25 MW operation. Operation at less than 1.25 MW ensures fuel temperature limits are not exceeded by a wide margin.

Limits on the maximum excess reactivity assure that operations during pulsing do not produce a power level (and generate the amount of energy) that would cause fuel-cladding temperature to exceed these limits; no other safety limit is required for pulsed operation.

4.8.3 Safety Margins

~~For 1,250 kW steady-state operations, the critical heat flux ratio indicated in Table 4.9 ranges from 5.8 for pool water at room temperature (27°C) to 4.1 at 60 °C (pool temperatures are controlled to less than 48°C for operational concerns). Even at pool water temperatures approaching boiling, the margin remains above 2. Therefore, margins to conditions that could cause excessive temperatures during steady state operations while cladding temperatures is below 500°C are extremely large.~~
For 1,250 kWth steady-state operations, the critical heat flux ratio remains above 2.0 for a core with 85 fuel elements and a maximum radial power peaking factor of 1.63 assuming a coolant inlet temperature of 49°C. The proposed Technical Specifications limit of 44°C on pool inlet temperature ensures that the DNBR will be at least 2.0 during steady-state operation. Limiting pool inlet water temperature to no greater than 44°C (or 37°C with an experiment installed in an interstitial flux-wire port) will ensure that the pool water does not reach temperatures associated with excessive amounts of nucleate boiling.

Normal pulsed operations initiated from power levels below 10 kW with a $3.00 reactivity insertion result in maximum hot spot temperatures of 746°C, a 34% margin to the fuel temperature limit. As indicated in Chapter 13, pulsed reactivity insertions of $3.00 from initial conditions of power operation can result in a maximum hot spot temperature of 869°C. Although administratively controlled and limited by an interlock, this pulse would still result in a 15% margin to the fuel temperature safety limit for cladding temperatures below 500°C.

Analysis shows that cladding temperatures will remain below 500°C when fuel is in water except following large pulses. However, mechanisms that can cause cladding temperature to achieve

500°C (invoking a 950°C fuel temperature limit) automatically limit fuel temperature as heat is transferred from the fuel to the cladding.

Immediately following a maximum pulsed reactivity additions, heat transfer driven by fuel temperature can cause cladding temperature to rise above 500°C, but the heat transfer simultaneously cools the fuel to much less than 950°C.

If fuel rods are placed in an air environment immediately following long-term, high power operation, cladding temperature can essentially equilibrate with fuel temperature. In worst-case air-cooling scenarios, cladding temperature can exceed 500°C, but fuel temperature is significantly lower than the temperature limit for cladding temperatures greater than 500°C.

4.9 Bibliography

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Appendix 4-A
Post-Pulse Fuel and Cladding Temperature

This discussion is reproduced from Safety Analysis Reports for the University of Texas Reactor Facility (UTA 1991) and the McClellan Nuclear Radiation Center (MNRC 1998).

The following discussion relates the element clad temperature and the maximum fuel temperature during a short time after a pulse. The radial temperature distribution in the fuel element immediately following a pulse is very similar to the power distribution shown in Figure 4A.1. This initial steep thermal gradient at the fuel surface results in some heat transfer during the time of the pulse so that the true peak temperature does not quite reach the adiabatic peak temperature. A large temperature gradient is also impressed upon the clad which can result in a high heat flux from the clad into the water. If the heat flux is sufficiently high, film boiling may occur and form an insulating jacket of steam around the fuel elements permitting the clad temperature to tend to approach the fuel temperature. Evidence has been obtained experimentally which shows that film boiling has occurred occasionally for some fuel elements in the Advanced TRIGA Prototype Reactor located at GA Technologies [Coffer 1964]. The consequence of this film boiling was discoloration of the clad surface.

Thermal transient calculations were made using the RAT computer code. RAT is a 2-D transient heat transport code developed to account for fluid flow and temperature dependent material properties. Calculations show that if film boiling occurs after a pulse it may take place either at the time of maximum heat flux from the clad, before the bulk temperature of the coolant has changed appreciably, or it may take place at a much later time when the bulk temperature of the coolant has approached the saturation temperature, resulting in a markedly reduced threshold for film boiling. Data obtained by Johnson et al. [1961] for transient heating of ribbons in 100°F water, showed burnout fluxes of 0.9 to 2.0 Mbtu ft^{-2} hr^{-1} for e-folding periods from 5 to 90 milliseconds. On the other hand, sufficient bulk heating of the coolant channel between fuel elements can take place in several tenths of a second to lower the departure from nucleate boiling (DNB) point to approximately 0.4 Mbtu ft^{-2} hr^{-1}. It is shown, on the basis of the following analysis, that the second mode is the most likely; i.e., when film boiling occurs it takes place under essentially steady-state conditions at local water temperatures near saturation.

A value for the temperature that may be reached by the clad if film boiling occurs was obtained in the following manner. A transient thermal calculation was performed using the radial and axial power distributions in Figures 4A.1and 4A.2, respectively, under the assumption that the thermal resistance at the fuel-clad interface was nonexistent. A boiling heat transfer model, as shown in Figure 4A.3, was used in order to obtain an upper limit for the clad temperature rise. The model used the data of McAdams [1954] for subcooled boiling and the work of Sparrow and Cess [1962] for the film boiling regime. A conservative estimate was obtained for the minimum heat flux in film boiling by using the correlations of Speigler et al. [1963], Zuber [1959], and Rohsenow and Choi [1961] to find the minimum temperature point at which film boiling could occur. This calculation gave an upper limit of 760°C clad temperature for a peak initial fuel temperature of 1000°C, as shown in Figure. 4A.4. Fuel temperature distributions for this case are shown in Figure 4A.5 and the heat flux into the water from the clad is shown in Figure 4A.6. In this limiting case, DNB occurred only 13 milliseconds after the pulse, conservatively calculated

assuming a steady-state DNB correlation. Subsequently, experimental transition and film boiling data were found to have been reported by Ellion [9] for water conditions similar to those for the TRIGA system. The Ellion data show the minimum heat flux, used in the limiting calculation described above, was conservative by a factor of 5. An appropriate correction was made which resulted in a more realistic estimate of 470°C as the maximum clad temperature expected if film boiling occurs. This result is in agreement with experimental evidence obtained for clad temperatures of 400°C to 500°C for TRIGA Mark F fuel elements which have been operated under film boiling conditions [Coffer et al. 1965].

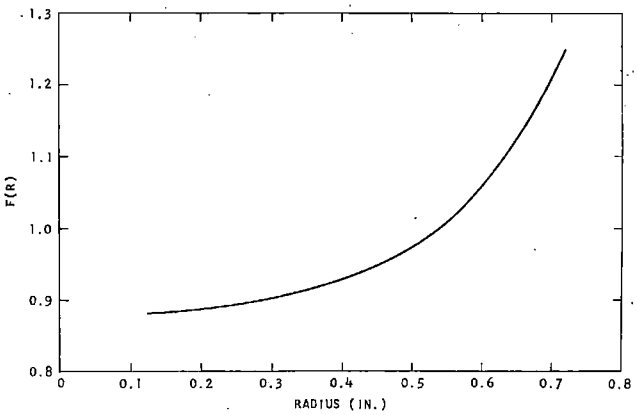

Figure 4A.1. Representative Radial Variation of Power Within the TRIGA Fuel Rod

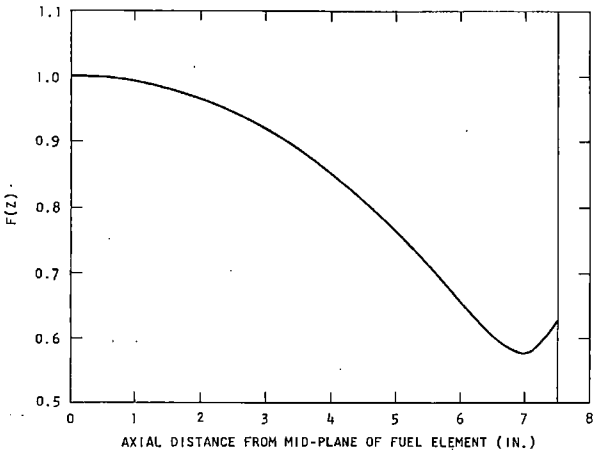

Figure 4A.2, Representative Axial Variation of Power Within the TRIGA Fuel Rod.

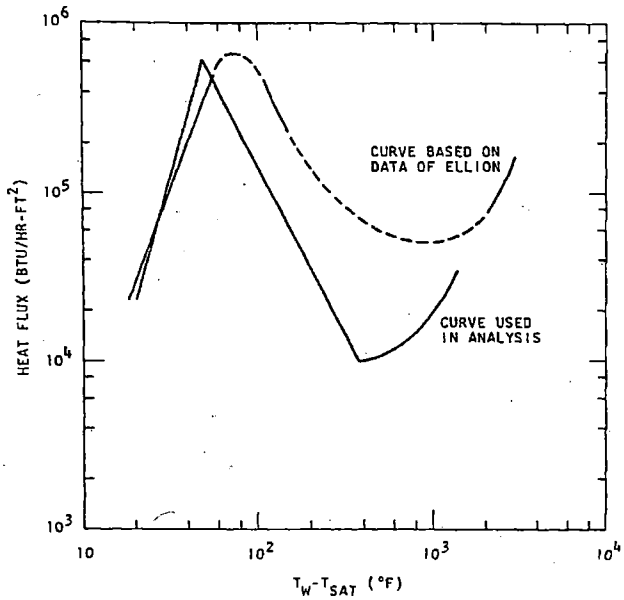

Figure 4A.3, Subcooled Boiling Heat Transfer for Water.

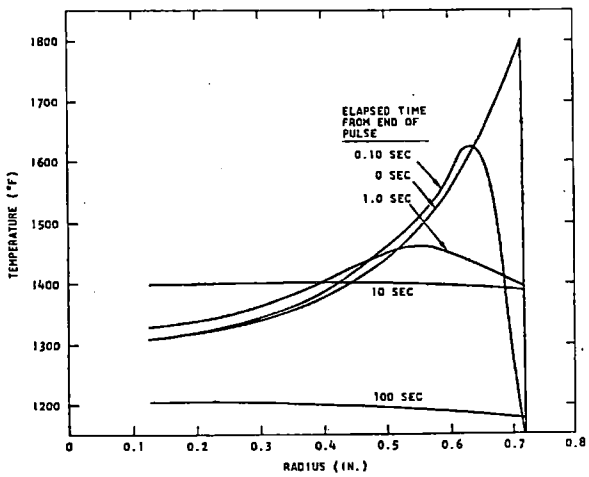

Figure 4A.4, Fuel Body Temperature at the Midplane of a Well-Bonded Fuel Element After Pulse.

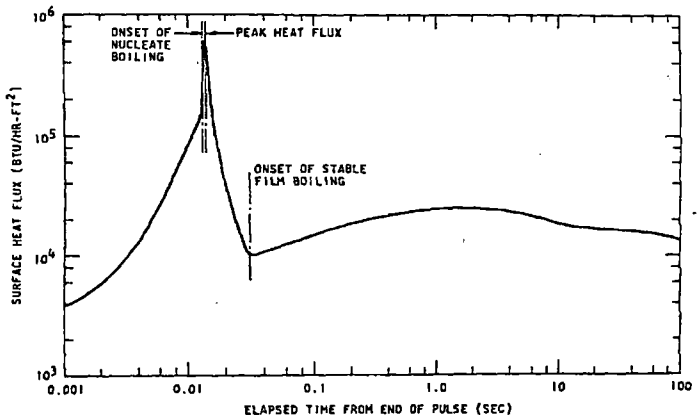

Figure 4A.5, Surface Heat Flux at the Midplane of a Well Bonded Fuel Element After a Pulse.

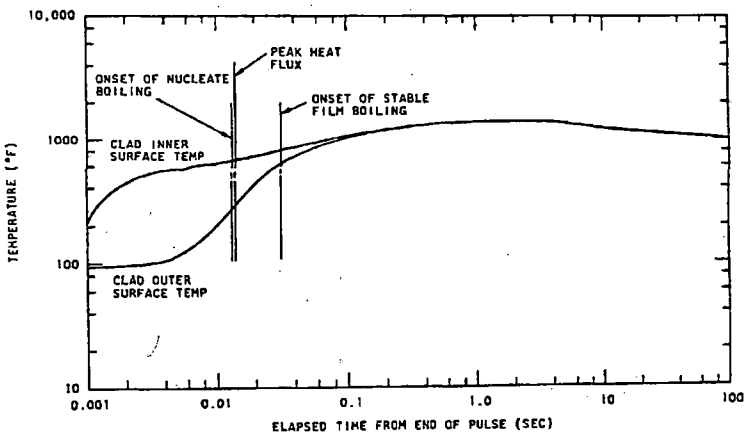

Figure 4A.6, Clad Temperature at Midpoint of Well-Bonded Fuel Element.

The preceding analysis assessing the maximum clad temperatures associated with film boiling

assumed no thermal resistance at fuel-clad interface. Measurements of fuel temperatures as a function of steady-state power level provide evidence that after operating at high fuel temperatures, a permanent gap is produced between the fuel body and the clad by fuel expansion. This gap exists at all temperatures below the maximum operating temperature. (See, for example, Figure 16 in the Coffer report [1965].) The gap thickness varies with fuel temperature and clad temperature so that cooling of the fuel or overheating of the clad tends to widen the gap and decrease the heat transfer rate. Additional thermal resistance due to oxide and other films on the fuel and clad surfaces is expected. Experimental and theoretical studies of thermal contact resistance have been reported [Fenech and Rohsenow 1959, Graff 1960, Fenech and Henry 1962] which provide insight into the mechanisms involved. They do not, however, permit quantitative prediction of this application because the basic data required for input are presently not fully known. Instead, several transient thermal computations were made using the RAT code. Each of these was made with an assumed value for the effective gap conductance, in order to determine the effective gap coefficient for which departure from nucleate boiling is incipient. These results were then compared with the incipient film boiling conditions of the 1000°C peak fuel temperature case.

For convenience, the calculations were made using the same initial temperature distribution as was used for the preceding calculation. The calculations assumed a coolant flow velocity of 1 ft per second, which is within the range of flow velocities computed for natural convection under various steady-state conditions for these reactors. The calculations did not use a complete boiling curve heat transfer model, but instead, included a convection cooled region (no boiling) and a subcooled nucleate boiling region without employing an upper DNB limit. The results were analyzed by inspection using the extended steady-state correlation of Bernath [1960] which has been reported by Spano [1964] to give agreement with SPERT II burnout results within the experimental uncertainties in flow rate.

The transient thermal calculations were performed using effective gap conductances of 500, 375, and 250 Btu ft^{-2} hr^{-1} $°F^{-1}$. The resulting wall temperature distributions were inspected to determine the axial wall position and time after the pulse which gave the closest approach between the local computed surface heat flux and the DNB heat flux according to Bernath. The axial distribution of the computed and critical heat fluxes for each of the three cases at the time of closest approach is given in Figures 4A.7 through 4A.9. If the minimum approach to DNB is corrected to TRIGA Mark F conditions and cross-plotted, an estimate of the effective gap conductance of 450 Btu ft^{-2} hr^{-1} $°F^{-1}$ is obtained for incipient burnout so that the case using 500 is thought to be representative of standard TRIGA fuel.

The surface heat flux at the midplane of the element is shown in Figure 4A.10 with gap conductance as a parameter. It may be observed that the maximum heat flux is approximately proportional to the heat transfer coefficient of the gap, and the time lag after the pulse for which the peak occurs is also increased by about the same factor. The closest approach to DNB in these calculations did not necessarily occur at these times and places, however, as indicated on the curves of Figures 4A.7 through 4A.9. The initial DNB point occurred near the core outlet for a local heat flux of about 340 kBtu ft^{-2} hr^{-1} $°F^{-1}$ according to the more conservative Bernath correlation at a local water temperature approaching saturation.

This analysis indicates that after operation of the reactor at steady-state power levels of 1 MW(t), or after pulsing to equivalent fuel temperatures, the heat flux through the clad is reduced and therefore reduces the likelihood of reaching a regime where there is a departure from nucleate boiling. From the foregoing analysis, a maximum temperature for the clad during a pulse which gives a peak adiabatic fuel temperature of 1000°C is conservatively estimated to be 470°C.

As can be seen from Figure 4.7, the ultimate strength of the clad at a temperature of 470°C is 59,000 psi. If the stress produced by the hydrogen over pressure in the can is less than 59,000 psi, the fuel element.will not undergo loss of containment. Referring to Figure 4.8, and considering U-ZrH fuel with a peak temperature of 1000°C, one finds the stress on the clad to be 12,600 psi. Further studies show that the hydrogen pressure that would result from a transient for which the peak fuel temperature is 1150°C would not produce a stress in the clad in excess of its ultimate strength. TRIGA fuel with a hydrogen to zirconium ratio of at least 1.65 has been pulsed to temperatures of about 1150°C without damage to the clad [Dee et al. 1966].

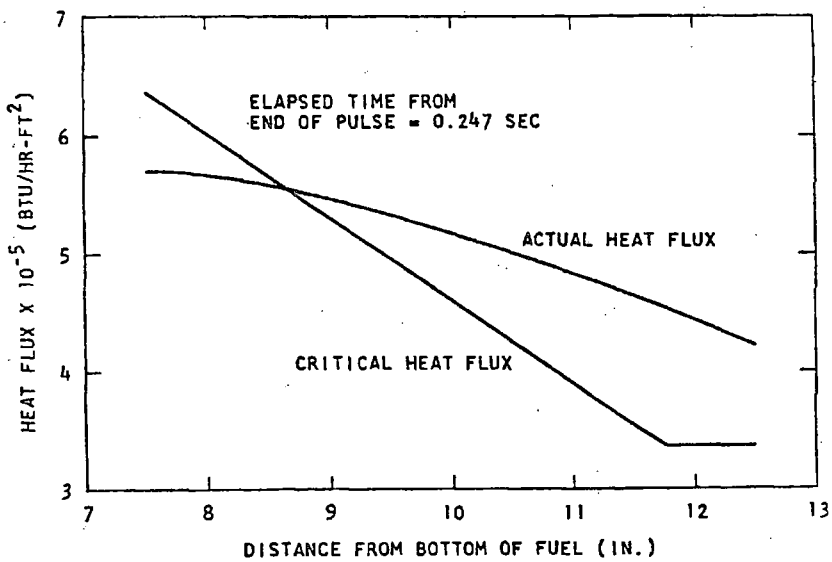

Figure 4A.7, Surface Heat Flux Distribution for Standard Non-Gapped (h_{gap} = 500 Btu/h ft^2 °F) Fuel Element After a Pulse.

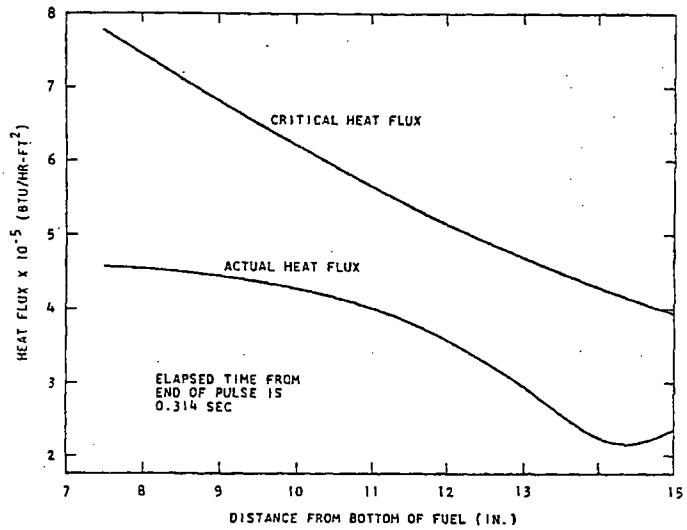

Figure 4A.8, Surface Heat-Flux Distribution for Standard Non-Gapped Fuel Element (h_{gap} = 375 Btu/h ft^2 °F) After a Pulse.

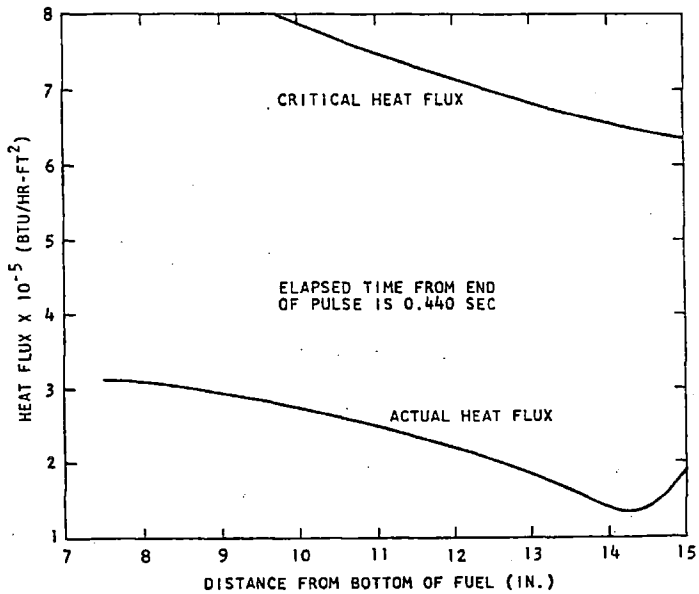

Figure 4A.9, Surface Heat-Flux Distribution for Standard Non-Gapped Fuel Element (h_{gap} = 250 Btu/h ft^2 °F) After a Pulse.

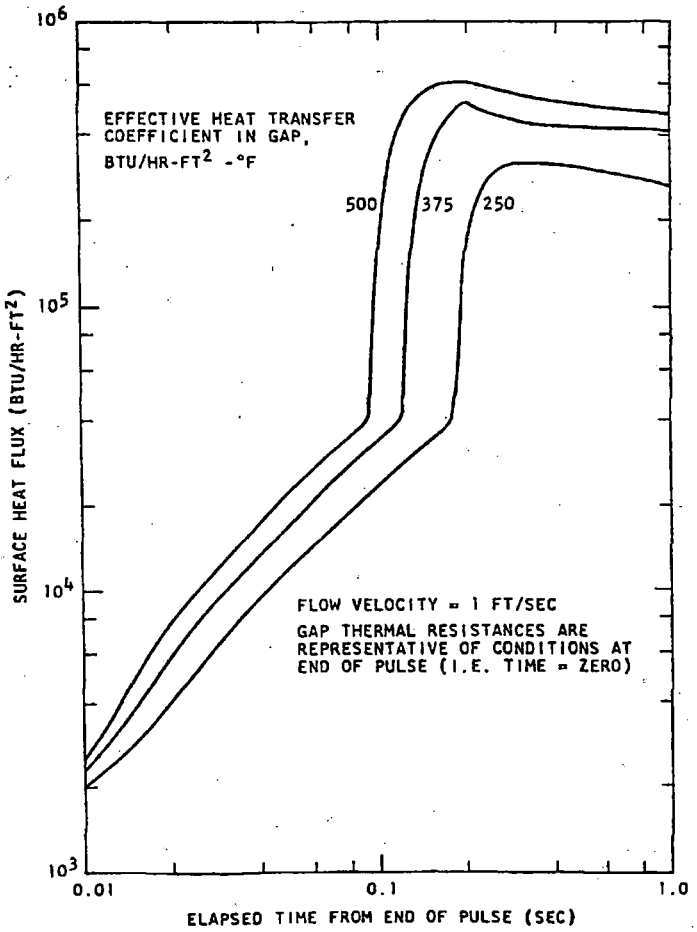

Figure 4A.10, Surface Heat Flux at Midpoint vs. Time for Standard Non-Gapped Fuel Element After a Pulse.

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Appendix B
Water Properties at Nominal Operating Conditions

Data for 16 Feet of Water over the Core

T_{pool} °C	ρ_T[1] kg m^{-3}	$\rho_{f,16}$[1] kg m^{-3}	$P_{h,16}$[2] kPa	$h_{f,16}$[1] kJ kg^{-1}	$h_{g,16}$[1] kJ kg^{-1}	$\rho_{g,16}$[1] kg m^{-3}	$T_{sat,16}$[1] °C	q"$_{sat,16}$[3] W m^{-2}	q"sub[4] W m^{-2}
15	999.21	950.00	47.79	465.10	2692.64	0.85	110.89	1553.842	7239.19
20	998.32	950.01	47.74	465.05	2692.63	0.85	110.88	1552.078	6931.74
25	997.16	950.02	47.69	465.01	2692.59	0.85	110.87	1549.496	6622.60
30	995.75	950.03	47.62	464.95	2692.59	0.85	110.86	1547.118	6311.82
35	994.12	950.04	47.54	464.89	2692.57	0.85	110.84	1543.981	5999.91
40	992.29	950.06	47.46	464.81	2692.54	0.85	110.83	1540.446	5688.24
45	990.27	950.07	47.36	464.73	2692.51	0.85	110.81	1536.512	5376.29
50	988.07	950.09	47.25	464.64	2692.48	0.85	110.78	1532.205	5064.36
55	985.70	950.11	47.14	464.54	2692.45	0.85	110.76	1527.561	4753.90
60	983.18	950.12	47.02	464.44	2692.41	0.85	110.74	1522.575	4444.85
65	980.50	950.14	46.89	464.33	2692.37	0.85	110.71	1517.255	4136.85
70	977.69	950.17	46.76	464.21	2692.33	0.84	110.68	1511.666	3830.73
75	974.74	950.19	46.62	464.09	2692.29	0.84	110.65	1505.778	3526.89
80	971.66	950.21	46.47	463.96	2692.24	0.84	110.62	1499.613	3225.47
85	968.45	950.23	46.32	463.83	2692.19	0.84	110.59	1493.199	2926.81
90	965.12	950.26	46.16	463.69	2692.15	0.84	110.56	1486.527	2631.05
95	961.68	950.29	45.99	463.55	2692.09	0.84	110.53	1479.626	2338.47
97	960.27	950.30	45.92	463.49	2692.07	0.84	110.51	1472.944	2216.11
99	958.84	950.31	45.86	463.43	2692.05	0.84	110.50	1466.058	2095.18

Data for 13 Feet of Water over the Core

T_{pool} °C	ρ_T[1] kg m^{-3}	$\rho_{f,13}$[1] kg m^{-3}	$P_{h,13}$[2] kPa	$h_{f,13}$[1] kJ kg^{-1}	$h_{g,13}$[1] kJ kg^{-1}	$\rho_{g,13}$[1] kg m^{-3}	$T_{sat,13}$[1] °C	q"$_{sat,13}$[3] W m^{-2}	q"sub[4] W m^{-2}
15	999.21	951.43	38.83	457.21	2689.85	0.80	109.03	1513.00	6964.74
20	998.32	951.43	38.79	457.18	2689.84	0.80	109.02	1511.32	6857.12
25	997.16	951.44	38.75	457.13	2689.82	0.80	109.01	1509.15	6543.62
30	995.75	951.45	38.69	457.09	2689.80	0.80	109.00	1505.85	6229.30
35	994.12	951.46	38.63	457.03	2689.78	0.80	108.99	1503.13	5913.58
40	992.29	951.47	38.56	456.96	2689.76	0.80	108.97	1500.16	5597.21
45	990.27	951.49	38.48	456.89	2689.74	0.80	108.96	1496.38	5281.90
50	988.07	951.50	38.39	456.82	2689.71	0.80	108.94	1492.25	4966.66
55	985.70	951.51	38.30	456.73	2689.68	0.80	108.92	1487.78	4652.39
60	983.18	951.53	38.20	456.64	2689.65	0.80	108.90	1482.99	4339.60
65	980.50	951.55	38.10	456.55	2689.61	0.80	108.87	1477.90	4027.94
70	977.69	951.57	37.99	456.45	2689.58	0.80	108.85	1472.52	3718.83
75	974.74	951.58	37.88	456.35	2689.54	0.80	108.83	1466.86	3412.07
80	971.66	951.60	37.76	456.24	2689.50	0.80	108.80	1460.95	3107.29
85	968.45	961.62	37.63	456.12	2689.46	0.80	108.77	1458.59	2812.63
90	965.12	951.64	37.50	456.01	2689.42	0.79	108.75	1448.37	2506.90
95	961.68	951.67	37.37	455.89	2689.38	0.79	108.72	1441.74	2211.19
97	960.27	951.68	37.31	455.84	2689.36	0.79	108.70	1435.27	2087.92
99	958.84	951.68	37.26	455.78	2689.34	0.79	108.69	1428.60	1966.16

Common Data

T °C	P_{atm} kPa	c_p[1] kJ kg^{-1} k^{-1}	σ N m^{-1}
15	99.83	4.23080	0.07149
20	99.83	4.23080	0.07120
25	99.83	4.23080	0.07083
30	99.83	4.23080	0.07039
35	99.83	4.23070	0.06989
40	99.83	4.23070	0.06932
45	99.83	4.23070	0.06869
50	99.83	4.23070	0.06800
55	99.83	4.23060	0.06727
60	99.83	4.23060	0.06649
65	99.83	4.23060	0.06566
70	99.83	4.23050	0.06480
75	99.83	4.23050	0.06390
80	99.83	4.23050	0.06297
85	99.83	4.23040	0.06201
90	99.83	4.23040	0.06102
95	99.83	4.23040	0.06001
97	99.83	4.23030	0.05898
99	99.83	4.23030	0.05793

NOTE[1]: 1967 ASME (IFC) Steam Tables & IAPWS-IF97

*NOTE[2]:kPa =Heigth(ft)*12(in/ft)*0.0254(meters/in)*Density(kg/m^3)*9.80665/1000*

NOTE[3]: $q''_{SAT} = 0.149 \cdot \rho_g^{0.5} \cdot \left(h_{g,sat} - h_{f,sat}\right) \cdot \left(g \cdot \sigma \cdot \left\{\rho_f - \rho_g\right\}\right)^{1/4}$

NOTE[4]: $q''_{sub} = q''_{SAT} \cdot \left(1 + 0.1 \cdot \left(\frac{\rho_f}{\rho_g}\right)^{3/4} \cdot \frac{c_{p,f} \cdot \left(T_{SAT} - T_{sub}\right)}{h_{g,sat} - h_{f,sat}}\right)$

NOTE:[5]:

$\sigma = 1.000E-11 \cdot T^4 + 7.370E-09 \cdot T^3 - 1.969E-06 \cdot T^2 + 4.709E-06 \cdot T + 7.1833E-02$

4. Reactor Description

4.1 Summary Description

The Kansas State University (KSU) Nuclear Reactor Facility, operated by the Department of Mechanical and Nuclear Engineering, is located in Ward Hall on the campus in Manhattan, Kansas. The Department is also the home of the Tate Neutron Activation Analysis Laboratory. The TRIGA reactor was obtained through a 1958 grant from the United States Atomic Energy Commission and is operated under Nuclear Regulatory Commission License R-88 and the regulations of Chapter 1, Title 10, Code of Federal Regulations. Chartered functions of the Nuclear Reactor Facility are to serve as: 1) an educational facility for all students at KSU and nearby universities and colleges, 2) an irradiation facility for researchers at KSU and for others in the central United States, 3) a facility for training nuclear reactor operators, and 4) a demonstration facility to increase public understanding of nuclear energy and nuclear reactor systems.

The KSU TRIGA reactor is a water-moderated, water-cooled thermal reactor operated in an open pool and fueled with heterogeneous elements consisting of nominally 20 percent enriched uranium in a zirconium hydride matrix and clad with stainless steel. Principal experimental features of the KSU TRIGA Reactor Facility are:

- Central thimble
- Rotary specimen rack
- Thermalizing column with bulk shielding tank
- Thermal column with removable door
- Beam ports
  - Radial (2)
  - Piercing (fast neutron) (1)
  - Tangential (thermal neutron) (1)

The reactor was licensed in 1962 to operate at a steady-state thermal power of 100 kilowatts (kW). The reactor has been licensed since 1968 to operate at a steady-state thermal power of 250 kW and a pulsing maximum thermal power of 250 MW. Application is made concurrently with license renewal to operate at a maximum of 1,250 kW, with fuel loading to support 500 kW steady state thermal power and with pulsing to $3.00 reactivity insertion. All cooling is by natural convection. The 250-kW core consists of 81 fuel elements typically (at least 83 planned for the 1,250-kW core), each containing as much as 39 grams of ^{235}U. The reactor core is in the form of a right circular cylinder about 23 cm (approximately 9 in.) radius and 38 cm (14.96 in.) depth, positioned with axis vertical near the base of a cylindrical water tank 1.98 m (6.5 ft.) diameter and 6.25 m (20.5 ft.) depth. Criticality is controlled and shutdown margin assured by three control rods in the form of aluminum or stainless-steel clad boron carbide or borated graphite. A fourth control rod would be used for 1,250-kW operation. A biological shield of reinforced concrete at least 2.5 m (8.2 ft) thick provides radiation shielding at the side and at the base the reactor tank. The tank and shield are in a 4078-m^3 (144,000 $ft.^3$) confinement building

made of reinforced concrete and structural steel, with composite sheathing and aluminum siding. Sectional views of the reactor are shown in Figures 4.1 and 4.2.

Criticality was first achieved on October 16, 1962 at 8:25 p.m. In 1968 pulsing capability was added and the maximum steady-state operating power was increased from 100 kW to 250 kW. The original aluminum-clad fuel elements were replaced with stainless-steel clad elements in 1973. Coolant system was replaced (and upgraded in 2000), the reactor operating console was replaced, and the control room was enlarged and modernized in 1993, with support from the U.S. Department of Energy. All neutronic instrumentation was replaced in 1994.

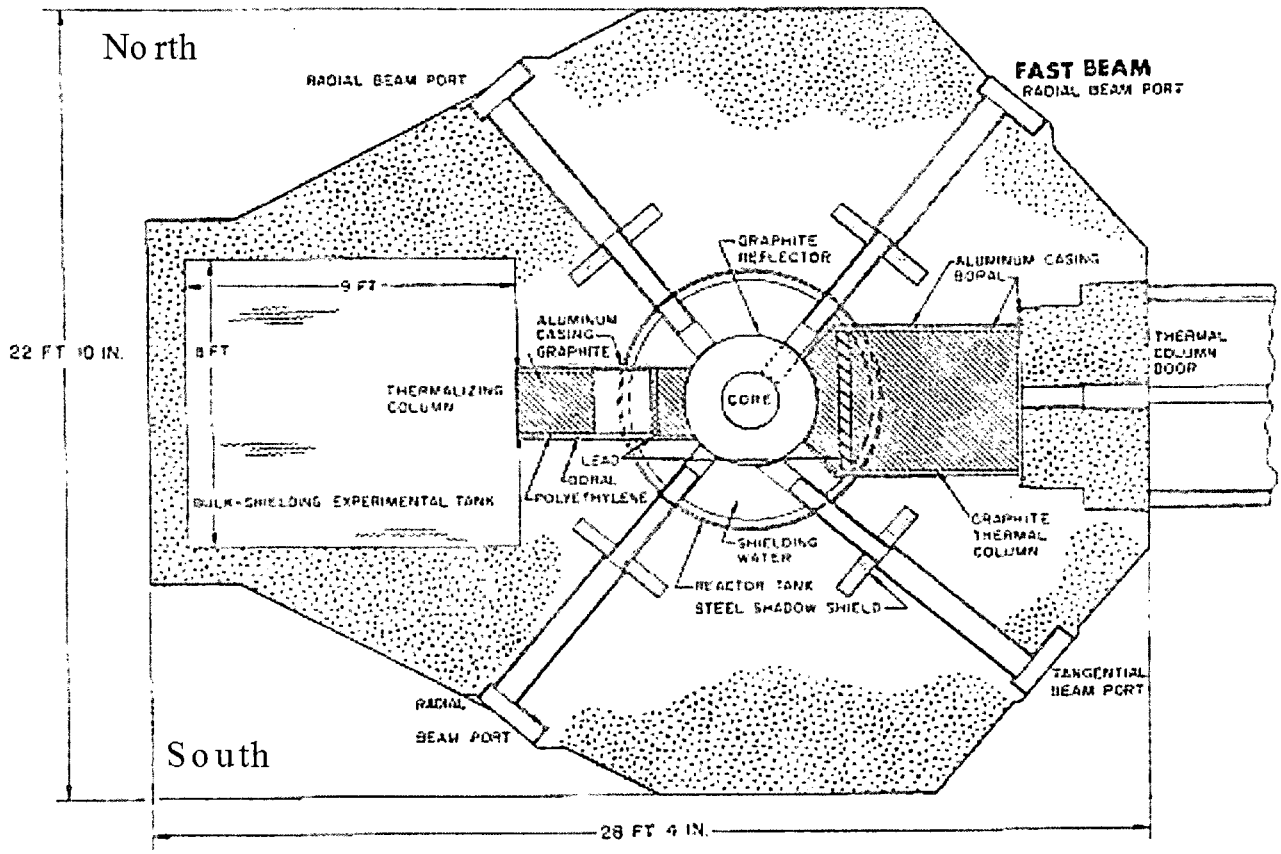


Figure 4. 1, Vertical Section Through the KSU TRIGA Reactor.

4.2 Reactor Core

The General Atomics TRIGA reactor design began in 1956. The original design goal was a completely and inherently safe reactor. Complete safety means that all the available excess reactivity of the reactor can be instantaneously introduced without causing an accident. Inherent safety means that an increase in the temperature of the fuel immediately and automatically results in decreased reactivity through a prompt negative temperature coefficient. These features were accomplished by using enriched uranium fuel in a zirconium hydride matrix.

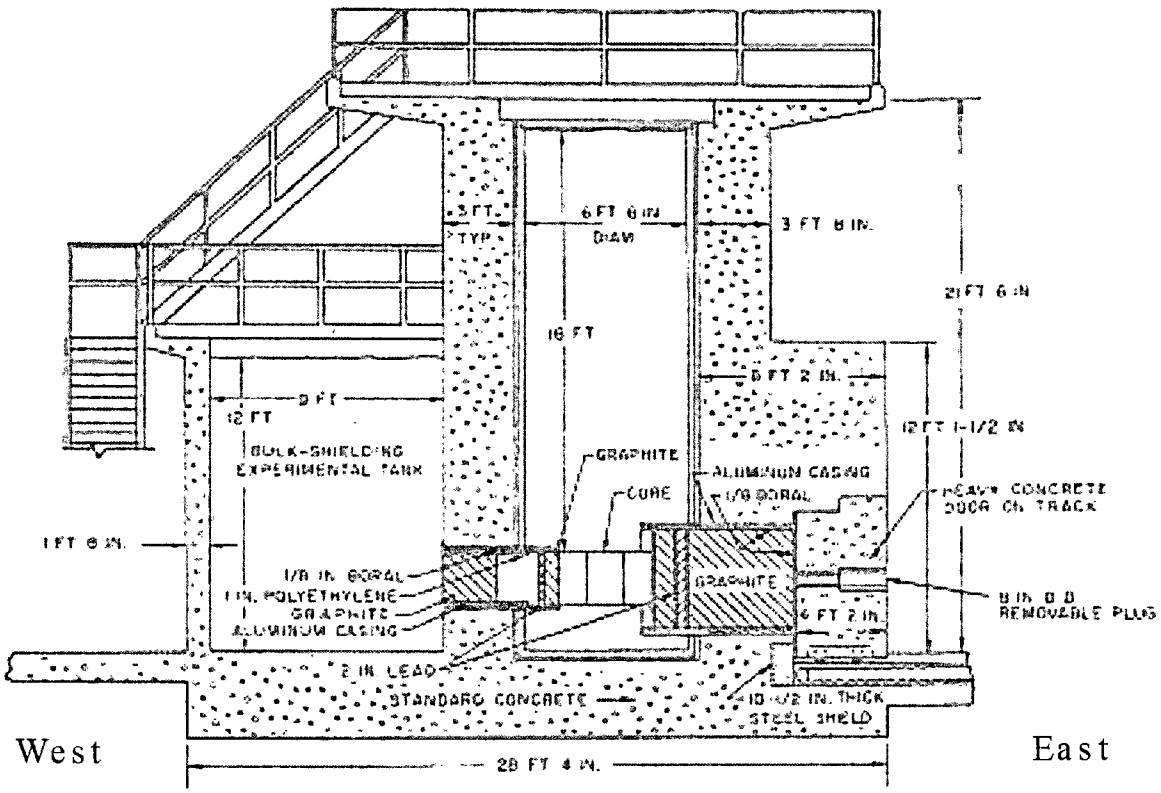


Figure 4. 2, Horizontal Section Through the KSU TRIGA Reactor.

The basic parameter providing the TRIGA system with a large safety factor in steady state and transient operations is a prompt negative temperature coefficient, relatively constant with temperature (-0.01% $\Delta k/k°C$). This coefficient is a function of the fuel composition and core geometry. As power and temperature increase, matrix changes cause a shift in the neutron energy spectrum in the fuel to higher energies. The uranium exhibits lower fission cross sections for the higher energy neutrons, thus countering the power increase. Therefore, fuel and clad temperature automatically limit operation of the reactor.

It is more convenient to set a power level limit that is based on temperature. The design bases analysis indicates that operation at up to 1900 kW (with an 83 element core and 120°F inlet water temperature) with natural convective flow will not allow film boiling, and therefore high fuel and clad temperatures which could cause loss of clad integrity could not occur. An 85-element core distributes the power over a larger volume of heat generating elements, and therefore results in a more favorable, more conservative, thermal hydraulic response.

4.2.1 Reactor Fuel[1]

TRIGA fuel was developed around the concept of inherent safety. A core composition was sought which had a large prompt negative temperature coefficient of reactivity such that if all the available excess reactivity were suddenly inserted into the core, the resulting fuel temperature would automatically cause the power excursion to terminate before any core damage resulted. Zirconium hydride was found to possess a basic mechanism to produce the desired characteristic. Additional advantages were that ZrH has a high heat capacity, results in relatively small core sizes and high flux values due to the high hydrogen content, and could be used effectively in a rugged fuel element size.

TRIGA fuel is designed to assure that fuel and cladding can withstand all credible environmental and radiation conditions during its lifetime at the reactor site. As described in 3.5.1 (Fuel System) and NUREG 1282, fuel temperature limits both steady-state and pulse-mode operation. The fuel temperature limit stems from potential hydrogen outgassing from the fuel and the subsequent stress produced in the fuel element clad material. The maximum temperature limits of 1150°C (with clad < 500°C) and 950°C (with clad > 500°C) for U-ZrH ($H/Zr_{1.65}$) have been set to limit internal fuel cladding stresses that might challenge clad integrity (NUREG 1282). These limits are the principal design bases for the safety analysis.

a. Dimensions and Physical Properties.

The KSU TRIGA reactor is fueled by stainless steel clad Mark III fuel-elements. Three instrumented aluminum-clad Mark II elements are still available for use in the core. General properties of TRIGA fuel are listed in Table 4.1. The Mark III elements are illustrated in Figure 4.3. To facilitate hydriding in the Mk III elements, a zirconium rod is inserted through a 0.635 cm. (1/4-in.) hole drilled through the center of the active fuel section.

Instrumented elements have three chromel-alumel thermocouples embedded to about 0.762 cm (0.3 in.) from the centerline of the fuel, one at the axial center plane, and one each at 2.54 cm. (1 in.) above and below the center plane. Thermocouple leadout wires pass through a seal in the upper end fixture, and a leadout tube provides a watertight conduit carrying the leadout wires above the water surface in the reactor tank.

[1]Unless otherwise indicated, fuel properties are taken from the General Atomics report of Simnad [1980] and from authorities cited by Simnad.

Graphite dummy elements may be used to fill grid positions in the core. The dummy elements are of the same general dimensions and construction as the fuel-moderator elements. They are clad in aluminum and have a graphite length of 55.88 cm (22 in.).

Table 4.1, Nominal Properties of Mark II and Mark III TRIGA Fuel Elements in use at the KSU Nuclear Reactor Facility.

Property	Mark II	Mark III
Dimensions		
Outside diameter, $D_o = 2r_o$	1.47 in. (3.7338 cm)	1.47 in. (3.7338 cm)
Inside diameter, $D_i = 2r_i$	1.41 in (3.6322 cm)	1.43 in. (3.6322 cm)
Overall length	28.4 in. (72.136 cm)	28.4 in. (72.136 cm)
Length of fuel zone, L	14 in. (35.56 cm)	15 in. (38.10 cm)
Length of graphite axial reflectors	4 in. (10.16 cm)	3.44 in (8.738 cm)
End fixtures and cladding	aluminum	304 stainless steel
Cladding thickness	0.030 in. (0.0762 cm)	0.020 in. (0.0508 cm)
Burnable poisons	Sm wafers	None
Uranium content		
Weight percent U	8.0	8.5
^{235}U enrichment percent	20	20
^{235}U content	36 g	38 g
Physical properties of fuel excluding cladding		
H/Zr atomic ratio	1.0	1.6
Thermal conductivity (W cm^{-1} K^{-1})	0.18	0.18
Heat capacity [T ≥0°C] (J cm^{-3} K^{-1})		2.04 + 0.00417T
Mechanical properties of delta phase U-ZrH[a]		
Elastic modulus at 20°C		9.1×10^6 psi
Elastic modulus at 650°C		6.0×10^6 psi
Ultimate tensile strength (to 650°C)		24,000 psi
Compressive strength (20°C)		60,000 psi
Compressive yield (20°C)		35,000 psi

[a]Source: Texas SAR [1991].

b. Composition and Phase Properties

The Mark III TRIGA fuel element in use at Kansas State University contains nominally 8.5% by weight of uranium, enriched to 20% ^{235}U, as a fine metallic dispersion in a zirconium hydride matrix. The H/Zr ratio is nominally 1.6 (in the face-centered cubic delta phase). The equilibrium hydrogen dissociation pressure is governed by the composition and temperature. For $ZrH_{1.6}$, the equilibrium hydrogen pressure is one atmosphere at about 760°C. The single-phase, high-hydride composition eliminates the problems of density changes associated with phase changes and with thermal diffusion of the hydrogen. Over 25,000 pulses have been performed with the TRIGA fuel elements at General Atomic, with fuel temperatures reaching peaks of about 1150°C.

The zirconium-hydrogen system, whose phase diagram is illustrated in Chapter 3, is essentially a simple eutectoid, with at least four separate hydride phases. The delta and epsilon phases are respectively face-centered cubic and face-centered tetragonal hydride phases. The two phase delta + epsilon region exists between $ZrH_{1.64}$ and $ZrH_{1.74}$ at room temperature, and closes at $ZrH_{1.7}$ at 455°C. From 455°C to about 1050°C, the delta phase is supported by a broadening range of H/Zr ratios.

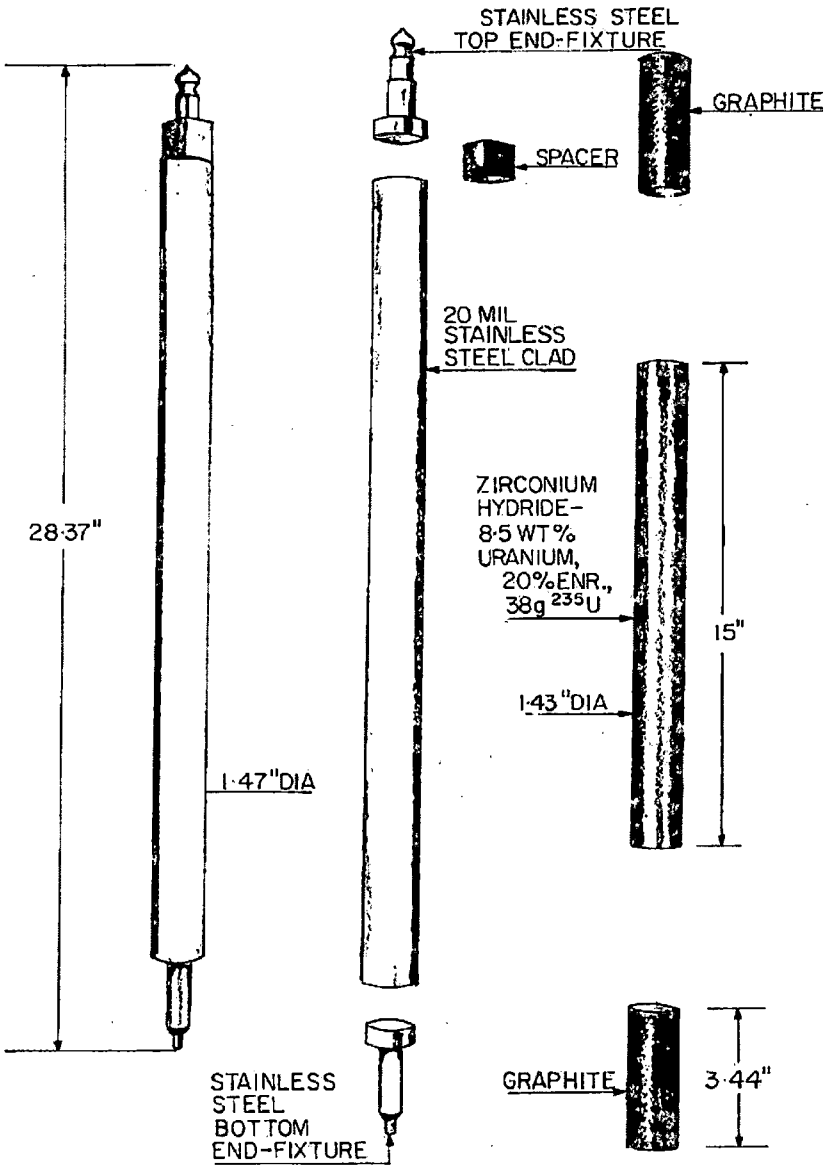


Figure 4.3, TRIGA Fuel Element.

c. Core Layout

A typical layout for a KSU TRIGA II 250-kW core (Core II-18) is illustrated in Figure 4.4. The layout for the 1,250-kW core is expected to be similar, except that the graphite elements will be replaced by fuel elements, one additional control rod will be added, and control rod positions will be adjusted..

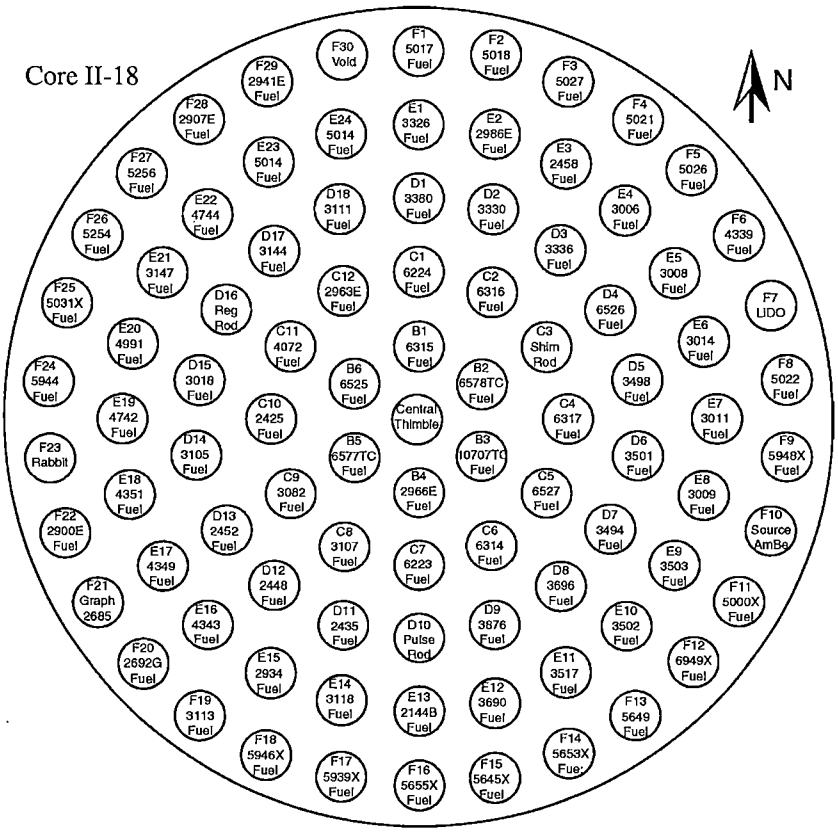


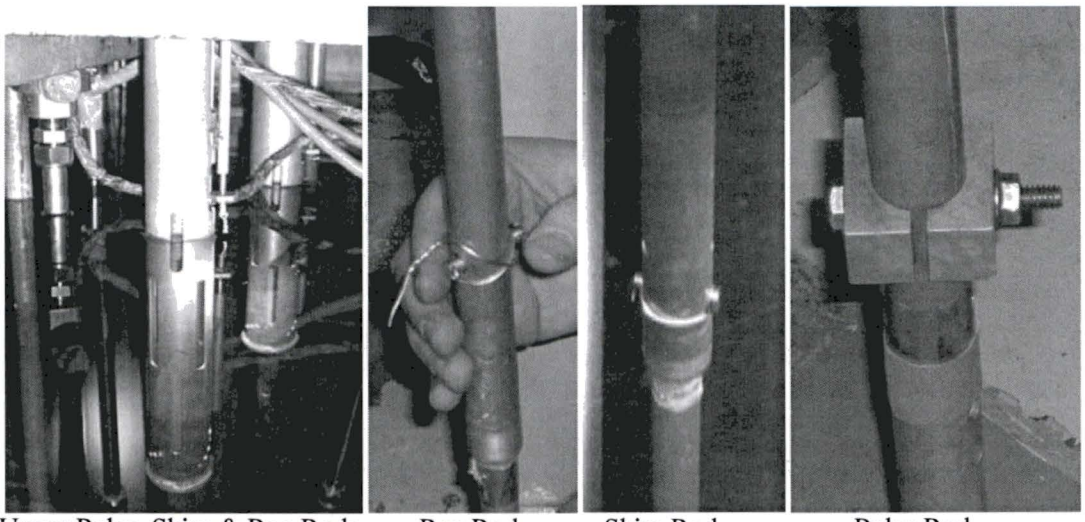
Figure 4.4, Core Layout (250 kW).

The additional fuel elements are required to compensate for higher operating temperatures from the higher maximum steady state power level. The additional control rod is required to meet reactivity control requirements at higher core reactivity associated with the additional fuel. The control rod positions will be different to allow a higher worth pulse rod (the 250 kW pulse rod reactivity worth is $2.00, the 1,250 kW core pulse rod reactivity worth is $3.00), balancing the remaining control rod's worth to meet minimum shutdown margin requirements, and meeting physical constraints imposed by the dimensions of the pool bridge

4.2.2 Control Rods

The pulse rod is 3.175 cm. (1.25 in.) diameter. Other rods are 2.225 cm (7/8 in.) diameter. Control rods are 50.8 cm. (20 in.) long boron carbide or borated graphite, clad with a 0.0762 cm. (30-mil) aluminum sheath.

The control rod drives are connected to the control rod clutches through three extension shafts. The clutch and upper extension shaft for standard rods extend through an assembly designed with slots that provides a hydraulic cushion (or buffer) for the rod during a scram, and also limits the bottom position of the control rods so that they do not impact the bottom of the control rod guide tube (in the core). The buffers for two standard rods are shown in the left hand picture below (slotted tubes on the right hand side) along with the top section of the pulse/transient rod extension. The pulse rod drive clutch connects to a solid extension shaft through a pneumatic cylinder; the dimensions of the cylinder limits bottom travel.



Upper Pulse, Shim & Reg Rods Reg Rod Shim Rod Pulse Rod

Figure 4.5, Control Rod Upper Extension Assemblies

The bottom of the pulse rod is shown on the left hand side of Figure 4.5. The upper extension shaft is a hollow tube, the middle extension is solid. The upper extension shaft is connected to the middle extension shaft with lock wire or a pin and lock wire for standard rods, with a bolted collar for the pulse rod (the mechanical shock during a pulse requires a more sturdy fastener). Securing the upper control rod extension to the middle extension at one of several holes drilled in the upper part of the middle extension (Figure 4.6) provides adjustment for the control rods necessary to ensure the control rod full in position is above the bottom of the guide tube.

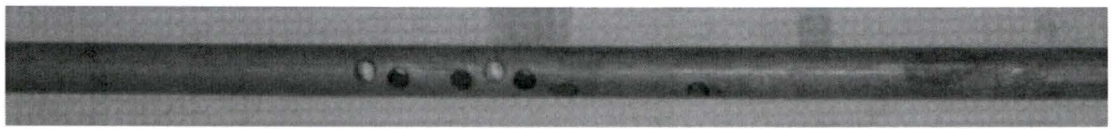


Figure 4.6, Middle Extension Rod Alignment Holes

The middle solid extension is similarly connected to the lower extension. The lower extension is hollow, the middle extension fits into the lower extension and a hole drilled in the overlap secures the lower extension to the middle extension. Typically the lower extension has a tighter fit than the upper extension because the lower and middle extension are not separated for inspections and because the interface with upper extension is used to set the bottom position of the control rod. Pictures of the lower connector for the pulse rod and one standard rod are shown at the left in Figure 4.7..

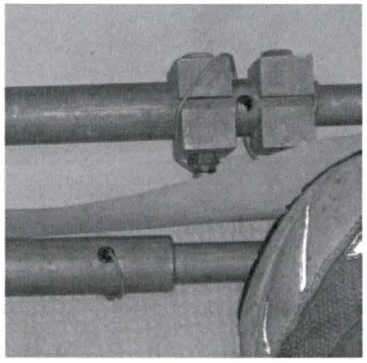
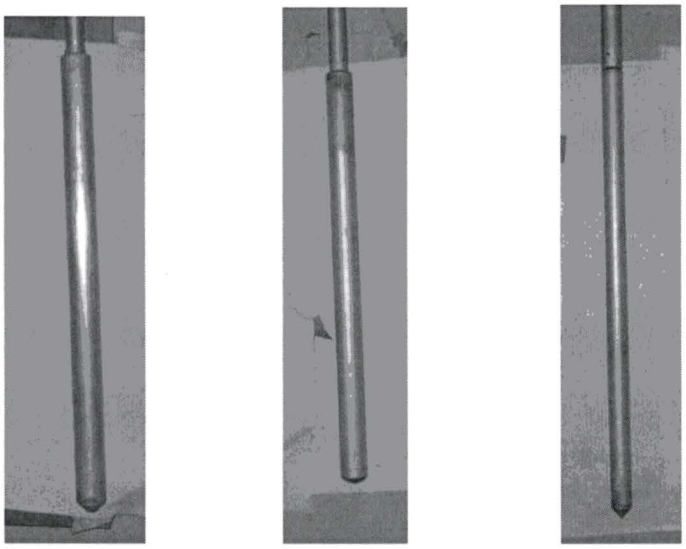


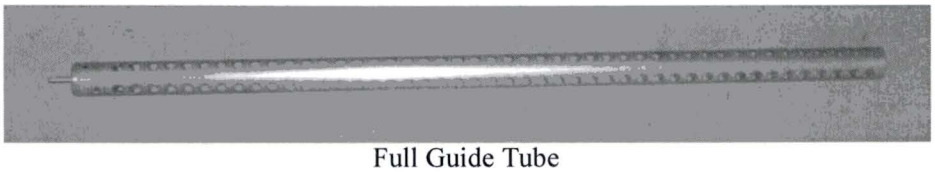
Figure 4.7, Standard & Pulse Rod Lower Coupling

The bottom of the lower extension attaches directly to the control rod. Pictures of the control rods taken during the 2003 control rod inspection are in Figure 4.8. The rods move within control rod guide tubes, shown in Figure 4.9. The guide tubes have perforated walls. The guide tubes have a small metal wire in the tip that fits into the lower grid plate; a setscrew inside the bottom of the guide tube pushes the wire against the lower grid plate to secure the guide tube.

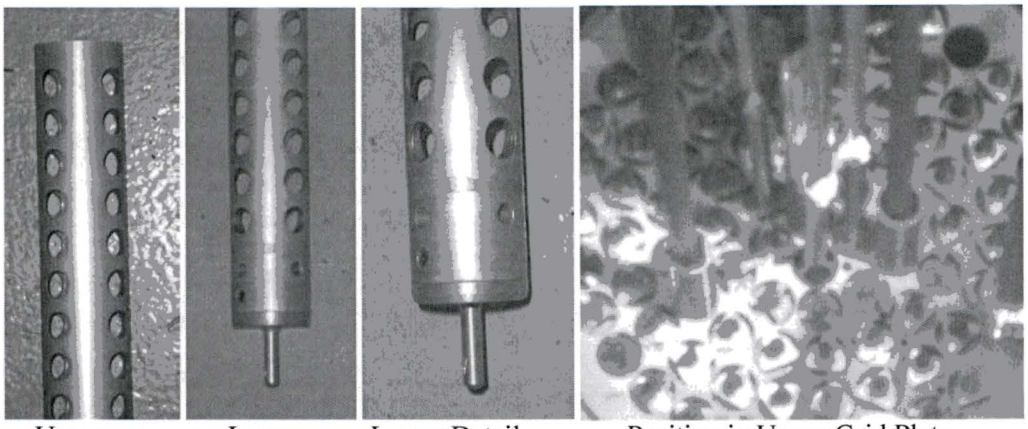


Pulse Rod Shim Rod Reg Rod

Figure 4.8, Control Rods During 2003 Inspection



Full Guide Tube



Upper Lower Lower Detail Position in Upper Grid Plate

Figure 4.9, Control Rod Guide Tubes

a. Control Function

While three control rods were adequate to meet Technical Specification requirements for reactivity control with the 100 kW and 250 kW cores, reactivity limits for operation at a maximum power level of 1,250 kW requires four control rods (three standard and one transient/pulsing control rod). The control-rod drives are mounted on a bridge at the top of the reactor tank. The control rod drives are coupled to the control rod through a connecting rod assembly that includes a clutch. The standard rod clutch is an electromagnet; the transient rod clutch is an air-operated shuttle. Scrams cause the clutch to release by de-energizing the magnetic clutch and venting air from the transient rod clutch; gravity causes the rod to fall back into the core. Interlocks ensure operation of the control rods remains within analyzed conditions for reactivity control, while scrams operation at limiting safety system settings. A detailed description of the control-rod system is provided in Chapter 7; a summary of interlocks and scrams is provided below in Table 4.2 and 4.3. Note that (1) the high fuel temperature and period scrams are not required, (2) the fuel temperature scram limiting setpoint depends on core location for the sensor, and (3) the period scram can be prevented by an installed bypass switch.

Table 4.2, Summary of Control Rod Interlocks

INTERLOCK	SETPOINT	FUNCTION/PURPOSE
Source Interlock	2 cps	Inhibit standard rod motion if nuclear instrument startup channel reading is less than instrument sensitivity/ensure nuclear instrument startup channel is operating
Pulse Rod Interlock	Pulse rod inserted	Prevent applying power to pulse rod unless rod inserted/prevent inadvertent pulse
Multiple Rod Withdrawal	Withdraw signal, more than 1 rod	Prevent withdrawal of more than 1 rod/Limit maximum reactivity addition rate
Pulse Mode Interlock	Mode switch in Hi Pulse	Prevent withdrawing standard control rods in pulse mode
Pulse-Power Interlock	10 kW	Prevent pulsing if power level is greater than 10 kW

NOTE: (1) Pulse-Power Interlock normally set at 1 kW, (2) only Pulse Mode Interlock required by Technical Specifications

b. Evaluation of Control Rod System

The reactivity worth and speed of travel for the control rods are adequate to allow complete control of the reactor system during operation from a shutdown condition to full power. The TRIGA system does not rely on speed of control as significant for safety of the reactor; scram times for the rods are measured periodically to monitor potential degradation of the control rod system. The inherent shutdown mechanism (temperature feedback) of the TRIGA prevents unsafe excursions and the control system is used only for the planned shutdown of the reactor and to control the power level in steady state operation.

Table 4.3, Summary of Reactor SCRAMs

Measuring Channel	Limiting Trip Setpoint		Actual Setpoint
	Steady State	Pulse	
Linear Channel High Power	110%	N/A	104%
Power Channel High power	110%	N/A	104%
Detector High Voltage	90%	90%	90%
High Fuel Temperature[1]	600°C B Ring element		450°C
	555°C C Ring element		
	480°C D Ring element		
	380°C E Ring element		350°C
Period [1]	N/A	N/A	3 sec

NOTE [1]: Period trip and temperature trip are not required by Technical Specifications

The reactivity worth of the control system can be varied by the placement of the control rods in the core. The control system may be configured to provide for the excess reactivity needed for 1,250 kW operations for eight hours per day (including xenon override) and will assure a shutdown margin of at least $0.50.

Nominal speed of the standard control rods is about 12 in. (30.5 cm) per minute (with the stepper motor specifically adjusted to this value), of the transient rod is about 24 in. (61 cm) per minute, with a total travel about 15 in. (38.1 cm). Maximum rate of reactivity change for standard control rods is specified in Technical Specifications.

4.2.3 Neutron Moderator and Reflector

Hydrogen in the Zr-H fuel serves as a neutron moderator. Demineralized light water in the reactor pool also provides neutron moderation (serving also to remove heat from operation of the reactor and as a radiation shield). Water occupies approximately 35% of the core volume. A graphite reflector surrounds the core, except for a cutout containing the rotary specimen rack (described in Chapter 10). Each fuel element contains graphite plugs above and below fuel approximately 3.4 in. in length, acting as top and bottom reflectors.

The radial reflector is a ring-shaped, aluminum-clad, block of graphite surrounding the core radially. The reflector is 0.457-m (18.7 in.) inside diameter, 1.066-m (42 in.) outside diameter, and 0.559-m (20 in.) height. Embedded as a circular well in the reflector is an aluminum housing for a rotary specimen rack, with 40 evenly spaced tubular containers, 3.18-cm (1.25 in.) inside diameter and 27.4-cm (10.8 in.) height. The rotary specimen rack housing is a watertight assembly located in a re-entrant well in the reflector.

The radial reflector assembly rests on an aluminum platform at the bottom of the reactor tank. Four lugs are provided for lifting the assembly. A radial void about 6 inches (15.24 cm) in diameter is located in the reflector such that it aligns with the radial piercing beam port (NE beam port). The reflector supports the core grid plates, with grid plate positions set by alignment fixtures. Graphite inserts within the fuel cladding provide additional reflection. Inserts are placed at both ends of the fuel meat, providing top and bottom reflection.

4.2.4 Neutron Startup Source

A 2-curie americium-beryllium startup source (approximately 2×10^6 n s^{-1}) is used for reactor startup. The source material is encapsulated in stainless steel and is housed in an aluminum-cylinder source holder of approximately the same dimensions as a fuel element. The source holder may be positioned in any one of the fuel positions defined by the upper and lower grid plates. A stainless-steel wire may be threaded through the upper end fixture of the holder for use in relocating the source manually from the 22-ft level (bridge level) of the reactor.

4.2.5 Core Support Structure

The fuel elements are spaced and supported by two 0.75-in. (1.9 cm) thick aluminum grid plates. The grid plates have a total of 91 spaces, up to 85 of which are filled with fuel-moderator elements and dummy elements, and the remaining spaces with control rods, the central thimble, the pneumatic transfer tube, the neutron source holder, and one or more voids. The bottom grid plate, which supports the weight of the fuel elements, has holes for receiving the lower end fixtures. Space is provided for the passage of cooling water around the sides of the bottom grid plate and through 36 special holes in it. The 1.5-in. (3.8 cm) diameter holes in the upper grid plate serve to space the fuel elements and to allow withdrawal of the elements from the core. Triangular-shaped spacers on the upper end fixtures allow the cooling water to pass through the upper grid plate when the fuel elements are in position. The reflector assembly supports both grid plates.

4.3 Reactor Tank

The KSU TRIGA reactor core support structure rests on the base of a continuous, cylindrical aluminum tank surrounded by a reinforced, standard concrete structure (with a minimum thickness of approximately 249 cm. or 8 ft 2 in), as illustrated in Figures 4.1 and 4.2. The tank is a welded aluminum structure with 0.635 cm. (1/4-in.) thick walls. The tank is approximately 198 cm (6.5-ft) in diameter and approximately 625 cm (20.5-ft) in depth. The exterior of the tank was coated with bituminous material prior to pouring concrete to retard corrosion. Each experiment facility penetration in the tank wall (described below) has a water collection plenum at the penetration. All collection plenums are connected to a leak-off volume through individual lines with isolation valves, with the leak-off volumes monitored by a pressure gauge. The bulk shield tank wall is known to have a small leak into the concrete at the thermalizing column plenum, therefore a separate individual leak-off volume (and pressure gauge) is installed for the bulk shield tank; all other plenums drain to a common volume. In the event of a leak from the pool

through an experiment facility, pressure in the volume will increase; isolating individual lines allows identification of the specific facility with the leak.

A bridge of steel plates mounted on two rails of structural steel provides support for control rod drives, central thimble, the rotary specimen rack, and instrumentation. The bridge is mounted directly over the core area, and spans the tank. Aluminum grating with clear plastic attached to the bottom is installed that can be lowered over the pool. The grating can be lowered when activities could cause objects or material to fall into the reactor pool. The grating normally remains up to reduce humidity at electro-mechanical components of the control rod drive system and to prevent the buildup of radioactive gasses at the pool surface during operations.

Four beam tubes run from the reactor wall to the outside of the concrete biological shield in the outward direction. Tubes welded to the inside of the wall run toward the reactor core. Three of the tubes (NW, SW, and SE) end at the radial reflector. The NE beam tube penetrates the radial reflector, extending to the outside of the core. Two penetrations in the tank allow neutron extraction into a thermal column and a thermalizing column (described in Chapter 10).

4.4 Biological Shield

The reactor tank is surrounded on all sides by a monolithic reinforced concrete biological shield. The shielding configuration is similar to those at other TRIGA facilities operating at power levels up to 1 MW. Above ground level, the thickness varies from approximately 249 cm. (8 ft 2 in.) at core level to approximately 91 cm. (3 ft.) at the top of the tank.

The massive concrete bulk shield structure provides additional radiation shielding for personnel working in and around the reactor laboratory and provides protection to the reactor core from potentially damaging natural phenomena.

4.5 Nuclear Design

The strong negative temperature coefficient is the principal method for controlling the maximum power (and consequently the maximum fuel temperature) for TRIGA reactors. This coefficient is a function of the fuel composition, core geometry, and temperature. For fuels with 8.5% U, 20% enrichment, the value is nearly constant at 0.01% $\Delta k/k$ per °C, and varies only weakly dependent on geometry and temperature.

Fuel and clad temperature define the safety limit. A power level limit is calculated that ensures that the fuel and clad temperature limits will not be exceeded. The design bases analysis indicates that operation at 1,250 kW thermal power with an 83-element across a broad range of core and coolant inlet temperatures with natural convective flow will not allow film boiling that could lead to high fuel and clad temperatures that could cause loss of clad integrity.

Increase in maximum thermal power from 250 to 1,250 kW does not affect fundamental aspects of TRIGA fuel and core design, including reactivity feedback coefficients, temperature safety

limits, and fission-product release rates. Thermal hydraulic performance is addressed in Section 4.6.

4.5.1 Design Criteria - Reference Core

The limiting core configuration for this analysis is a compact core defined by the TRIGA Mk II grid plates (Section 4.2.5). The grid plates have a total of 91 spaces, up to 85 of which are filled with fuel-moderator elements and graphite dummy elements, and the remaining spaces with control rods, the central thimble, the pneumatic transfer tube, the neutron source holder, and one or more voids in the E or F (outermost two rings) as required to support experiment operations or limit excess reactivity. The bottom grid plate, which supports the weight of the fuel elements, has holes for receiving the lower end fixtures.

4.5.2 Reactor Core Physics Parameters

The limiting core configuration differs from the configuration prior to upgrade only in the addition of a fourth control rod, taking the place of a graphite dummy element or void experimental position. For this reason, core physics is not affected by the upgrade except for linear scaling with power of neutron fluxes and gamma-ray dose rates.

For comparison purposes, a tabulation of total rod worth for each control element from the K-State reactor from a recent rod worth measurement is provided with the values from the Cornell University TRIGA reactor as listed in NUREG 0984 (Safety Evaluation Report Related to the Renewal of the Operating license for the Cornell University TRIGA Research Reactor).

Table 4.4, 250 kW Core Parameters.

β (effective delayed neutron fraction)	0.007
ℓ (effective neutron lifetime)	43 :S
α_{Tf} (prompt temperature coefficient)	-\$0.017 EC^{-1} @ 250kW ~275EC
α_V (void coefficient)	-0.003 $1\%^{-1}$ void
α_p (power temperature coefficient – weighted ave)	-\$0.006 kW^{-1} to – \$0.01 kW^{-1}

Table 4.5, Comparison of Control Rod Worths.

	KSU TRIGA Mark II (250 kW)				Cornell University (500 kW)	
	Core II-19		Core III-1			
Pulse	D-10	\$1.96	C-4	\$2.12	D-10	\$1.88
Shim	C-3	\$2.88	D-4	\$1.85	D-16	\$2.20
Safety	NA	\$0.0	D-16	\$1.82	D-4	\$1.99
Regulating	D-16	\$1.58	E-1	\$0.79	E-1	\$0.58
TOTAL	NA	\$6.42	NA	\$6.58	NA	\$6.65

NOTE: Core III-1 has an experiment positioned to control the worth of the pulse rod

The pulse rod is similar to a standard control rod, and the worth of the pulse rod compares well with the comparable standard control rods in similar ring positions. A maximum pulse is analyzed for thermal hydraulic response and maximum fuel temperature.

4.5.3 Fuel and Clad Temperatures

This section analyzes expected fuel and cladding temperatures with realistic modeling of the fuel-cladding gap. Analysis of steady state conditions reveals maximum heat fluxes well below the critical heat flux associated with departure from nucleate boiling. Analysis of pulsed-mode behavior reveals that film boiling is not expected, even during or after pulsing leading to maximum adiabatic fuel temperatures.

Chapter 4, Appendix A of this chapter reproduces a commonly cited analysis of TRIGA fuel and cladding temperatures associated with pulsing operations. The analysis addresses the case of a fuel element at an average temperature of 1000°C immediately following a pulse and estimates the cladding temperature and surface heat flux as a function of time after the pulse. The analysis predicts that, if there is no gap resistance between cladding and fuel, film boiling can occur very shortly after a pulse, with cladding temperature reaching 470°C, but with stresses to the cladding well below the ultimate tensile strength of the stainless steel. However, through comparisons with experimental results, the analysis concludes that an effective gap resistance of 450 Btu hr^{-1} ft^{-2} $°F^{-1}$ (2550 W m^{-2} K^{-1}) is representative of standard TRIGA fuel and, with that gap resistance, film boiling is not expected. This section provides an independent assessment of the expected fuel and cladding thermal conditions associated with both steady-state and pulse-mode operations.

a. Spatial Power Distribution

The following conservative approximations are made in characterizing the spatial distribution of the power during steady-state operations.

- The hottest fuel element delivers twice the power of the average.

  Classically, the radial hot-channel factor for a cylindrical reactor (using R as the physical radius and R_e as the physical radius and the extrapolation distance) is given[2] by:

$$F^N_{R_0} = \frac{1.202 * \left(R/R_e \right)}{J_1\left[2.4048 * \left(R/R_e \right) \right]}$$

  with a radial peaking factor of 1.93 for the KSU TRIGA II geometry,. However, TRIGA fuel elements are on the order of a mean free path of thermal neutrons, and there is a significant change in thermal neutron flux across a fuel element.

[2] Elements of Nuclear Reactor Design, 2nd Edition (1983), J. Weisman, Section 6.3

Calculated thermal neutron flux data[3] indicates that the ratio of peak to average neutron flux (peaking factor) for TRIGA cores under a range of conditions (temperature, fuel type, water and graphite reflection) has a small range of 1.36 to 1.40.

Actual power produced in the most limiting actual case is 14% less than power calculated using the assumption; therefore using a peaking factor of 2.0 to determine calculated temperatures and will bound actual temperatures by a large margin, and is extremely conservative.

- The axial distribution of power in the hottest fuel element is sinusoidal, with the peak power a factor of $\pi/2$ times the average, and heat conduction radial only.

  The axial factor for power produced within a fuel element is given by:

$$g(z) = 1.514 * \cos\left(\frac{\pi}{2} * \frac{z}{2 * \ell + \ell_{ext}}\right), \tag{6}$$

  in which $\ell = L/2$ and ℓ_{ext} is the extrapolation length in graphite, namely, 0.0275 m. The value used to calculate power in the limiting location within the fuel element is therefore 4% higher a power calculated with the actual peaking factor. Actual power produced in the most limiting actual case is 4% less than power calculated using the assumption; therefore calculated temperatures will bound actual temperatures.

- The location on the fuel rod producing the most thermal power with thermal power distributed over 83 fuel rods is therefore:

$$q''_{\max} = \frac{P}{83 \cdot \pi \cdot D_0 \cdot L} \cdot \frac{\pi}{2} \cdot 2 = \frac{P}{83 \cdot D_0 \cdot L} = P \cdot 0.8469 \tag{7}$$

- The radial and axial distribution of the power within a fuel element is given by

$$q'''(r,z) = q'''_{avg} f(r) g(z), \tag{5}$$

  in which r is measured from the vertical axis of the fuel element and z is measured along the axis, from the center of the fuel element. The axial peaking factor follows from the previous assumption of the core axial peaking factor, but (since there is a significant flux depression across a TRIGA fuel element) distribution of power produced across the radius of the fuel the radial peaking factor requires a different approach than the previous radial peaking factor for the core.

[3] GA-4361, Calculated Fluxes and Cross Sections for TRIGA Reactors (8/14/1963), G. B. West

- The radial factor is given by:

$$f(r) = \frac{a + cr + er^2}{1 + br + dr^2}, \tag{7}$$

in which the parameters of the rational polynomial approximation are derived from flux-depression calculations for the TRIGA fuel (Ahrens 1999a). Values are: a = 0.82446, b = -0.26315, c = -0.21869, d = -0.01726, and e = +0.04679. The fit is illustrated in Figure 4.11.

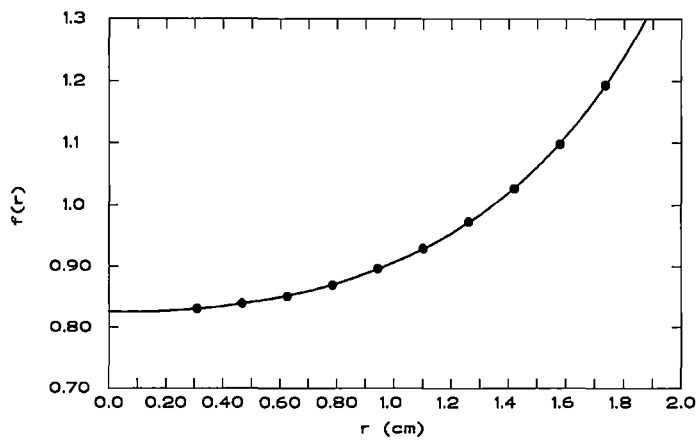


Figure 4.12, Radial Variation of Power Within a TRGIA Fuel Rod.
(Data Points from Monte Carlo Calculations [Ahrens 1999a])

a. Heat Transfer Models

The overall heat transfer coefficient relating heat flux at the surface of the cladding to the difference between the maximum fuel (centerline) temperature and the coolant temperature can be calculated as the sum of the temperature changes through each element from the centerline of the fuel rod to the water coolant, where the subscripts for each of the ΔT's represent changes between bulk water temperature and cladding outer surface, (br_0), changes between cladding outer surface and cladding inner surface (r_0r_i), cladding inner surface and fuel outer surface – gap (g), and the fuel outer surface to centerline (r_icl):

$$T_{cl} = T_b + \Delta T_{br_0} + \Delta T_{r_0r_i} + \Delta T_g + \Delta T_{r_icl} \qquad \text{Eq. 1}$$

A standard heat resistance model for this system is:

$$T_{cl} = T_b + q''\left[\frac{1}{h} + \frac{r_o \ln\left(r_o/r_i\right)}{k_c} + \frac{r_o}{r_i h_g} + \frac{r_o}{2k_f}\right] \qquad \text{Eq. 2}$$

and heat flux is calculated directly as:

$$q'' = U\Delta T = \frac{T_{\max} - T_b}{\frac{1}{h} + \frac{r_o \ln(r_o/r_i)}{k_c} + \frac{r_o}{r_i h_g} + \frac{r_o}{2k_f}}, \qquad (2)$$

in which r_o and r_i are cladding inner and outer radii, h_g is the gap conductivity, h is the convective heat transfer coefficient, and k_f is the fuel thermal conductivity. The gap conductivity of 2840 W m^{-2} K^{-1} (500 Btu h^{-1} ft^{-2} °F^{-1}) is taken from Appendix A. The convective heat transfer coefficient is mode dependent and is determined in context. Parameters are cross-referenced to source in Table 4.6.

Table 4.6: Thermodynamic Values

Parameter	Symbol	Value	Units	Reference
Fuel conductivity	k_f	18	W m^{-1} K^{-1}	Table 13.3
Clad conductivity	k_g	14.9	W m^{-1} K^{-1} (300 K)	Table 13.3
		16.6	W m^{-1} K^{-1} (400 K)	Table 13.3
		19.8	W m^{-1} K^{-1} (600 K)	Table 13.3
Gap resistance	h_g	2840	W m^{-2} K^{-1}	Appendix A
Clad outer radius	r_0	0.018161	M	Table 13.1
Fuel outer radius	r_i	0.018669	M	Table 13.1
Active fuel length	L_f	0.381	M	Table 13.1
No. fuel elements	N	83	N/A	Chap 13
Axial peaking factor	APF	$\pi/2$	N/A	Table 13.4

General Atomics reports that fuel conductivity over the range of interest has little temperature dependence, so that:

$$\frac{r_o}{2k_f} = 5.1858\text{E-}04\,\frac{\text{m}^2 K}{\text{W}}$$

Gap resistance has been experimentally determined as indicated, so that:

$$\frac{r_o}{r_i h_g} = 3.6196\text{E-}04\,\frac{\text{m}^2 K}{\text{W}}$$

Temperature change across the cladding is temperature dependent, with values quoted at 300 K, 400 K and 600 K. Under expected conditions, the value for 127°C applies so that:

$$\frac{r_o \ln \frac{r_o}{r_i}}{k_c} = 3.103e-5 \frac{m^2 K}{W}$$

Table 4.7, Cladding Heat Transfer Coefficient

Temp (°K)	Temp (°C)	m^2 K w^{-1}
300	27	3.457e-5
400	127	3.103e-5
600	327	2.601e-5

It should be noted that, since these values are less than 10% of the resistance to heat flow attributed to the other components, any errors attributed to calculating this factor are small.

The convection heat transfer coefficient was calculated at various steady state power levels. A graph of the calculated values results in a nearly linear response function.

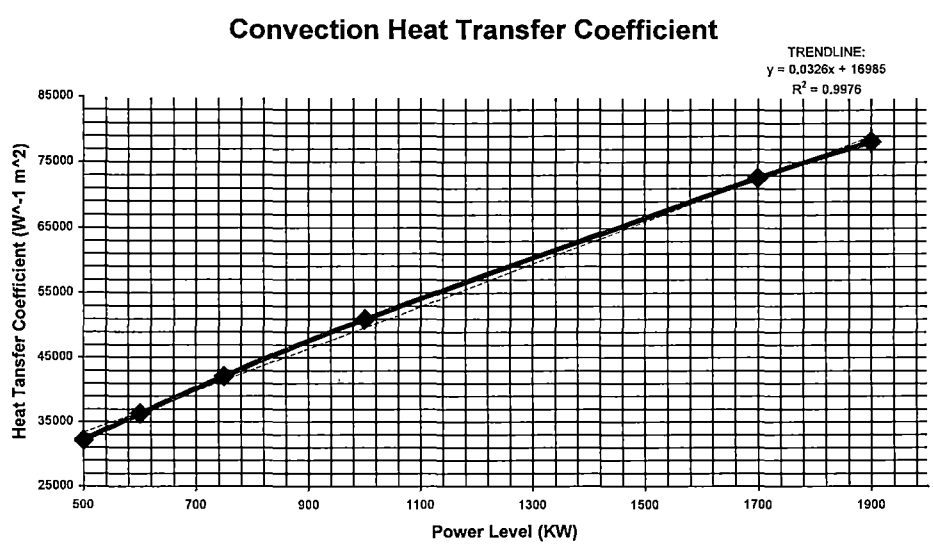


Figure 4.10, Convection Hear Transfer Coefficient versus Power Level

$$\frac{1}{h} = \frac{1}{0.0326P(watts) + 16985}$$

Core centerline temperature for the fuel rod producing the maximum heat as a function of power can be calculated as:

$$T_{cl} = T_b + 0.423P\left[\frac{1}{0.0326P + 16985} + 3.103e-5 + 3.620e-4 + 5.186e-4\right] \quad (10)$$

c. Steady-State Mode of Operation

Centerline temperature calculations were performed on a "reference core" using the model as described above for the hottest location in the core were made. The reference core contains 83 fuel elements; temperature calculations using the reference core are conservative because at least 83 elements are required for steady state 500 kW operations, while analysis assumes 1.25 MW operation. A core with more than 83 elements will distribute heat production across a larger number of fuel elements, resulting in a lower heat flux per fuel rod than calculations based on the reference core. Since actual heat production will be less than heat calculated in analysis, actual temperatures will be lower. A power level of 1.25 MW steady state power at 20°C and 100°C was assumed with the following results:

Table 4.8, Calculated Temperature Data for 1,250 kW Operation

Fuel Centerline °C	Fuel/Gap Interface °C	Gap/Clad Interface °C	Clad/Water Interface °C	Bulk Water °C
503.2	229.0	37.7	21.2	20.0
582.0	307.8	116.4	100.0	100.0

For the purposes of calculation, the two extremes of cladding thermal conductivity were assumed (300 K value and 600 K value) to determine expected centerline temperature as a function of power level. Calculations show the effects of thermal conductivity changes are minimal. The graph also shows that fuel temperature remains below about 750 °C at power levels up to 1900 kW with pool temperature at 27 °C (300 K), and 1700 kW with pool temperatures at 100 °C.

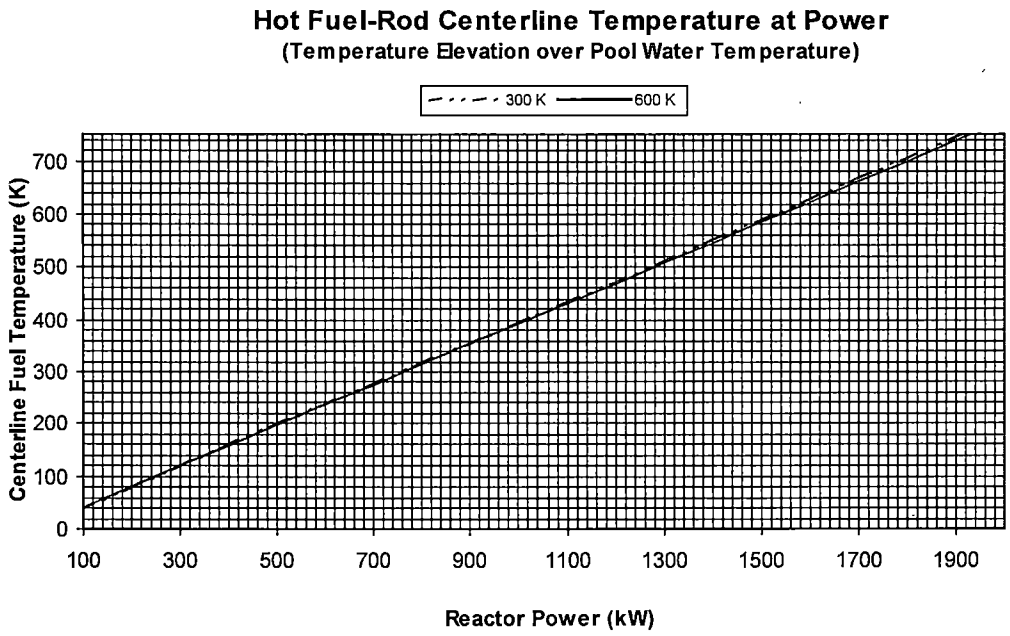


Figure 4.11, Hot Fuel-Rod Centerline Temperature

For the analysis of critical heat flux, a single channel model was built in RELAP-5/MOD 3.3 patch 04 (Feldman 2008). A snapshot of the model is presented in Figure 4.12. It has two time-dependent volumes, enforcing the pressure boundary conditions, and two pipes, simulating the cold and hot channel connected via a single junction component of RELAP. Heat is added to the fluid by incorporating the heat structure component (simulating a fuel element) of RELAP with an appropriate axial power profile and power level. In this analysis, the power level for the B ring is at 24 kW (corresponding to an 85-element core with a ring-to-average peaking factor of 1.63). This power level is applied to the heat structure within the single channel. The model assumes an operating pressure of 143 kPa, and an operating temperature of 322.15 K (49.15°C).

The version of the RELAP code licensed to KSU uses PG-CHF correlation which is a state of the art best estimate CHF correlation developed by Nuclear Research institute of Rez in the Czech Republic. It is based on data in the Czech Republic data bank from 173 different sets of tube data, 23 sets of annular data, and 153 sets of rod bundle data. There are four forms of the PG-CHF correlation `Basic', `Flux', `Geometry', and `Power'. For the rod bundle it is applicable in the pressure range of 0.28 MPa to 18.73 MPa, for a mass flux of 34.1 to 7478 kg/s-m2, for 0.4-7.0 m length and for a diameter of 0.00241 to 0.07813 m. TRIGA has an operating pressure of 0.143 MPa and fuel rod length of 0.381 m, thus the operating conditions fall outside the range of the applicability of the PG-CHF correlation, and a different correlation is required to assess the departure from nucleate boiling ratio (DNBR ratio). One such correlation which is applicable for the low pressure range observed in TRIGA reactor facility is the Bernath correlation. The functional form of the Bernath correlation can be presented in the following equations.

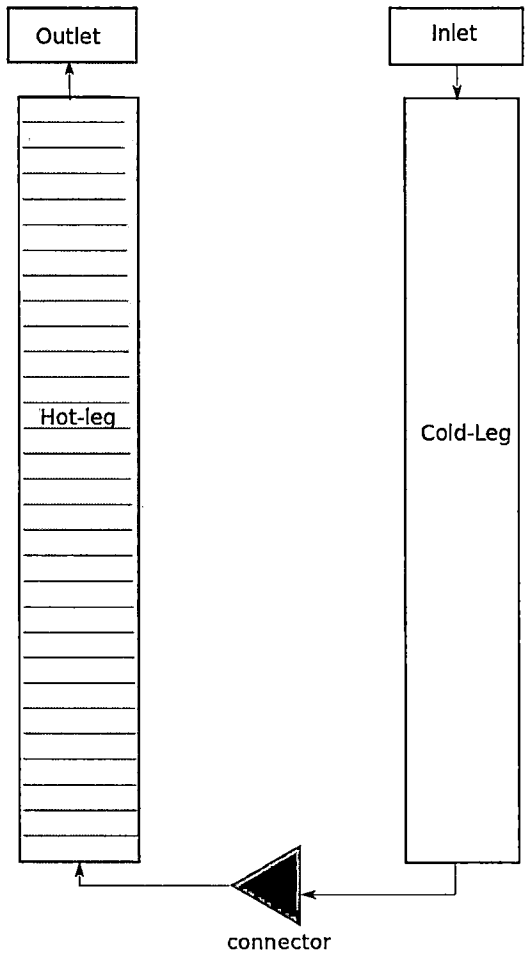


Figure 4.12 - RELAP single channel model used in CHF analysis

$$Q''_{BO} = h_{BO}\left(T_{wBO} - T_b\right) \quad (8)$$

$$h_{BO} = 10890\left(\frac{D_h}{D_h + D_H}\right) + \Delta v \quad (8)$$

$$\Delta = \frac{48}{D_h^{0.6}}, \text{if } D_h \leq 0.1 ft$$

$$\Delta = \frac{10}{D_h} + 90, \text{if } D_h \geq 0.1 ft$$

h_{BO} = film coefficient at CHF

D_h = hydraulic diameter (ft)

v = coolant velocity (ft / s)

T_{wBO} = wall temperature at burnout (°C)

D_H = heated diameter (ft)

The RELAP simulations were performed for the hot channel, i.e., a channel with a radial peaking factor of 1.63, assuming an 85-element core load and a power of 1.25 MWth, in order to obtain the pressure, temperature, and velocity distribution at different axial locations. With these calculations and the functional form of the Bernath correlation, the axial distribution of CHF was estimated in the hot channel. The methodology adopted for this analysis is described in literature (Feldman 2008). The hot channel model was based on the smallest hydraulic diameter in the core (between the A-ring and two B-ring elements) and the highest radial peaking factor. In the KSU TRIGA, the A-ring is occupied by the central thimble, not a fuel element. Since the actual hot channel would be between two B-ring elements and a C-ring element, the real hydraulic diameter will be slightly larger and the real heat flux into the channel will be slightly lower than the values assumed in the model. Therefore, this model is conservative in this regard.

The axial CHF results from the PG and Bernath heat flux models are shown in Figure 4.13 and Figure 4.14. The DNBR ratio exceeds 2.0 for all locations along the heated length of the hot channel.

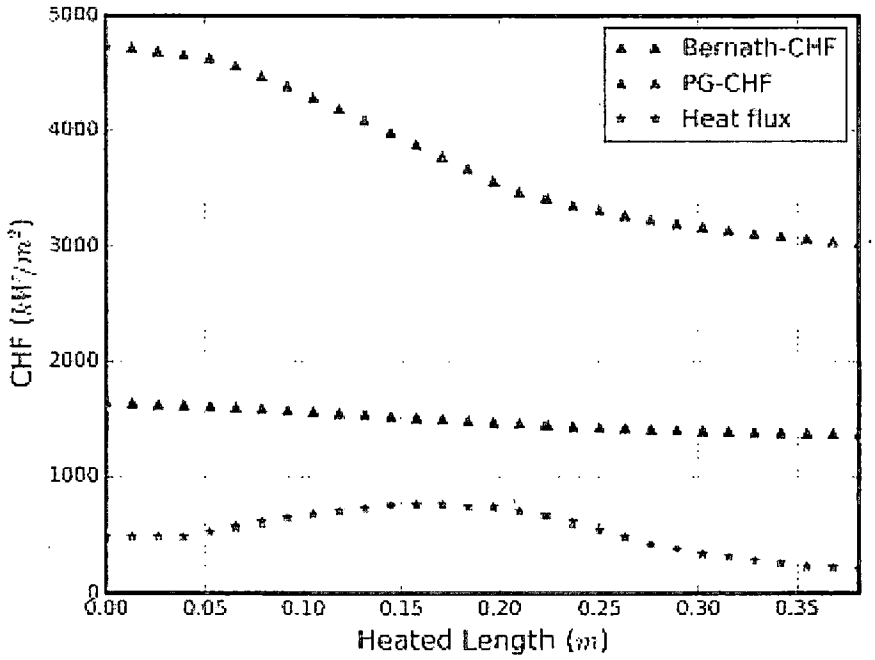


Figure 4.13 - CHF versus heated length

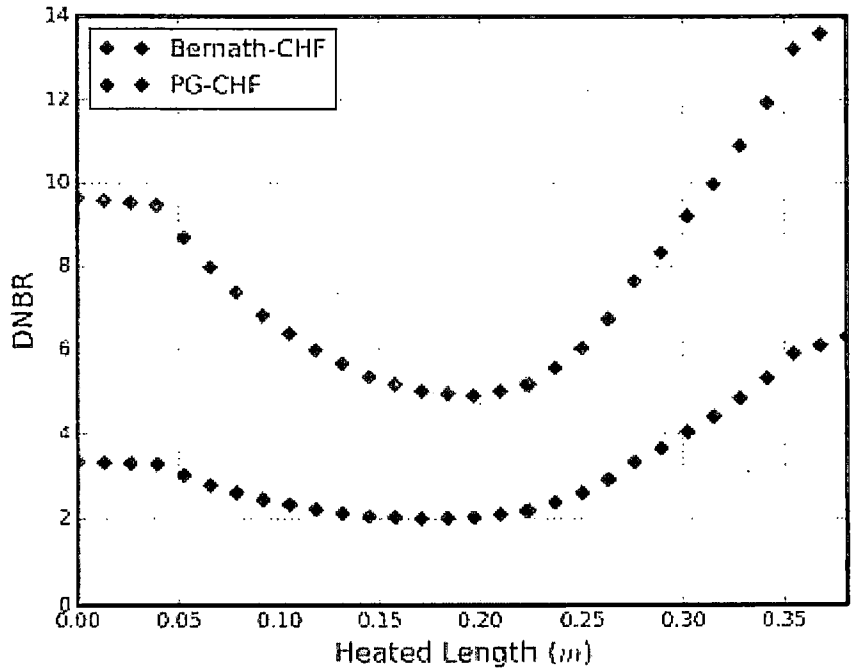


Figure 4.14 - DNBR versus heated length

d. Pulsed Mode of Operation

Transient calculations have been performed using a custom computer code TASCOT for transient and steady state two-dimensional conduction calculations (Ahrens 1999). For these calculations, the initial axial and radial temperature distribution of fuel temperature was based on Eqs. (9) and (10), with the peak fuel temperature set to 746 °C, i.e., a temperature rise of 719 °C above 27 °C ambient temperature. The temperature rise is computed in Chapter 13, Section 13.2.3 for a 2.1% ($3.00) pulse from zero power and a 0.7% ($1.00) pulse from power operation. In the TASCOT calculations, thermal conductivity was set to 0.18 W cm^{-1} K^{-1} (Table 4.1) and the overall heat transfer coefficient U was set to 0.21 W cm^{-1} K^{-1}. The convective heat transfer coefficient was based on the boiling heat transfer coefficient computed using the formulation (Chen 1963, Collier and Thome 1994)

$$q'' = h_b(T_w - T_{sat}) = h(T_w - T_b). \tag{9}$$

The boiling heat transfer coefficient is given by the correlation (Forster & Zuber 1955)

$$h_b = 0.00122 * \left[\frac{k_f^{0.79} * c_{pf}^{0.45} * \lambda^{0.51}}{\sigma^{0.5} * \mu_f^{0.29} * \rho_g^{0.24} * \left(v_g - v_v\right)^{0.75} * T_{sat}^{0.75}} \right] * \left(T_w - T_{sat}\right)^{0.99}, \tag{10}$$

in which T_w is the cladding outside temperature, T_{sat} the saturation temperature (111.9 °C), and T_b the coolant ambient temperature (27°C). Fluid-property symbols and values are given in Appendix B. Subscripts f and g refer respectively to liquid and vapor phases. The overall heat transfer coefficient U varies negligibly for ambient temperatures from 20 to 60 °C, and has the value 0.21 W cm^{-1} K^{-1} at $T_b = 27$ °C.

Figure 4.15 illustrates the radial variation of temperature within the fuel, at the midplane of the core, as a function of time after the pulse. Table 4.10 lists temperatures and heat fluxes as function of time after a 2.1% ($3.00) reactivity insertion in a reactor initially at zero power. The CHFR is based on the critical heat flux of 1.49 MW m^{-1} from Eqs. (3) and (4) and from Table 4.2 for saturated boiling. Figure 4A.3 of Appendix A, using the Ellion data, indicates a Leidenfrost temperature in excess of 500°C. Thus transition boiling, but not fully developed film boiling might be expected for a short time after the end of a pulse.

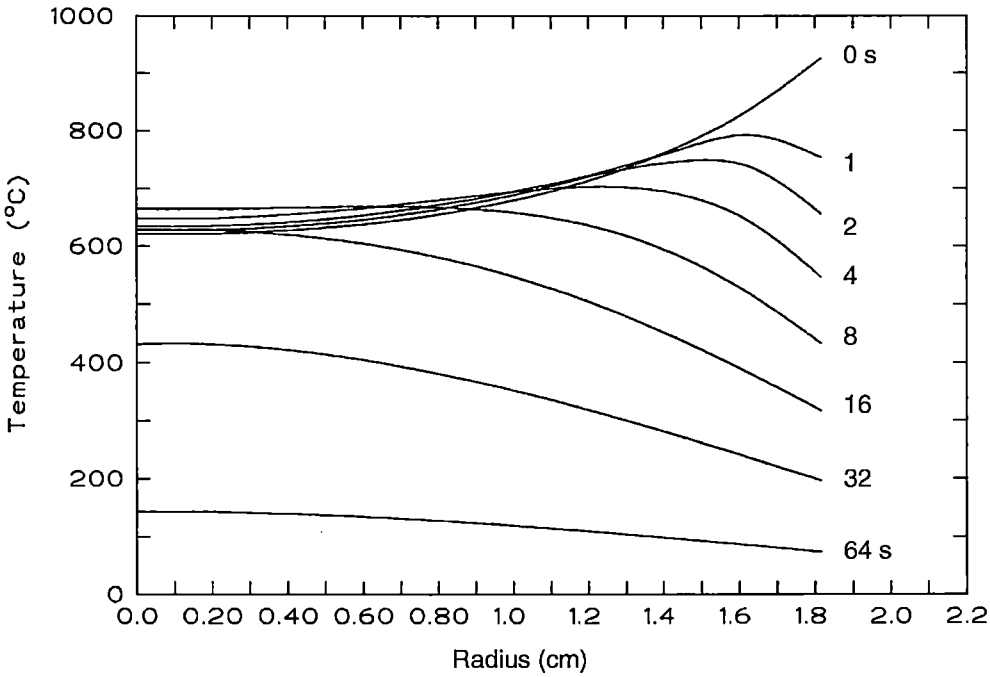


Figure 4. 15, Midplane Radial Variation of Temperature Within the Fuel Subsequent to a $3.00 Pulse.

Table 4.10, Heat Flux and Fuel Temperatures Following a $3.00 Pulse from Zero Power, with 27(°C) Coolant Ambient Temperature.

Time (s)	Q'' (W m^{-2})	CHFR	Fuel outside Temp. (°C)	Clad surface Temp. (°C)
0	-	-	953	-
1	3.57×10^5	4.2	781	224
2	7.34×10^5	2.0	683	432
4	8.52×10^5	1.7	574	498
8	7.54×10^5	2.0	461	443
16	5.71×10^5	2.6	344	342
32	3.46×10^5	4.3	224	218
64	1.04×10^5	14.4	100	84

4.6 Thermal Hydraulic Design and Analysis

A balance between the buoyancy driven pressure gain and the frictional and acceleration pressure losses accrued by the coolant in its passage through the core determines the coolant mass flow rate through the core, and the corresponding coolant temperature rise. The buoyancy pressure gain is given by

$$\Delta p_g = \rho_o \beta_o \Delta T g L, \tag{11}$$

in which ρ_o and β_o are the density and volumetric expansion coefficient at core inlet conditions (27°C, 0.15285 Mpa), g is the acceleration of gravity, 9.8 cm^2 s^{-1}, ΔT is the temperature rise through the core, and L is the height of the core (between gridplates), namely, 0.556 m. The frictional pressure loss is given by

$$\Delta p_f = \frac{\dot{m}^2 f L}{2 A^2 D_h \rho_o}, \tag{12}$$

in which $\dot{m}$ is the coolant mass flow rate (kg s^{-1}) in a unit cell approximated as the equivalent annulus surrounding a single fuel element, A is the flow area, namely, 0.00062 m^2, and D_h is the hydraulic diameter, namely, 0.02127 m. The friction factor f for laminar flow through the annular area is given by 100 Re^{-1} (Shah & London 1978), in which the Reynolds number is given by $D_h \dot{m} / A \mu_o$ in which μ_o is the dynamic viscosity at core inlet conditions.

Entrance of coolant into the core is from the side, above the lower grid plate (see Section 4.2.5), and the entrance pressure loss would be expected to be negligible. The exit contraction loss is given by

$$\Delta p_e = \frac{\dot{m}^2 K}{2 \rho_o A^2}. \tag{13}$$

The coefficient K is calculated from geometry of an equilateral-triangle spacer in a circular opening, for which

$$K \cong \left[\frac{A_t}{A_c}\right]^2 = \left[\frac{3 * R^2 \sin 60^o \cos 60^o}{\pi * R^2}\right] = 0.171, \qquad (14)$$

where R is the radius of the opening in the upper grid plate. Equations (12) through (14), solved simultaneously yield the mass flow rates per fuel element, and coolant temperature rises through the core listed in Table 4.11.

Table 4.11, Coolant Flow Rate and Temperature Rise for Natural-Convection Cooling the TRIGA Reactor During Steady-State Operations.

P (kWt)	$\dot{m}$ (kg s^{-1})	ΔT (°C)
50	0.047	3.1
100	0.061	4.7
200	0.077	7.5
300	0.090	9.6
400	0.100	11.5
500	0.108	13.3
750	0.125	17.2
1000	0.139	20.6
1250	0.150	23.8

4.7 Safety Limit

As described in 3.5.1 (Fuel System) and NUREG 1282, fuel temperature limits both steady-state and pulse-mode operation. The fuel temperature limit stems from potential hydrogen outgassing from the fuel and the subsequent stress produced in the fuel element clad material by heated hydrogen gas. Yield strength of cladding material decreases at a temperature of 500°C; consequently, limits on fuel temperature change for cladding temperatures greater than 500°C. A maximum temperature of 1150°C (with clad < 500°C) and 950°C (with clad > 500°C) for U-ZrH ($H/Zr_{1.65}$) will limit internal fuel cladding stresses that might lead to clad integrity (NUREG 1282) challenges.

4.8 Operating Limits

4.8.1 Operating Parameters

The main safety consideration is to maintain the fuel temperature below the value that would result in fuel damage. Setting limits on other operating parameters, that is, limiting safety system settings, controls the fuel temperature. The operating parameters established for the KSU TRIGA reactor are:

- Steady-state power level
- Fuel temperature measured by thermocouple during pulsing operations
- Maximum step reactivity insertion of transient rod

4.8.2 Limiting Safety System Settings

Heat transfer characteristics (from the fuel to the pool) controls fuel temperature during normal operations. As long as thermal hydraulic conditions do not cause critical heat flux to be exceeded, fuel temperature remains well below any limiting value. Figure 4.13 illustrates that critical heat flux is not reached over a wide range of pool temperatures and power levels. As indicated in Figure 4.14, the ratio of actual to critical heat flux is at least 2.0 for temperatures less than 100°C bulk pool water temperature for 1.25 MW operation. Operation at less than 1.25 MW ensures fuel temperature limits are not exceeded by a wide margin.

Limits on the maximum excess reactivity assure that operations during pulsing do not produce a power level (and generate the amount of energy) that would cause fuel-cladding temperature to exceed these limits; no other safety limit is required for pulsed operation.

4.8.3 Safety Margins

For 1,250 kWth steady-state operations, the critical heat flux ratio remains above 2.0 for a core with 85 fuel elements and a maximum radial power peaking factor of 1.63 assuming a coolant inlet temperature of 49°C. The proposed Technical Specifications limit of 44°C on pool inlet temperature ensures that the DNBR will be at least 2.0 during steady-state operation. Limiting pool inlet water temperature to no greater than 44°C (or 37°C with an experiment installed in an interstitial flux-wire port) will ensure that the pool water does not reach temperatures associated with excessive amounts of nucleate boiling.

Normal pulsed operations initiated from power levels below 10 kW with a $3.00 reactivity insertion result in maximum hot spot temperatures of 746°C, a 34% margin to the fuel temperature limit. As indicated in Chapter 13, pulsed reactivity insertions of $3.00 from initial conditions of power operation can result in a maximum hot spot temperature of 869°C. Although administratively controlled and limited by an interlock, this pulse would still result in a 15% margin to the fuel temperature safety limit for cladding temperatures below 500°C.

Analysis shows that cladding temperatures will remain below 500°C when fuel is in water except following large pulses. However, mechanisms that can cause cladding temperature to achieve 500°C (invoking a 950°C fuel temperature limit) automatically limit fuel temperature as heat is transferred from the fuel to the cladding.

Immediately following a maximum pulsed reactivity additions, heat transfer driven by fuel temperature can cause cladding temperature to rise above 500°C, but the heat transfer simultaneously cools the fuel to much less than 950°C.

If fuel rods are placed in an air environment immediately following long-term, high power operation, cladding temperature can essentially equilibrate with fuel temperature. In worst-case air-cooling scenarios, cladding temperature can exceed 500°C, but fuel temperature is significantly lower than the temperature limit for cladding temperatures greater than 500°C.

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Appendix 4-A
Post-Pulse Fuel and Cladding Temperature

This discussion is reproduced from Safety Analysis Reports for the University of Texas Reactor Facility (UTA 1991) and the McClellan Nuclear Radiation Center (MNRC 1998).

The following discussion relates the element clad temperature and the maximum fuel temperature during a short time after a pulse. The radial temperature distribution in the fuel element immediately following a pulse is very similar to the power distribution shown in Figure 4A.1. This initial steep thermal gradient at the fuel surface results in some heat transfer during the time of the pulse so that the true peak temperature does not quite reach the adiabatic peak temperature. A large temperature gradient is also impressed upon the clad which can result in a high heat flux from the clad into the water. If the heat flux is sufficiently high, film boiling may occur and form an insulating jacket of steam around the fuel elements permitting the clad temperature to tend to approach the fuel temperature. Evidence has been obtained experimentally which shows that film boiling has occurred occasionally for some fuel elements in the Advanced TRIGA Prototype Reactor located at GA Technologies [Coffer 1964]. The consequence of this film boiling was discoloration of the clad surface.

Thermal transient calculations were made using the RAT computer code. RAT is a 2-D transient heat transport code developed to account for fluid flow and temperature dependent material properties. Calculations show that if film boiling occurs after a pulse it may take place either at the time of maximum heat flux from the clad, before the bulk temperature of the coolant has changed appreciably, or it may take place at a much later time when the bulk temperature of the coolant has approached the saturation temperature, resulting in a markedly reduced threshold for film boiling. Data obtained by Johnson et al. [1961] for transient heating of ribbons in 100°F water, showed burnout fluxes of 0.9 to 2.0 Mbtu ft^{-2} hr^{-1} for e-folding periods from 5 to 90 milliseconds. On the other hand, sufficient bulk heating of the coolant channel between fuel elements can take place in several tenths of a second to lower the departure from nucleate boiling (DNB) point to approximately 0.4 Mbtu ft^{-2} hr^{-1}. It is shown, on the basis of the following analysis, that the second mode is the most likely; i.e., when film boiling occurs it takes place under essentially steady-state conditions at local water temperatures near saturation.

A value for the temperature that may be reached by the clad if film boiling occurs was obtained in the following manner. A transient thermal calculation was performed using the radial and axial power distributions in Figures 4A.1and 4A.2, respectively, under the assumption that the thermal resistance at the fuel-clad interface was nonexistent. A boiling heat transfer model, as shown in Figure 4A.3, was used in order to obtain an upper limit for the clad temperature rise. The model used the data of McAdams [1954] for subcooled boiling and the work of Sparrow and Cess [1962] for the film boiling regime. A conservative estimate was obtained for the minimum heat flux in film boiling by using the correlations of Speigler et al. [1963], Zuber [1959], and Rohsenow and Choi [1961] to find the minimum temperature point at which film boiling could occur. This calculation gave an upper limit of 760°C clad temperature for a peak initial fuel temperature of 1000°C, as shown in Figure. 4A.4. Fuel temperature distributions for this case are shown in Figure 4A.5 and the heat flux into the water from the clad is shown in Figure 4A.6. In this limiting case, DNB occurred only 13 milliseconds after the pulse, conservatively calculated

assuming a steady-state DNB correlation. Subsequently, experimental transition and film boiling data were found to have been reported by Ellion [9] for water conditions similar to those for the TRIGA system. The Ellion data show the minimum heat flux, used in the limiting calculation described above, was conservative by a factor of 5. An appropriate correction was made which resulted in a more realistic estimate of 470°C as the maximum clad temperature expected if film boiling occurs. This result is in agreement with experimental evidence obtained for clad temperatures of 400°C to 500°C for TRIGA Mark F fuel elements which have been operated under film boiling conditions [Coffer et al. 1965].

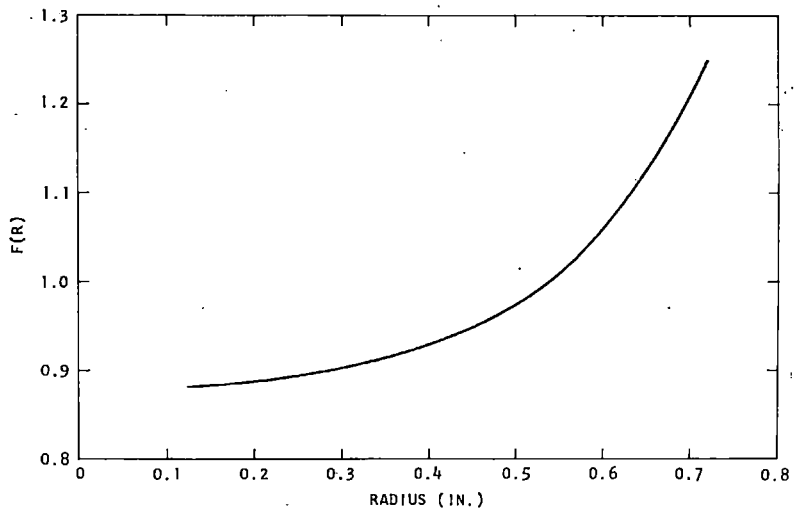

Figure 4A.1. Representative Radial Variation of Power Within the TRIGA Fuel Rod

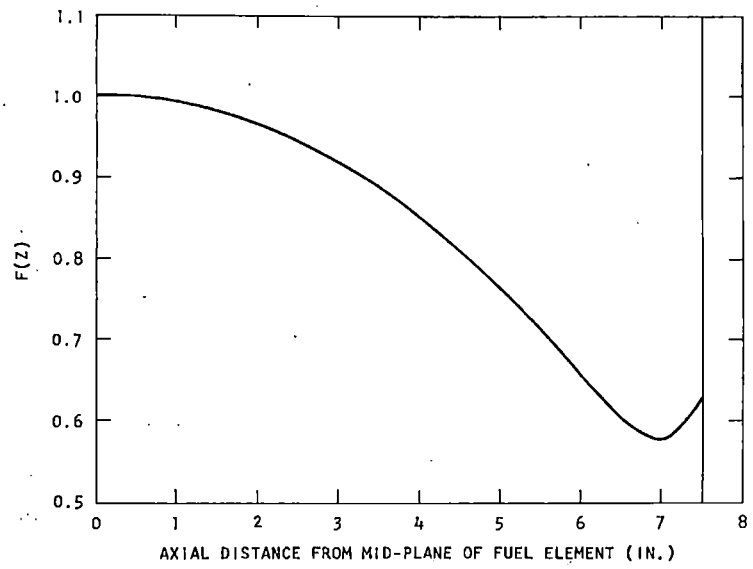

Figure 4A.2, Representative Axial Variation of Power Within the TRIGA Fuel Rod.

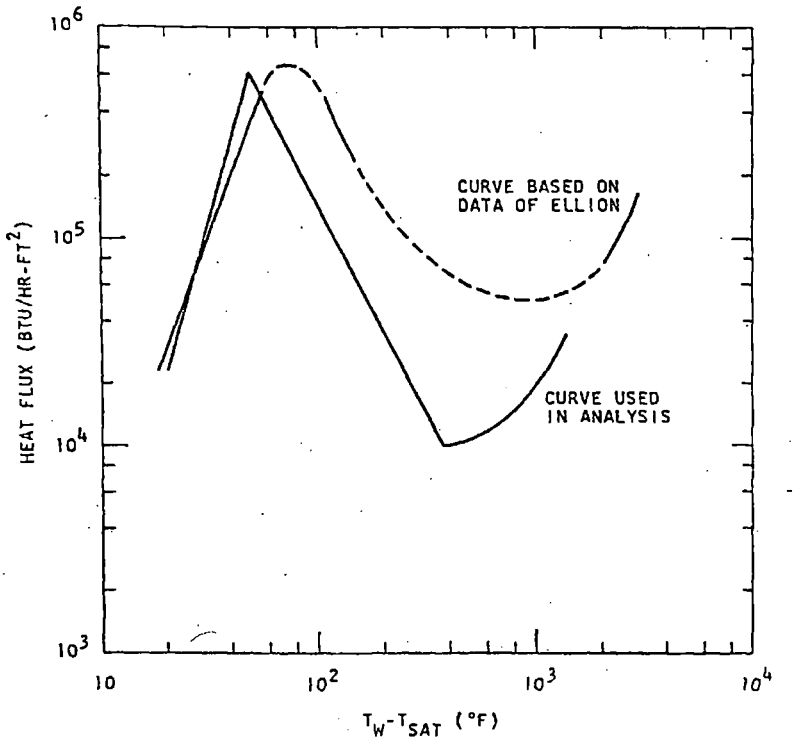

Figure 4A.3, Subcooled Boiling Heat Transfer for Water.

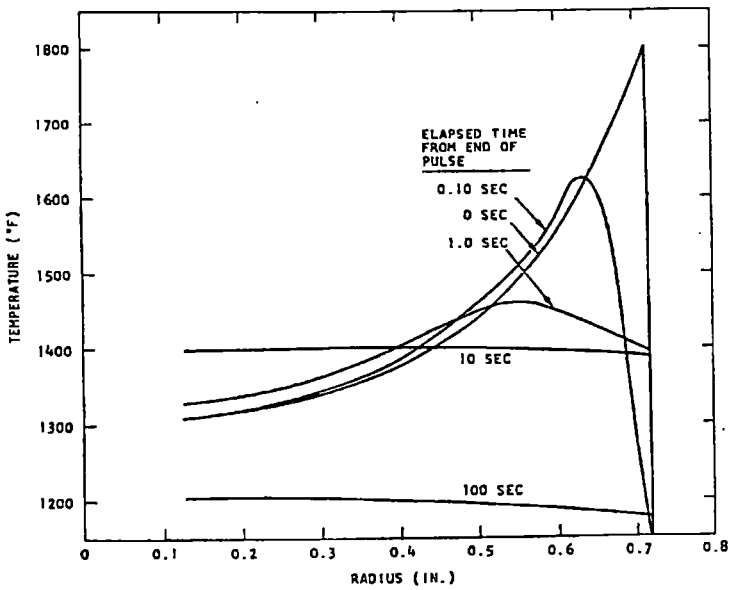

Figure 4A.4, Fuel Body Temperature at the Midplane of a Well-Bonded Fuel Element After Pulse.

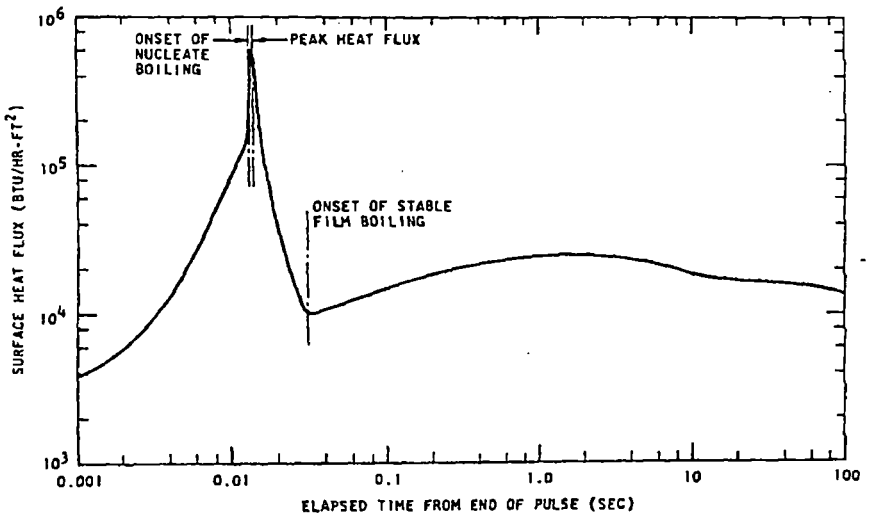

Figure 4A.5, Surface Heat Flux at the Midplane of a Well Bonded Fuel Element After a Pulse.

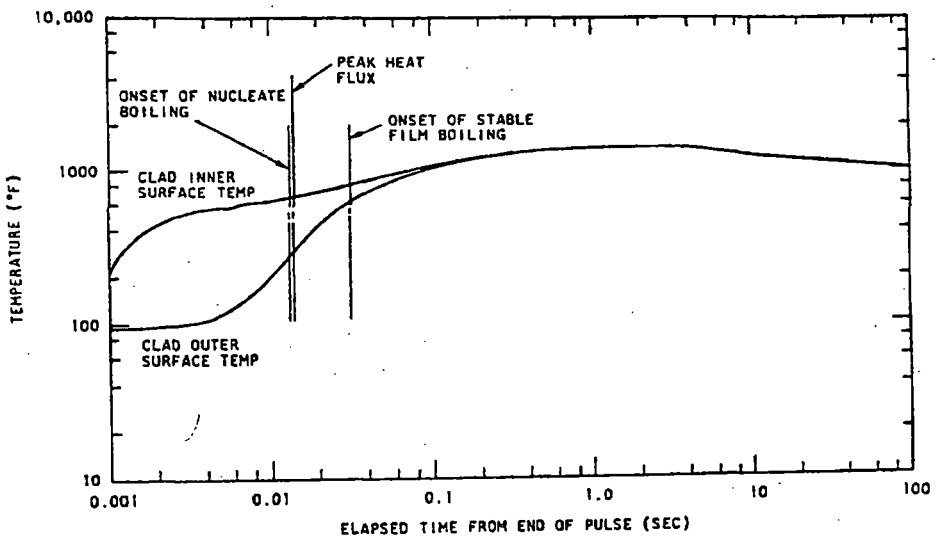

Figure 4A.6, Clad Temperature at Midpoint of Well-Bonded Fuel Element.

The preceding analysis assessing the maximum clad temperatures associated with film boiling

assumed no thermal resistance at fuel-clad interface. Measurements of fuel temperatures as a function of steady-state power level provide evidence that after operating at high fuel temperatures, a permanent gap is produced between the fuel body and the clad by fuel expansion. This gap exists at all temperatures below the maximum operating temperature. (See, for example, Figure 16 in the Coffer report [1965].) The gap thickness varies with fuel temperature and clad temperature so that cooling of the fuel or overheating of the clad tends to widen the gap and decrease the heat transfer rate. Additional thermal resistance due to oxide and other films on the fuel and clad surfaces is expected. Experimental and theoretical studies of thermal contact resistance have been reported [Fenech and Rohsenow 1959, Graff 1960, Fenech and Henry 1962] which provide insight into the mechanisms involved. They do not, however, permit quantitative prediction of this application because the basic data required for input are presently not fully known. Instead, several transient thermal computations were made using the RAT code. Each of these was made with an assumed value for the effective gap conductance, in order to determine the effective gap coefficient for which departure from nucleate boiling is incipient. These results were then compared with the incipient film boiling conditions of the 1000°C peak fuel temperature case.

For convenience, the calculations were made using the same initial temperature distribution as was used for the preceding calculation. The calculations assumed a coolant flow velocity of 1 ft per second, which is within the range of flow velocities computed for natural convection under various steady-state conditions for these reactors. The calculations did not use a complete boiling curve heat transfer model, but instead, included a convection cooled region (no boiling) and a subcooled nucleate boiling region without employing an upper DNB limit. The results were analyzed by inspection using the extended steady-state correlation of Bernath [1960] which has been reported by Spano [1964] to give agreement with SPERT II burnout results within the experimental uncertainties in flow rate.

The transient thermal calculations were performed using effective gap conductances of 500, 375, and 250 Btu ft^{-2} hr^{-1} $°F^{-1}$. The resulting wall temperature distributions were inspected to determine the axial wall position and time after the pulse which gave the closest approach between the local computed surface heat flux and the DNB heat flux according to Bernath. The axial distribution of the computed and critical heat fluxes for each of the three cases at the time of closest approach is given in Figures 4A.7 through 4A.9. If the minimum approach to DNB is corrected to TRIGA Mark F conditions and cross-plotted, an estimate of the effective gap conductance of 450 Btu ft^{-2} hr^{-1} $°F^{-1}$ is obtained for incipient burnout so that the case using 500 is thought to be representative of standard TRIGA fuel.

The surface heat flux at the midplane of the element is shown in Figure 4A.10 with gap conductance as a parameter. It may be observed that the maximum heat flux is approximately proportional to the heat transfer coefficient of the gap, and the time lag after the pulse for which the peak occurs is also increased by about the same factor. The closest approach to DNB in these calculations did not necessarily occur at these times and places, however, as indicated on the curves of Figures 4A.7 through 4A.9. The initial DNB point occurred near the core outlet for a local heat flux of about 340 kBtu ft^{-2} hr^{-1} $°F^{-1}$ according to the more conservative Bernath correlation at a local water temperature approaching saturation.

This analysis indicates that after operation of the reactor at steady-state power levels of 1 MW(t), or after pulsing to equivalent fuel temperatures, the heat flux through the clad is reduced and therefore reduces the likelihood of reaching a regime where there is a departure from nucleate boiling. From the foregoing analysis, a maximum temperature for the clad during a pulse which gives a peak adiabatic fuel temperature of 1000°C is conservatively estimated to be 470°C.

As can be seen from Figure 4.7, the ultimate strength of the clad at a temperature of 470°C is 59,000 psi. If the stress produced by the hydrogen over pressure in the can is less than 59,000 psi, the fuel element.will not undergo loss of containment. Referring to Figure 4.8, and considering U-ZrH fuel with a peak temperature of 1000°C, one finds the stress on the clad to be 12,600 psi. Further studies show that the hydrogen pressure that would result from a transient for which the peak fuel temperature is 1150°C would not produce a stress in the clad in excess of its ultimate strength. TRIGA fuel with a hydrogen to zirconium ratio of at least 1.65 has been pulsed to temperatures of about 1150°C without damage to the clad [Dee et al. 1966].

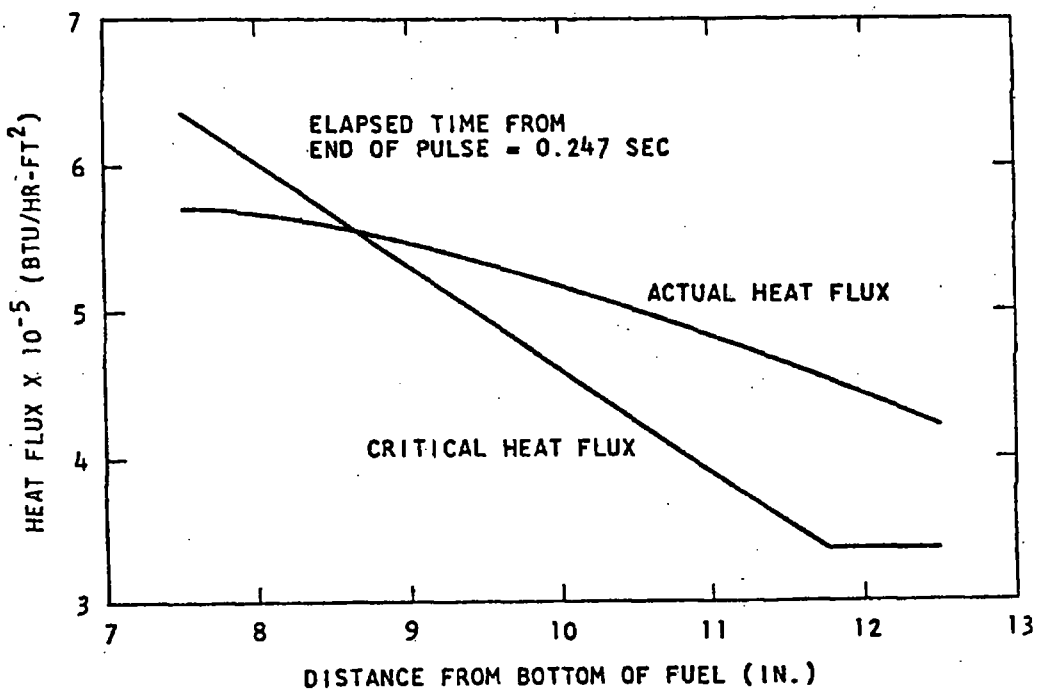

Figure 4A.7, Surface Heat Flux Distribution for Standard Non-Gapped (h_{gap}= 500 Btu/h ft^2 °F) Fuel Element After a Pulse.

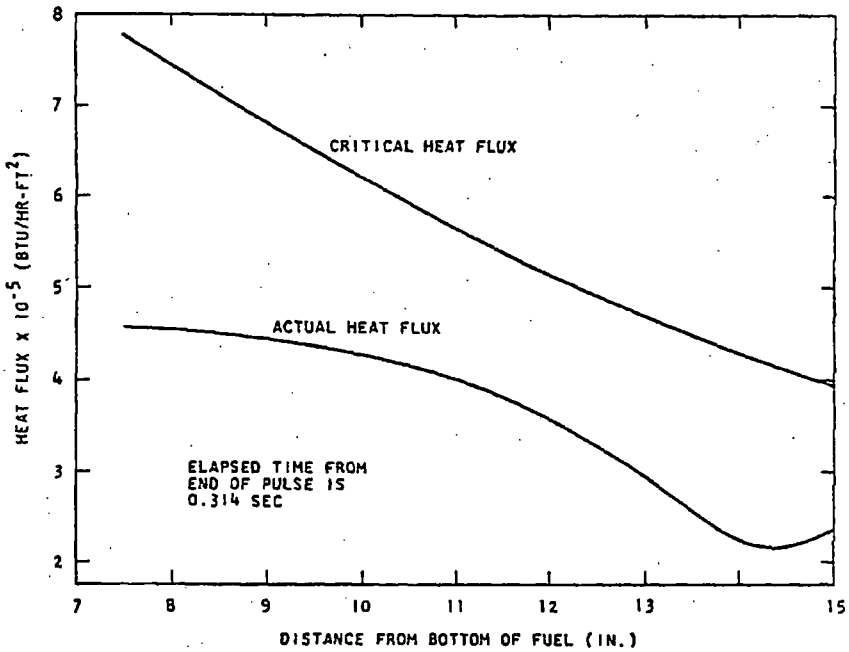

Figure 4A.8, Surface Heat-Flux Distribution for Standard Non-Gapped Fuel Element (h_{gap} = 375 Btu/h ft^2 °F) After a Pulse.

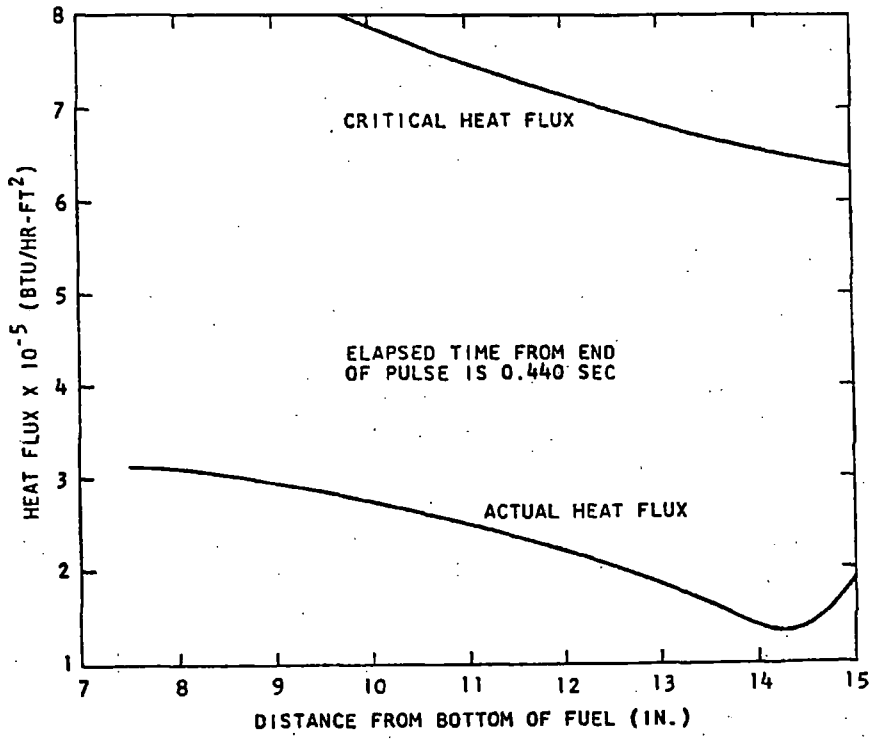

Figure 4A.9, Surface Heat-Flux Distribution for Standard Non-Gapped Fuel Element (h_{gap} = 250 Btu/h ft^2 °F) After a Pulse.

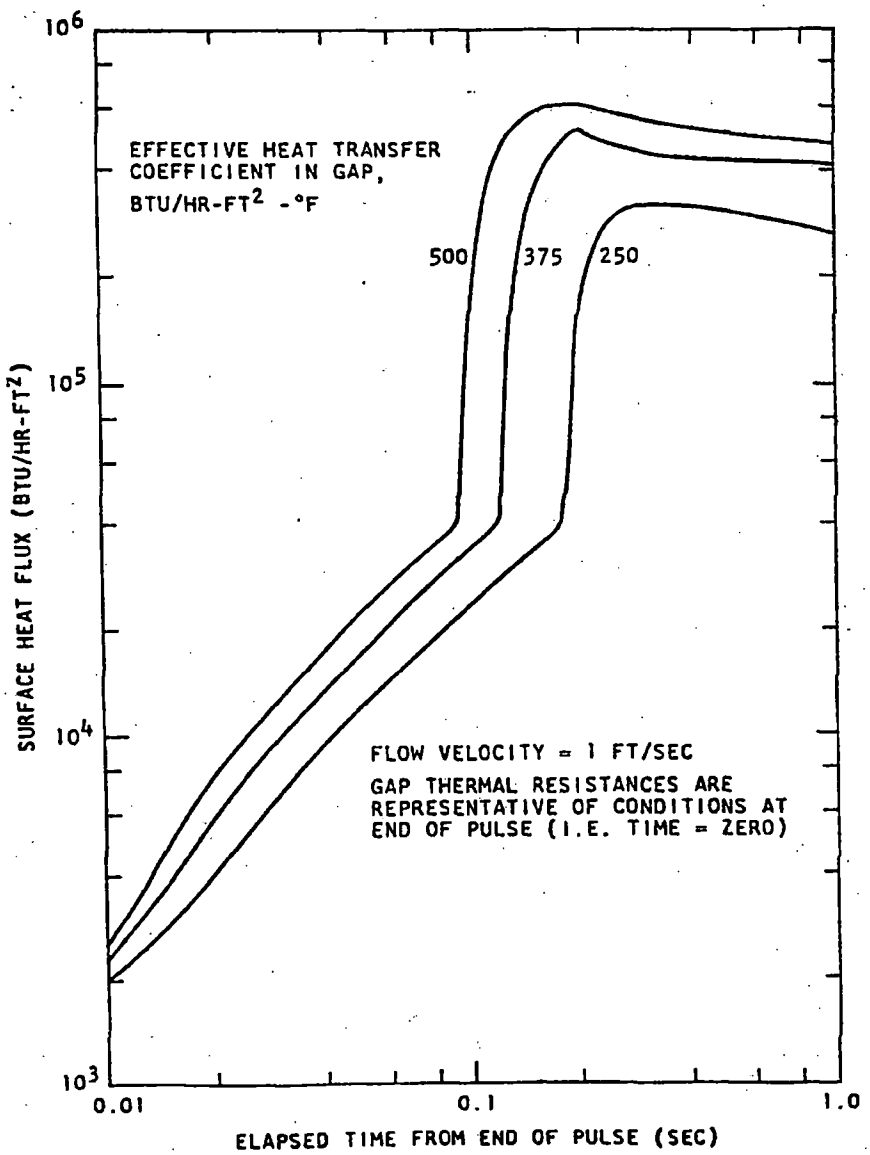

Figure 4A.10, Surface Heat Flux at Midpoint vs. Time for Standard Non-Gapped Fuel Element After a Pulse.

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Appendix B
Water Properties at Nominal Operating Conditions

Data for 16 Feet of Water over the Core

T_{pool} °C	ρ_T[1] kg m^{-3}	$\rho_{f,16}$[1] kg m^{-3}	$P_{h,16}$[2] kPa	$h_{f,16}$[1] kJ kg^{-1}	$h_{g,16}$[1] kJ kg^{-1}	$\rho_{g,16}$[1] kg m^{-3}	$T_{sat,16}$[1] °C	$q''_{sat,16}$[3] W m^{-2}	q"sub[4] W m^{-2}
15	999.21	950.00	47.79	465.10	2692.64	0.85	110.89	1553.842	7239.19
20	998.32	950.01	47.74	465.05	2692.63	0.85	110.88	1552.078	6931.74
25	997.16	950.02	47.69	465.01	2692.59	0.85	110.87	1549.496	6622.60
30	995.75	950.03	47.62	464.95	2692.59	0.85	110.86	1547.118	6311.82
35	994.12	950.04	47.54	464.89	2692.57	0.85	110.84	1543.981	5999.91
40	992.29	950.06	47.46	464.81	2692.54	0.85	110.83	1540.446	5688.24
45	990.27	950.07	47.36	464.73	2692.51	0.85	110.81	1536.512	5376.29
50	988.07	950.09	47.25	464.64	2692.48	0.85	110.78	1532.205	5064.36
55	985.70	950.11	47.14	464.54	2692.45	0.85	110.76	1527.561	4753.90
60	983.18	950.12	47.02	464.44	2692.41	0.85	110.74	1522.575	4444.85
65	980.50	950.14	46.89	464.33	2692.37	0.85	110.71	1517.255	4136.85
70	977.69	950.17	46.76	464.21	2692.33	0.84	110.68	1511.666	3830.73
75	974.74	950.19	46.62	464.09	2692.29	0.84	110.65	1505.778	3526.89
80	971.66	950.21	46.47	463.96	2692.24	0.84	110.62	1499.613	3225.47
85	968.45	950.23	46.32	463.83	2692.19	0.84	110.59	1493.199	2926.81
90	965.12	950.26	46.16	463.69	2692.15	0.84	110.56	1486.527	2631.05
95	961.68	950.29	45.99	463.55	2692.09	0.84	110.53	1479.626	2338.47
97	960.27	950.30	45.92	463.49	2692.07	0.84	110.51	1472.944	2216.11
99	958.84	950.31	45.86	463.43	2692.05	0.84	110.50	1466.058	2095.18

Data for 13 Feet of Water over the Core

T_{pool} °C	ρ_T[1] kg m^{-3}	$\rho_{f,13}$[1] kg m^{-3}	$P_{h,13}$[2] kPa	$h_{f,13}$[1] kJ kg^{-1}	$h_{g,13}$[1] kJ kg^{-1}	$\rho_{g,13}$[1] kg m^{-3}	$T_{sat,13}$[1] °C	$q''_{sat,13}$[3] W m^{-2}	q"sub[4] W m^{-2}
15	999.21	951.43	38.83	457.21	2689.85	0.80	109.03	1513.00	6964.74
20	998.32	951.43	38.79	457.18	2689.84	0.80	109.02	1511.32	6857.12
25	997.16	951.44	38.75	457.13	2689.82	0.80	109.01	1509.15	6543.62
30	995.75	951.45	38.69	457.09	2689.80	0.80	109.00	1505.85	6229.30
35	994.12	951.46	38.63	457.03	2689.78	0.80	108.99	1503.13	5913.58
40	992.29	951.47	38.56	456.96	2689.76	0.80	108.97	1500.16	5597.21
45	990.27	951.49	38.48	456.89	2689.74	0.80	108.96	1496.38	5281.90
50	988.07	951.50	38.39	456.82	2689.71	0.80	108.94	1492.25	4966.66
55	985.70	951.51	38.30	456.73	2689.68	0.80	108.92	1487.78	4652.39
60	983.18	951.53	38.20	456.64	2689.65	0.80	108.90	1482.99	4339.60
65	980.50	951.55	38.10	456.55	2689.61	0.80	108.87	1477.90	4027.94
70	977.69	951.57	37.99	456.45	2689.58	0.80	108.85	1472.52	3718.83
75	974.74	951.58	37.88	456.35	2689.54	0.80	108.83	1466.86	3412.07
80	971.66	951.60	37.76	456.24	2689.50	0.80	108.80	1460.95	3107.29
85	968.45	961.62	37.63	456.12	2689.46	0.80	108.77	1458.59	2812.63
90	965.12	951.64	37.50	456.01	2689.42	0.79	108.75	1448.37	2506.90
95	961.68	951.67	37.37	455.89	2689.38	0.79	108.72	1441.74	2211.19
97	960.27	951.68	37.31	455.84	2689.36	0.79	108.70	1435.27	2087.92
99	958.84	951.68	37.26	455.78	2689.34	0.79	108.69	1428.60	1966.16

Common Data

T °C	P_{atm} kPa	c_p[1] kJ kg^{-1} k^{-1}	σ N m^{-1}
15	99.83	4.23080	0.07149
20	99.83	4.23080	0.07120
25	99.83	4.23080	0.07083
30	99.83	4.23080	0.07039
35	99.83	4.23070	0.06989
40	99.83	4.23070	0.06932
45	99.83	4.23070	0.06869
50	99.83	4.23070	0.06800
55	99.83	4.23060	0.06727
60	99.83	4.23060	0.06649
65	99.83	4.23060	0.06566
70	99.83	4.23050	0.06480
75	99.83	4.23050	0.06390
80	99.83	4.23050	0.06297
85	99.83	4.23040	0.06201
90	99.83	4.23040	0.06102
95	99.83	4.23040	0.06001
97	99.83	4.23030	0.05898
99	99.83	4.23030	0.05793

NOTE[1]: 1967 ASME (IFC) Steam Tables & IAPWS-IF97

*NOTE[2]:kPa =Heigth(ft)*12(in/ft)*0.0254(meters/in)*Density(kg/m^3)*9.80665/1000*

NOTE[3]: $q''_{SAT} = 0.149 \cdot \rho_g^{0.5} \cdot \left(h_{g,sat} - h_{f,sat}\right) \cdot \left(\cdot\, g \cdot \sigma \cdot \left\{\rho_f - \rho_g\right\}\right)^{1/4}$

NOTE[4]: $q''_{sub} = q''_{SAT} \cdot \left(1 + 0.1 \cdot \left(\frac{\rho_f}{\rho_g}\right)^{3/4} \cdot \frac{c_{p,f} \cdot \left(T_{SAT} - T_{sub}\right)}{h_{g,sat} - h_{f,sat}}\right)$

NOTE:[5]:

$\sigma = 1.000\text{E-}11 \cdot \text{T}^4 + 7.370\text{E-}09 \cdot \text{T}^3 - 1.969\text{E-}06 \cdot \text{T}^2 + 4.709\text{E-}06 \cdot \text{T} + 7.1833\text{E-}02$

Table of Contents

1. **DEFINITIONS** TS-1

2. **SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS** TS-8
   - 2.1 Fuel Element Temperature Safety Limit TS-8
     - 2.1.1. Applicability TS-8
     - 2.1.2. Objective TS-8
     - 2.1.3. Specification TS-8
     - 2.1.4. Actions TS-8
     - 2.1.5. Basis TS-8
   - 2.2 Limiting Safety System Settings TS-10
     - 2.2.1. Applicability TS-10
     - 2.2.3. Objective TS-10
     - 2.2.4. Specification TS-10
     - 2.2.5. Actions TS-10
     - 2.2.6. Basis TS-10

3. **LIMITING CONDITIONS FOR OPERATIONS** TS-11
   - 3.1 CORE REACTIVITY TS-11
     - 3.1.1. Applicability TS-11
     - 3.1.3. Objective TS-11
     - 3.1.4. Specification TS-11
     - 3.1.5. Actions TS-12
     - 3.1.6. Basis TS-13
   - 3.2 PULSED MODE OPERATIONS TS-13
     - 3.2.1. Applicability TS-13
     - 3.2.3. Objective TS-13
     - 3.2.4. Specification TS-13
     - 3.2.5. Actions TS-13
     - 3.2.6. Basis TS-13
   - 3.3 MEASURING CHANNELS TS-14
     - 3.3.1. Applicability TS-14
     - 3.3.3. Objective TS-14
     - 3.3.4. Specification TS-14
     - 3.3.5. Actions TS-14
     - 3.3.6. Bases TS-16
   - 3.4. SAFETY CHANNEL AND CONTROL ROD OPERABILITY TS-18
     - 3.4.1. Applicability TS-18
     - 3.4.3. Objective TS-18
     - 3.4.4. Specification TS-18
     - 3.4.5. Actions TS-18
     - 3.4.6. Basis TS-19
   - 3.5 GASEOUS EFFLUENT CONTROL TS-20
     - 3.5.1. Applicability TS-20
     - 3.5.3. Objective TS-20
     - 3.5.4. Specification TS-20
     - 3.5.5. Actions TS-20
     - 3.5.6. Basis TS-21
   - 3.6 LIMITATIONS ON EXPERIMENTS TS-22
     - 3.6.1. Applicability TS-22
     - 3.6.3. Objective TS-22

3.6.4. Specification TS-22
3.6.5. Actions TS-22
3.6.6. Basis TS-23
3.7 FUEL INTEGRITY TS-24
3.7.1. Applicability TS-24
3.7.3. Objective TS-24
3.7.4. Specification TS-24
3.7.5. Actions TS-24
3.7.6. Basis TS-24
3.8 REACTOR POOL WATER TS-25
3.8.1. Applicability TS-25
3.8.3. Objective TS-25
3.8.4. Specification TS-25
3.8.5. Actions TS-25
3.8.6. Basis TS-26
3.9 Maintenance Retest Requirements TS-27
3.9.1. Applicability TS-27
3.9.3. Objective TS-27
3.9.4. Specification TS-27
3.9.5. Actions TS-27
3.9.6. Basis TS-27

4. **SURVIELLANCES** TS-28
4.1 CORE REACTIVITY TS-28
4.1.1. Objective TS-28
4.1.2. Specification TS-28
4.1.3. Basis TS-28
4.2 PULSE MODE TS-29
4.2.1. Objective TS-29
4.2.2. Specification TS-29
4.2.3. Basis TS-29
4.3 MEASURING CHANNELS TS-30
4.3.1. Objective TS-30
4.3.2. Specification TS-30
4.3.3. Basis TS-30
4.4 SAFETY CHANNEL AND CONTROL ROD OPERABILITY TS-31
4.4.1. Objective TS-31
4.4.2. Specification TS-31
4.4.3. Basis TS-32
4.5 GASEOUS EFFLUENT CONTROL TS-33
4.5.1. Objective TS-33
4.5.2. Specification TS-33
4.5.3. Basis TS-33
4.6 LIMITATIONS ON EXPERIMENTS TS-34
4.6.1. Objective TS-34
4.6.2. Specification TS-34
4.6.3. Basis TS-34
4.7 FUEL INTEGRITY TS-35
4.7.1. Objective TS-35
4.7.2. Specification TS-35
4.7.3. Basis TS-35
4.8 REACTOR POOL WATER TS-36
4.8.1. Objective TS-36

4.8.2. Specification TS-36
4.8.3. Basis TS-36
4.9 MAINTENANCE RETEST REQUIREMENTS TS-37
4.9.1. Objective TS-37
4.9.2. Specification TS-37
4.10.3. Basis TS-37
5. **DESIGN FEATURES** TS-38
5.1 REACTOR FUEL TS-38
5.1.1. Applicability TS-38
5.1.2. Objective TS-38
5.1.3. Specification TS-38
5.1.4. Basis TS-38
5.2 REACTOR FUEL AND FUELED DEVICES IN STORAGE TS-38
5.2.1. Applicability TS-38
5.2.2. Objective TS-39
5.2.3. Specification TS-39
5.2.4. Basis TS-39
5.3 REACTOR BUILDING TS-39
5.3.1. Applicability TS-39
5.3.2. Objective TS-39
5.3.3. Specification TS-39
5.3.4. Basis TS-40
5.4 EXPERIMENTS TS-40
5.4.1. Applicability TS-40
5.4.2. Objective TS-40
5.4.3. Specification TS-40
5.4.4. Basis TS-41

6. **ADMINISTRATIVE CONTROLS** TS-42
6.1 ORGANIZATION AND RESPONSIBILITIES OF PERSONNEL TS-44
6.2 REVIEW AND AUDIT TS-45
6.3 PROCEDURES TS-45
6.4 REVIEW OF PROPOSALS FOR EXPERIMENTS TS-47
6.5 EMERGENCY PLAN AND PROCEDURES TS-48
6.6 OPERATOR REQUALIFICATION TS-48
6.7 PHYSICAL SECURITY PLAN TS-48
6.8 ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS VIOLATED TS-48
6.9 ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE OCCURRENCE TS-48
6.10 PLANT OPERATING RECORDS TS-49
6.11 REPORTING REQUIREMENTS TS-50

1. DEFINITIONS

The following frequently used terms are defined to aid in the uniform interpretation of these specifications. Capitalization is used in the body of the Technical Specifications to identify defined terms.

ACTION	Actions are steps to be accomplished in the event a required condition identified in a "Specification" section is not met, as stated in the "Condition" column of "Actions." In using Action Statements, the following guidance applies: • Where multiple conditions exist in an LCO, actions are linked to the (failure to meet a "Specification") "Condition" by letters and number. • Where multiple action steps are required to address a condition, COMPLETION TIME for each action is linked to the action by letter and number. • AND in an Action Statement means all steps need to be performed to complete the action; OR indicates options and alternatives, only one of which needs to be performed to complete the action. • If a "Condition" exists, the "Action" consists of completing all steps associated with the selected option (if applicable) except where the "Condition" is corrected prior to completion of the steps
ANNUAL	12 months, not to exceed 15 months
CHANNEL CALIBRATION	A channel calibration is an adjustment of the channel to that its output responds, with acceptable range and accuracy, to known values of the parameter that the channel measures.
BIENNIAL	Every two years, not to exceed a 28 month interval
CHANNEL CHECK	A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification shall include comparison of the channel with expected values, other independent channels, or other methods of measuring the same variable.
CHANNEL TEST	A channel test is the introduction of an input signal into a channel to verify that it is operable. A functional test of operability is a channel test.
CONTROL ROD (STANDARD)	A standard control rod is one having an electric motor drive and scram capability.
CONTROL ROD (TRANSIENT)	A transient rod is one that is pneumatically operated and has scram capability.
DAILY	Prior to initial operation each day (when the reactor is operated), or before

	an operation extending more than 1 day
ENSURE	Verify existence of specified condition or (if condition does not meet criteria) take action necessary to meet condition
EXHAUST PLENUM	The air volume in the reactor bay atmosphere between the pool surface and the reactor bay exhaust fan
EXPERIMENT	An EXPERIMENT is (1) any apparatus, device, or material placed in the reactor core region (in an EXPERIMENTAL FACILITY associated with the reactor, or in line with a beam of radiation emanating from the reactor) or (2) any in-core operation designed to measure reactor characteristics.
EXPERIMENTAL FACILITY	Experimental facilities are the beamports, thermal column, pneumatic transfer system, central thimble, rotary specimen rack, and the in-core facilities (including non-contiguous single-element positions, and, in the E and F rings, as many as three contiguous fuel-element positions).
IMMEDIATE	Without delay, and not exceeding one hour. *NOTE:* *IMMEDIATE permits activities to restore required conditions for up to one hour; this does not permit or imply deferring or postponing action*
INDEPENDENT EXPERIMENT	INDEPENDENT Experiments are those not connected by a mechanical, chemical, or electrical link to another experiment
LIMITING CONDITION FOR OPERATION (LCO)	The lowest functional capability or performance levels of equipment required for safe operation of the facility.
LIMITING SAFETY SYSTEM SETTING (LSSS)	Settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit placed, the setting shall be chosen so that the automatic protective action will correct the abnormal situation before a safety limit is exceeded.
MEASURED VALUE	The measured value of a parameter is the value as it appears at the output of a MEASURING CHANNEL.
MEASURING CHANNEL	A MEASURING CHANNEL is the combination of sensor, lines, amplifiers, and output devices that are connected for the purpose of measuring the value of a process variable.
MOVABLE EXPERIMENT	A MOVABLE EXPERIMENT is one that may be moved into, out-of or near the reactor while the reactor is OPERATING.
NONSECURED EXPERIMENT	NONSECURED Experiments are these that should not move while the reactor is OPERATING, but are held in place with less restraint than a secured experiment.

OPERABLE	A system or component is OPERABLE when it is capable of performing its intended function in a normal manner
OPERATING	A system or component is OPERATING when it is performing its intended function in a normal manner.
PULSE MODE	The reactor is in the PULSE MODE when the reactor mode selection switch is in the pulse position and the key switch is in the "on" position. *NOTE:* *In the PULSE MODE, reactor power may be increased on a period of much less than l second by motion of the transient control rod.*
REACTOR SAFETY SYSTEM	The REACTOR SAFETY SYSTEM is that combination of MEASURING CHANNELS and associated circuitry that is designed to initiate reactor scram or that provides information that requires manual protective action to be initiated.
REACTOR SECURED MODE	The reactor is secured when the conditions of either item (1) or item (2) are satisfied: (1) There is insufficient moderator or insufficient fissile material in the reactor to attain criticality under optimum available conditions of moderation and reflection (2) All of the following: a. The console key is it the OFF position and the key is removed from the lock b. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives (unless the drive is physically decoupled from the control rod) c. No experiments are being moved or serviced that have, on movement, a reactivity worth greater than $1.00
REACTOR SHUTDOWN	The reactor is shutdown if it is subcritical by at least $1.00 in the REFERENCE CORE CONDITION with the reactivity worth of all experiments included.
RING	A ring is one of the five concentric bands of fuel elements surrounding the central opening (thimble) of the core. The letters B through F, with the letter B used to designate the innermost ring,
REFERENCE CORE CONDITION	The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible (<$0.30)
SAFETY CHANNEL	A safety channel is a MEASURING CHANNEL in the REACTOR SAFETY SYSTEM
SECURED EXPERIMENT	A secured EXPERIMENT is an EXPERIMENT held firmly in place by a mechanical device or by gravity providing that the weight of the EXPERIMENT is such that it cannot be moved by force of less than 60 lb.

SECURED EXPERIMENT WITH MOVABLE PARTS	A secured EXPERIMENT with movable parts is one that contains parts that are intended to be moved while the reactor is OPERATING.
SHALL (SHALL NOT)	Indicates specified action is required/(not to be performed)
SEMIANNUAL	Every six months, with intervals not greater than 8 months
SHUTDOWN MARGIN	The shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems, starting from any permissible operating condition, and that the reactor will remain subcritical without further operator action
STANDARD THERMOCOUPLE FUEL ELEMENT	A standard thermocouple fuel element is stainless steel clad fuel element containing three sheathed thermocouples embedded in the fuel element.
STEADY-STATE MODE	The reactor is in the steady-state mode when the reactor mode selector switch is in either the manual or automatic position and the key switch is in the "on" position.
TECHNICAL SPECIFICATION VIOLATION	A violation of a Safety Limit occurs when the Safety Limit value is exceeded. A violation of a Limiting Safety System Setting or Limiting Condition for Operation) occurs when a "Condition" exists which does not meet a "Specification" and the corresponding "Action" has not been met within the required "Completion Time." If the "Action" statement of an LSSS or LCO is completed or the "Specification" is restored within the prescribed "Completion Time," a violation has not occurred. *NOTE* *"Condition," "Specification," "Action," and "Completion Time" refer to applicable titles of sections in individual Technical Specifications*

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Fuel Element Temperature Safety Limit

2.1.1 Applicability

This specification applies when the reactor in STEADY STATE MODE and the PULSE MODE.

2.1.2 Objective

This SAFETY LIMIT ensures fuel element cladding integrity

2.1.3 Specification

(1)	Stainless steel clad, high-hydride fuel element temperature SHALL NOT exceed 1150°C.
(2)	Steady state fuel temperature shall not exceed 750°C.

2.1.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A Stainless steel clad, high-hydride fuel element temperature exceeds 1150°C. OR Fuel temperature exceeds 750°C in steady state conditions	A.1 Establish SHUTDOWN condition AND A.2 Report per Section 6.8	A.1 IMMEDIATE A.2 Within 24 hours

2.1.5 Bases

Safety Analysis Report, Section 3.5.1 (Fuel System) identifies design and operating constraints for TRIGA fuel that will ensure cladding integrity is not challenged.

NUREG 1282 identifies the safety limit for the high-hydride ($ZrH_{1.7}$) fuel elements with stainless steel cladding based on the stress in the cladding (resulting from the hydrogen pressure from the dissociation of the zirconium hydride). This stress will remain below the yield strength of the stainless steel cladding with fuel temperatures below 1,150°C. A change in yield strength occurs for stainless steel cladding temperatures of 500°C, but there is no scenario for fuel cladding to achieve 500°C while submerged; consequently the safety limit during reactor operations is 1,150°C.

Therefore, the important process variable for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification, and it is readily measured. During operation, fission product gases and dissociation of the hydrogen and zirconium builds up gas inventory in internal components and spaces of the fuel elements. Fuel temperature acting on these gases controls fuel element internal pressure. Limiting the maximum temperature prevents excessive internal pressures that could be generated by heating these gases.

Fuel growth and deformation can occur during normal operations, as described in General Atomics technical report E-117-833. Damage mechanisms include fission recoils and fission gases, strongly influenced by thermal gradients. Operating with maximum long-term, steady state fuel temperature of 750°C does not have significant time- and temperature-dependent fuel growth.

2.2 Limiting Safety System Settings (LSSS)

2.2.1 Applicability

This specification applies when the reactor in STEADY STATE MODE

2.2.2 Objective

The objective of this specification is to ensure the safety limit is not exceeded.

2.2.3 Specifications

(1)	Power level SHALL NOT exceed 1,250 kW (th) in STEADY STATE MODE of operation

2.2.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Steady state power level exceeds 1,250 kW (th)	A.1 Reduce power to less than 1,250 kW (th) OR A.2. Establish REACTOR SHUTDOWN condition	A.1 IMMEDIATE A.2. IMMEDIATE

2.2.5 Bases

Analysis in Chapter 4 demonstrates that if operating thermal (th) power is 1,250 kW, the maximum steady state fuel temperature is less than the safety limit for steady state operations by a large margin. For normal pool temperature, calculations in Chapter 4 demonstrate that the heat flux of the hottest area of the fuel rod generating the highest power level in the core during operations is less than the critical heat flux by a large margin up to the maximum permitted cooling temperatures; margin remains even at temperatures approaching bulk boiling for atmospheric conditions. Therefore, steady state operations at a maximum of 1,250 kW meet requirements for safe operation with respect to maximum fuel temperature and thermal hydraulics by a wide margin. Steady state operation of 1,250 kW was assumed in analyzing the loss of cooling and maximum hypothetical accidents. The analysis assumptions are protected by assuring that the maximum steady state operating power level is 1,250 kW.

In 1968 the reactor was licensed to operate at 250 kW with a minimum reactor safety system scram set point required by Technical Specifications at 110% of rated full power, with the scram set point set conservatively at 104%. In 1993 the original TRIGA power level channels were replaced with more reliable, solid state instrumentation. The actual safety system setting will be chosen to ensure that a scram will occur at a level that does not exceed 1,250 kW.

3. Limiting Conditions for Operation (LCO)

3.1 Core Reactivity

3.1.1 Applicability

These specifications are required prior to entering STEADY STATE MODE or PULSING MODE in OPERATING conditions; reactivity limits on experiments are specified in Section 3.8.

3.1.2 Objective

This LCO ensures the reactivity control system is OPERABLE, and that an accidental or inadvertent pulse does not result in exceeding the safety limit.

3.1.3 Specification

(1)	The maximum available core reactivity (excess reactivity) with all control rods fully withdrawn is less than $4.00 when: 1. REFERENCE CORE CONDITIONS exists 2. No experiments with net negative reactivity worth are in place
(2)	The reactor is capable of being made subcritical by a SHUTDOWN MARGIN more than $0.50 under REFERENCE CORE CONDITIONS and under the following conditions: 1. The highest worth control rod is fully withdrawn 2. The highest worth NONSECURED EXPERIMENT is in its most positive reactive state, and each SECURED EXPERIMENT with movable parts is in its most reactive state.

3.1.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactivity with all control rods fully withdrawn exceeds $4.00	A.1 ENSURE REACTOR SHUTDOWN AND A.2 Configure reactor to meet LCO	A.1 IMMEDIATE A.2 Prior to continued operations

B. The reactor is not subcritical by more than $0.50 under specified conditions	B.1.a ENSURE control rods fully inserted AND B.1.b Secure electrical power to the control rod circuits AND B.1.c Secure all work on in-core experiments or installed control rod drives AND B.2 Configure reactor to meet LCO	B.1 IMMEDIATE B.2 Prior to continued operations

3.1.5 Bases

The value for excess reactivity was used in establishing core conditions for calculations (Table 13.4) that demonstrate fuel temperature limits are met during potential accident scenarios under extremely conservative conditions of analysis. Since the fundamental protection for the KSU reactor is the maximum power level and fuel temperature that can be achieved with the available positive core reactivity, experiments with positive reactivity are included in determining excess reactivity. Since experiments with negative reactivity will increase available reactivity if they are removed during operation, they are not credited in determining excess reactivity.

Analysis (Chapter 13) shows fuel temperature will not exceed 1,150°C for the stainless-steel-clad fuel in the event of inadvertent or accidental pulsing of the reactor. Section 13.2 demonstrates that a $3.00 reactivity insertion from critical, zero power conditions leads to maximum fuel temperature of 746°C, while a $1.00 reactivity insertion from a worst-case steady state operation at 107 kW leads to a maximum fuel temperature of 869°C, well below the safety limit.

The limiting SHUTDOWN MARGIN is necessary so that the reactor can be shut down from any operating condition, and will remain shut down after cool down and xenon decay, even if one control rod (including the transient control rod) should remain in the fully withdrawn position.

3.2 PULSED MODE Operations

3.2.1 Applicability

These specifications apply to operation of the reactor in the PULSE MODE.

3.2.2 Objective

This Limiting Condition for Operation prevents fuel temperature safety limit from being exceeded during PULSE MODE operation.

3.2.3 Specification

(1)	The transient rod drive is positioned for reactivity insertion (upon withdrawal) less than or equal to $3.00

3.2.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. With all stainless steel clad fuel elements, the worth of the pulse rod in the transient rod drive position is greater than $3.00 in the PULSE MODE	A.1 Position the transient rod drive for pulse rod worth less than or equal to $3.00 OR A.2 Place reactor in STEADY STATE MODE	A.1 IMMEDIATE OR A.2 IMMEDIATE

3.2.5 Bases

The value for pulsed reactivity with all stainless steel elements in the core was used in establishing core conditions for calculations (Table 13.4) that demonstrate fuel temperature limits are met during potential accident scenarios under extremely conservative conditions of analysis.

3.3 MEASURING CHANNELS

3.3.1 Applicability

This specification applies to the reactor MEASURING CHANNELS during STEADY STATE MODE and PULSE MODE operations.

3.3.2 Objective

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

3.3.3 Specifications

(1)	The MEASURING CHANNELS specified in TABLE 1 SHALL be OPERATING
(2)	The neutron count rate on the startup channel is greater than the minimum detector sensitivity

TABLE 1: MINIMUM MEASURING CHANNEL COMPLEMENT

MEASURING CHANNEL	Minimum Number Operable	
	STEADY STATE MODE	PULSE MODE
Reactor power level[1]	2	1
Primary Pool Water Temperature	1	1
Reactor Bay Differential Pressure	1	1
Fuel Temperature	1	1
22 foot Area radiation monitor	1	1
0 or 12 foot Area monitor	1	1
Continuous air radiation monitor[2]	1	1
EXHAUST PLENUM radiation monitor[2]	1	1

NOTE[1]: One "Startup Channel" required to have range that indicates <10 W
NOTE[2]: High-level alarms audible in the control room may be used

3.3.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A.1 Reactor power channels not OPERATING (min 2 for STEADY STATE, 1 PULSE MODE)	A.1.1 Restore channel to operation OR A.1.2 ENSURE reactor is SHUTDOWN	A.1.1 IMMEDIATE A.1.2 IMMEDIATE

CONDITION	REQUIRED ACTION	COMPLETION TIME
A.2 High voltage to reactor power level detector less than 90% of required operating value	A.2.1 Establish REACTOR SHUTDOWN condition AND A.2.2 Enter REACTOR SECURED mode	A.2. IMMEDIATE
B. Primary water temperature, reactor bay differential pressure or fuel temperature CHANNEL not operable	B.1 Restore channel to operation OR B.2 ENSURE reactor is SHUTDOWN	A.1 IMMEDIATE A.2 IMMEDIATE
C. 22 foot Area radiation monitor is not OPERATING	C.1 Restore MEASURING CHANNEL OR C.2 ENSURE reactor is shutdown OR C.3 ENSURE personnel are not on the 22 foot level OR C.4 ENSURE personnel on 22 foot level are using portable survey meters to monitor dose rates	C.1 IMMEDIATE C.2 IMMEDIATE C.3 IMMEDIATE C.4 IMMEDIATE
D. 0 or 12 foot Area monitor is not OPERATING	D.1 Restore MEASURING CHANNEL OR D.2 ENSURE reactor is shutdown OR D.3 ENSURE personnel are not in the reactor bay OR D.4 ENSURE personnel entering reactor bay are using portable survey meters to monitor dose rates	D.1 IMMEDIATE D.2 IMMEDIATE D.3 IMMEDIATE D.4 IMMEDIATE

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Continuous air radiation monitor is not OPERATING	E.1 Restore MEASURING CHANNEL OR E.2 ENSURE reactor is shutdown OR E.3.a ENSURE EXHAUST PLENUM radiation monitor is OPERATING AND E.3.b Restore MEASURING CHANNEL	E.1 IMMEDIATE E.2. IMMEDIATE E.3.a. IMMEDIATE E.3.b Within 30 days
F. Exhaust plenum radiation monitor is not OPERATING	F.1 Restore MEASURING CHANNEL OR F.2 ENSURE reactor is shutdown OR F.3.a ENSURE continuous air radiation monitor is OPERATING AND F.3.b Restore MEASURING CHANNEL	F.1 IMMEDIATE F.2. IMMEDIATE F.3.a. IMMEDIATE F.3.b Within 30 days
G. The neutron count rate on the startup channel is not greater than the minimum detector sensitivity	G.1 Do not perform a reactor startup OR G.2 Perform a neutron-source check on the startup channel prior to startup	G.1 IMMEDIATE G.2 IMMEDIATE

3.3.5 Bases

Maximum steady state power level is 1,250 kW; neutron detectors measure reactor power level. Chapter 4 and 13 discuss normal and accident heat removal capabilities. Chapter 7 discusses radiation detection and monitoring systems, and neutron and power level detection systems.

According to General Atomics, detector voltages less than 90% of required operating value do not provide reliable, accurate nuclear instrumentation. Therefore, if operating voltage falls below the minimum value the power level channel is inoperable.

Primary water temperature indication is required to assure water temperature limits are met, protecting primary cleanup resin integrity. The reactor bay differential pressure indictor is required to control reactor bay atmosphere radioactive contaminants. Fuel temperature indication provides a means of observing that the safety limits are met.

The 22-foot and 0-foot area radiation monitors provide information about radiation hazards in the reactor bay. A loss of reactor pool water (Chapter 13), changes in shielding effectiveness (Chapter 11), and releases of radioactive material to the restricted area (Chapter 11) could cause changes in radiation levels within the reactor bay detectable by these monitors. Portable survey instruments will detect changes in radiation levels.

The air monitors (continuous air- and exhaust plenum radiation-monitor) provide indication of airborne contaminants in the reactor bay prior to discharge of gaseous effluent. Iodine channels provide evidence of fuel element failure. The air monitors provide similar information on independent channels; the continuous air monitor (CAM) has maximum sensitivity to iodine and particulate activity, while the air monitoring system (AMS) has individual channels for radioactive particulate, iodine, noble gas and iodine.

When filters in the air monitoring system begin to load, there are frequent, sporadic trips of the AMS alarms. Although the filters are changed on a regular basis, changing air quality makes these trips difficult to prevent. Short outages of the AMS system have resulted in unnecessary shutdowns, exercising the shutdown mechanisms unnecessarily, creating stressful situations, and preventing the ability to fully discharge the mission of the facility while the CAM also monitors conditions of airborne contamination monitored by the AMS. The AMS detector has failure modes than cannot be corrected on site; AMS failures have caused longer outages at the K-State reactor. The facility has experienced approximately two-week outages, with one week dedicated to testing and troubleshooting and (sometimes) one-week for shipment and repair at the vendor facility.

Permitting operation using a single channel of atmospheric monitoring will reduce unnecessary shutdowns while maintaining the ability to detect abnormal conditions as they develop. Relative indications ensure discharges are routine; abnormal indications trigger investigation or action to prevent the release of radioactive material to the surrounding environment. Ensuring the alternate airborne contamination monitor is functioning during outages of one system provides the contamination monitoring required for detecting abnormal conditions. Limiting the outage for a single unit to a maximum of 30 days ensures radioactive atmospheric contaminants are monitored while permitting maintenance and repair outages on the other system.

Chapter 13 discusses inventories and releases of radioactive material from fuel element failure into the reactor bay, and to the environment. Particulate and noble gas channels monitor more routine discharges. Chapter 11 and SAR Appendix A discuss routine discharges of radioactive gasses generated from normal operations into the reactor bay and into the environment. Chapter 3 identifies design bases for the confinement and ventilation system. Chapter 7 discusses air-monitoring systems.

Experience has shown that subcritical multiplication with the neutron source used in the reactor does not provide enough neutron flux to correspond to an indicated power level of 10 Watts. Therefore an indicated power of 10 Watts or more indicates operating in a potential critical condition, and at least one neutron channel is required with sensitivity at a neutron flux level corresponding to reactor power levels less than 10 Watts ("Startup Channel"). If the indicated neutron level is less than the minimum sensitivity for both the log-wide range and the multirange linear power level channels, a neutron source will be used to determine that at least one of the channels is responding to neutrons to ensure that the channel is functioning prior to startup.

3.4 Safety Channel and Control Rod Operability

3.4.1 Applicability

This specification applies to the reactor MEASURING Channels during STEADY STATE MODE and PULSE MODE operations.

3.4.2 Objective

The objectives are to require the minimum number of REACTOR SAFETY SYSTEM channels that must be OPERABLE in order to ensure that the fuel temperature safety limit is not exceeded, and to ensure prompt shutdown in the event of a scram signal.

3.4.3 Specifications

(1)	The SAFETY SYSTEM CHANNELS specified in TABLE 2 are OPERABLE
(2)	CONTROL RODS (STANDARD) are capable of 90% of full reactivity insertion from the fully withdrawn position in less than 1 sec.
(3)	A minimum of three CONTROL RODS must be OPERABLE. Inoperable CONTROL RODS must be fully inserted.

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TABLE 2: REQUIRED SAFETY SYSTEM CHANNELS				
Safety System Channel or Interlock	Minimum Number Operable	Function	Required OPERATING Mode	
			STEADY STATE MODE	PULSE MODE
Reactor power level	2	Scram	YES	NA
Manual scram bar	1	Scram	YES	YES
CONTROL ROD (STANDARD) position interlock	1	Prevent withdrawal of standard rods in the PULSE MODE	NA	YES
Pulse rod interlock	1	Prevent inadvertent pulsing while in STEADY STATE MODE	YES	NA

3.4.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any required SAFETY SYSTEM CHANNEL or interlock function is not OPERABLE	A.1 Restore channel or interlock to operation OR A.2 ENSURE reactor is SHUTDOWN	A1. IMMEDIATE A2. IMMEDIATE

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. A control rod is not OPERABLE.	B.1 ENSURE inoperable control rod is fully inserted OR B.~~3~~2 ENSURE reactor is SHUTDOWN	B1. IMMEDIATE B2. IMMEDIATE

3.4.5 Bases

The power level scram is provided to ensure that reactor operation stays within the licensed limits of 1,250 kW, preventing abnormally high fuel temperature. The power level scram is not credited in analysis, but provides defense in depth to assure that the reactor is not operated in conditions beyond the assumptions used in analysis (Table 13.2.1.4).

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

The CONTROL ROD (STANDARD) interlock function is to prevent withdrawing control rods (other than the pulse rod) when the reactor is in the PULSE MODE. This will ensure the reactivity addition rate during a pulse is limited to the reactivity added by the pulse rod.

The pulse rod interlock function prevents air from being applied to the transient rod drive when it is withdrawn while disconnected from the control rod to prevent inadvertent pulses during STEADY STATE MODE operations. The control rod interlock prevents inadvertent pulses which would be likely to exceed the maximum range of the power level instruments configured for steady state operations.

Inoperable control rods that are fully inserted in the reactor will not negatively affect the minimum safety shutdown margin or maximum excess reactivity of the core. Operating with a fully-inserted control rod may cause power peaking to shift, however, in this case calculations have demonstrated that the maximum element-to-average power peaking of 2.0 assumed in SAR Chapter 13 is still bounding, and the reduction in maximum core power by having an inoperable control rod fully inserted means that the highest temperature in any fuel element with a fully-inserted inoperable control rod will be lower than the highest temperature in the B-ring with all rods withdrawn. Therefore the reactor can be safely operated with an inoperable control rod provided that the rod is fully inserted into the core.

3.5 Gaseous Effluent Control

3.5.1 Applicability

This specification applies to gaseous effluent in STEADY STATE MODE and PULSE MODE.

3.5.2 Objective

The objective is to ensure that exposures to the public resulting from gaseous effluents released during normal operations and accident conditions are within limits and ALARA.

3.5.3 Specification

(1)	The reactor bay ventilation exhaust system SHALL maintain in-leakage to the reactor bay
(2)	Releases of Ar-41 from the reactor bay exhaust plenum to an unrestricted environment SHALL NOT exceed 30 Ci per year.

3.5.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The reactor bay ventilation exhaust system is not OPERABLE	A.1 ENSURE reactor is SHUTDOWN	A.1 IMMEDIATE
	OR	
	A.2.a Do not OPERATE in the PULSE MODE	A.2.a IMMEDIATE
	AND	
	A.2.b Secure EXPERIMENT operations for EXPERIMENT with failure modes that could result in the release of radioactive gases or aerosols.	A.2.b IMMEDIATE
	A.2.c ENSURE no irradiated fuel handing	A.2.b IMMEDIATE
	AND	
	A.2.d Restore the reactor bay ventilation exhaust system to OPERABLE	A.2.d Within 30 days

CONDITION	REQUIRED ACTION	COMPLETION TIME
Calculated releases of Ar-41 from the reactor bay exhaust plenum exceed 30 Ci per year.	Do not operate.	IMMEDIATE

3.5.5 Bases

The confinement and ventilation system is described in Section 3.5.4. Routine operations produce radioactive gas, principally Argon 41, in the reactor bay. If the reactor bay ventilation system is secured, SAR Chapter 11 Appendix A demonstrates reactor bay concentration of 0.746 Bq ml^{-1} (2.01×10^{-5} μCi ml^{-1}), well below the 10CFR20 annual limit of 2000 DAC hours of Argon 41 at 6×10^{-3} μCi h/mL. Therefore, the reduction in concentration of Argon 41 from operation of the confinement and ventilation system is a defense in depth measure, and not required to assure meeting personnel exposure limits. Consequently, the ventilation system can be secured without causing significant personnel hazard from normal operations. Thirty days for a confinement and ventilation system outage is selected as a reasonable interval to allow major repairs and work to be accomplished, if required. During this interval, experiment activities that might cause airborne radionuclide levels to be elevated are prohibited.

It is shown in Section 13.2.2 of the Safety Analysis Report that, if the reactor were to be operating at full steady-state power, fuel element failure would not occur even if all the reactor tank water were to be lost instantaneously.

Section 13.2.4 addresses the maximum hypothetical fission product inventory release. Using unrealistically conservative assumptions, concentrations for a few nuclides of iodine would be in excess of occupational derived air concentrations for a matter of hours or days. ^{90}Sr activity available for release from fuel rods previously used at other facilities is estimated to be at most about 4 times the ALI. In either case (radio-iodine or -Sr), there is no credible scenario for accidental inhalation or ingestion of the undiluted nuclides that might be released from a damaged fuel element. Finally, fuel element failure during a fuel handling accident is likely to be observed and mitigated immediately.

SAR Appendix A shows the release of 30 Ci per year of Ar-41 from normal operations would result in less than 10 mrem annual exposure to any person in unrestricted areas.

3.6 Limitations on Experiments

3.6.1 Applicability

This specification applies to operations in STEADY STATE MODE and PULSE MODE.

3.6.2 Objectives

These Limiting Conditions for Operation prevent reactivity excursions that might cause the fuel temperature to exceed the safety limit (with possible resultant damage to the reactor), and the excessive release of radioactive materials in the event of an EXPERIMENT failure

3.6.3 Specifications

(1)	If all fuel elements are stainless steel clad, the reactivity worth of any individual EXPERIMENT SHALL NOT exceed $2.00
(2)	If two or more experiments in the reactor are interrelated so that operation or failure of one can induce reactivity-affecting change in the other(s), the sum of the absolute reactivity of such experiments SHALL NOT exceed $2.00.
(3)	Irradiation holders and vials SHALL prevent release of encapsulated material in the reactor pool and core area

3.6.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. INDEPENDENT EXPERIMENT worth is greater than $2.00	A.1 ENSURE the reactor is SHUTDOWN AND A.2 Remove the experiment	A.1 IMMEDIATE A.2 Prior to continued operations
C. An irradiation holder or vial releases material capable of causing damage to the reactor fuel or structure into the pool or core area	C.1 ENSURE the reactor is SHUTDOWN AND C.2 Inspect the affected area AND C.3 Obtain RSC review and approval	C.1 IMMEDIATE C.2 Prior to continued operation C.3 Prior to continued operation

3.6.5 Bases

Specifications 3.7(1) through 3.7(3) are conservatively chosen based on prior operation at 250 kW to limit reactivity additions to maximum values that are less than an addition which could cause temperature to challenge the safety limit.

Experiments are approved with expectations that there is reasonable assurance the facility will not be damaged during normal or failure conditions. If an irradiation capsule which contains material with potential for challenging the fuel cladding or pool wall, the facility will be inspected to ensure that continued operation is acceptable.

3.7 Fuel Integrity

3.7.1 Applicability

This specification applies to operations in STEADY STATE MODE and PULSE MODE.

3.7.2 Objective

The objective is to prevent the use of damaged fuel in the KSU TRIGA reactor.

3.7.3 Specifications

(1)	Fuel elements in the reactor core SHALL NOT be elongated more than 1/8 in. over manufactured length
(2)	Fuel elements in the reactor core SHALL NOT be laterally bent more than 1/8 in.

3.7.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any fuel element is elongated greater than 1/8 in. over manufactured length, or bent laterally greater than 1/8 in.	Do not insert the fuel element into the upper core grid plate.	IMMEDIATE

3.7.5 Bases

The above limits on the allowable distortion of a fuel element have been shown to correspond to strains that are considerably lower than the strain expected to cause rupture of a fuel element and have been successfully applied at TRIGA installations. Fuel cladding integrity is important since it represents the only process barrier for fission product release from the TRIGA reactor.

3.8 Reactor Pool Water

3.8.1 Applicability

This specification applies to operations in STEADY STATE MODE, PULSE MODE, and SECURED MODE.

3.8.2 Objective

The objective is to set acceptable limits on the water quality, temperature, conductivity, and level in the reactor pool.

3.8.3 Specifications

(1)	~~Water temperature at the exit of the reactor pool~~~~Pool inlet~~Bulk water temperature SHALL NOT exceed 445°C (1113°F)~~130°F with flow through the primary cleanup loop~~
(2)	Water conductivity SHALL be less than 5 μmho/cm
(3)	Water level above the core SHALL be at least 13 ft from the top of the core
(4)	~~Pool inlet~~Bulk water temperature SHALL NOT exceed 4037°C (99104°F) with an experiment installed in an interstitial flux wire port.

3.8.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ~~Water temperature at the exit~~~~inlet~~Bulk water temperature~~ of the reactor pool~~ exceeds ~~130~~454°~~F~~C	A.1 ENSURE the reactor is SHUTDOWN AND ~~A.2 Secure flow through the demineralizer~~ ~~AND~~ A.~~3~~2 Reduce bulk water temperature to less than ~~130~~454°~~F~~C	A.1 IMMEDIATE ~~A.2 IMMEDIATE~~ A.3 IMMEDIATE

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ~~Water temperature at the inlet of the reactor pool~~Bulk water temperature exceeds ~~40~~37°C with an experiment installed in an interstitial flux wire port.	B.1 ENSURE the reactor is SHUTDOWN AND B.2 Reduce bulk water temperature to less than ~~40~~ 37°C. OR B.3 Remove experiment from flux wire port	B.1 IMMEDIATE B.2 IMMEDIATE B.3 IMMEDIATE
B. Water conductivity is greater than 5 μmho/cm	B.1 ENSURE the reactor is SHUTDOWN AND B.2 Restore conductivity to less than 5 μmho/cm	B.1 IMMEDIATE B.2 Within 4 weeks
C. Water level above the core SHALL be at least 13 ft from the top of the core for all operating conditions	C.1 ENSURE the reactor is SHUTDOWN AND C.2 Restore water level	C.1 IMMEDIATE C.2 IMMEDIATE

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3.8.5 Bases

The resin used in the mixed bed deionizer limits the water temperature of the reactor pool. Resin in use (as described in Section 5.4) maintains mechanical and chemical integrity at temperatures below 130°F (54.4°C). While the integrity of the ion exchange resin requires water temperature to remain below 54.4°C, it is necessary to maintain water temperature below 4~~5~~4°C to ensure that the departure from nucleate boiling ratio (DNBR) will remain at least 2.0 for the hot channel while operating at 1250 kWth in STEADY STATE MODE and that excessive amounts of nucleate boiling will not occur. Insertion of an experiment into an interstitial flux wire port between fuel elements necessitates a further reduction in water temperature to a maximum of ~~40~~37°C in order to preclude excessive nucleate boiling of the water ~~ensure a DNBR of at least 2.0~~.

Maintaining low water conductivity over a prolonged period prevents possible corrosion, deionizer degradation, or slow leakage of fission products from degraded cladding. Although fuel degradation does not occur over short time intervals, long-term integrity of the fuel is important, and a 4-week interval was selected as an appropriate maximum time for high conductivity.

The top of the core is 16 feet below the top of the primary coolant tank. The lowest suction of primary cooling flow into the forced cooling loop is 3.5 feet below the top of the primary coolant tank (water level is maintained about 6 inches below the top of the tank).

The principle contributor to radiation dose rates at the pool surface is Nitrogen 16 generated in the reactor core and dispersed in the pool. Calculations in Chapter 11 show the pool surface radiation dose rates from Nitrogen 16 with 13 feet of water above the core are acceptable.

For normal pool temperature, calculations in Chapter 4 assuming 16 feet and 13 feet above the core demonstrate that the heat flux of the hottest area of the fuel rod generating the highest power level in the core during operations is less than the critical heat flux by a large margin up to the maximum permitted cooling temperatures; margin remains even at temperatures approaching bulk boiling for atmospheric conditions. Therefore, pool levels greater than 13 feet above the core meet requirements for safe operation with respect to maximum fuel temperature and thermal hydraulics by a wide margin.

Therefore, a minimum pool level of 13 feet above the core is adequate to provide shielding and support the core cooling.

3.9 Maintenance Retest Requirements

3.9.1 Applicability

This specification applies to operations in STEADY STATE MODE and PULSE MODE.

3.9.2 Objective

The objective is to ensure Technical Specification requirements are met following maintenance that occurs within surveillance test intervals.

3.9.3 Specifications

Maintenance activities SHALL NOT change, defeat or alter equipment or systems in a way that prevents the systems or equipment from being OPERABLE or otherwise prevent the systems or equipment from fulfilling the safety basis

3.9.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
Maintenance is performed that has the potential to change a setpoint, calibration, flow rate, or other parameter that is measured or verified in meeting a surveillance or operability requirement	Perform surveillance OR Operate only to perform retest	Prior to continued, normal operation in STEADY STATE MODE or PULSE MODE

3.9.5 Bases

Operation of the K-State reactor will comply with the requirements of Technical Specifications. This specification ensures that if maintenance might challenge a Technical Specifications requirement, the requirement is verified prior to resumption of normal operations.

3.10 Maximum Steady State Power

3.9.1 Applicability

This specification applies to operations in STEADY STATE MODE.

3.9.2 Objective

The objective is to ensure that the reactor has adequate margin to critical heat flux (CHF) and operates below the Limiting Safety System Setting of 1,250 kWth.

3.9.3 Specifications

(1)	Maximum OPERATING thermal power SHALL NOT exceed 1,000 kWth in STEADY STATE MODE.
(2)	A required reactor power level scram is set to a value no greater than 1,250 kWth.

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3.9.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Thermal power exceeds 1,050 kWth in STEADY STATE MODE	Reduce power to a level no greater than 1,050 kWth.	IMMEDIATE
B. A required reactor power level scram is set to a value above 1,250 kWth or above the maximum readable value on a required channel.	B.1 SHUT DOWN the reactor. AND B.2 Adjust reactor power level scram setpoint to a readable value less than or equal to 1,250 kWth.	B.1. IMMEDIATE AND B.2. Prior to resuming operations.

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3.9.5 Bases

The reactor control panel instrumentation is designed to measure up to 1,000 kWth of thermal power. The Limiting Safety System Setting ensures that automatic protective functions, i.e., high power scrams, are set to no greater than 1,250 kWth. However, by specifying the maximum OPERATING power level as 1,000 kWth in STEADY STATE MODE, the reactor will have additional margin to critical heat flux and will still be allowed to operate at up to the maximum power readable on the reactor console instruments. Action to reduce power is not required until power exceeds 1050kWth in STEADY STATE MODE to allow for slight variation in power level that is typical during normal operation.

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4. Surveillance Requirements

4.1 Core Reactivity

4.1.1 Objective

This surveillance ensures that the minimum SHUTDOWN MARGIN requirements and maximum excess reactivity limits of section 3.1 are met.

4.1.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SHUTDOWN MARGIN Determination	SEMIANNUAL
Excess Reactivity Determination	SEMIANNUAL
	Following Insertion of experiments with measurable positive reactivity
Control Rod Reactivity Worth determination	BIENNIAL

4.1.3 Basis

Experience has shown verification of the minimum allowed SHUTDOWN MARGIN at the specified frequency is adequate to assure that the limiting safety system setting is met

When core reactivity parameters are affected by operations or maintenance, additional activity is required to ensure changes are incorporated in reactivity evaluations.

4.2 PULSE MODE

4.2.1 Objectives

The verification that the pulse rod position does not exceed a reactivity value corresponding to $3.00 assures that the limiting condition for operation is met.

4.2.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
ENSURE Transient Pulse Rod position corresponds to reactivity not greater than $3.00	Prior to pulsing operations

4.2.3 Basis

Verifying pulse rod position corresponds to less than $3.00 ensures that the maximum pulsed reactivity meets the limiting condition for operation.

4.3 MEASURING CHANNELS

4.3.1 Objectives

Surveillances on MEASURING CHANNELS at specified frequencies ensure instrument problems are identified and corrected before they can affect operations.

4.3.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Reactor power level MEASURING CHANNEL	
CHANNEL TEST	DAILY
Calorimetric calibration	ANNUAL
CHANNEL CHECK high voltage to required power level instruments	DAILY
Primary pool water temperature CHANNEL CALIBRATION	ANNUAL
Reactor Bay differential pressure CHANNEL CALIBRATION	ANNUAL
Fuel temperature CHANNEL CALIBRATION	ANNUAL
22 Foot Area radiation monitor	
CHANNEL CHECK	DAILY
CHANNEL CALIBRATION	ANNUAL
0 or 12 Foot Area Radiation Monitor	
CHANNEL CHECK	DAILY
CHANNEL CALIBRATION	ANNUAL
Continuous Air Radiation Monitor	
CHANNEL CHECK	DAILY
CHANNEL CALIBRATION	ANNUAL
EXHAUST PLENUM Radiation Monitor	
CHANNEL CHECK	DAILY
CHANNEL CALIBRATION	ANNUAL
Startup Count Rate	DAILY

4.3.3 Basis

The DAILY CHANNEL CHECKS will ensure that the SAFETY SYSTEM and MEASURING CHANNELS are operable. The required periodic calibrations and verifications will permit any long-term drift of the channels to be corrected.

4.4 Safety Channel and Control Rod Operability

4.4.1 Objective

The objectives of these surveillance requirements are to ensure the REACTOR SAFETY SYSTEM will function as required. Surveillances related to safety system MEASURING CHANNELS ensure appropriate signals are reliably transmitted to the shutdown system; the surveillances in this section ensure the control rod system is capable of providing the necessary actions to respond to these signals.

4.4.2 Specifications

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Manual scram SHALL be tested by releasing partially withdrawn CONTROL RODS (STANDARD)	DAILY
CONTROL ROD (STANDARD) drop times SHALL be measured to have a drop time from the fully withdrawn position of less than 1 sec.	ANNUAL
The control rods SHALL be visually inspected for corrosion and mechanical damage at intervals	BIENNIAL
CONTROL ROD (STANDARD) position interlock functional test	SEMIANNUAL
Pulse rod interlock functional test	SEMIANNUAL
On each day that PULSE MODE operation of the reactor is planned, a functional performance check of the CONTROL ROD (TRANSIENT) system SHALL be performed.	Prior to pulsing operations each day a pulse is planned
The CONTROL ROD (TRANSIENT) rod drive cylinder and the associated air supply system SHALL be inspected, cleaned, and lubricated, as necessary.	SEMIANNUAL

4.4.3 Basis

Manual and automatic scrams are not credited in accident analysis, although the systems function to assure long-term safe shutdown conditions. The manual scram and control rod drop timing surveillances are intended to monitor for potential degradation that might interfere with the operation of the control rod systems. The verification of high voltage to the power level monitoring channels assures that the instrument channel providing an overpower trip will function on demand.

The control rod inspections (visual inspections and transient drive system inspections) are similarly intended to identify potential degradation that lead to control rod degradation or inoperability.

A test of the interlock that prevents the pulse rod from coupling to the drive in the state state mode unless the drive is fully down assures that pulses will occur only when in pulsing mode. A

test of the interlock that prevents standard control rod motion while in the pulse mode assures that the interlock will function as required.

The functional checks of the control rod drive system assure the control rod drive system operates as intended for any pulsing operations. The inspection of the pulse rod mechanism will assure degradation of the pulse rod drive will be detected prior to malfunctions.

4.5 Gaseous Effluent Control

4.5.1 Objectives

These surveillances ensure that routine releases are normal, and (in conjunction with MEASURING CHANNEL surveillances) that instruments will alert the facility if conditions indicate abnormal releases.

4.5.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL TEST of air monitor	ANNUAL
Verify negative reactor bay differential pressure	DAILY

4.5.3 Basis

The continuous air monitor provides indication that levels of radioactive airborne contamination in the reactor bay are normal.

If the reactor bay differential pressure gage indicates a negative pressure, the reactor bay exhaust fan is controlling airflow by directing effluent out of confinement.

4.6 Limitations on Experiments

4.6.1 Objectives

This surveillance ensures that experiments do not have significant negative impact on safety of the public, personnel or the facility.

4.6.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Experiments SHALL be evaluated and approved prior to implementation.	Prior to inserting a new experiment for purposes other than determination of reactivity worth
Measure and record experiment worth of the EXPERIMENT (where the absolute value of the estimated worth is greater than $0.40).	Initial insertion of a new experiment where absolute value of the estimated worth is greater than $0.40

4.6.3 Basis

These surveillances allow determination that the limits of 3.7 are met.

Experiments with an absolute value of the estimated significant reactivity worth (greater than $0.40) will be measured to assure that maximum experiment reactivity worths are met. If an absolute value of the estimate indicates less than $0.40 reactivity worth, even a 100% error will result in actual reactivity less than the assumptions used in analysis for inadvertent pulsing at low power operations in the Safety Analysis Report (13.2.3, Case I).

4.7 Fuel Integrity

4.7.1 Objective

The objective is to ensure that the dimensions of the fuel elements remain within acceptable limits.

4.7.2 Applicability

This specification applies to the surveillance requirements for the fuel elements in the reactor core.

4.7.3 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
The standard fuel elements SHALL be visually inspected for corrosion and mechanical damage, and measured for length and bend	500 pulses of magnitude equal to or less than a pulse insertion of 3.00$ AND Following the exceeding of a limited safety system set point with potential for causing degradation
B, C, D, E, and F RING elements comprising approximately 1/3 of the core SHALL be visually inspected annually for corrosion and mechanical damage such that the entire core SHALL be inspected at 3-year intervals, but not to exceed 38 months	ANNUAL

4.7.4 Basis

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.

Triennial visual inspection of fuel elements combined with measurements at intervals determined by pulsing as described is considered adequate to identify potential degradation of fuel prior to catastrophic fuel element failure.

4.8 Reactor Pool Water

This specification applies to the water contained in the KSU TRIGA reactor pool.

4.8.1 Objective

The objective is to provide surveillance of reactor primary coolant water quality, pool level, temperature and (in conjunction with MEASURING CHANNEL surveillances), and conductivity.

4.8.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify reactor pool water level above the inlet line vacuum breaker	DAILY
Verify reactor pool water temperature channel operable	DAILY
Measure reactor Pool water conductivity	DAILY
	At least every 20 days

4.9.3 Bases

Surveillance of the reactor pool will ensure that the water level is adequate before reactor operation. Evaporation occurs over longer periods of time, and daily checks are adequate to identify the need for water replacement.

Water temperature must be monitored to ensure that the limit of the deionizer will not be exceeded. A daily check on the instrument prior to reactor operation is adequate to ensure the instrument is operable when it will be needed.

Water conductivity must be checked to ensure that the deionizer is performing properly and to detect any increase in water impurities. A daily check is adequate to verify water quality is appropriate and also to provide data useful in trend analysis. If the reactor is not operated for long periods of time, the requirement for checks at least every 20 days will ensure water quality is maintained in a manner that does not permit fuel degradation.

4.9 Maintenance Retest Requirements

4.9.1 Objective

The objective is to ensure that a system is OPERABLE within specified limits before being used after maintenance has been performed.

4.9.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Evaluate potential for maintenance activities to affect operability and function of equipment required by Technical Specifications	Following maintenance of systems of equipment required by Technical Specifications
Perform surveillance to assure affected function meets requirements	Prior to resumption of normal operations

4.9.3 Bases

This specification ensures that work on the system or component has been properly carried out and that the system or component has been properly reinstalled or reconnected before reliance for safety is placed on it.

5. Design Features

5.1 Reactor Fuel

5.1.1 Applicability

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

5.1.2 Objective

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

5.1.3 Specification

(1) The high-hydride fuel element shall contain uranium-zirconium hydride, clad in 0.020 in. of 304 stainless steel. It shall contain a maximum of ~~9.0~~12.5 weight percent uranium which has a maximum enrichment of 20%. There shall be 1.55 to 1.80 hydrogen atoms to 1.0 zirconium atom.

(2) For the loading process, the elements shall be placed in a close packed array except for experimental facilities or for single positions occupied by control rods and a neutron startup source.

(3) Up to four elements with greater than 9.0 weight percent uranium may be installed in the core. These elements may only be placed in the E- and F-rings of the core lattice, and may not be ~~adjacent to control rods or water channels.~~located in the following positions: E2, E4, E5, E6, E20, E21, E22, E24, F1, F2, F30.

5.1.4 Bases

These types of fuel elements have a long history of successful use in TRIGA reactors. Calculations show that 12%-load fuel in the E- and F-rings will not exceed the temperature of 8%-load instrumented elements in the B-ring. Additionally the power peaking and fission product inventory assumptions in the SAR will not be challenged by 12% fuel in the E- and F-rings. Local power and temperature peaking effects during pulsing are avoided by prohibiting placement of the 12%-load fuel near water and control rod channels.

5.2 Reactor Fuel and Fueled Devices in Storage

5.2.1 Applicability

This specification applies to reactor fuel elements in storage

5.2.2 Objective

The objective is to ensure fuel elements or fueled devices in storage are maintained Subcritical in a safe condition.

5.2.3 Specification

(1) All fuel elements or fueled devices shall be in a safe, stable geometry

(2) The k_{eff} of all fuel elements or fueled devices in storage is less than 0.8

(3) Irradiated fuel elements or fueled devices will be stored in an array which will permit sufficient natural convection cooling by air or water such that the fuel element or fueled device will not exceed design values.

5.2.4 Bases

This specification is based on American Nuclear Society standard 15.1, section 5.4.

5.3 Reactor Building

5.3.1 Applicability

This specification applies to the building that houses the TRIGA reactor facility.

5.3.2 Objective

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

5.3.3 Specification

(1) The reactor shall be housed in a closed room designed to restrict leakage when the reactor is in operation, when the facility is unmanned, or when spent fuel is being handled exterior to a cask.

(2) The minimum free volume of the reactor room shall be approximately 144,000 cubic feet.

(3) The building shall be equipped with a ventilation system capable of exhausting air or other gases from the reactor room at a minimum of 30 ft. above ground level.

5.3.4 Bases

To control the escape of gaseous effluent, the reactor room contains no windows that can be opened. The room air is exhausted through an independent exhaust system, and discharged at roof level to provide dilution.

5.4 Experiments

5.4.1 Applicability

This specification applies to the design of experiments.

5.4.2 Objective

The objective is to ensure that experiments are designed to meet criteria.

5.4.3 Specifications

(1) EXPERIMENT with a design reactivity worth greater than $1.00 SHALL be securely fastened (as defined in Section l, Secured Experiment).

(2) Design shall ensure that failure of an EXPERIMENT SHALL NOT lead to a direct failure of a fuel element or of other experiments that could result in a measurable increase in reactivity or a measurable release of radioactivity due to the associated failure.

(3) EXPERIMENT SHALL be designed so that it does not cause bulk boiling of core water

(4) EXPERIMENT design SHALL ensure no interference with control rods or shadowing of reactor control instrumentation.

(5) EXPERIMENT design shall minimize the potential for industrial hazards, such as fire or the release of hazardous and toxic materials.

(6) Each fueled experiment shall be limited such that the total inventory of iodine isotopes 131 through 135 in the experiment is not greater than 5 millicuries except as the fueled experiment is a standard TRIGA instrumented element in which instance the iodine inventory limit is removed.

(7) Where the possibility exists that the failure of an EXPERIMENT (except fueled EXPERIMENTS) could release radioactive gases or aerosols to the reactor bay or atmosphere, the quantity and type of material shall be limited such that the airborne concentration of radioactivity averaged over a year will not exceed the limits of Table II of Appendix B of 10 CFR Part 20 assuming 100% of the gases or aerosols escape.

(8) The following assumptions shall be used in experiment design:

a. If effluents from an experimental facility exhaust through a hold-up tank which closes automatically at a high radiation level, at least 10% of the gaseous activity or aerosols produced will escape.

b. If effluents from an experimental facility exhaust through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of the aerosols produced will escape.

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

(9) Use of explosive solid or liquid material with a National Fire Protection Association Reactivity (Stability) index of 2, 3, or 4 in the reactor pool or biological shielding SHALL NOT exceed the equivalent of 25 milligrams of TNT without prior NRC approval.

5.4.4 Basis

Designing the experiment to reactivity and thermal-hydraulic conditions ensure that the experiment is not capable of breaching fission product barriers or interfering with the control systems (interferences from other - than reactivity - effects with the control and safety systems are also prohibited). Design constraints on industrial hazards ensure personnel safety and continuity of operations. Design constraints limiting the release of radioactive gasses prevent unacceptable personnel exposure during off-normal experiment conditions.

6. Administrative Controls

6.1 Organization and Responsibilities of Personnel

a) Structure.

The reactor organization is related to the University structure as shown in SAR Figure 12.1 and Technical Specifications Figure TS.1 below.

Kansas State University (KSU) holds the license for the KSU TRIGA Reactor, located in the KSU Nuclear Reactor Facility in Ward Hall on the campus of Kansas State University. The chief administrating officer for KSU is the President. Environment, safety and health oversight functions are administered through the Vice President for Administration and Finance, while reactor line management functions are through the Provost Chief Academic Officer.

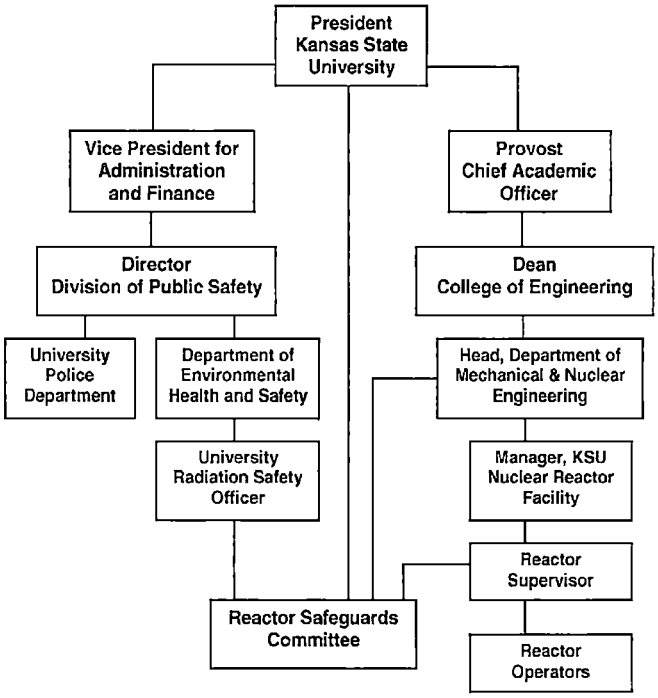

Figure TS.1: Organization and Management Structure for the K-State Reactor

Radiation protection functions are divided between the University Radiation Safety Officer (RSO) and the reactor staff and management, with management and authority for the RSO separate from line management and authority for facility operations. Day-to-day radiation protection functions implemented by facility staff and management are guided

by approved administrative controls (Reactor Radiation Protection Program or RPP, Facility Operating Manual, operating and experiment procedures); these controls are reviewed and approved by the RSO as part of the Reactor Safeguards Committee (with specific veto authority). The RSO has specific oversight functions assigned though the RPP. The RSO provides routine support for personnel monitoring, radiological analysis, and radioactive material inventory control. The RSO provides guidance on request for non-routine operations such as transportation and implementation of new experiments.

b) Responsibility.

The President of the University shall be responsible for the appointment of responsible and competent persons as members of the TRIGA Reactor Safeguards Committee upon the recommendation of the *ex officio* Chairperson of the Committee.

The KSU Nuclear Reactor Facility shall be under the supervision of the Nuclear Reactor Facility Manager, who shall have the overall responsibility for safe, efficient, and competent use of its facilities in conformity with all applicable laws, regulations, terms of facility licenses, and provisions of the Reactor Safeguards Committee. The Manager also has responsibility for maintenance and modification of laboratories associated with the Reactor Facility. The Manager shall have education and/or experience commensurate with the responsibilities of the position and shall report to the Head of the Department of Mechanical and Nuclear Engineering.

A Reactor Supervisor may serve as the deputy of the Nuclear Reactor Facility Manager in all matters relating to the enforcement of established rules and procedures (but not in matters such as establishment of rules, appointments, and similar administrative functions). The Supervisor should have at least two years of technical training beyond high school and shall possess a Senior Reactor Operator's license. The Supervisor shall have had reactor OPERATING experience and have a demonstrated competence in supervision. The Supervisor is appointed by the Nuclear Reactor Facility Manager and is responsible for enforcing all applicable rules, procedures, and regulations, for ensuring adequate exchange of information between OPERATING personnel when shifts change, and for reporting all malfunctions, accidents, and other potentially hazardous occurrences and situations to the Reactor Nuclear Reactor Facility Manager. The Nuclear Reactor Facility Manager may also serve as Reactor Supervisor.

The Reactor Operator shall be responsible for the safe and proper operation of the reactor, under the direction of the Reactor Supervisor. Reactor Operators shall possess an Operator's or Senior Operator's license and shall be appointed by the Nuclear Reactor Facility Manager.

The University Radiation Safety Officer (RSO), or a designated alternate, shall (in addition to other duties defined by the Director of Environmental Health and Safety, Division of Public Safety) be responsible for overseeing the safety of Reactor Facility operations from the standpoint of radiation protection. The RSO and/or designated alternate shall be appointed by the Director of Environmental Health and Safety, Division of Public Safety, with the approval of the University Radiation Safety Committee, and shall report to the Director of Environmental Health and Safety, whose organization is independent of the Reactor Facility organization, as shown on SAR Figure 12.1.

The Nuclear Reactor Facility Manager, with the approval of the Reactor Safeguards Committee, may designate an appropriately qualified member of the Facility organization as Reactor Facility Safety Officer (RFSO) with duties including those of an intra-Facility

Radiation Safety Officer. The University Radiation Safety Officer may, with the concurrence of the Nuclear Reactor Facility Manager, authorize the RFSO to perform some of the specific duties of the RSO at the Nuclear Reactor Facility.

c). Staffing.

Whenever the reactor is not secured, the reactor shall be under the direction of a (USNRC licensed) Senior Operator (designated as Reactor Supervisor). The Supervisor shall be on call, within twenty minutes travel time to the facility.

Whenever the reactor is not secured, a (USNRC licensed) Reactor Operator (or Senior Reactor Operator) who meets requirements of the Operator Requalification Program shall be at the reactor control console, and directly responsible for control manipulations.

In addition to the above requirements, during fuel movement a senior operator shall be inside the reactor bay directing fuel operations.

6.2 Review and Audit

a) There will be a Reactor Safeguards Committee which shall review TRIGA reactor operations to assure that the reactor facility is operated and used in a manner within the terms of the facility license and consistent with the safety of the public and of persons within the Laboratory.

b) The responsibilities of the Committee include, but are not limited to, the following:

1. Review and approval of rules, procedures, and proposed Technical Specifications;

2. Review and approval of all proposed changes in the facility that could have a significant effect on safety and of all proposed changes in rules, procedures, and Technical Specifications, in accordance with procedures in Section 6.3;

3. Review and approval of experiments using the reactor in accordance with procedures and criteria in Section 6.4;

4. Determine whether changes in the facility as described in the safety analysis report (as updated), changes in the procedures as described in the final safety analysis report (as updated), and the conduct of tests or experiments not described in the safety analysis report (as updated) may be accomplished in accordance with 10 CFR 50.59 without obtaining prior NRC approval via license amendment pursuant to 10 CFR Sec. 50.90.

5. Review of abnormal performance of plant equipment and OPERATING anomalies;

6. Review of unusual or abnormal occurrences and incidents which are reportable under 10 CFR 20 and 10 CFR50;

7. Inspection of the facility, review of safety measures, and audit of operations at a frequency not less than once a year, including operation and operations records of the facility;

8. Requalification of the Nuclear Reactor Facility Manager and/or the Reactor Supervisor,

9. Review of container failures where released materials have the potential for damaging reactor fuel or structural components including:

   a) results of physical inspection

   b) evaluation of consequences

   c) need for corrective actions

c) The Committee shall be composed of:

1. one or more persons proficient in reactor and nuclear science or engineering,

2. one or more persons proficient in chemistry, geology, or chemical engineering,

3. one person proficient in biological effects of radiation,

4. the Nuclear Reactor Facility Manager, *ex officio*,

5. the University Radiation Safety Officer, *ex officio*, and,

6. The Head of the Department of Mechanical and Nuclear Engineering, *ex officio*, or a designated deputy, to serve as chairperson of the Committee.

The same individual may serve under more than one category above, but the minimum membership shall be seven. At least five members shall be faculty members. The Reactor Supervisor, if other than the Nuclear Reactor Facility Manager, shall attend and participate in Committee meetings, but shall not be a voting member.

d) The Committee shall have a written statement defining its authority and responsibilities, the subjects within its purview, and other such administrative provisions as are required for its effective functioning. Minutes of all meetings and records of all formal actions of the Committee shall be kept.

e) A quorum shall consist of not less than a majority of the full Committee and shall include all *ex officio* members.

f) Any permissive action of the Committee requires affirmative vote of the University Radiation Safety Officer as well as a majority vote of the members present.

g) The Committee shall meet a minimum of two times a year. Additional meetings may be called by any member, and the Committee may be polled in lieu of a meeting. Such a poll shall constitute Committee action subject to the same requirements as for an actual meeting.

6.3 Procedures

a) Written procedures, reviewed and approved by the Reactor Safeguards Committee, shall be followed for the activities listed below. The procedures shall be adequate to

assure the safety of the reactor, persons within the Laboratory, and the public, but should not preclude the use of independent judgment and action should the situation require it. The activities are:

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

   (a) startup checkout procedures to test the reactor instrumentation and safety systems, area monitors, and continuous air monitors,

   (b) prohibition of routine operations with failed (or leaking) fuel except to find leaking elements, and

   (b) shutdown procedures to assure that the reactor is secured before OPERATING personnel go off duty.

2. Installation or removal of fuel elements, control rods, and other core components that significantly affect reactivity or reactor safety.

3. Preventive or corrective maintenance activities which could have a significant effect on the safety of the reactor or personnel.

4. Periodic inspection, testing or calibration of auxiliary systems or instrumentation that relate to reactor operation.

b) Substantive changes in the above procedures shall be made only with the approval of the Reactor Safeguards Committee, and shall be issued to the OPERATING personnel in written form. The Nuclear Reactor Facility Manager may make temporary changes that do not change the original intent. The change and the reasons thereof shall be noted in the log book, and shall be subsequently reviewed by the Reactor Safeguards Committee.

c) Determination as to whether a proposed activity in categories (1), (2) and (3) in Section 6.2b above does or does not have a significant safety effect and therefore does or does not require approved written procedures shall require the concurrence of

   1. the Nuclear Reactor Facility Manager, and

   2. at least one other member of the Reactor Safeguards Committee, to be selected for relevant expertise by the Nuclear Reactor Facility Manager. If the Manager and the Committee member disagree, or if in their judgment the case warrants it, the proposal shall be submitted to the full Committee, and

   3. the University Radiation Safety Officer, or his/her deputy, who may withhold agreement until approval by the University Radiation Safety Committee is obtained.

   The Rector Safeguards Committee shall subsequently review determinations that written procedures are not required. The time at which determinations are made, and the review and approval of written procedures, if required, are carried out, shall be a reasonable interval before the proposed activity is to be undertaken.

d) Determination that a proposed change in the facility does or does not have a significant safety effect and therefore does or does not require review and approval by the full Reactor Safeguards Committee shall be made in the same manner as for proposed activities under (c) above.

6.4 Review of Proposals for Experiments

a) All proposals for new experiments involving the reactor shall be reviewed with respect to safety in accordance with the procedures in (b) below and on the basis of criteria in (c) below.

b) Procedures:

1. Proposed reactor operations by an experimenter are reviewed by the Reactor Supervisor, who may determine that the operation is described by a previously approved EXPERIMENT or procedure. If the Reactor Supervisor determines that the proposed operation has not been approved by the Reactor Safeguards Committee, the experimenter shall describe the proposed EXPERIMENT in written form in sufficient detail for consideration of safety aspects. If potentially hazardous operations are involved, proposed procedures and safety measures including protective and monitoring equipment shall be described.

2. If the experimenter is a student, approval by his/her research supervisor is required. If the experimenter is a staff or faculty member, his/her own signature is sufficient.

3. The proposal is then to be submitted to the Reactor Safeguards Committee for consideration and approval. The Committee may find that the experiment, or portions thereof, may only be performed in the presence of the University Radiation Safety Officer or Deputy thereto.

4. The scope of the EXPERIMENT and the procedures and safety measures as described in the approved proposal, Including any amendments or conditions added by those reviewing and approving it, shall be binding on the experimenter and the OPERATING personnel. Minor deviations shall be allowed only in the manner described in Section 6 above. Recorded affirmative votes on proposed new or revised experiments or procedures must indicated that the Committee determines that the proposed actions do not involve changes in the facility as designed, changes in Technical Specifications, changes that under the guidance of 10 CFR 50.59 require prior approval of the NRC, and could be taken without endangering the health and safety of workers or the public or constituting a significant hazard to the integrity of the reactor core.

5. Transmission to the Reactor Supervisor for scheduling.

c) Criteria that shall be met before approval can be granted shall include:

1. The EXPERIMENT must meet the applicable Limiting Conditions for Operation and Design Description specifications.

2. It must not involve violation of any condition of the facility license or of Federal, State, University, or Facility regulations and procedures.

3. The conduct of tests or experiments not described in the safety analysis report (as updated) must be evaluated in accordance with 10 CFR 50.59 to determine if the test

or experiment can be accomplished without obtaining prior NRC approval via license amendment pursuant to 10 CFR Sec. 50.90.

4. In the safety review the basic criterion is that there shall be no hazard to the reactor, personnel or public. The review SHALL determine that there is reasonable assurance that the experiment can be performed with no significant risk to the safety of the reactor, personnel or the public.

6.5 Emergency Plan and Procedures

An emergency plan shall be established and followed in accordance with NRC regulations. The plan shall be reviewed and approved by the Reactor Safeguards Committee prior to its submission to the NRC. In addition, emergency procedures that have been reviewed and approved by the Reactor Safeguards Committee shall be established to cover all foreseeable emergency conditions potentially hazardous to persons within the Laboratory or to the public, including, but not limited to, those involving an uncontrolled reactor excursion or an uncontrolled release of radioactivity.

6.6 Operator Requalification

An operator requalification program shall be established and followed in accordance with NRC regulations.

6.7 Physical Security Plan

Administrative controls for protection of the reactor plant shall be established and followed in accordance with NRC regulations.

6.8 Action To Be Taken In The Event A Safety Limit Is Exceeded

In the event a safety limit is exceeded:

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

b) An immediate report of the occurrence shall be made to the Chair of the Reactor Safeguards Committee, and reports shall be made to the NRC in accordance with Section 6.11 of these specifications.

c) A report shall be made to include an analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to Reactor Safeguards Committee for review, and a suitable similar report submitted to the NRC when authorization to resume operation of the reactor is sought.

6.9 Action To Be Taken In The Event Of A Reportable Occurrence

a) A reportable occurrence is any of the following conditions:

1. any actual safety system setting less conservative than specified in Section 2.2, Limiting Safety System Settings;

2. VIOLATION OF SL, LSSS OR LCO;

NOTES

Violation of an LSSS or LCO occurs through failure to comply with an "Action" statement when "Specification" is not met; failure to comply with the "Specification" is not by itself a violation.

Surveillance Requirements must be met for all equipment/components/conditions to be considered operable.

Failure to perform a surveillance within the required time interval or failure of a surveillance test shall result in the /component/condition being inoperable

3. incidents or conditions that prevented or could have prevented the performance of the intended safety functions of an engineered safety feature or the REACTOR SAFETY SYSTEM;

4. release of fission products from the fuel that cause airborne contamination levels in the reactor bay to exceed 10CFR20 limits for releases to unrestricted areas;

5. an uncontrolled or unanticipated change in reactivity greater than $1.00;

6. an observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy has caused the existence or development of an unsafe condition in connection with the operation of the reactor;

7. an uncontrolled or unanticipated release of radioactivity.

b) In the event of a reportable occurrence, the following actions shall be taken:

1. The reactor shall be shut down at once. The Reactor Supervisor shall be notified and corrective action taken before operations are resumed; the decision to resume shall require approval following the procedures in Section 6.3.

2. A report shall be made to include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to the Reactor Safeguards Committee for review.

3. A report shall be submitted to the NRC in accordance with Section 6.11 of these specifications.

6.10 Plant Operating Records

a) In addition to the requirements of applicable regulations, in 10 CFR 20 and 50, records and logs shall be prepared and retained for a period of at least 5 years for the following items as a minimum.

1. normal plant operation, including power levels;

3. principal maintenance activities;

4. reportable occurrences;

5. equipment and component surveillance activities;

6. experiments performed with the reactor;

7. all emergency reactor scrams, including reasons for emergency shutdowns.

b) The following records shall be maintained for the life of the facility:

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

2. offsite environmental monitoring surveys;

3. fuel inventories and transfers;

4. facility radiation and contamination surveys;

5. radiation exposures for all personnel;

6. updated, corrected, and as-built drawings of the facility.

6.11 Reporting Requirements

All written reports shall be sent within the prescribed interval to the United States Nuclear Regulatory Commission, Washington, D.C., 20555, Attn: Document Control Desk.

In addition to the requirements of applicable regulations, and in no way substituting therefor, reports shall be made to the US. Nuclear Regulatory Commission (NRC) as follows:

a) A report within 24 hours by telephone and fax or electronic mail to the NRC Operations Center and the USNRC Region IV of;

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

2. any violation of a safety limit;

3. any reportable occurrences as defined in Section 6.9 of these specifications.

b) A report within 10 days in writing of:

1. any accidental release of radioactivity above permissible limits in unrestricted areas, whether or not the release resulted in property damage, personal injury or exposure; the written report (and, to the extent possible, the preliminary telephone and

telegraph report) shall describe, analyze, and evaluate safety implications, and outline the corrective measures taken or planned to prevent recurrence of the event;

2. any violation of a safety limit;

3. any reportable occurrence as defined in Section 1.1 of these specifications.

c) A report within 30 days in writing of:

1. any significant variation of a MEASURED VALUE from a corresponding predicted or previously MEASURED VALUE of safety-connected OPERATING characteristics occurring during operation of the reactor;

2. any significant change in the transient or accident analysis as described in the Safety Analysis Report.

3. a change in personnel for the Department of Mechanical and Nuclear Engineering Chair, or a change in reactor manager

d) A report within 60 days after criticality of the reactor in writing to the US Nuclear Regulatory Commission, resulting from a receipt of a new facility license or an amendment to the license authorizing an increase in reactor power level or the installation of a new core, describing the MEASURED VALUE of the OPERATING conditions or characteristics of the reactor under the new conditions.

e) A routine report in writing to the US. Nuclear Regulatory Commission within 60 days after completion of the first calendar year of OPERATING and at intervals not to exceed 12 months, thereafter, providing the following information:

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

2. a tabulation showing the energy generated by the reactor (in megawatt-hours);

3. the number of emergency shutdowns and inadvertent scrams, including the reasons thereof and corrective action, if any, taken;

4. discussion of the major maintenance operations performed during the period, including the effects, if any, on the safe operation of the reactor, and the reasons for any corrective maintenance required;

5. a summary of each change to the facility or procedures, tests, and experiments carried out under the conditions of 10 CFR 50.59;

6. a summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or before the point of such release or discharge;

7. a description of any environmental surveys performed outside the facility;

8. a summary of radiation exposures received by facility personnel and visitors, including the dates and time of significant exposure, and a brief summary of the results of radiation and contamination surveys performed within the facility.

Table of Contents

1. **DEFINITIONS** TS-1

2. **SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS** TS-8
   2.1 Fuel Element Temperature Safety Limit TS-8
      2.1.1. Applicability TS-8
      2.1.2. Objective TS-8
      2.1.3. Specification TS-8
      2.1.4. Actions TS-8
      2.1.5. Basis TS-8
   2.2 Limiting Safety System Settings TS-10
      2.2.1. Applicability TS-10
      2.2.3. Objective TS-10
      2.2.4. Specification TS-10
      2.2.5. Actions TS-10
      2.2.6. Basis TS-10

3. **LIMITING CONDITIONS FOR OPERATIONS** TS-11
   3.1 CORE REACTIVITY TS-11
      3.1.1. Applicability TS-11
      3.1.3. Objective TS-11
      3.1.4. Specification TS-11
      3.1.5. Actions TS-12
      3.1.6. Basis TS-13
   3.2 PULSED MODE OPERATIONS TS-13
      3.2.1. Applicability TS-13
      3.2.3. Objective TS-13
      3.2.4. Specification TS-13
      3.2.5. Actions TS-13
      3.2.6. Basis TS-13
   3.3 MEASURING CHANNELS TS-14
      3.3.1. Applicability TS-14
      3.3.3. Objective TS-14
      3.3.4. Specification TS-14
      3.3.5. Actions TS-14
      3.3.6. Bases TS-16
   3.4. SAFETY CHANNEL AND CONTROL ROD OPERABILITY TS-18
      3.4.1. Applicability TS-18
      3.4.3. Objective TS-18
      3.4.4. Specification TS-18
      3.4.5. Actions TS-18
      3.4.6. Basis TS-19
   3.5 GASEOUS EFFLUENT CONTROL TS-20
      3.5.1. Applicability TS-20
      3.5.3. Objective TS-20
      3.5.4. Specification TS-20
      3.5.5. Actions TS-20
      3.5.6. Basis TS-21
   3.6 LIMITATIONS ON EXPERIMENTS TS-22
      3.6.1. Applicability TS-22
      3.6.3. Objective TS-22

3.6.4. Specification TS-22
3.6.5. Actions TS-22
3.6.6. Basis TS-23
3.7 FUEL INTEGRITY TS-24
3.7.1. Applicability TS-24
3.7.3. Objective TS-24
3.7.4. Specification TS-24
3.7.5. Actions TS-24
3.7.6. Basis TS-24
3.8 REACTOR POOL WATER TS-25
3.8.1. Applicability TS-25
3.8.3. Objective TS-25
3.8.4. Specification TS-25
3.8.5. Actions TS-25
3.8.6. Basis TS-26
3.9 Maintenance Retest Requirements TS-27
3.9.1. Applicability TS-27
3.9.3. Objective TS-27
3.9.4. Specification TS-27
3.9.5. Actions TS-27
3.9.6. Basis TS-27

4. **SURVIELLANCES** TS-28
4.1 CORE REACTIVITY TS-28
4.1.1. Objective TS-28
4.1.2. Specification TS-28
4.1.3. Basis TS-28
4.2 PULSE MODE TS-29
4.2.1. Objective TS-29
4.2.2. Specification TS-29
4.2.3. Basis TS-29
4.3 MEASURING CHANNELS TS-30
4.3.1. Objective TS-30
4.3.2. Specification TS-30
4.3.3. Basis TS-30
4.4 SAFETY CHANNEL AND CONTROL ROD OPERABILITY TS-31
4.4.1. Objective TS-31
4.4.2. Specification TS-31
4.4.3. Basis TS-32
4.5 GASEOUS EFFLUENT CONTROL TS-33
4.5.1. Objective TS-33
4.5.2. Specification TS-33
4.5.3. Basis TS-33
4.6 LIMITATIONS ON EXPERIMENTS TS-34
4.6.1. Objective TS-34
4.6.2. Specification TS-34
4.6.3. Basis TS-34
4.7 FUEL INTEGRITY TS-35
4.7.1. Objective TS-35
4.7.2. Specification TS-35
4.7.3. Basis TS-35
4.8 REACTOR POOL WATER TS-36
4.8.1. Objective TS-36

4.8.2. Specification TS-36
4.8.3. Basis TS-36
4.9 MAINTENANCE RETEST REQUIREMENTS TS-37
4.9.1. Objective TS-37
4.9.2. Specification TS-37
4.10.3. Basis TS-37
5. **DESIGN FEATURES** TS-38
5.1 REACTOR FUEL TS-38
5.1.1. Applicability TS-38
5.1.2. Objective TS-38
5.1.3. Specification TS-38
5.1.4. Basis TS-38
5.2 REACTOR FUEL AND FUELED DEVICES IN STORAGE TS-38
5.2.1. Applicability TS-38
5.2.2. Objective TS-39
5.2.3. Specification TS-39
5.2.4. Basis TS-39
5.3 REACTOR BUILDING TS-39
5.3.1. Applicability TS-39
5.3.2. Objective TS-39
5.3.3. Specification TS-39
5.3.4. Basis TS-40
5.4 EXPERIMENTS TS-40
5.4.1. Applicability TS-40
5.4.2. Objective TS-40
5.4.3. Specification TS-40
5.4.4. Basis TS-41

6. **ADMINISTRATIVE CONTROLS** TS-42
6.1 ORGANIZATION AND RESPONSIBILITIES OF PERSONNEL TS-44
6.2 REVIEW AND AUDIT TS-45
6.3 PROCEDURES TS-45
6.4 REVIEW OF PROPOSALS FOR EXPERIMENTS TS-47
6.5 EMERGENCY PLAN AND PROCEDURES TS-48
6.6 OPERATOR REQUALIFICATION TS-48
6.7 PHYSICAL SECURITY PLAN TS-48
6.8 ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS VIOLATED TS-48
6.9 ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE OCCURRENCE TS-48
6.10 PLANT OPERATING RECORDS TS-49
6.11 REPORTING REQUIREMENTS TS-50

1. DEFINITIONS

The following frequently used terms are defined to aid in the uniform interpretation of these specifications. Capitalization is used in the body of the Technical Specifications to identify defined terms.

ACTION	Actions are steps to be accomplished in the event a required condition identified in a "Specification" section is not met, as stated in the "Condition" column of "Actions." In using Action Statements, the following guidance applies: • Where multiple conditions exist in an LCO, actions are linked to the (failure to meet a "Specification") "Condition" by letters and number. • Where multiple action steps are required to address a condition, COMPLETION TIME for each action is linked to the action by letter and number. • AND in an Action Statement means all steps need to be performed to complete the action; OR indicates options and alternatives, only one of which needs to be performed to complete the action. • If a "Condition" exists, the "Action" consists of completing all steps associated with the selected option (if applicable) except where the "Condition" is corrected prior to completion of the steps
ANNUAL	12 months, not to exceed 15 months
CHANNEL CALIBRATION	A channel calibration is an adjustment of the channel to that its output responds, with acceptable range and accuracy, to known values of the parameter that the channel measures.
BIENNIAL	Every two years, not to exceed a 28 month interval
CHANNEL CHECK	A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification shall include comparison of the channel with expected values, other independent channels, or other methods of measuring the same variable.
CHANNEL TEST	A channel test is the introduction of an input signal into a channel to verify that it is operable. A functional test of operability is a channel test.
CONTROL ROD (STANDARD)	A standard control rod is one having an electric motor drive and scram capability.
CONTROL ROD (TRANSIENT)	A transient rod is one that is pneumatically operated and has scram capability.
DAILY	Prior to initial operation each day (when the reactor is operated), or before

	an operation extending more than 1 day
ENSURE	Verify existence of specified condition or (if condition does not meet criteria) take action necessary to meet condition
EXHAUST PLENUM	The air volume in the reactor bay atmosphere between the pool surface and the reactor bay exhaust fan
EXPERIMENT	An EXPERIMENT is (1) any apparatus, device, or material placed in the reactor core region (in an EXPERIMENTAL FACILITY associated with the reactor, or in line with a beam of radiation emanating from the reactor) or (2) any in-core operation designed to measure reactor characteristics.
EXPERIMENTAL FACILITY	Experimental facilities are the beamports, thermal column, pneumatic transfer system, central thimble, rotary specimen rack, and the in-core facilities (including non-contiguous single-element positions, and, in the E and F rings, as many as three contiguous fuel-element positions).
IMMEDIATE	Without delay, and not exceeding one hour. *NOTE:* *IMMEDIATE permits activities to restore required conditions for up to one hour; this does not permit or imply deferring or postponing action*
INDEPENDENT EXPERIMENT	INDEPENDENT Experiments are those not connected by a mechanical, chemical, or electrical link to another experiment
LIMITING CONDITION FOR OPERATION (LCO)	The lowest functional capability or performance levels of equipment required for safe operation of the facility.
LIMITING SAFETY SYSTEM SETTING (LSSS)	Settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit placed, the setting shall be chosen so that the automatic protective action will correct the abnormal situation before a safety limit is exceeded.
MEASURED VALUE	The measured value of a parameter is the value as it appears at the output of a MEASURING CHANNEL.
MEASURING CHANNEL	A MEASURING CHANNEL is the combination of sensor, lines, amplifiers, and output devices that are connected for the purpose of measuring the value of a process variable.
MOVABLE EXPERIMENT	A MOVABLE EXPERIMENT is one that may be moved into, out-of or near the reactor while the reactor is OPERATING.
NONSECURED EXPERIMENT	NONSECURED Experiments are these that should not move while the reactor is OPERATING, but are held in place with less restraint than a secured experiment.

OPERABLE

A system or component is OPERABLE when it is capable of performing its intended function in a normal manner

OPERATING

A system or component is OPERATING when it is performing its intended function in a normal manner.

PULSE MODE

The reactor is in the PULSE MODE when the reactor mode selection switch is in the pulse position and the key switch is in the "on" position.

NOTE:
In the PULSE MODE, reactor power may be increased on a period of much less than l second by motion of the transient control rod.

REACTOR SAFETY SYSTEM

The REACTOR SAFETY SYSTEM is that combination of MEASURING CHANNELS and associated circuitry that is designed to initiate reactor scram or that provides information that requires manual protective action to be initiated.

REACTOR SECURED MODE

The reactor is secured when the conditions of either item (1) or item (2) are satisfied:

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

(2) All of the following:

a. The console key is it the OFF position and the key is removed from the lock

b. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives (unless the drive is physically decoupled from the control rod)

c. No experiments are being moved or serviced that have, on movement, a reactivity worth greater than $1.00

REACTOR SHUTDOWN

The reactor is shutdown if it is subcritical by at least $1.00 in the REFERENCE CORE CONDITION with the reactivity worth of all experiments included.

RING

A ring is one of the five concentric bands of fuel elements surrounding the central opening (thimble) of the core. The letters B through F, with the letter B used to designate the innermost ring,

REFERENCE CORE CONDITION

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

SAFETY CHANNEL

A safety channel is a MEASURING CHANNEL in the REACTOR SAFETY SYSTEM

SECURED EXPERIMENT

A secured EXPERIMENT is an EXPERIMENT held firmly in place by a mechanical device or by gravity providing that the weight of the EXPERIMENT is such that it cannot be moved by force of less than 60 lb.

SECURED EXPERIMENT WITH MOVABLE PARTS	A secured EXPERIMENT with movable parts is one that contains parts that are intended to be moved while the reactor is OPERATING.
SHALL (SHALL NOT)	Indicates specified action is required/(not to be performed)
SEMIANNUAL	Every six months, with intervals not greater than 8 months
SHUTDOWN MARGIN	The shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems, starting from any permissible operating condition, and that the reactor will remain subcritical without further operator action
STANDARD THERMOCOUPLE FUEL ELEMENT	A standard thermocouple fuel element is stainless steel clad fuel element containing three sheathed thermocouples embedded in the fuel element.
STEADY-STATE MODE	The reactor is in the steady-state mode when the reactor mode selector switch is in either the manual or automatic position and the key switch is in the "on" position.
TECHNICAL SPECIFICATION VIOLATION	A violation of a Safety Limit occurs when the Safety Limit value is exceeded. A violation of a Limiting Safety System Setting or Limiting Condition for Operation) occurs when a "Condition" exists which does not meet a "Specification" and the corresponding "Action" has not been met within the required "Completion Time." If the "Action" statement of an LSSS or LCO is completed or the "Specification" is restored within the prescribed "Completion Time," a violation has not occurred. *NOTE* *"Condition," "Specification," "Action," and "Completion Time" refer to applicable titles of sections in individual Technical Specifications*

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Fuel Element Temperature Safety Limit

2.1.1 Applicability

This specification applies when the reactor in STEADY STATE MODE and the PULSE MODE.

2.1.2 Objective

This SAFETY LIMIT ensures fuel element cladding integrity

2.1.3 Specification

(1)	Stainless steel clad, high-hydride fuel element temperature SHALL NOT exceed 1150°C.
(2)	Steady state fuel temperature shall not exceed 750°C.

2.1.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A Stainless steel clad, high-hydride fuel element temperature exceeds 1150°C.	A.1 Establish SHUTDOWN condition	A.1 IMMEDIATE
OR	AND	
Fuel temperature exceeds 750°C in steady state conditions	A.2 Report per Section 6.8	A.2 Within 24 hours

2.1.5 Bases

Safety Analysis Report, Section 3.5.1 (Fuel System) identifies design and operating constraints for TRIGA fuel that will ensure cladding integrity is not challenged.

NUREG 1282 identifies the safety limit for the high-hydride ($ZrH_{1.7}$) fuel elements with stainless steel cladding based on the stress in the cladding (resulting from the hydrogen pressure from the dissociation of the zirconium hydride). This stress will remain below the yield strength of the stainless steel cladding with fuel temperatures below 1,150°C. A change in yield strength occurs for stainless steel cladding temperatures of 500°C, but there is no scenario for fuel cladding to achieve 500°C while submerged; consequently the safety limit during reactor operations is 1,150°C.

Therefore, the important process variable for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification, and it is readily measured. During operation, fission product gases and dissociation of the hydrogen and zirconium builds up gas inventory in internal components and spaces of the fuel elements. Fuel temperature acting on these gases controls fuel element internal pressure. Limiting the maximum temperature prevents excessive internal pressures that could be generated by heating these gases.

Fuel growth and deformation can occur during normal operations, as described in General Atomics technical report E-117-833. Damage mechanisms include fission recoils and fission gases, strongly influenced by thermal gradients. Operating with maximum long-term, steady state fuel temperature of 750°C does not have significant time- and temperature-dependent fuel growth.

2.2 Limiting Safety System Settings (LSSS)

2.2.1 Applicability

This specification applies when the reactor in STEADY STATE MODE

2.2.2 Objective

The objective of this specification is to ensure the safety limit is not exceeded.

2.2.3 Specifications

(1)	Power level SHALL NOT exceed 1,250 kW (th) in STEADY STATE MODE of operation

2.2.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Steady state power level exceeds 1,250 kW (th)	A.1 Reduce power to less than 1,250 kW (th) OR A.2. Establish REACTOR SHUTDOWN condition	A.1 IMMEDIATE A.2. IMMEDIATE

2.2.5 Bases

Analysis in Chapter 4 demonstrates that if operating thermal (th) power is 1,250 kW, the maximum steady state fuel temperature is less than the safety limit for steady state operations by a large margin. For normal pool temperature, calculations in Chapter 4 demonstrate that the heat flux of the hottest area of the fuel rod generating the highest power level in the core during operations is less than the critical heat flux by a large margin up to the maximum permitted cooling temperatures; margin remains even at temperatures approaching bulk boiling for atmospheric conditions. Therefore, steady state operations at a maximum of 1,250 kW meet requirements for safe operation with respect to maximum fuel temperature and thermal hydraulics by a wide margin. Steady state operation of 1,250 kW was assumed in analyzing the loss of cooling and maximum hypothetical accidents. The analysis assumptions are protected by assuring that the maximum steady state operating power level is 1,250 kW.

In 1968 the reactor was licensed to operate at 250 kW with a minimum reactor safety system scram set point required by Technical Specifications at 110% of rated full power, with the scram set point set conservatively at 104%. In 1993 the original TRIGA power level channels were replaced with more reliable, solid state instrumentation. The actual safety system setting will be chosen to ensure that a scram will occur at a level that does not exceed 1,250 kW.

3. Limiting Conditions for Operation (LCO)

3.1 Core Reactivity

3.1.1 Applicability

These specifications are required prior to entering STEADY STATE MODE or PULSING MODE in OPERATING conditions; reactivity limits on experiments are specified in Section 3.8.

3.1.2 Objective

This LCO ensures the reactivity control system is OPERABLE, and that an accidental or inadvertent pulse does not result in exceeding the safety limit.

3.1.3 Specification

(1)	The maximum available core reactivity (excess reactivity) with all control rods fully withdrawn is less than $4.00 when: 1. REFERENCE CORE CONDITIONS exists 2. No experiments with net negative reactivity worth are in place
(2)	The reactor is capable of being made subcritical by a SHUTDOWN MARGIN more than $0.50 under REFERENCE CORE CONDITIONS and under the following conditions: 1. The highest worth control rod is fully withdrawn 2. The highest worth NONSECURED EXPERIMENT is in its most positive reactive state, and each SECURED EXPERIMENT with movable parts is in its most reactive state.

3.1.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactivity with all control rods fully withdrawn exceeds $4.00	A.1 ENSURE REACTOR SHUTDOWN AND A.2 Configure reactor to meet LCO	A.1 IMMEDIATE A.2 Prior to continued operations

B. The reactor is not subcritical by more than $0.50 under specified conditions	B.1.a ENSURE control rods fully inserted AND B.1.b Secure electrical power to the control rod circuits AND B.1.c Secure all work on in-core experiments or installed control rod drives AND B.2 Configure reactor to meet LCO	B.1 IMMEDIATE B.2 Prior to continued operations

3.1.5 Bases

The value for excess reactivity was used in establishing core conditions for calculations (Table 13.4) that demonstrate fuel temperature limits are met during potential accident scenarios under extremely conservative conditions of analysis. Since the fundamental protection for the KSU reactor is the maximum power level and fuel temperature that can be achieved with the available positive core reactivity, experiments with positive reactivity are included in determining excess reactivity. Since experiments with negative reactivity will increase available reactivity if they are removed during operation, they are not credited in determining excess reactivity.

Analysis (Chapter 13) shows fuel temperature will not exceed 1,150°C for the stainless-steel-clad fuel in the event of inadvertent or accidental pulsing of the reactor. Section 13.2 demonstrates that a $3.00 reactivity insertion from critical, zero power conditions leads to maximum fuel temperature of 746°C, while a $1.00 reactivity insertion from a worst-case steady state operation at 107 kW leads to a maximum fuel temperature of 869°C, well below the safety limit.

The limiting SHUTDOWN MARGIN is necessary so that the reactor can be shut down from any operating condition, and will remain shut down after cool down and xenon decay, even if one control rod (including the transient control rod) should remain in the fully withdrawn position.

3.2 PULSED MODE Operations

3.2.1 Applicability

These specifications apply to operation of the reactor in the PULSE MODE.

3.2.2 Objective

This Limiting Condition for Operation prevents fuel temperature safety limit from being exceeded during PULSE MODE operation.

3.2.3 Specification

(1)	The transient rod drive is positioned for reactivity insertion (upon withdrawal) less than or equal to $3.00

3.2.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. With all stainless steel clad fuel elements, the worth of the pulse rod in the transient rod drive position is greater than $3.00 in the PULSE MODE	A.1 Position the transient rod drive for pulse rod worth less than or equal to $3.00 OR A.2 Place reactor in STEADY STATE MODE	A.1 IMMEDIATE OR A.2 IMMEDIATE

3.2.5 Bases

The value for pulsed reactivity with all stainless steel elements in the core was used in establishing core conditions for calculations (Table 13.4) that demonstrate fuel temperature limits are met during potential accident scenarios under extremely conservative conditions of analysis.

3.3 MEASURING CHANNELS

3.3.1 Applicability

This specification applies to the reactor MEASURING CHANNELS during STEADY STATE MODE and PULSE MODE operations.

3.3.2 Objective

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

3.3.3 Specifications

(1)	The MEASURING CHANNELS specified in TABLE 1 SHALL be OPERATING
(2)	The neutron count rate on the startup channel is greater than the minimum detector sensitivity

TABLE 1: MINIMUM MEASURING CHANNEL COMPLEMENT

MEASURING CHANNEL	Minimum Number Operable	
	STEADY STATE MODE	PULSE MODE
Reactor power level[1]	2	1
Primary Pool Water Temperature	1	1
Reactor Bay Differential Pressure	1	1
Fuel Temperature	1	1
22 foot Area radiation monitor	1	1
0 or 12 foot Area monitor	1	1
Continuous air radiation monitor[2]	1	1
EXHAUST PLENUM radiation monitor[2]	1	1

NOTE[1]: One "Startup Channel" required to have range that indicates <10 W
NOTE[2]: High-level alarms audible in the control room may be used

3.3.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A.1 Reactor power channels not OPERATING (min 2 for STEADY STATE, 1 PULSE MODE)	A.1.1 Restore channel to operation OR A.1.2 ENSURE reactor is SHUTDOWN	A.1.1 IMMEDIATE A.1.2 IMMEDIATE

CONDITION	REQUIRED ACTION	COMPLETION TIME
A.2 High voltage to reactor power level detector less than 90% of required operating value	A.2.1 Establish REACTOR SHUTDOWN condition AND A.2.2 Enter REACTOR SECURED mode	A.2. IMMEDIATE
B. Primary water temperature, reactor bay differential pressure or fuel temperature CHANNEL not operable	B.1 Restore channel to operation OR B.2 ENSURE reactor is SHUTDOWN	A.1 IMMEDIATE A.2 IMMEDIATE
C. 22 foot Area radiation monitor is not OPERATING	C.1 Restore MEASURING CHANNEL OR C.2 ENSURE reactor is shutdown OR C.3 ENSURE personnel are not on the 22 foot level OR C.4 ENSURE personnel on 22 foot level are using portable survey meters to monitor dose rates	C.1 IMMEDIATE C.2 IMMEDIATE C.3 IMMEDIATE C.4 IMMEDIATE
D. 0 or 12 foot Area monitor is not OPERATING	D.1 Restore MEASURING CHANNEL OR D.2 ENSURE reactor is shutdown OR D.3 ENSURE personnel are not in the reactor bay OR D.4 ENSURE personnel entering reactor bay are using portable survey meters to monitor dose rates	D.1 IMMEDIATE D.2 IMMEDIATE D.3 IMMEDIATE D.4 IMMEDIATE

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Continuous air radiation monitor is not OPERATING	E.1 Restore MEASURING CHANNEL OR E.2 ENSURE reactor is shutdown OR E.3.a ENSURE EXHAUST PLENUM radiation monitor is OPERATING AND E.3.b Restore MEASURING CHANNEL	E.1 IMMEDIATE E.2. IMMEDIATE E.3.a. IMMEDIATE E.3.b Within 30 days
F. Exhaust plenum radiation monitor is not OPERATING	F.1 Restore MEASURING CHANNEL OR F.2 ENSURE reactor is shutdown OR F.3.a ENSURE continuous air radiation monitor is OPERATING AND F.3.b Restore MEASURING CHANNEL	F.1 IMMEDIATE F.2. IMMEDIATE F.3.a. IMMEDIATE F.3.b Within 30 days
G. The neutron count rate on the startup channel is not greater than the minimum detector sensitivity	G.1 Do not perform a reactor startup OR G.2 Perform a neutron-source check on the startup channel prior to startup	G.1 IMMEDIATE G.2 IMMEDIATE

3.3.5 Bases

Maximum steady state power level is 1,250 kW; neutron detectors measure reactor power level. Chapter 4 and 13 discuss normal and accident heat removal capabilities. Chapter 7 discusses radiation detection and monitoring systems, and neutron and power level detection systems.

According to General Atomics, detector voltages less than 90% of required operating value do not provide reliable, accurate nuclear instrumentation. Therefore, if operating voltage falls below the minimum value the power level channel is inoperable.

Primary water temperature indication is required to assure water temperature limits are met, protecting primary cleanup resin integrity. The reactor bay differential pressure indictor is required to control reactor bay atmosphere radioactive contaminants. Fuel temperature indication provides a means of observing that the safety limits are met.

The 22-foot and 0-foot area radiation monitors provide information about radiation hazards in the reactor bay. A loss of reactor pool water (Chapter 13), changes in shielding effectiveness (Chapter 11), and releases of radioactive material to the restricted area (Chapter 11) could cause changes in radiation levels within the reactor bay detectable by these monitors. Portable survey instruments will detect changes in radiation levels.

The air monitors (continuous air- and exhaust plenum radiation-monitor) provide indication of airborne contaminants in the reactor bay prior to discharge of gaseous effluent. Iodine channels provide evidence of fuel element failure. The air monitors provide similar information on independent channels; the continuous air monitor (CAM) has maximum sensitivity to iodine and particulate activity, while the air monitoring system (AMS) has individual channels for radioactive particulate, iodine, noble gas and iodine.

When filters in the air monitoring system begin to load, there are frequent, sporadic trips of the AMS alarms. Although the filters are changed on a regular basis, changing air quality makes these trips difficult to prevent. Short outages of the AMS system have resulted in unnecessary shutdowns, exercising the shutdown mechanisms unnecessarily, creating stressful situations, and preventing the ability to fully discharge the mission of the facility while the CAM also monitors conditions of airborne contamination monitored by the AMS. The AMS detector has failure modes than cannot be corrected on site; AMS failures have caused longer outages at the K-State reactor. The facility has experienced approximately two-week outages, with one week dedicated to testing and troubleshooting and (sometimes) one-week for shipment and repair at the vendor facility.

Permitting operation using a single channel of atmospheric monitoring will reduce unnecessary shutdowns while maintaining the ability to detect abnormal conditions as they develop. Relative indications ensure discharges are routine; abnormal indications trigger investigation or action to prevent the release of radioactive material to the surrounding environment. Ensuring the alternate airborne contamination monitor is functioning during outages of one system provides the contamination monitoring required for detecting abnormal conditions. Limiting the outage for a single unit to a maximum of 30 days ensures radioactive atmospheric contaminants are monitored while permitting maintenance and repair outages on the other system.

Chapter 13 discusses inventories and releases of radioactive material from fuel element failure into the reactor bay, and to the environment. Particulate and noble gas channels monitor more routine discharges. Chapter 11 and SAR Appendix A discuss routine discharges of radioactive gasses generated from normal operations into the reactor bay and into the environment. Chapter 3 identifies design bases for the confinement and ventilation system. Chapter 7 discusses air-monitoring systems.

Experience has shown that subcritical multiplication with the neutron source used in the reactor does not provide enough neutron flux to correspond to an indicated power level of 10 Watts. Therefore an indicated power of 10 Watts or more indicates operating in a potential critical condition, and at least one neutron channel is required with sensitivity at a neutron flux level corresponding to reactor power levels less than 10 Watts ("Startup Channel"). If the indicated neutron level is less than the minimum sensitivity for both the log-wide range and the multirange linear power level channels, a neutron source will be used to determine that at least one of the channels is responding to neutrons to ensure that the channel is functioning prior to startup.

3.4 Safety Channel and Control Rod Operability

3.4.1 Applicability

This specification applies to the reactor MEASURING Channels during STEADY STATE MODE and PULSE MODE operations.

3.4.2 Objective

The objectives are to require the minimum number of REACTOR SAFETY SYSTEM channels that must be OPERABLE in order to ensure that the fuel temperature safety limit is not exceeded, and to ensure prompt shutdown in the event of a scram signal.

3.4.3 Specifications

(1)	The SAFETY SYSTEM CHANNELS specified in TABLE 2 are OPERABLE
(2)	CONTROL RODS (STANDARD) are capable of 90% of full reactivity insertion from the fully withdrawn position in less than 1 sec.
(3)	A minimum of three CONTROL RODS must be OPERABLE. Inoperable CONTROL RODS must be fully inserted.

TABLE 2: REQUIRED SAFETY SYSTEM CHANNELS

Safety System Channel or Interlock	Minimum Number Operable	Function	Required OPERATING Mode	
			STEADY STATE MODE	PULSE MODE
Reactor power level	2	Scram	YES	NA
Manual scram bar	1	Scram	YES	YES
CONTROL ROD (STANDARD) position interlock	1	Prevent withdrawal of standard rods in the PULSE MODE	NA	YES
Pulse rod interlock	1	Prevent inadvertent pulsing while in STEADY STATE MODE	YES	NA

3.4.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any required SAFETY SYSTEM CHANNEL or interlock function is not OPERABLE	A.1 Restore channel or interlock to operation OR A.2 ENSURE reactor is SHUTDOWN	A1. IMMEDIATE A2. IMMEDIATE

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. A control rod is not OPERABLE.	B.1 ENSURE inoperable control rod is fully inserted OR B.2 ENSURE reactor is SHUTDOWN	B1. IMMEDIATE B2. IMMEDIATE

3.4.5 Bases

The power level scram is provided to ensure that reactor operation stays within the licensed limits of 1,250 kW, preventing abnormally high fuel temperature. The power level scram is not credited in analysis, but provides defense in depth to assure that the reactor is not operated in conditions beyond the assumptions used in analysis (Table 13.2.1.4).

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

The CONTROL ROD (STANDARD) interlock function is to prevent withdrawing control rods (other than the pulse rod) when the reactor is in the PULSE MODE. This will ensure the reactivity addition rate during a pulse is limited to the reactivity added by the pulse rod.

The pulse rod interlock function prevents air from being applied to the transient rod drive when it is withdrawn while disconnected from the control rod to prevent inadvertent pulses during STEADY STATE MODE operations. The control rod interlock prevents inadvertent pulses which would be likely to exceed the maximum range of the power level instruments configured for steady state operations.

Inoperable control rods that are fully inserted in the reactor will not negatively affect the minimum safety shutdown margin or maximum excess reactivity of the core. Operating with a fully-inserted control rod may cause power peaking to shift, however, in this case calculations have demonstrated that the maximum element-to-average power peaking of 2.0 assumed in SAR Chapter 13 is still bounding, and the reduction in maximum core power by having an inoperable control rod fully inserted means that the highest temperature in any fuel element with a fully-inserted inoperable control rod will be lower than the highest temperature in the B-ring with all rods withdrawn. Therefore the reactor can be safely operated with an inoperable control rod provided that the rod is fully inserted into the core.

3.5 Gaseous Effluent Control

3.5.1 Applicability

This specification applies to gaseous effluent in STEADY STATE MODE and PULSE MODE.

3.5.2 Objective

The objective is to ensure that exposures to the public resulting from gaseous effluents released during normal operations and accident conditions are within limits and ALARA.

3.5.3 Specification

(1)	The reactor bay ventilation exhaust system SHALL maintain in-leakage to the reactor bay
(2)	Releases of Ar-41 from the reactor bay exhaust plenum to an unrestricted environment SHALL NOT exceed 30 Ci per year.

3.5.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The reactor bay ventilation exhaust system is not OPERABLE	A.1 ENSURE reactor is SHUTDOWN	A.1 IMMEDIATE
	OR	
	A.2.a Do not OPERATE in the PULSE MODE	A.2.a IMMEDIATE
	AND	
	A.2.b Secure EXPERIMENT operations for EXPERIMENT with failure modes that could result in the release of radioactive gases or aerosols.	A.2.b IMMEDIATE
	A.2.c ENSURE no irradiated fuel handing	A.2.b IMMEDIATE
	AND	
	A.2.d Restore the reactor bay ventilation exhaust system to OPERABLE	A.2.d Within 30 days

CONDITION	REQUIRED ACTION	COMPLETION TIME
Calculated releases of Ar-41 from the reactor bay exhaust plenum exceed 30 Ci per year.	Do not operate.	IMMEDIATE

3.5.5 Bases

The confinement and ventilation system is described in Section 3.5.4. Routine operations produce radioactive gas, principally Argon 41, in the reactor bay. If the reactor bay ventilation system is secured, SAR Chapter 11 Appendix A demonstrates reactor bay concentration of 0.746 Bq ml^{-1} ($2.01x10^{-5}$ μCi ml^{-1}), well below the 10CFR20 annual limit of 2000 DAC hours of Argon 41 at 6×10^{-3} μCi h/mL. Therefore, the reduction in concentration of Argon 41 from operation of the confinement and ventilation system is a defense in depth measure, and not required to assure meeting personnel exposure limits. Consequently, the ventilation system can be secured without causing significant personnel hazard from normal operations. Thirty days for a confinement and ventilation system outage is selected as a reasonable interval to allow major repairs and work to be accomplished, if required. During this interval, experiment activities that might cause airborne radionuclide levels to be elevated are prohibited.

It is shown in Section 13.2.2 of the Safety Analysis Report that, if the reactor were to be operating at full steady-state power, fuel element failure would not occur even if all the reactor tank water were to be lost instantaneously.

Section 13.2.4 addresses the maximum hypothetical fission product inventory release. Using unrealistically conservative assumptions, concentrations for a few nuclides of iodine would be in excess of occupational derived air concentrations for a matter of hours or days. ^{90}Sr activity available for release from fuel rods previously used at other facilities is estimated to be at most about 4 times the ALI. In either case (radio-iodine or -Sr), there is no credible scenario for accidental inhalation or ingestion of the undiluted nuclides that might be released from a damaged fuel element. Finally, fuel element failure during a fuel handling accident is likely to be observed and mitigated immediately.

SAR Appendix A shows the release of 30 Ci per year of Ar-41 from normal operations would result in less than 10 mrem annual exposure to any person in unrestricted areas.

3.6 Limitations on Experiments

3.6.1 Applicability

This specification applies to operations in STEADY STATE MODE and PULSE MODE.

3.6.2 Objectives

These Limiting Conditions for Operation prevent reactivity excursions that might cause the fuel temperature to exceed the safety limit (with possible resultant damage to the reactor), and the excessive release of radioactive materials in the event of an EXPERIMENT failure

3.6.3 Specifications

(1)	If all fuel elements are stainless steel clad, the reactivity worth of any individual EXPERIMENT SHALL NOT exceed $2.00
(2)	If two or more experiments in the reactor are interrelated so that operation or failure of one can induce reactivity-affecting change in the other(s), the sum of the absolute reactivity of such experiments SHALL NOT exceed $2.00.
(3)	Irradiation holders and vials SHALL prevent release of encapsulated material in the reactor pool and core area

3.6.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. INDEPENDENT EXPERIMENT worth is greater than $2.00	A.1 ENSURE the reactor is SHUTDOWN AND A.2 Remove the experiment	A.1 IMMEDIATE A.2 Prior to continued operations
C. An irradiation holder or vial releases material capable of causing damage to the reactor fuel or structure into the pool or core area	C.1 ENSURE the reactor is SHUTDOWN AND C.2 Inspect the affected area AND C.3 Obtain RSC review and approval	C.1 IMMEDIATE C.2 Prior to continued operation C.3 Prior to continued operation

3.6.5 Bases

Specifications 3.7(1) through 3.7(3) are conservatively chosen based on prior operation at 250 kW to limit reactivity additions to maximum values that are less than an addition which could cause temperature to challenge the safety limit.

Experiments are approved with expectations that there is reasonable assurance the facility will not be damaged during normal or failure conditions. If an irradiation capsule which contains material with potential for challenging the fuel cladding or pool wall, the facility will be inspected to ensure that continued operation is acceptable.

3.7 Fuel Integrity

3.7.1 Applicability

This specification applies to operations in STEADY STATE MODE and PULSE MODE.

3.7.2 Objective

The objective is to prevent the use of damaged fuel in the KSU TRIGA reactor.

3.7.3 Specifications

(1)	Fuel elements in the reactor core SHALL NOT be elongated more than 1/8 in. over manufactured length
(2)	Fuel elements in the reactor core SHALL NOT be laterally bent more than 1/8 in.

3.7.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any fuel element is elongated greater than l/8 in. over manufactured length, or bent laterally greater than l/8 in.	Do not insert the fuel element into the upper core grid plate.	IMMEDIATE

3.7.5 Bases

The above limits on the allowable distortion of a fuel element have been shown to correspond to strains that are considerably lower than the strain expected to cause rupture of a fuel element and have been successfully applied at TRIGA installations. Fuel cladding integrity is important since it represents the only process barrier for fission product release from the TRIGA reactor.

3.8 Reactor Pool Water

3.8.1 Applicability

This specification applies to operations in STEADY STATE MODE, PULSE MODE, and SECURED MODE.

3.8.2 Objective

The objective is to set acceptable limits on the water quality, temperature, conductivity, and level in the reactor pool.

3.8.3 Specifications

(1)	Bulk water temperature SHALL NOT exceed 44°C (111°F)
(2)	Water conductivity SHALL be less than 5 μmho/cm
(3)	Water level above the core SHALL be at least 13 ft from the top of the core
(4)	Bulk water temperature SHALL NOT exceed 37°C (99°F) with an experiment installed in an interstitial flux wire port.

3.8.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Bulk water temperature exceeds 44°C	A.1 ENSURE the reactor is SHUTDOWN AND A.2 Reduce bulk water temperature to less than 44°C	A.1 IMMEDIATE A.3 IMMEDIATE

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Bulk water temperature exceeds 37°C with an experiment installed in an interstitial flux wire port.	B.1 ENSURE the reactor is SHUTDOWN AND B.2 Reduce bulk water temperature to less than 37°C. OR B.3 Remove experiment from flux wire port	B.1 IMMEDIATE B.2 IMMEDIATE B.3 IMMEDIATE
B. Water conductivity is greater than 5 μmho/cm	B.1 ENSURE the reactor is SHUTDOWN AND B.2 Restore conductivity to less than 5 μmho/cm	B.1 IMMEDIATE B.2 Within 4 weeks
C. Water level above the core SHALL be at least 13 ft from the top of the core for all operating conditions	C.1 ENSURE the reactor is SHUTDOWN AND C.2 Restore water level	C.1 IMMEDIATE C.2 IMMEDIATE

3.8.5 Bases

The resin used in the mixed bed deionizer limits the water temperature of the reactor pool. Resin in use (as described in Section 5.4) maintains mechanical and chemical integrity at temperatures below 130°F (54.4°C). While the integrity of the ion exchange resin requires water temperature to remain below 54.4°C, it is necessary to maintain water temperature below 44°C to ensure that the departure from nucleate boiling ratio (DNBR) will remain at least 2.0 for the hot channel while operating at 1250 kWth in STEADY STATE MODE and that excessive amounts of nucleate boiling will not occur. Insertion of an experiment into an interstitial flux wire port between fuel elements necessitates a further reduction in water temperature to a maximum of 37°C in order to preclude excessive nucleate boiling of the water.

Maintaining low water conductivity over a prolonged period prevents possible corrosion, deionizer degradation, or slow leakage of fission products from degraded cladding. Although fuel degradation does not occur over short time intervals, long-term integrity of the fuel is important, and a 4-week interval was selected as an appropriate maximum time for high conductivity.

The top of the core is 16 feet below the top of the primary coolant tank. The lowest suction of primary cooling flow into the forced cooling loop is 3.5 feet below the top of the primary coolant tank (water level is maintained about 6 inches below the top of the tank).

The principle contributor to radiation dose rates at the pool surface is Nitrogen 16 generated in the reactor core and dispersed in the pool. Calculations in Chapter 11 show the pool surface radiation dose rates from Nitrogen 16 with 13 feet of water above the core are acceptable.

For normal pool temperature, calculations in Chapter 4 assuming 16 feet and 13 feet above the core demonstrate that the heat flux of the hottest area of the fuel rod generating the highest power level in the core during operations is less than the critical heat flux by a large margin up to the maximum permitted cooling temperatures; margin remains even at temperatures approaching bulk boiling for atmospheric conditions. Therefore, pool levels greater than 13 feet above the core meet requirements for safe operation with respect to maximum fuel temperature and thermal hydraulics by a wide margin.

Therefore, a minimum pool level of 13 feet above the core is adequate to provide shielding and support the core cooling.

3.9 Maintenance Retest Requirements

3.9.1 Applicability

This specification applies to operations in STEADY STATE MODE and PULSE MODE.

3.9.2 Objective

The objective is to ensure Technical Specification requirements are met following maintenance that occurs within surveillance test intervals.

3.9.3 Specifications

Maintenance activities SHALL NOT change, defeat or alter equipment or systems in a way that prevents the systems or equipment from being OPERABLE or otherwise prevent the systems or equipment from fulfilling the safety basis

3.9.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
Maintenance is performed that has the potential to change a setpoint, calibration, flow rate, or other parameter that is measured or verified in meeting a surveillance or operability requirement	Perform surveillance OR Operate only to perform retest	Prior to continued, normal operation in STEADY STATE MODE or PULSE MODE

3.9.5 Bases

Operation of the K-State reactor will comply with the requirements of Technical Specifications. This specification ensures that if maintenance might challenge a Technical Specifications requirement, the requirement is verified prior to resumption of normal operations.

3.10 Maximum Steady State Power

3.9.1 Applicability

This specification applies to operations in STEADY STATE MODE.

3.9.2 Objective

The objective is to ensure that the reactor has adequate margin to critical heat flux (CHF) and operates below the Limiting Safety System Setting of 1,250 kWth.

3.9.3 Specifications

(1)	Maximum OPERATING thermal power SHALL NOT exceed 1,000 kWth in STEADY STATE MODE.
(2)	A required reactor power level scram is set to a value no greater than 1,250 kWth.

3.9.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Thermal power exceeds 1,050 kWth in STEADY STATE MODE	Reduce power to a level no greater than 1,050 kWth.	IMMEDIATE
B. A required reactor power level scram is set to a value above 1,250 kWth or above the maximum readable value on a required channel.	B.1 SHUT DOWN the reactor. AND B.2 Adjust reactor power level scram setpoint to a readable value less than or equal to 1,250 kWth.	B.1. IMMEDIATE AND B.2. Prior to resuming operations.

3.9.5 Bases

The reactor control panel instrumentation is designed to measure up to 1,000 kWth of thermal power. The Limiting Safety System Setting ensures that automatic protective functions, i.e., high power scrams, are set to no greater than 1,250 kWth. However, by specifying the maximum OPERATING power level as 1,000 kWth in STEADY STATE MODE, the reactor will have additional margin to critical heat flux and will still be allowed to operate at up to the maximum power readable on the reactor console instruments. Action to reduce power is not required until power exceeds 1050kWth in STEADY STATE MODE to allow for slight variation in power level that is typical during normal operation.

4. Surveillance Requirements

4.1 Core Reactivity

4.1.1 Objective

This surveillance ensures that the minimum SHUTDOWN MARGIN requirements and maximum excess reactivity limits of section 3.1 are met.

4.1.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SHUTDOWN MARGIN Determination	SEMIANNUAL
Excess Reactivity Determination	SEMIANNUAL
	Following Insertion of experiments with measurable positive reactivity
Control Rod Reactivity Worth determination	BIENNIAL

4.1.3 Basis

Experience has shown verification of the minimum allowed SHUTDOWN MARGIN at the specified frequency is adequate to assure that the limiting safety system setting is met

When core reactivity parameters are affected by operations or maintenance, additional activity is required to ensure changes are incorporated in reactivity evaluations.

4.2 PULSE MODE

4.2.1 Objectives

The verification that the pulse rod position does not exceed a reactivity value corresponding to $3.00 assures that the limiting condition for operation is met.

4.2.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
ENSURE Transient Pulse Rod position corresponds to reactivity not greater than $3.00	Prior to pulsing operations

4.2.3 Basis

Verifying pulse rod position corresponds to less than $3.00 ensures that the maximum pulsed reactivity meets the limiting condition for operation.

4.3 MEASURING CHANNELS

4.3.1 Objectives

Surveillances on MEASURING CHANNELS at specified frequencies ensure instrument problems are identified and corrected before they can affect operations.

4.3.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Reactor power level MEASURING CHANNEL	
CHANNEL TEST	DAILY
Calorimetric calibration	ANNUAL
CHANNEL CHECK high voltage to required power level instruments	DAILY
Primary pool water temperature CHANNEL CALIBRATION	ANNUAL
Reactor Bay differential pressure CHANNEL CALIBRATION	ANNUAL
Fuel temperature CHANNEL CALIBRATION	ANNUAL
22 Foot Area radiation monitor	
CHANNEL CHECK	DAILY
CHANNEL CALIBRATION	ANNUAL
0 or 12 Foot Area Radiation Monitor	
CHANNEL CHECK	DAILY
CHANNEL CALIBRATION	ANNUAL
Continuous Air Radiation Monitor	
CHANNEL CHECK	DAILY
CHANNEL CALIBRATION	ANNUAL
EXHAUST PLENUM Radiation Monitor	
CHANNEL CHECK	DAILY
CHANNEL CALIBRATION	ANNUAL
Startup Count Rate	DAILY

4.3.3 Basis

The DAILY CHANNEL CHECKS will ensure that the SAFETY SYSTEM and MEASURING CHANNELS are operable. The required periodic calibrations and verifications will permit any long-term drift of the channels to be corrected.

4.4 Safety Channel and Control Rod Operability

4.4.1 Objective

The objectives of these surveillance requirements are to ensure the REACTOR SAFETY SYSTEM will function as required. Surveillances related to safety system MEASURING CHANNELS ensure appropriate signals are reliably transmitted to the shutdown system; the surveillances in this section ensure the control rod system is capable of providing the necessary actions to respond to these signals.

4.4.2 Specifications

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Manual scram SHALL be tested by releasing partially withdrawn CONTROL RODS (STANDARD)	DAILY
CONTROL ROD (STANDARD) drop times SHALL be measured to have a drop time from the fully withdrawn position of less than 1 sec.	ANNUAL
The control rods SHALL be visually inspected for corrosion and mechanical damage at intervals	BIENNIAL
CONTROL ROD (STANDARD) position interlock functional test	SEMIANNUAL
Pulse rod interlock functional test	SEMIANNUAL
On each day that PULSE MODE operation of the reactor is planned, a functional performance check of the CONTROL ROD (TRANSIENT) system SHALL be performed.	Prior to pulsing operations each day a pulse is planned
The CONTROL ROD (TRANSIENT) rod drive cylinder and the associated air supply system SHALL be inspected, cleaned, and lubricated, as necessary.	SEMIANNUAL

4.4.3 Basis

Manual and automatic scrams are not credited in accident analysis, although the systems function to assure long-term safe shutdown conditions. The manual scram and control rod drop timing surveillances are intended to monitor for potential degradation that might interfere with the operation of the control rod systems. The verification of high voltage to the power level monitoring channels assures that the instrument channel providing an overpower trip will function on demand.

The control rod inspections (visual inspections and transient drive system inspections) are similarly intended to identify potential degradation that lead to control rod degradation or inoperability.

A test of the interlock that prevents the pulse rod from coupling to the drive in the state state mode unless the drive is fully down assures that pulses will occur only when in pulsing mode. A

test of the interlock that prevents standard control rod motion while in the pulse mode assures that the interlock will function as required.

The functional checks of the control rod drive system assure the control rod drive system operates as intended for any pulsing operations. The inspection of the pulse rod mechanism will assure degradation of the pulse rod drive will be detected prior to malfunctions.

4.5 Gaseous Effluent Control

4.5.1 Objectives

These surveillances ensure that routine releases are normal, and (in conjunction with MEASURING CHANNEL surveillances) that instruments will alert the facility if conditions indicate abnormal releases.

4.5.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL TEST of air monitor	ANNUAL
Verify negative reactor bay differential pressure	DAILY

4.5.3 Basis

The continuous air monitor provides indication that levels of radioactive airborne contamination in the reactor bay are normal.

If the reactor bay differential pressure gage indicates a negative pressure, the reactor bay exhaust fan is controlling airflow by directing effluent out of confinement.

4.6 Limitations on Experiments

4.6.1 Objectives

This surveillance ensures that experiments do not have significant negative impact on safety of the public, personnel or the facility.

4.6.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Experiments SHALL be evaluated and approved prior to implementation.	Prior to inserting a new experiment for purposes other than determination of reactivity worth
Measure and record experiment worth of the EXPERIMENT (where the absolute value of the estimated worth is greater than $0.40).	Initial insertion of a new experiment where absolute value of the estimated worth is greater than $0.40

4.6.3 Basis

These surveillances allow determination that the limits of 3.7 are met.

Experiments with an absolute value of the estimated significant reactivity worth (greater than $0.40) will be measured to assure that maximum experiment reactivity worths are met. If an absolute value of the estimate indicates less than $0.40 reactivity worth, even a 100% error will result in actual reactivity less than the assumptions used in analysis for inadvertent pulsing at low power operations in the Safety Analysis Report (13.2.3, Case I).

4.7 Fuel Integrity

4.7.1 Objective

The objective is to ensure that the dimensions of the fuel elements remain within acceptable limits.

4.7.2 Applicability

This specification applies to the surveillance requirements for the fuel elements in the reactor core.

4.7.3 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
The standard fuel elements SHALL be visually inspected for corrosion and mechanical damage, and measured for length and bend	500 pulses of magnitude equal to or less than a pulse insertion of 3.00$ AND Following the exceeding of a limited safety system set point with potential for causing degradation
B, C, D, E, and F RING elements comprising approximately 1/3 of the core SHALL be visually inspected annually for corrosion and mechanical damage such that the entire core SHALL be inspected at 3-year intervals, but not to exceed 38 months	ANNUAL

4.7.4 Basis

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.

Triennial visual inspection of fuel elements combined with measurements at intervals determined by pulsing as described is considered adequate to identify potential degradation of fuel prior to catastrophic fuel element failure.

4.8 Reactor Pool Water

This specification applies to the water contained in the KSU TRIGA reactor pool.

4.8.1 Objective

The objective is to provide surveillance of reactor primary coolant water quality, pool level, temperature and (in conjunction with MEASURING CHANNEL surveillances), and conductivity.

4.8.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify reactor pool water level above the inlet line vacuum breaker	DAILY
Verify reactor pool water temperature channel operable	DAILY
Measure reactor Pool water conductivity	DAILY
	At least every 20 days

4.9.3 Bases

Surveillance of the reactor pool will ensure that the water level is adequate before reactor operation. Evaporation occurs over longer periods of time, and daily checks are adequate to identify the need for water replacement.

Water temperature must be monitored to ensure that the limit of the deionizer will not be exceeded. A daily check on the instrument prior to reactor operation is adequate to ensure the instrument is operable when it will be needed.

Water conductivity must be checked to ensure that the deionizer is performing properly and to detect any increase in water impurities. A daily check is adequate to verify water quality is appropriate and also to provide data useful in trend analysis. If the reactor is not operated for long periods of time, the requirement for checks at least every 20 days will ensure water quality is maintained in a manner that does not permit fuel degradation.

4.9 Maintenance Retest Requirements

4.9.1 Objective

The objective is to ensure that a system is OPERABLE within specified limits before being used after maintenance has been performed.

4.9.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Evaluate potential for maintenance activities to affect operability and function of equipment required by Technical Specifications	Following maintenance of systems of equipment required by Technical Specifications
Perform surveillance to assure affected function meets requirements	Prior to resumption of normal operations

4.9.3 Bases

This specification ensures that work on the system or component has been properly carried out and that the system or component has been properly reinstalled or reconnected before reliance for safety is placed on it.

5. Design Features

5.1 Reactor Fuel

5.1.1 Applicability

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

5.1.2 Objective

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

5.1.3 Specification

(1) The high-hydride fuel element shall contain uranium-zirconium hydride, clad in 0.020 in. of 304 stainless steel. It shall contain a maximum of 12.5 weight percent uranium which has a maximum enrichment of 20%. There shall be 1.55 to 1.80 hydrogen atoms to 1.0 zirconium atom.

(2) For the loading process, the elements shall be placed in a close packed array except for experimental facilities or for single positions occupied by control rods and a neutron startup source.

(3) Up to four elements with greater than 9.0 weight percent uranium may be installed in the core. These elements may only be placed in the E- and F-rings of the core lattice, and may not be located in the following positions: E2, E4, E5, E6, E20, E21, E22, E24, F1, F2, F30.

5.1.4 Bases

These types of fuel elements have a long history of successful use in TRIGA reactors. Calculations show that 12%-load fuel in the E- and F-rings will not exceed the temperature of 8%-load instrumented elements in the B-ring. Additionally the power peaking and fission product inventory assumptions in the SAR will not be challenged by 12% fuel in the E- and F-rings. Local power and temperature peaking effects during pulsing are avoided by prohibiting placement of the 12%-load fuel near water and control rod channels.

5.2 Reactor Fuel and Fueled Devices in Storage

5.2.1 Applicability

This specification applies to reactor fuel elements in storage

5.2.2 Objective

The objective is to ensure fuel elements or fueled devices in storage are maintained Subcritical in a safe condition.

5.2.3 Specification

(1) All fuel elements or fueled devices shall be in a safe, stable geometry

(2) The k_{eff} of all fuel elements or fueled devices in storage is less than 0.8

(3) Irradiated fuel elements or fueled devices will be stored in an array which will permit sufficient natural convection cooling by air or water such that the fuel element or fueled device will not exceed design values.

5.2.4 Bases

This specification is based on American Nuclear Society standard 15.1, section 5.4.

5.3 Reactor Building

5.3.1 Applicability

This specification applies to the building that houses the TRIGA reactor facility.

5.3.2 Objective

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

5.3.3 Specification

(1) The reactor shall be housed in a closed room designed to restrict leakage when the reactor is in operation, when the facility is unmanned, or when spent fuel is being handled exterior to a cask.

(2) The minimum free volume of the reactor room shall be approximately 144,000 cubic feet.

(3) The building shall be equipped with a ventilation system capable of exhausting air or other gases from the reactor room at a minimum of 30 ft. above ground level.

5.3.4 Bases

To control the escape of gaseous effluent, the reactor room contains no windows that can be opened. The room air is exhausted through an independent exhaust system, and discharged at roof level to provide dilution.

5.4 Experiments

5.4.1 Applicability

This specification applies to the design of experiments.

5.4.2 Objective

The objective is to ensure that experiments are designed to meet criteria.

5.4.3 Specifications

(1) EXPERIMENT with a design reactivity worth greater than $1.00 SHALL be securely fastened (as defined in Section l, Secured Experiment).

(2) Design shall ensure that failure of an EXPERIMENT SHALL NOT lead to a direct failure of a fuel element or of other experiments that could result in a measurable increase in reactivity or a measurable release of radioactivity due to the associated failure.

(3) EXPERIMENT SHALL be designed so that it does not cause bulk boiling of core water

(4) EXPERIMENT design SHALL ensure no interference with control rods or shadowing of reactor control instrumentation.

(5) EXPERIMENT design shall minimize the potential for industrial hazards, such as fire or the release of hazardous and toxic materials.

(6) Each fueled experiment shall be limited such that the total inventory of iodine isotopes 131 through 135 in the experiment is not greater than 5 millicuries except as the fueled experiment is a standard TRIGA instrumented element in which instance the iodine inventory limit is removed.

(7) Where the possibility exists that the failure of an EXPERIMENT (except fueled EXPERIMENTS) could release radioactive gases or aerosols to the reactor bay or atmosphere, the quantity and type of material shall be limited such that the airborne concentration of radioactivity averaged over a year will not exceed the limits of Table II of Appendix B of 10 CFR Part 20 assuming 100% of the gases or aerosols escape.

(8) The following assumptions shall be used in experiment design:

a. If effluents from an experimental facility exhaust through a hold-up tank which closes automatically at a high radiation level, at least 10% of the gaseous activity or aerosols produced will escape.

b. If effluents from an experimental facility exhaust through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of the aerosols produced will escape.

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

(9) Use of explosive solid or liquid material with a National Fire Protection Association Reactivity (Stability) index of 2, 3, or 4 in the reactor pool or biological shielding SHALL NOT exceed the equivalent of 25 milligrams of TNT without prior NRC approval.

5.4.4 Basis

Designing the experiment to reactivity and thermal-hydraulic conditions ensure that the experiment is not capable of breaching fission product barriers or interfering with the control systems (interferences from other - than reactivity - effects with the control and safety systems are also prohibited). Design constraints on industrial hazards ensure personnel safety and continuity of operations. Design constraints limiting the release of radioactive gasses prevent unacceptable personnel exposure during off-normal experiment conditions.

6. Administrative Controls

6.1 Organization and Responsibilities of Personnel

a) Structure.

The reactor organization is related to the University structure as shown in SAR Figure 12.1 and Technical Specifications Figure TS.1 below.

Kansas State University (KSU) holds the license for the KSU TRIGA Reactor, located in the KSU Nuclear Reactor Facility in Ward Hall on the campus of Kansas State University. The chief administrating officer for KSU is the President. Environment, safety and health oversight functions are administered through the Vice President for Administration and Finance, while reactor line management functions are through the Provost Chief Academic Officer.

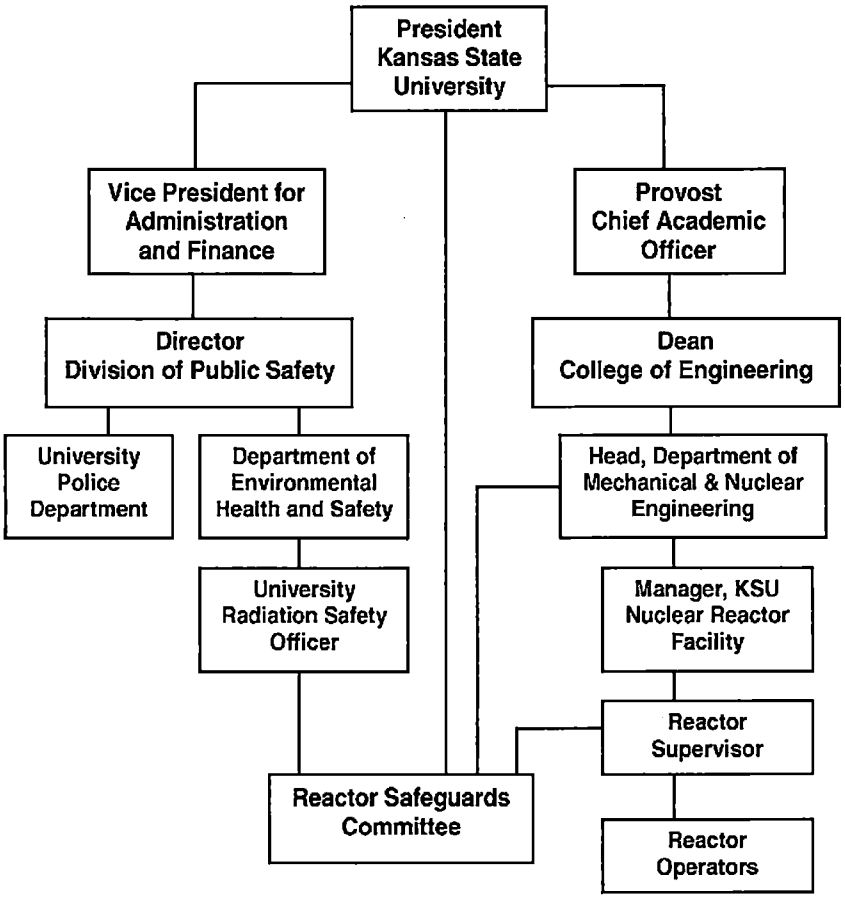

Figure TS.1: Organization and Management Structure for the K-State Reactor

Radiation protection functions are divided between the University Radiation Safety Officer (RSO) and the reactor staff and management, with management and authority for the RSO separate from line management and authority for facility operations. Day-to-day radiation protection functions implemented by facility staff and management are guided

by approved administrative controls (Reactor Radiation Protection Program or RPP, Facility Operating Manual, operating and experiment procedures); these controls are reviewed and approved by the RSO as part of the Reactor Safeguards Committee (with specific veto authority). The RSO has specific oversight functions assigned though the RPP. The RSO provides routine support for personnel monitoring, radiological analysis, and radioactive material inventory control. The RSO provides guidance on request for non-routine operations such as transportation and implementation of new experiments.

b) Responsibility.

The President of the University shall be responsible for the appointment of responsible and competent persons as members of the TRIGA Reactor Safeguards Committee upon the recommendation of the *ex officio* Chairperson of the Committee.

The KSU Nuclear Reactor Facility shall be under the supervision of the Nuclear Reactor Facility Manager, who shall have the overall responsibility for safe, efficient, and competent use of its facilities in conformity with all applicable laws, regulations, terms of facility licenses, and provisions of the Reactor Safeguards Committee. The Manager also has responsibility for maintenance and modification of laboratories associated with the Reactor Facility. The Manager shall have education and/or experience commensurate with the responsibilities of the position and shall report to the Head of the Department of Mechanical and Nuclear Engineering.

A Reactor Supervisor may serve as the deputy of the Nuclear Reactor Facility Manager in all matters relating to the enforcement of established rules and procedures (but not in matters such as establishment of rules, appointments, and similar administrative functions). The Supervisor should have at least two years of technical training beyond high school and shall possess a Senior Reactor Operator's license. The Supervisor shall have had reactor OPERATING experience and have a demonstrated competence in supervision. The Supervisor is appointed by the Nuclear Reactor Facility Manager and is responsible for enforcing all applicable rules, procedures, and regulations, for ensuring adequate exchange of information between OPERATING personnel when shifts change, and for reporting all malfunctions, accidents, and other potentially hazardous occurrences and situations to the Reactor Nuclear Reactor Facility Manager. The Nuclear Reactor Facility Manager may also serve as Reactor Supervisor.

The Reactor Operator shall be responsible for the safe and proper operation of the reactor, under the direction of the Reactor Supervisor. Reactor Operators shall possess an Operator's or Senior Operator's license and shall be appointed by the Nuclear Reactor Facility Manager.

The University Radiation Safety Officer (RSO), or a designated alternate, shall (in addition to other duties defined by the Director of Environmental Health and Safety, Division of Public Safety) be responsible for overseeing the safety of Reactor Facility operations from the standpoint of radiation protection. The RSO and/or designated alternate shall be appointed by the Director of Environmental Health and Safety, Division of Public Safety, with the approval of the University Radiation Safety Committee, and shall report to the Director of Environmental Health and Safety, whose organization is independent of the Reactor Facility organization, as shown on SAR Figure 12.1.

The Nuclear Reactor Facility Manager, with the approval of the Reactor Safeguards Committee, may designate an appropriately qualified member of the Facility organization as Reactor Facility Safety Officer (RFSO) with duties including those of an intra-Facility

Radiation Safety Officer. The University Radiation Safety Officer may, with the concurrence of the Nuclear Reactor Facility Manager, authorize the RFSO to perform some of the specific duties of the RSO at the Nuclear Reactor Facility.

c). Staffing.

Whenever the reactor is not secured, the reactor shall be under the direction of a (USNRC licensed) Senior Operator (designated as Reactor Supervisor). The Supervisor shall be on call, within twenty minutes travel time to the facility.

Whenever the reactor is not secured, a (USNRC licensed) Reactor Operator (or Senior Reactor Operator) who meets requirements of the Operator Requalification Program shall be at the reactor control console, and directly responsible for control manipulations.

In addition to the above requirements, during fuel movement a senior operator shall be inside the reactor bay directing fuel operations.

6.2 Review and Audit

a) There will be a Reactor Safeguards Committee which shall review TRIGA reactor operations to assure that the reactor facility is operated and used in a manner within the terms of the facility license and consistent with the safety of the public and of persons within the Laboratory.

b) The responsibilities of the Committee include, but are not limited to, the following:

1. Review and approval of rules, procedures, and proposed Technical Specifications;

2. Review and approval of all proposed changes in the facility that could have a significant effect on safety and of all proposed changes in rules, procedures, and Technical Specifications, in accordance with procedures in Section 6.3;

3. Review and approval of experiments using the reactor in accordance with procedures and criteria in Section 6.4;

4. Determine whether changes in the facility as described in the safety analysis report (as updated), changes in the procedures as described in the final safety analysis report (as updated), and the conduct of tests or experiments not described in the safety analysis report (as updated) may be accomplished in accordance with 10 CFR 50.59 without obtaining prior NRC approval via license amendment pursuant to 10 CFR Sec. 50.90.

5. Review of abnormal performance of plant equipment and OPERATING anomalies;

6. Review of unusual or abnormal occurrences and incidents which are reportable under 10 CFR 20 and 10 CFR50;

7. Inspection of the facility, review of safety measures, and audit of operations at a frequency not less than once a year, including operation and operations records of the facility;

8. Requalification of the Nuclear Reactor Facility Manager and/or the Reactor Supervisor,

9. Review of container failures where released materials have the potential for damaging reactor fuel or structural components including:

   a) results of physical inspection

   b) evaluation of consequences

   c) need for corrective actions

c) The Committee shall be composed of:

1. one or more persons proficient in reactor and nuclear science or engineering,

2. one or more persons proficient in chemistry, geology, or chemical engineering,

3. one person proficient in biological effects of radiation,

4. the Nuclear Reactor Facility Manager, *ex officio*,

5. the University Radiation Safety Officer, *ex officio*, and,

6. The Head of the Department of Mechanical and Nuclear Engineering, *ex officio*, or a designated deputy, to serve as chairperson of the Committee.

The same individual may serve under more than one category above, but the minimum membership shall be seven. At least five members shall be faculty members. The Reactor Supervisor, if other than the Nuclear Reactor Facility Manager, shall attend and participate in Committee meetings, but shall not be a voting member.

d) The Committee shall have a written statement defining its authority and responsibilities, the subjects within its purview, and other such administrative provisions as are required for its effective functioning. Minutes of all meetings and records of all formal actions of the Committee shall be kept.

e) A quorum shall consist of not less than a majority of the full Committee and shall include all *ex officio* members.

f) Any permissive action of the Committee requires affirmative vote of the University Radiation Safety Officer as well as a majority vote of the members present.

g) The Committee shall meet a minimum of two times a year. Additional meetings may be called by any member, and the Committee may be polled in lieu of a meeting. Such a poll shall constitute Committee action subject to the same requirements as for an actual meeting.

6.3 Procedures

a) Written procedures, reviewed and approved by the Reactor Safeguards Committee, shall be followed for the activities listed below. The procedures shall be adequate to

assure the safety of the reactor, persons within the Laboratory, and the public, but should not preclude the use of independent judgment and action should the situation require it. The activities are:

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

   (a) startup checkout procedures to test the reactor instrumentation and safety systems, area monitors, and continuous air monitors,

   (b) prohibition of routine operations with failed (or leaking) fuel except to find leaking elements, and

   (b) shutdown procedures to assure that the reactor is secured before OPERATING personnel go off duty.

2. Installation or removal of fuel elements, control rods, and other core components that significantly affect reactivity or reactor safety.

3. Preventive or corrective maintenance activities which could have a significant effect on the safety of the reactor or personnel.

4. Periodic inspection, testing or calibration of auxiliary systems or instrumentation that relate to reactor operation.

b) Substantive changes in the above procedures shall be made only with the approval of the Reactor Safeguards Committee, and shall be issued to the OPERATING personnel in written form. The Nuclear Reactor Facility Manager may make temporary changes that do not change the original intent. The change and the reasons thereof shall be noted in the log book, and shall be subsequently reviewed by the Reactor Safeguards Committee.

c) Determination as to whether a proposed activity in categories (1), (2) and (3) in Section 6.2b above does or does not have a significant safety effect and therefore does or does not require approved written procedures shall require the concurrence of

   1. the Nuclear Reactor Facility Manager, and

   2. at least one other member of the Reactor Safeguards Committee, to be selected for relevant expertise by the Nuclear Reactor Facility Manager. If the Manager and the Committee member disagree, or if in their judgment the case warrants it, the proposal shall be submitted to the full Committee, and

   3. the University Radiation Safety Officer, or his/her deputy, who may withhold agreement until approval by the University Radiation Safety Committee is obtained.

   The Rector Safeguards Committee shall subsequently review determinations that written procedures are not required. The time at which determinations are made, and the review and approval of written procedures, if required, are carried out, shall be a reasonable interval before the proposed activity is to be undertaken.

d) Determination that a proposed change in the facility does or does not have a significant safety effect and therefore does or does not require review and approval by the full Reactor Safeguards Committee shall be made in the same manner as for proposed activities under (c) above.

6.4 Review of Proposals for Experiments

a) All proposals for new experiments involving the reactor shall be reviewed with respect to safety in accordance with the procedures in (b) below and on the basis of criteria in (c) below.

b) Procedures:

1. Proposed reactor operations by an experimenter are reviewed by the Reactor Supervisor, who may determine that the operation is described by a previously approved EXPERIMENT or procedure. If the Reactor Supervisor determines that the proposed operation has not been approved by the Reactor Safeguards Committee, the experimenter shall describe the proposed EXPERIMENT in written form in sufficient detail for consideration of safety aspects. If potentially hazardous operations are involved, proposed procedures and safety measures including protective and monitoring equipment shall be described.

2. If the experimenter is a student, approval by his/her research supervisor is required. If the experimenter is a staff or faculty member, his/her own signature is sufficient.

3. The proposal is then to be submitted to the Reactor Safeguards Committee for consideration and approval. The Committee may find that the experiment, or portions thereof, may only be performed in the presence of the University Radiation Safety Officer or Deputy thereto.

4. The scope of the EXPERIMENT and the procedures and safety measures as described in the approved proposal, Including any amendments or conditions added by those reviewing and approving it, shall be binding on the experimenter and the OPERATING personnel. Minor deviations shall be allowed only in the manner described in Section 6 above. Recorded affirmative votes on proposed new or revised experiments or procedures must indicated that the Committee determines that the proposed actions do not involve changes in the facility as designed, changes in Technical Specifications, changes that under the guidance of 10 CFR 50.59 require prior approval of the NRC, and could be taken without endangering the health and safety of workers or the public or constituting a significant hazard to the integrity of the reactor core.

5. Transmission to the Reactor Supervisor for scheduling.

c) Criteria that shall be met before approval can be granted shall include:

1. The EXPERIMENT must meet the applicable Limiting Conditions for Operation and Design Description specifications.

2. It must not involve violation of any condition of the facility license or of Federal, State, University, or Facility regulations and procedures.

3. The conduct of tests or experiments not described in the safety analysis report (as updated) must be evaluated in accordance with 10 CFR 50.59 to determine if the test

or experiment can be accomplished without obtaining prior NRC approval via license amendment pursuant to 10 CFR Sec. 50.90.

4. In the safety review the basic criterion is that there shall be no hazard to the reactor, personnel or public. The review SHALL determine that there is reasonable assurance that the experiment can be performed with no significant risk to the safety of the reactor, personnel or the public.

6.5 Emergency Plan and Procedures

An emergency plan shall be established and followed in accordance with NRC regulations. The plan shall be reviewed and approved by the Reactor Safeguards Committee prior to its submission to the NRC. In addition, emergency procedures that have been reviewed and approved by the Reactor Safeguards Committee shall be established to cover all foreseeable emergency conditions potentially hazardous to persons within the Laboratory or to the public, including, but not limited to, those involving an uncontrolled reactor excursion or an uncontrolled release of radioactivity.

6.6 Operator Requalification

An operator requalification program shall be established and followed in accordance with NRC regulations.

6.7 Physical Security Plan

Administrative controls for protection of the reactor plant shall be established and followed in accordance with NRC regulations.

6.8 Action To Be Taken In The Event A Safety Limit Is Exceeded

In the event a safety limit is exceeded:

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

b) An immediate report of the occurrence shall be made to the Chair of the Reactor Safeguards Committee, and reports shall be made to the NRC in accordance with Section 6.11 of these specifications.

c) A report shall be made to include an analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to Reactor Safeguards Committee for review, and a suitable similar report submitted to the NRC when authorization to resume operation of the reactor is sought.

6.9 Action To Be Taken In The Event Of A Reportable Occurrence

a) A reportable occurrence is any of the following conditions:

1. any actual safety system setting less conservative than specified in Section 2.2, Limiting Safety System Settings;

2. VIOLATION OF SL, LSSS OR LCO;

NOTES

Violation of an LSSS or LCO occurs through failure to comply with an "Action" statement when "Specification" is not met; failure to comply with the "Specification" is not by itself a violation.

Surveillance Requirements must be met for all equipment/components/conditions to be considered operable.

Failure to perform a surveillance within the required time interval or failure of a surveillance test shall result in the /component/condition being inoperable

3. incidents or conditions that prevented or could have prevented the performance of the intended safety functions of an engineered safety feature or the REACTOR SAFETY SYSTEM;

4. release of fission products from the fuel that cause airborne contamination levels in the reactor bay to exceed 10CFR20 limits for releases to unrestricted areas;

5. an uncontrolled or unanticipated change in reactivity greater than $1.00;

6. an observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy has caused the existence or development of an unsafe condition in connection with the operation of the reactor;

7. an uncontrolled or unanticipated release of radioactivity.

b) In the event of a reportable occurrence, the following actions shall be taken:

1. The reactor shall be shut down at once. The Reactor Supervisor shall be notified and corrective action taken before operations are resumed; the decision to resume shall require approval following the procedures in Section 6.3.

2. A report shall be made to include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to the Reactor Safeguards Committee for review.

3. A report shall be submitted to the NRC in accordance with Section 6.11 of these specifications.

6.10 Plant Operating Records

a) In addition to the requirements of applicable regulations, in 10 CFR 20 and 50, records and logs shall be prepared and retained for a period of at least 5 years for the following items as a minimum.

1. normal plant operation, including power levels;

3. principal maintenance activities;

4. reportable occurrences;

5. equipment and component surveillance activities;

6. experiments performed with the reactor;

7. all emergency reactor scrams, including reasons for emergency shutdowns.

b) The following records shall be maintained for the life of the facility:

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

2. offsite environmental monitoring surveys;

3. fuel inventories and transfers;

4. facility radiation and contamination surveys;

5. radiation exposures for all personnel;

6. updated, corrected, and as-built drawings of the facility.

6.11 Reporting Requirements

All written reports shall be sent within the prescribed interval to the United States Nuclear Regulatory Commission, Washington, D.C., 20555, Attn: Document Control Desk.

In addition to the requirements of applicable regulations, and in no way substituting therefor, reports shall be made to the US. Nuclear Regulatory Commission (NRC) as follows:

a) A report within 24 hours by telephone and fax or electronic mail to the NRC Operations Center and the USNRC Region IV of;

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

2. any violation of a safety limit;

3. any reportable occurrences as defined in Section 6.9 of these specifications.

b) A report within 10 days in writing of:

1. any accidental release of radioactivity above permissible limits in unrestricted areas, whether or not the release resulted in property damage, personal injury or exposure; the written report (and, to the extent possible, the preliminary telephone and

telegraph report) shall describe, analyze, and evaluate safety implications, and outline the corrective measures taken or planned to prevent recurrence of the event;

2. any violation of a safety limit;

3. any reportable occurrence as defined in Section 1.1 of these specifications.

c) A report within 30 days in writing of:

1. any significant variation of a MEASURED VALUE from a corresponding predicted or previously MEASURED VALUE of safety-connected OPERATING characteristics occurring during operation of the reactor;

2. any significant change in the transient or accident analysis as described in the Safety Analysis Report.

3. a change in personnel for the Department of Mechanical and Nuclear Engineering Chair, or a change in reactor manager

d) A report within 60 days after criticality of the reactor in writing to the US Nuclear Regulatory Commission, resulting from a receipt of a new facility license or an amendment to the license authorizing an increase in reactor power level or the installation of a new core, describing the MEASURED VALUE of the OPERATING conditions or characteristics of the reactor under the new conditions.

e) A routine report in writing to the US. Nuclear Regulatory Commission within 60 days after completion of the first calendar year of OPERATING and at intervals not to exceed 12 months, thereafter, providing the following information:

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

2. a tabulation showing the energy generated by the reactor (in megawatt-hours);

3. the number of emergency shutdowns and inadvertent scrams, including the reasons thereof and corrective action, if any, taken;

4. discussion of the major maintenance operations performed during the period, including the effects, if any, on the safe operation of the reactor, and the reasons for any corrective maintenance required;

5. a summary of each change to the facility or procedures, tests, and experiments carried out under the conditions of 10 CFR 50.59;

6. a summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or before the point of such release or discharge;

7. a description of any environmental surveys performed outside the facility;

8. a summary of radiation exposures received by facility personnel and visitors, including the dates and time of significant exposure, and a brief summary of the results of radiation and contamination surveys performed within the facility.