TEXAS A&M UNIVERSITY
NUCLEAR SCIENCE CENTER REACTOR
LICENSE NO. R-83
DOCKET NO. 50-128

SAFETY AND ACCIDENT ANALYSES REPORT FOR LICENSE AMENDMENT NO. 16 AUTHORIZING CHANGES TO TECHNICAL SPECIFICATIONS FOR THE CONVERSION FROM HIGH-ENRICHED URANIUM TO LOW-ENRICHED URANIUM FUEL

REDACTED VERSION

SECURITY RELATED INFORMATION REMOVED

Redacted text and figures blacked out or denoted by brackets

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U.S. Nuclear Regulatory Commission
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Subject: Docket No. 50-128, Facility License R-83
Submittal of Safety and Accident Analyses and Report for License Amendment 16 Authorizing Changes to Technical Specification for the Conversion from HEU Fuel to LEU Fuel at the Texas A&M University Nuclear Science Center Reactor

The Texas A&M University Nuclear Science Center Reactor (NSCR) proposes changes to our Technical Specifications (License R-83) to facilitate the conversion of its reactor from high enriched uranium (HEU) fuel to low enriched uranium (LEU) fuel. This would represent the 16th amendment to the NSCR license. Attached (Attachment I) is a Safety and Analysis Report formatted in accordance with the NRC's SAR Review Plan and Acceptance Criteria "Chapter 18 - High Enriched Uranium to Low Enriched Uranium Conversions". The technical analysis for this report was provided by General Atomics.

If there are any questions or concerns, with this response please contact Jim Remlinger at (979) 845-7551, and/or e-mail him at jaremlinger@tamu.edu .

Respectfully Submitted,

W. D. Reece
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Attachment I

cc: Marvin M. Mendonca - NRC
Theresa Maldonado - Deputy Director TAMU/TEES w/o Attachment

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ATTACHMENT I

SUBMITTAL REPORT TO COVER SAFETY AND ACCIDENT ANALYSES FOR Texas A&M UNIVERSITY (TAMU) CONVERSION FROM HEU TO LEU FUEL

SUBMITTAL REPORT
Safety and Accident Analyses And Report
Texas A&M University (TAMU)
Conversion from HEU to LEU Fuel

Submitted by

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

Technical Analysis Provided by

General Atomics

December 2005

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Summary

This report contains the results of design, safety, and accident analyses performed for the Texas A&M University by General Atomics for conversion of the Texas A&M University 1 MW TRIGA research reactor from the use of highly-enriched uranium (FLIP HEU) fuel to low-enriched uranium (LEU 30/20) fuel. This study investigates the performance and safety margins of the proposed LEU core under nominal and accident conditions. It identifies any suggested changes to the TAMU Final Safety Analysis Report (FSAR) and Technical Specifications (Ref. 1).

1. General Description of the Facility

1.1 Introduction

This section provides an overview of the changes to the physical, nuclear and operational characteristics of the facility required by the HEU to LEU conversion of the TAMU fuel.

The HEU to LEU conversion only requires the installation of TRIGA LEU (30/20) fuel, and does not require any changes to the remainder of the facility.

The proposed LEU critical core contains 90 fuel elements including four control rod fuel followers. Based on this core configuration, it is concluded: i) the shutdown margin meets the required limit; ii) the reactivity coefficients remain essentially the same as for the FLIP HEU core; iii) fuel integrity is maintained under all operating conditions; and iv) dose to public from the Maximum Hypothetical Accident (MHA) and Fuel Handling Accident (DBA) remain essentially unchanged from the HEU core and below the maximum permissible limits.

The HEU to LEU conversion requires changes to the Technical Specifications as discussed in Section 14.

1.2 Summary and Conclusions of Principal Safety Considerations

The LEU core meets all the safety requirements as specified in FSAR.

1.3 Summary of Reactor Facility Changes

The LEU (30/20) 4-rod fuel cluster has the same design as the present FLIP HEU 4-rod cluster. The clad for the LEU (30/20) fuel and the construction of the 4-rod cluster is identical with that for use with the FLIP HEU fuel. This LEU (30/20) fuel has been approved by the Nuclear Regulatory Commission (NRC) for use in non-power reactors (Ref. 2).

1.4 Summary of Operating License, Technical Specifications, and Procedural Changes

In addition to the updated LEU fuel parameters, the maximum pulsed reactivity in the Technical Specifications may change (see Section 14).

1.5 Comparison with Similar Facilities Already Converted

No 4-rod TRIGA reactor using FLIP HEU fuel has been previously converted.

2. Site Characteristics

The HEU to LEU conversion does not impact the site characteristics.

3. Design of Structures, Systems, and Components

The HEU to LEU conversion does not require any changes to the design of structure, systems, and components.

4. Reactor Description

4.1 Reactor Facility

The HEU to LEU conversion of the TAMU facility requires only changes in the fuel type. All the following aspects of the facility remain unchanged:

- Control Rods
- Neutron Reflector
- Neutron Source and Holder
- Reactor Tank and Biological Shielding
- Core Support Structure
- Functional Design of the Reactivity Control System

The HEU and LEU cores contain different type enrichment and fuel loading per rod. Note that the proposed LEU core configuration may differ when the actual fuel loading is performed.

Table 4-1 provides a comparison of the key design safety features of the HEU and LEU fuel bundles and a comparison of the key reactor and safety parameters that were calculated for each core. The results show that the TAMU reactor facility can be operated as safely with the new LEU fuel as with the present HEU fuel bundles.

The evaluation of the Puerto Rico Nuclear Center (PRNC) TRIGA HEU (FLIP) core provides an opportunity to benchmark the computational technique to be used for evaluating the TRIGA LEU (30/20) fuel in the Texas A&M University (TAMU) TRIGA 4-rod cluster core. It also provides the information required for the performance comparison of the fresh HEU (FLIP) and fresh LEU (30/20) fuel for the HEU-to-LEU Conversion Report.

The computations produced operational parameters to be compared with the actual measured values from the commissioning tests conducted by GA for the PRNC TRIGA core loaded with FLIP (HEU) fuel. The experimentally measured parameters included the 1/M approach to critical tests; the reactivity for the fully loaded core (95 fuel elements plus 1 stainless steel dummy with full water reflection); the control rod calibration values; the reactivity loss and peak fuel temperatures as a function of reactor

Table 4-1.

HEU and LEU Design Data, Core Physics, and Safety Parameters for Conversion of the TAMU Research Reactor. (Note: The HEU data are taken from the Puerto Rico Nuclear Center FLIP and TAMU Core.)

DESIGN DATA	**HEU (FLIP) PRNC Core**	**LEU (30/20) NSCR Core**
Number of Fuel Rods	95	90
Fuel Type	UZrH-Er	UZrH-Er
Uranium Enrichment, %	70	19.75
Erbium, wt %	1.48	0.90
Zirconium Rod Outer Diameter, mm	6.35	6.35
Fuel Meat Outer Diameter, mm	34.823	34.823
Fuel Meat Length, mm	381	381
Clad Thickness, mm	0.508	0.508
Clad Material	304 SS	304 SS
REACTOR PARAMETERS		
Reactor Steady State Operation		
Routine Power, MW	1.4	1.0
Testing, Maintenance, MW	-	1.3
Maximum Fuel Temperature at 1 MW, °C	362	373
Maximum Pulsing Operation with $\hat{T}$ Limited to 830°C, MW	2400	1325
Cold Clean Excess Reactivity, Δk/kβ ($)	6.26	7.73
Prompt Negative Temperature Coefficient of Reactivity, -Δk/k-°C 23-700°C	0.47 to 1.79×10^{-4}	0.53 to 1.31×10^{-4}
Coolant Void Coefficient, Δk/k % void	1.07×10^{-3}	1.30×10^{-3}
Maximum Rod Power at 1 MW, kW/element	10.5	17.4
Average Rod Power at 1 MW, kW/element	16.3	11.1
Maximum Rod Power at 1.4 MW, kW/element	22.8	-
Average Rod Power at 1.4 MW, kW/element	14.7	-
Maximum Rod Power at DNB = 1.0, kW/element	44	42
DNB Ratio at Operating Power	1.93	2.42
Prompt Neutron Lifetime, μsec	22.5	26.3
Effective Delayed Neutron Fraction	0.0071	0.0070
Shutdown Margin, Δk/kβ ($)		
(with most reactive rod stuck out)	-2.12	
(with most reactive rod and Reg. Rod stuck out)		-1.15

Table 4-1. (cont.)

SAFETY LIMITS	HEU (FLIP) PRNC Core	LEU (30/20) NSCR Core
Limiting Safety System Setting		
Reactor Power, MW	2.2	1.25
Measured Fuel Temperature, °C	600	525
Minimum DNB ratio at 1.0 MW	-	2.42
Minimum DNB ratio at 1.4 MW	1.93	-
Maximum Positive Pulsed Reactivity Insertion to reach $\hat{T}$=830°C, $\Delta k/k\beta$ ($)	2.05	2.10
Peak Pulsed Power, MW	2400	1325
Peak Pulsed Fuel Temperature, °C	830	830

power; and pulsing performance including peak power, peak fuel temperature, and energy production all as a function of prompt reactivity insertion. In addition, the computation produced a plot of the prompt, negative temperature coefficient of reactivity ($\Delta k/k$-°C) versus reactor fuel temperature that can be compared with the value in the SAR (1972) for the same parameter. (Figure 4-14)

The steady state parameters for the TAMU LEU (30/20) core were calculated using the same computational procedures adapted to the TAMU 4-rod configuration.

Two basically similar reactor geometries were evaluated in this report: the AMF reactor in PRNC and the GE reactor at TAMU. Both reactors had been converted to operate with TRIGA 4-rod fuel clusters. The FLIP fuel was originally installed and operated in the PRNC reactor before this same fuel was then transferred to the converted GE reactor at TAMU.

4.2 Reactor Core

This chapter provides a detailed description of the components and structures in the reactor core. Comparisons between the HEU and LEU cores are presented when the conversion requires changes in some characteristics.

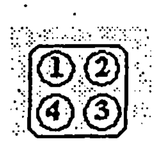
The TAMU conversion reactor is a primarily homogeneous, light water moderated and cooled, pool-type reactor fueled with a core containing either a full core of HEU FLIP (70% enriched $UZrH_x$) 4-rod TRIGA fuel or a mixture of HEU FLIP fuel and Standard TRIGA fuel (20% enriched) in a 4-rod cluster configuration. The fuel clusters are supported by a 5-inch thick aluminum grid plate. Figure 4.1 shows the TAMU movable reactor core against the graphite coupler box in the Stall location. At TAMU, the two most frequently used reactor locations are: (1) against the coupler box or (2) 50 to 75 cm removed from the coupler box with light water intervening. A typical TAMU FLIP core configuration is shown in Figure 4.2 that contains 86 fuel elements, 4 fuel-followed control rods, a water-followed regulating rod, and an air-followed transient rod. Graphite reflector blocks are used in-core for reflection in addition to the graphite in the coupler box and thermal column.

The reactor is supported from the top of the pool wall and is moveable along the central, long axis of the reactor pool. The reactor is controlled by four fuel-followed control rods plus one water-followed regulating rod and one air-followed transient rod. All six control rods are supported from the bridge structure at the top of the reactor pool wall.

Figure 4.1: Stall Beam Port Installation at TAMU with Graphite Coupler Box and Thermal Column Extension

Figure 4.2: Core Configuration

Note: Each core location is designated by cluster location (e.g., 5C) and by one of four positions in the cluster. See following example:



4.2.1 Fuel Elements

The HEU and LEU fuel elements have similar overall designs, i.e., they are both TRIGA fuel rods mounted in 4-rod, 3-rod, or 2-rod clusters.

Fuel Description

The geometries, materials, and fissile loadings of the current HEU 4-rod clusters and the replacement LEU (30/20) 4-rod fuel clusters are summarized in Table 4-2.

TABLE 4-2. Descriptions of the HEU and LEU Fuel Elements

Design Data	PRNC (HEU)	TAMU (LEU)
Number of Fuel Elements		
Critical Test	62	62
Full Load	95	90
Fuel Type	U-ZrH (FLIP)	U-ZrH (30/20)
Enrichment, %	70.00	19.75
Uranium Density,		
g/cm^3	0.50	2.14
Wt-%	8.42	30
Number of Fuel Elements per Bundle	4	4
^{235}U per Fuel Bundle, g	494.6	597.26
^{235}U per Fuel Element, g	123.65	149.32
^{166}Er per Fuel Element, g	10.27	7.46
^{167}Er per Fuel Element, g	7.09	5.15
Erbium, wt-%	1.48	0.90
Zirconium Rod Outer Diameter, mm	6.35	6.35
Fuel Meat Outer Diameter, mm	34.823	34.823
Fuel Meat Length, mm	381	381
Cladding Thickness, mm	0.508	0.508
Cladding Material	304 SS	304 SS

Figure 4.3 shows a 4-rod fuel cluster; Figure 4.4 shows a 3-rod cluster modified to accommodate a fuel-followed control rod. Figure 4.5 shows the nominal cluster and fuel rod spacing for the 4-rod clusters. This figure shows more water between fuel clusters in one direction compared to another. It is useful to note that at PRNC, the extra water lay in sheets parallel to the face of the thermal column. At TAMU, the extra water lies in sheets perpendicular to the face of the thermal column.

The heat removal system for the TAMU LEU core remains unchanged from that used with the HEU (FLIP) core. The primary cooling system circulates heated water from the reactor tank through the heat exchanger and returns the cooled water to the reactor tank. The secondary cooling system circulates water from the heat exchanger to the cooling tower. The core itself is cooled by natural convection.

The LEU (30/20) TRIGA fuel to be installed in the TAMU core consists of 4-rod, and modified 3-rod fuel clusters exactly the same as for the present FLIP core. Figures 4.6(a) and 4.6(b) show detailed illustrations of the fuel element in the 4-rod cluster and the instrumented (integrated thermocouple) fuel element. The FLIP (HEU) and 30/20 (LEU) fuel differ only in the alloy in the fuel sections of these illustrations; the dimensions are the same for these two types of TRIGA fuel.

The above description applies to the TAMU facility. However, the specific details used as input for the calculations are very similar to those for the PRNC reactor. Both used a heavy aluminum grid plate to support the 4-rod fuel clusters and both were supported by a moveable bridge mounted at the top of the reactor pool wall. The 4-rod fuel cluster was suitable for both reactors. In fact, the same FLIP (HEU) 4-rod clusters were transferred from PRNC and installed in the TAMU facility. Apart from small differences, the major geometrical difference in these facilities lies in the fact that extra holes were drilled in the TAMU grid plate to accommodate fuel-followed or air-followed control rods rather than the water-followed control rods used for all control rods in the PRNC reactor core.

Figure 4.3: Four Element Bundle

Figure 4.4: Three-Element Bundle with Fuel-Followed Control-Rod

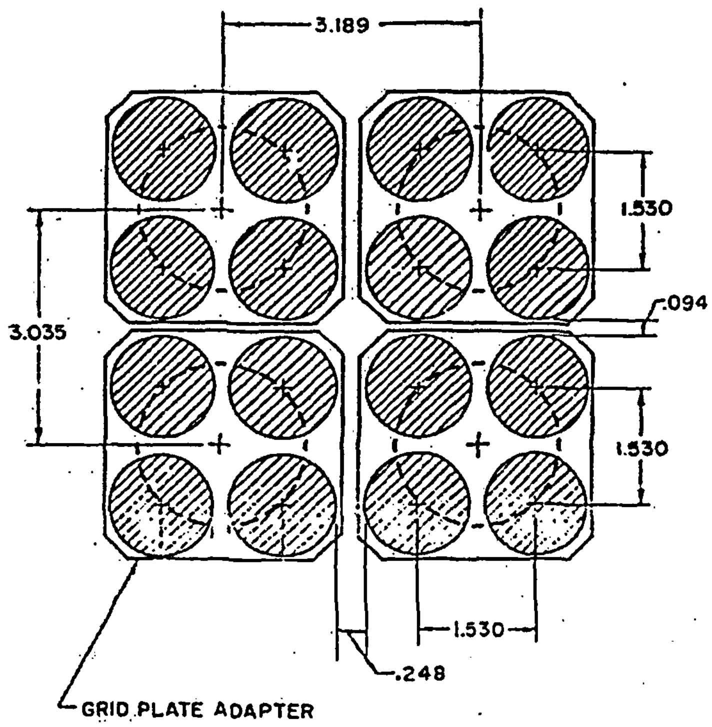


Figure 4.5: Nominal Fuel Rod Spacing in the TAMU Core

Figure 4.6(a): Detailed Drawing of Fuel Rod

Figure 4.6(b): Integrated Filament Thermocouple Fuel Rod

4.2.2 Control Rods

The TAMU reactivity control system consists of six standard TRIGA control rods: four fuel follower control rods (Figure. 4-4), one water-followed regulating rod, and one air-followed transient rod both located as shown in Figure 4.2.

4.2.3 Neutron Reflector

There is no plan to change the reflector during the HEU to LEU conversion. The TAMU reactor uses nuclear-grade graphite blocks (as well as water) in the core as reflector and also in the coupler box. Figures 4-1 (coupler box) and 4-2 (in-core) show different reflector regions as modeled.

4.2.4 Neutron Source and Holder

The proposed HEU to LEU conversion of the TAMU core does not require any changes in the existing neutron source location.

4.2.5 In-Core Experimental Facilities

There are no in-core experiments in the TAMU reactor.

4.2.6 Reactor Materials

The TAMU conversion to LEU requires a change in the fuel rod composition but no change in the fuel clad. Table 4-3 presents the material composition used in the computational models.

TABLE 4-3.

Material Composition used in the MCNP Models.

Material	Nuclide	Nuc. Den. (atoms/b-cm)	Physical Density (g/cc)
SS 304 (clad)	Cr-50	0.000778	7.98
	Cr-52	0.015003	
	Cr-53	0.001701	
	Fe-56	0.056730	
	Ni-58	0.007939	
	Mn-55	0.001697	
Graphite (TC)	C		1.7
Graphite (reflector in fuel)	C		1.75
Zirconium (rod)	Zr		6.51
6061 Al (grid plate and control rod clad)	Al-27	0.058693	
	Fe-56	0.000502	
90% B4C (control rod)	B-10	0.020950	
	B-11	0.084310	
	C	0.026320	
Boral (35wt% B4C 65 wt% Al) (detector channel)	B-10	0.06031	2.64
	B-11	0.24489	
	C	0.08725	
	Al-27	0.63581	
Al+Water Mix 1 (2" lower cluster adapter)	H	0.028748	
	O	0.014374	
	AL-27	0.033455	
	FE-56	0.000286	
Al+Water Mix 2 (5" grid plate)	H	0.030788	
	O	0.015394	
	AL-27	0.031663	
	FE-56	0.000271	
Water			1.0
Air			0.000123

4.3 Reactor Pool and Biological Shielding

The proposed HEU to LEU conversion of the TAMU core does not require any changes in the reactor pool or biological shielding.

4.4 Core Support Structure

The proposed HEU to LEU conversion of the TAMU core does not require any changes in the core support structure.

4.5 Dynamic Design

4.5.1 Calculation Models; Nuclear Analysis Codes

Two-dimensional and three-dimensional calculations are performed using both diffusion theory and Monte Carlo codes. In general, multi-group diffusion theory is used for design calculations since it gives adequate results for systems of this kind and its multi-group fluxes and cross sections are easily utilized in nuclide burnup calculations. The Monte Carlo calculations are used to evaluate the facilities around the core and also to compute the worth of core components and different core configurations.

The diffusion theory code is DIF3D (Ref. 3,4), a multi-group code which solves the neutron diffusion equations with arbitrary group scattering.

The Monte Carlo code is MCNP5 (Ref. 5) that contains its own cross section library.

MCNP5 Monte Carlo Code

This section discusses the MCNP5 models developed for these analyses and the benchmark calculations for the HEU core, and determines a reference critical LEU core.

Reactor calculations were performed in three dimensions for the initial criticality and the full core loading of the PRNC (HEU) core and the TAMU (LEU) core using the MCNP 5, Version 1.3, continuous energy Monte Carlo code. The nuclide cross sections were based on ENDF/B VI data included in the MCNP 5 data libraries.

The PRNC (FLIP) and TAMU (30/20) fuel meat nuclide densities used in the two models are shown in Table 4-4.

The other materials beside the fuel used in the PRNC and TAMU MCNP models are listed in Table 4-3.

TABLE 4-4
The nuclide densities of the fuel meat used in the PRNC (FLIP) and TAMU (30/20, LEU) MCNP Models.

		FLIP		LEU 30/20	
Nuclide	Atomic Mass	Nuc. Den. (atoms/b-cm)	Mass (g)	Nuc. Den. (atoms/b-cm)	Mass (g)
H	1.0079	0.05459307	32.05	0.04915763	28.86
C	12.011	0.00149606	10.47	0.00178701	12.50
Zr	91.224	0.03529874	1875.81	0.03227955	1715.37
Er-166	165.93	0.00010624	10.27	0.00007717	7.46
Er-167	166.932	0.00007295	7.09	0.00005299	5.15
U-234	234.041	0.00000659	0.90	0.00000715	0.97
U-235	235.0439	0.00090116	123.39	0.00108821	149.00
U-236	236.0456	0.00000423	0.58	0.00000627	0.86
U-238	238.0508	0.00037065	51.40	0.00432194	599.33
Hf	178.49	2.11792E-06	0.22	1.93677E-06	0.20
Total		0.09285181	2111.96	0.08877792	2519.51

Geometrical Models

Each fuel rod was explicitly modeled such that 15 cells and 6 surface cards were constructed to properly represent one fuel rod. A total of 930 cells and 372 surface cards were made for the PRNC 62 fuel rods, and a similar number of cells and surface cards for the TAMU 62 fuel rods in the approach-to-critical core.

For the full core, the model for PRNC with 95 HEU fuel rods has a total of 1425 cells and 570 surface cards. The full core TAMU core model has 90 fuel rods with a total of 1350 cells and 540 surface cards.

PRNC Reactor Model, Approach-to-Critical

A detailed MCNP model of the reactor was made including 62 FLIP fuel rods, 5 water-followed control rods, 1 rabbit system, 2 inches thick lower cluster adapter and the 5 inches thick grid plate below the fuel rods. This number of FLIP fuel elements was chosen to match the number and geometry used by GA in the approach-to-critical tests at PRNC in 1972. The reactor went critical on 59 fuel rods.

Figures 4.7(a) and 4.7(b) are the XY and XZ plots of the MCNP model of the PRNC cold critical case. In this case the whole core model was infinite water reflected since the initial approach to critical measurements during commissioning used this configuration.

TAMU Reactor Model, Approach-to-Critical

A detailed MCNP model of the TAMU reactor was also made including 58 LEU (30/20) fuel rods, 4 fuel-followed rods, 1 void followed transient rod, 1 water-followed regulating rod, 11 graphite blocks around the core and 4 detector assemblies. This number of fuel rods was chosen to be close to the 59 fuel rods in the critical configuration at PRNC.

For the initial critical case the TAMU core was modeled to be close to the coupler box, 0.5 inch of water gap between the core and the coupler box (Ref. 6). This configuration has been chosen since this is the most reactive arrangement. Figures 4.8(a) and 4.8(b) are the XY and XZ plots of the MCNP model of the TAMU cold critical case.

PRNC Full Core

The full core for the PRNC reactor had 95 fuel rods and was fully water reflected. A detailed MCNP model included 95 fuel rods, 5 water-followed control rods (3 shim/safety rods, 1 transient rod, and 1 regulating rod), 1 rabbit system, 2 inches thick lower cluster adapter, and the 5 inches thick aluminum grid plate below the fuel rods.

Figures 4.9(a) and 4.9(b) are the XY and XZ plots of the MCNP model of the full unrodded core PRNC reactor with infinite water reflector. Figures 4.10(a) and 4.10(b) are the XY and XZ plots of the MCNP model of the full core PRNC reactor with all control rods inserted. The reactor has full water reflector.

TAMU Full Core

The full core for the TAMU reactor has 90 fuel rods and will be adjacent to the coupler box (graphite). A detailed MCNP model includes 86 fuel rods, 4 fuel-followed control rods, 1 void followed transient rod, 1 water-followed regulating rod, 11 graphite blocks around the core, a coupler box (graphite), 2 inches thick lower cluster adapter, and the 5 inches thick aluminum grid plate below the fuel rods.

Figures 4.11(a) and 4.11(b) are the XY and XZ plots of the MCNP model of the full unrodded core TAMU reactor.

Figures 4.12(a) and 4.12(b) are the XY and XZ plots of the MCNP model of the full core TAMU reactor with all fuel-followed rods inserted.

Figure 4.7(a) PRNC Radial Model for Monte Carlo (MCNP5) Transport Calculations – 62 Fuel Rods

Figure 4.7(b) PRNC Axial Model for Monte Carlo (MCNP5) Transport Calculations – 62 Fuel Rods

Figure 4.8(a) TAMU Radial Model for Monte Carlo (MCNP5) Transport Calculations - 62 Fuel Rods

Figure 4.8(b) TAMU Axial Model for Monte Carlo (MCNP5) Transport Calculations - 62 Fuel Rods

Figure 4.9(a) PRNC Radial Model Monte Carlo (MCNP5) Transport Calculations-
95 Fuel Rods, Unrodded

Figure 4.9(b) PRNC Axial Model Monte Carlo (MCNP5) Transport Calculations-
95 Fuel Rods, Unrodded

Figure 4.10(a) PRNC Radial Model Monte Carlo (MCNP5) Transport Calculations-
95 Fuel Rods, All Control Rods Inserted

Figure 4.10(b) PRNC Axial Model Monte Carlo (MCNP5) Transport Calculations-
95 Fuel Rods, All Control Rods Inserted

Figure 4.11(a) TAMU Radial Model Monte Carlo (MCNP5) Transport Calculations-
90 Fuel Rods, Unrodded

Figure 4.11(b) TAMU Axial Model Monte Carlo (MCNP5) Transport Calculations-
90 Fuel Rods, Unrodded

**Figure 4.12(a) TAMU Radial Model Monte Carlo (MCNP5) Transport Calculations-
90 Fuel Rods, All Control Rods Inserted**

**Figure 4.12(b) TAMU Radial Model Monte Carlo (MCNP5) Transport Calculations-
90 Fuel Rods, All Control Rods Inserted**

Benchmark of FLIP HEU Core

The evaluation of the Puerto Rico Nuclear Center (PRNC) TRIGA HEU (FLIP) core provided an opportunity to benchmark the computational techniques to be used for evaluating the TRIGA LEU (30/20) fuel in the Texas A&M University (TAMU) TRIGA 4-rod cluster core. It also provides the information required for the performance comparison of the fresh HEU (FLIP) and fresh LEU (30/20) fuel for the HEU-to-LEU Conversion Report.

Approach-to-Critical – PRNC

The calculated k_{eff} with one sigma uncertainty value for the PRNC core with 62 FLIP (HEU) fuel elements is:

$$k_{eff} = 1.00097 \pm 0.00018$$

This computed value is consistent with just critical ($k_{eff} = 1.0$) with 62 fuel elements with infinite water reflector. The present value gives a slight excess reactivity of about $0.14.

The initial approach to critical conducted by GA in 1972 using the 1/M approach indicated that the PRNC reactor went critical with 59 fuel elements in the same infinite water reflector configuration; the 62 fuel elements in this configuration gave a measured excess reactivity of $0.78. Thus, the MCNP code predicts that three (3) more fuel elements are required for initial criticality, an error of about 5%.

The small difference (5%) between calculated and measured just critical fuel loading may possibly be due to a slightly different actual U-235 and Erbium content in the PRNC core. Impurities in actual materials could also contribute to the reactivity difference. The values used as input for the MCNP code were the best values available from the fuel manufacturing records.

Full Unrodded Core Loading – PRNC

The full core loading in the PRNC water reflected core contained 95 fuel rods and one in-core stainless steel dummy. The MCNP calculation gave an unrodded k_{eff} value with one sigma uncertainty:

$$k_{eff} = 1.04649 \pm 0.00017$$

This is equivalent to a reactivity of $6.26. The experimentally determined measured value was $7.12.

The MCNP 5 code value for full core reactivity is about $0.86 less than actually measured for the PRNC core. This is consistent with the prediction of a slightly larger (3 fuel elements) critical mass compared to the actual measurement. This difference may be due to a slightly different actual U-235 and Erbium content in the PRNC core and/or also impurities in the actual materials.

Full Core Loading, All Control Rods Inserted – PRNC

The full core loading in the PRNC water reflected core contained 95 fuel rods and one in-core stainless steel dummy. The MCNP calculation with all control rods inserted gave a k_{eff} value with one sigma uncertainty:

$$k_{eff} = 0.96079 \pm 0.00018$$

This is equivalent to a reactivity shutdown of -\$5.75. The five control rods have a calculated worth of \$12.00. It may be useful to note that the total worth of the experimentally determined individual five control rods was \$12.07.

4.5.2 Critical Core Configuration; Excess Reactivity

For comparison with the proposed LEU (30/20) core, the critical core configuration was evaluated for the clean FLIP HEU fuel in the PRNC reactor. The number of FLIP HEU fuel rods was 95, including 4 fuel follower control rods.

The proposed core loading for the TAMU LEU core will be 90, including 4 fuel follower control rods.

Full Unrodded Core Loading – PRNC

The full core loading in the PRNC water reflected core contained 95 fuel rods and one in-core stainless steel dummy. The MCNP calculation gave an unrodded k_{eff} value with one sigma uncertainty:

$$k_{eff} = 1.04649 \pm 0.00017$$

This is equivalent to a reactivity of \$6.26. The experimentally determined measured value was \$7.12.

Full, Unrodded Core Loading – TAMU

The full core loading in the TAMU core (against the coupler box (graphite)) contains 90 fuel elements including four control rod fuel followers. The MCNP calculation gives an unrodded k_{eff} value with one sigma uncertainty of:

$$k_{eff} = 1.05722 \pm 0.00018$$

This corresponds to a core reactivity of \$7.73.

With the core assembly moved a distance of about 250 mm from the coupler box (graphite), the embedded k_{eff} value with one sigma uncertainty is:

$$k_{eff} = 1.04553 \pm 0.00017$$

This corresponds to a core reactivity of \$6.22, a loss of \$1.51 with water reflector substituted for the coupler box (graphite).

4.5.3 Worth of Control Rods

Full Core Loading, All Control Rods Inserted – PRNC

The full core loading in the PRNC water reflected core contained 95 fuel rods and one in-core stainless steel dummy. The MCNP calculation with all control rods inserted gave a k_{eff} value with one sigma uncertainty:

$$k_{eff} = 0.96079 \pm 0.00018$$

This is equivalent to a reactivity shutdown of -\$5.75. The five control rods have a calculated worth of \$12.00. It may be useful to note that the total worth of the experimentally determined individual five control rods was \$12.07.

Full Core Loading, All Control Rods Inserted – TAMU

The full core loading in the TAMU reactor contains 90 fuel rods including 4 fuel follower control rods. The reactor core is positioned against the coupler box (graphite). The MCNP calculation with all control rods inserted gives a k_{eff} value with one sigma uncertainty:

$$k_{eff} = 0.94314 \pm 0.00017$$

This is equivalent to reactivity shutdown of -\$8.61. The six control rods have a combined reactivity worth of \$16.34.

4.5.4 Shutdown Margin for HEU and LEU Cores

Shutdown Margin, PRNC Core

As stated in the applicable Technical Specifications, the reactor shall not be operated unless the shutdown margin provided by the control rods is greater than \$0.25 with:

a) The highest worth non-secured experiment in its most reactive state,
b) The highest worth control rod fully withdrawn, and
c) The reactor in the cold condition without xenon.

The MCNP code has been used to evaluate the individual worth of the five control rods. The maximum worth control rod thus determined was the same rod identified from experimental control rod calibration curves as the most reactive.

The MCNP 5 code with appropriate input files has been used to calculate the shutdown reactivity with the most reactive rod stuck out and all other rods in the core. The k_{eff} with one sigma uncertainty for the shutdown core was:

$$k_{eff} = 0.98599 \pm 0.00018$$

This corresponds to a reactivity of -\$2.00. This is considerably more negative than -\$0.25. It agrees well with the -\$2.12 determined from the calibrated control rod reactivity values measured during startup.

Shutdown Margin, TAMU Core

As stated in the applicable Technical Specifications, the reactor shall not be operated unless the shutdown margin provided by control rods is greater than \$0.25 with:

a) The highest worth control rod and the regulating rod (if not scrammable) fully withdrawn, and
b) The highest worth control rod and the regulating rod (if not scrammable) fully withdrawn, and
c) The reactor in the cold condition without xenon.

The MCNP 5 code was run for the case with 90 fuel elements and the most reactive rod plus the non-scrammable regulating rod up, with three control rods with fuel followers and the transient rod inserted in the core. The MCNP results gave a k_{eff} with one sigma uncertainty of:

$$k_{eff} = 0.99198 \pm 0.00017$$

This corresponds to a reactivity of -\$1.15. This is clearly more negative than -\$0.25.

4.5.5 Additional Core Physics Parameters for HEU and LEU Cores

Effective Delayed Neutron Fraction, βeff, for PRNC Core

The effective delayed neutron fraction, β_{eff}, was derived from diffusion theory reactor calculations where the reactivity is first computed with the prompt fission spectrum alone and then recalculated with the fission spectrum of both prompt and delayed neutrons. Seventeen groups in the fast energy range and the standard four thermal groups were used for these 3-D calculations in order to represent the two fission spectra in greater detail than is possible with only three fast groups. The results of previous GA SAR calculations indicate that detail in the group structure is of much greater importance than geometric detail. This is not unexpected since the calculation of β_{eff} is directly related to neutron energy effects.

The prompt fission spectrum is obtained from the GGC-5 spectrum calculation (Ref. 7). The delayed fission spectrum is obtained by integrating over the broad energy groups.

The prompt and total fission spectra for each of the broad energy groups in the calculation are given in Table 4-4.

The computed values of K_t and K_p are used in the following expression to obtain β_{eff}:

$$\beta_{eff} = [\, K_t(1+\beta_o)/K_p \,]-1$$

where:

K_t = core reactivity using prompt and delayed fission spectrum,
K_p = core reactivity using prompt fission spectrum,
β_o = intrinsic delayed neutron fraction for U-235 (0.0065)

The 3-D model used 21 total groups and very tight convergence criteria (1.0×10^{-8} on k_{eff}, 1.0×10^{-6} point flux). The cases were run cold (23°C) with fresh fuel.

The result for PRNC FLIP fuel was

$$\beta_{eff} = 0.0071$$

TABLE 4-4.
Fission Spectra Used for Calculation of βeff

Group	Energy Interval (eV)	Prompt χ_P	Delayed χ_t
1	$15.0 \times 10^6 - 10.0 \times 10^6$	0.00104	0.00103
2	$10.0 \times 10^6 - 8.19 \times 10^6$	0.00345	0.00343
3	$8.19 \times 10^6 - 6.70 \times 10^6$	0.00979	0.00973
4	$6.70 \times 10^6 - 5.49 \times 10^6$	0.02149	0.02135
5	$5.49 \times 10^6 - 4.49 \times 10^6$	0.03825	0.03800
6	$4.49 \times 10^6 - 3.68 \times 10^6$	0.05745	0.05708
7	$3.68 \times 10^6 - 3.01 \times 10^6$	0.07525	0.07476
8	$3.01 \times 10^6 - 2.02 \times 10^6$	0.18292	0.18180
9	$2.02 \times 10^6 - 1.50 \times 10^6$	0.14006	0.13943
10	$1.50 \times 10^6 - 1.00 \times 10^6$	0.16006	0.15972
11	$1.00 \times 10^6 - 6.08 \times 10^5$	0.14012	0.14046
12	$6.08 \times 10^5 - 3.02 \times 10^5$	0.10357	0.10463
13	$3.02 \times 10^5 - 1.11 \times 10^5$	0.05119	0.05235
14	$1.11 \times 10^5 - 4.09 \times 10^4$	0.01248	0.01305
15	$4.09 \times 10^4 - 9.12 \times 10^3$	0.00288	0.00317
16	$9.12 \times 10^3 - 4.54 \times 10^2$	0	0
17	$4.54 \times 10^2 - 1.125$	0	0
		1.00000	1.00000

Effective Delayed Neutron Fraction, βeff, for TAMU Core

The effective delayed neutron fraction, β_{eff}, for the TAMU core was calculated exactly as for the PRNC core above but with the TAMU input parameters.

The 3-D model used 21 total groups and very tight convergence (1.0×10^{-8} on keff, 1.0×10^{-6} point flux). The cases were run cold (23°C) with fresh 30/20 fuel.

The result for TRIGA LEU (30/20) fuel was

$$\beta eff = 0.0070$$

Prompt Neutron Life (ℓ) for PRNC Core

The prompt neutron lifetime, ℓ, was computed by the 1/v absorber method where a very small amount of boron is distributed homogeneously throughout the system and the resulting change in reactivity is related to the neutron lifetime. This calculation was done using the 3-D diffusion theory model for the core to allow very tight convergence of the problems. The boron cross sections used in the core were generated over a homogenized core spectrum. Boron cross sections used in all other zones were generated over a water spectrum.

The neutron lifetime, ℓ, is defined as follows:

$$\ell = \Delta k_{eff}/\omega$$

where Δk_{eff} is the change in reactivity due to the addition of boron and ω is related to the boron atom density and,

$$N_B = \omega/\delta_o\, v_o = 6.0205 \times 10^{-7}$$

where N_B = boron density (atoms/b-cm)
ω = integer = 100 (the calculation is insensitive to changes in ω between 1 and 100),
v_o = 2200 m/sec,
δ_o = 755 barns = δ_a^B at 2200 m/sec

As described in the β_{eff} section above, the 3-D model used very tight convergence criteria (1.0×10^{-8} of k_{eff}, 1.0×10^{-6} point flux). The cases were run cold (23°C) with fresh FLIP fuel. The result for prompt neutron lifetime in the unrodded core is the following:

$$\ell = 22.5\ \mu\text{sec}$$

Prompt Neutron Life (ℓ) for TAMU Core

Using the same 1/v absorber method described above for the PRNC HEU core, the prompt neutron life (ℓ) has been evaluated for the TRIGA LEU (30/20) fuel in the TAMU core.

The result for the prompt neutron life (ℓ) in the unrodded TAMU core is the following:

$$\ell = 26.3\ \mu\text{sec}$$

Prompt Negative Temperature Coefficient of Reactivity, α, for PRNC Core

The definition of α, the prompt negative temperature coefficient of reactivity, is given as

$$\alpha = \frac{d\rho}{dT}$$

where ρ = reactivity
= (k-1)/k
T = reactor temperature (°C)

$$\alpha = \frac{1}{k^2}\frac{dk}{dT}$$

To evaluate ($\Delta \rho$) from reactivity as a function of reactor core temperature, the finite differences can be written as follows:

$$\Delta\rho_{1,2} = \frac{k_2 - 1}{k_2} - \frac{k_1 - 1}{k_1}$$

$$= \frac{k_2 - k_1}{k_1 k_2}$$

Thus,

$$\alpha_{1,2} \cong \frac{k_2 - k_1}{k_1 k_2} \text{ x } \frac{1}{\Delta T_{1,2}}$$

The data in Table 4-5 were produced by DIF3D for the listed core temperatures.

Figure 4.13 is a histogram plot of the computed values of α as a function of reactor temperature. Also shown in the same figure is the representations of α given in the 1969 SAR. It can be seen that the presently computed values of α are somewhat smaller for each core temperature.

Table 4-5. Reactivity Change With Temperature, PRNC

Avg. Core Temperature (°C)	k_{eff}	Δk_{eff}	$\frac{k_a - k_b}{k_a k_b}$	$\alpha_{a,b}$
23	1.04171			
		0.00909	0.00845	4.77×10^{-5}
200	1.03262			
		0.00628	0.005926	7.41×10^{-5}
280	1.02364			
		0.01212	0.011643	9.70×10^{-5}
400	1.01422			
		0.042256	0.042256	14.1×10^{-5}
700	0.97254			
		0.04822	0.053641	17.9×10^{-5}
1000	0.92432			

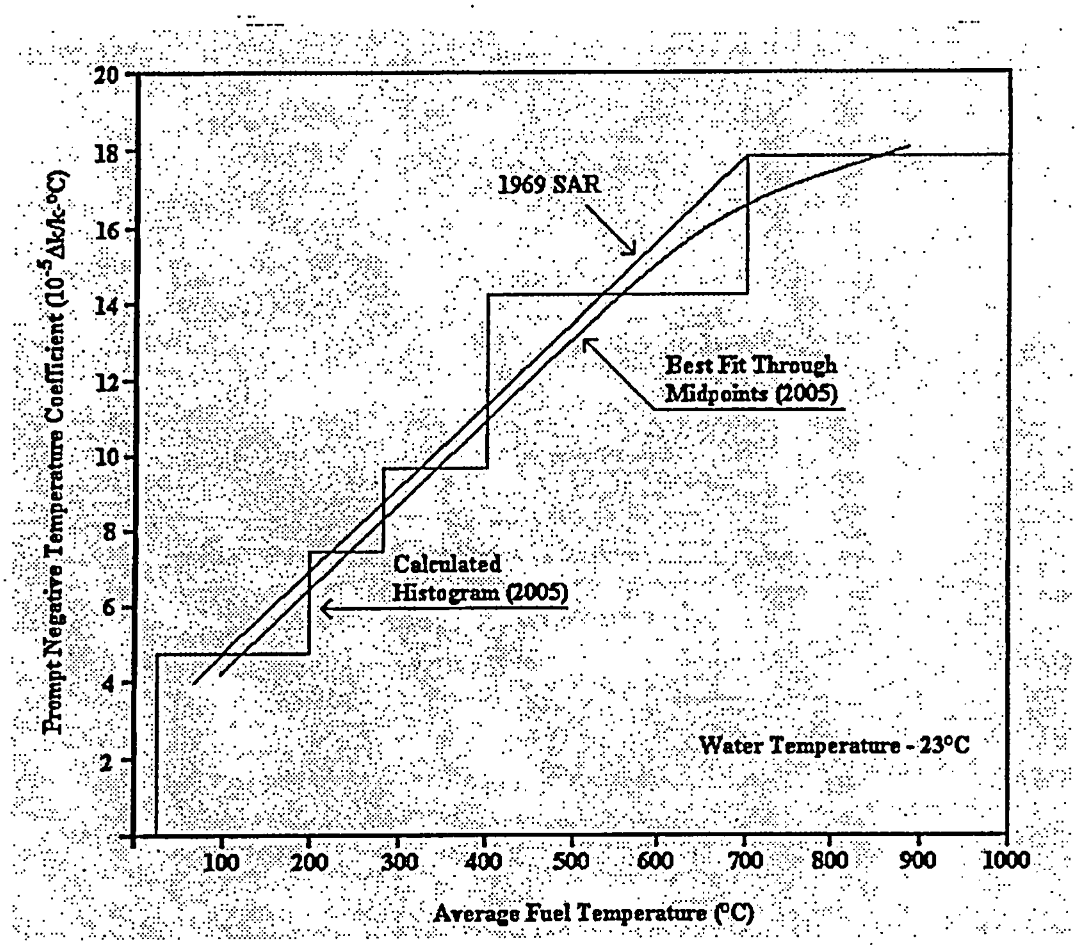


Figure 4.13 Calculated Prompt Negative Temperature Coefficient vs. Fuel Temperature at Beginning of Life for FLIP Fuel. The Straight Line is Included from the 1969 SAR for PRNC for Comparison.

Prompt Negative Temperature Coefficient of Reactivity, α, for TAMU core
Following the procedure set forth in the section for the PRNC core, the computer results from DIF3D for the TAMU reactor are listed in Table 4-6(a) for Beginning-of-Life (BOL) and in Table 4-6(b) for End-of-Life (EOL).

Figure 4.14 is a histogram plot of the computed values for α in Table 4-6 as a function of core temperature for both BOL and EOL..

In Figure 4.14, it can be seen that the prompt negative temperature coefficient (α) for LEU (30/20) fuel has only a modest decrease in values at 2000 MWD burnup (EOL) (e.g., 13.1 x 10-4 to 9.9 x 10-4 Δk/°C at 700-1000°C). As illustrated in the TAMU SAR, the corresponding decrease for FLIP fuel is much larger, as an example, 17 x 10-4 to 4 x 10-4 Δk/°C at 800°C for 2000 MW burnup.

The relatively small change in α for LEU 30/20 fuel is expected due to the 80 wt-% U-238 in LEU (30/20) fuel compared to 30 wt-% U-238 in FLIP fuel. It will be

remembered that it is the 80 wt-% U-238 in standard TRIGA fuel that is responsible for the nearly temperature independent α value of ~10 x 10^{-5} $\Delta k/°C$.

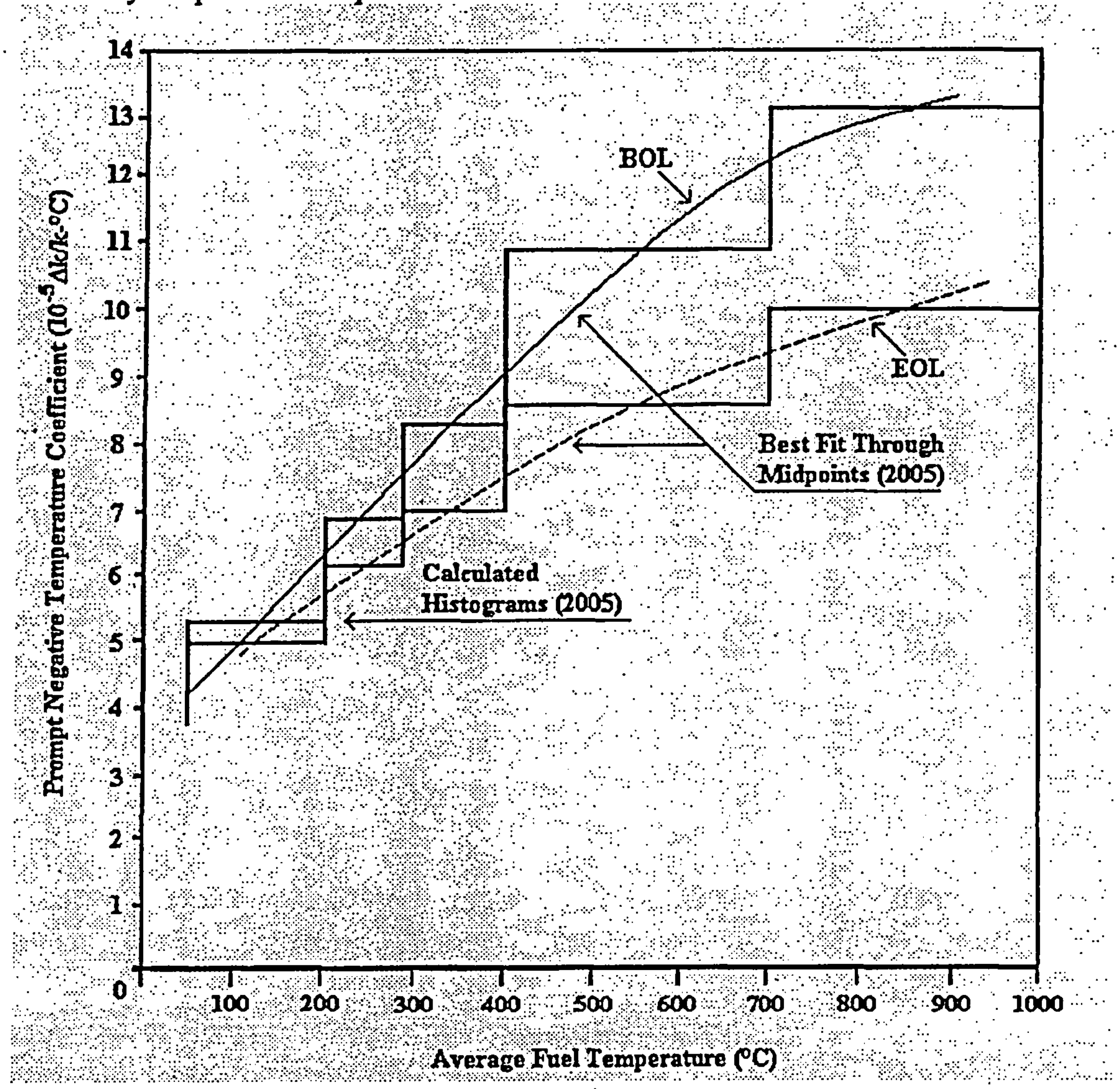


Figure 4.14 Prompt Negative Temperature Coefficient for TRIGA LEU (30/20) Fuel, Beginning of Life (BOL) and End of Life (EOL), TAMU Core

TABLE 4-6(a) Reactivity Change With Temperature, TAMU, BOL				
Avg. Core Temperature (°C)	k_{eff}	Δk_{eff}	$\frac{k_a - k_b}{k_a k_b}$	$\alpha_{a,b}$ ($\Delta k/k$-°C)
23	1.04496			
		0.01015	0.009387	5.303×10^{-5}
200	1.03481			
		0.00585	0.005494	6.87×10^{-5}
280	1.02896			
		0.01039	0.009913	8.26×10^{-5}
400	1.01857			
		0.03271	0.032574	10.86×10^{-5}
700	0.98586			
		0.03673	0.039254	13.1×10^{-5}
1000	0.94913			

TABLE 4-6(b) Reactivity Change With Temperature, TAMU, EOL				
Avg. Core Temperature (°C)	k_{eff}	Δk_{eff}	$\frac{k_a - k_b}{k_a k_b}$	$\alpha_{a,b}$ ($\Delta k/k$-°C)
23	1.01876			
		0.0090	0.008759	4.95×10^{-5}
200	1.00975			
		0.0050	0.004889	6.11×10^{-5}
280	1.00479			
		0.0084	0.008431	7.03×10^{-5}
400	0.99635			
		0.0250	0.02580	8.60×10^{-5}
700	0.97138			
		0.0275	0.029949	9.98×10^{-5}
1000	0.94392			

Void Coefficient – PRNC Core

The "void" coefficient of reactivity is defined for a TRIGA reactor as the negative reactivity per 1% void in the reactor core water. For the PRNC reactor with FLIP fuel, the calculated void coefficient is about 0.107% Δk/k per 1% water void. This void coefficient is not normally considered a safety concern for TRIGA reactors. The reason is the relatively small size of this coefficient and the fact that all TRIGA reactors are significantly undermoderated. Therefore, if a portion of the core water is replaced with a low density material (i.e., steam, gas including air, etc.), a negative reactivity will occur. An example would be a dry, experimental tube introduced into the core [e.g., (B4,1)] with a volume of 242 cc in the 381 cm core fueled height. The calculated loss in core reactivity would be about $0.15.

A safety effect of rapid reactivity insertion to be considered is the effect of accidental flooding of an in-core dry experimental tube such as postulated above. In this case, the

rapid reactivity insertion would be only about \$0.15. The insertion of \$0.15 reactivity is far less than \$1.00 (prompt critical).

The conclusion is that the very small void coefficient of reactivity is not a source of safety concern.

Void Coefficient – TAMU

The void coefficient for the TAMU reactor is 0.130% $\Delta k/k$ per 1% water void. If a dry experimental region (of 281 cc in the 381 cm of fuel height) is inserted near core center (replacing a fuel rod) and is accidentally flooded with water, the prompt gain in reactivity is about \$0.26. This reactivity addition is far less than \$1.00 required for prompt critical.

The conclusion is that the very small void coefficient is not a source of safety concern.

4.5.6 Core Burnup – TAMU LEU Fuel

Burnup analyses were performed using the DIF3D multi-dimensional diffusion theory code along with the BURP depletion code. All burnup analyses used the cross-sections generated for Beginning-of-Life (BOL) concentrations at the approximate average fuel temperature of 200°C, the closest nuclear data available. The burnup curve was then adjusted to 237°C using α.

Figure 4.15 shows the results from design calculations for core excess reactivity as a function of burnup of the initial core. The time steps used for the burnup calculation started with 3 days (to evaluate equilibrium xenon poisoning) and then 50 day intervals from time 0 at full power (1.0 MW). The LEU burnup curve in Figure 4.14 gives a lifetime of the initial core (with no fuel shuffling) of about 2000 MWD at 1.0 MW, full equilibrium xenon poisoning, and about \$0.60 reactivity left for burnup or experiments. For comparison, a FLIP burnup curve for a 90 fuel element core was calculated and is also shown in Figure 4.15. The FLIP burnup curve gives a lifetime of the initial core (with no fuel shuffling) of about 2350 MWD at 1.0 MW, full equilibrium xenon poisoning, and about \$0.60 reactivity remaining.

The data on burnup at the 3-day interval indicates a xenon equilibrium poison value of about \$1.31. The magnitude of this xenon poison may seem small for a 1.0 MW reactor ($\sim 10^{13}$ n/cm$^2 \bullet$s). However, xenon is produced interior to the TRIGA fuel elements where the thermal neutron flux is severely depressed due to 30 wt% uranium and erbium burnable poison. At the end of core life, an independent calculation gives an equilibrium xenon poison value of \$1.51. This value is larger at EOL because the thermal flux in the fuel is larger due to burnup of a portion of the U-235 and erbium.

With TAMU operation for 35-70 hours/week, full equilibrium xenon will not be built into the core (a process taking more than 60 continuous hours). Consequently, this increased core excess reactivity will permit operation to about 2600 MWD (an additional 600 MWD). The burnup curve in Figure 4.15 assumes no reactivity requirement for experiments but full equilibrium xenon poison. In view of the fact that far less xenon poison will be present, one can assume that a portion of the \$1.31 reactivity for xenon

poison could be used as needed to cover reactivity loss due to experiments with the same 2000 MWD core life.

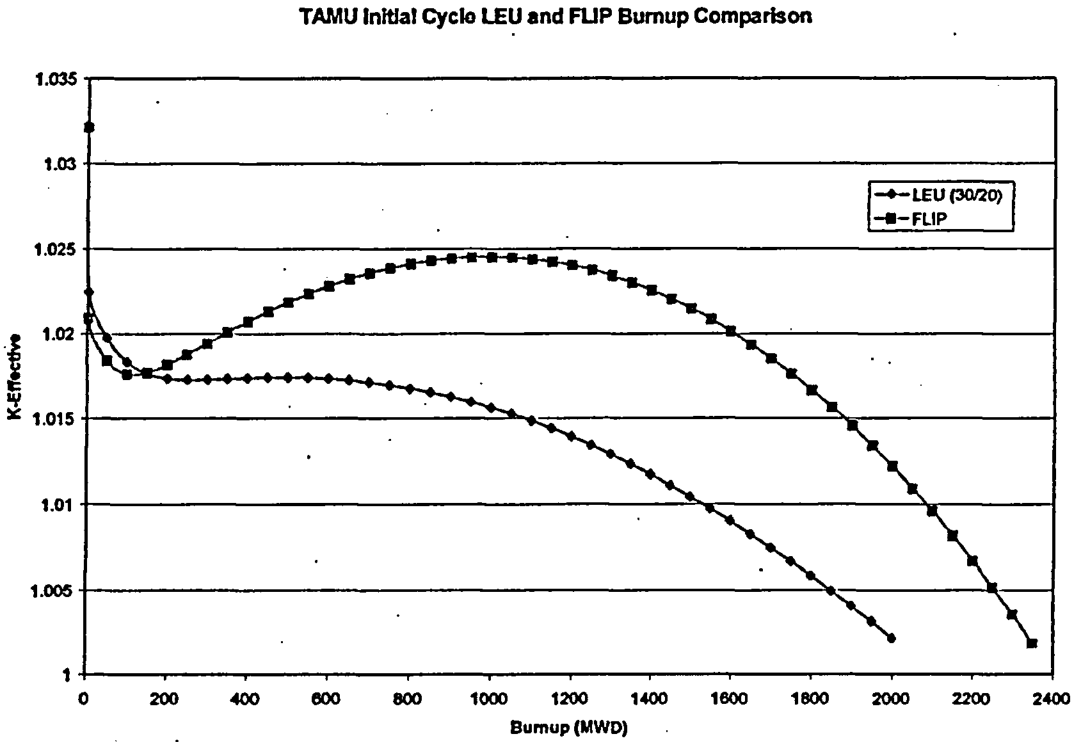


Figure 4.15 Reactivity versus Burnup for 1 MW TRIGA FLIP and LEU (30/20) Cores

The above comments lend credence to the assertion that the TRIGA LEU 30/20 core is typically a "lifetime core". Since TAMU operates the core between 35 and 70 MW Hrs per week, this corresponds to 1.46 to 2.9 MWD/week, or about 73 to 146 MWD/year. With only partial Xenon poisoning expected during the planned operating schedule (≤70 MWH/wk), the initial core loading is expected to provide operation at 1 MW as required for a time period ranging from 18 to 35 years.

With the initial LEU core expected to provide full power operation on demand for an extended period of time (18 to 35 years), a detailed reload schedule would not be useful at this time. However, the addition of either a fresh fuel rod or a full cluster of fresh fuel rods to the center of a heavily burned core raises concern for excessive power peaking. Unless a detailed calculation is performed to evaluate the magnitude of the power peaking, the addition of fresh fuel must be limited to the outer perimeter of the core. However, it may be noted that shuffling the outer, less burned fuel clusters to the core center during the 2000 MWD burnup will add significantly to the core life without concern for excessive power peaking. Typically, experience has shown that a shuffled core can add an additional 20% to the life of an unshuffled core.

Additional information on power peaking during fresh fuel reloading is presented in Section 12.6, Reload Considerations.

Reactor Parameters at 2000 MWD Burnup

Using the procedure set forth above for the delayed neutron fraction, the effective delayed neutron fraction has been evaluated for the LEU (30/20) core at 2000 MWD burnup. The value obtained is

$$\beta = 0.0070,$$

unchanged from the beginning of core life.

Similarly, using the procedure set forth above for the prompt neutron life, the prompt neutron life (ℓ) at 2000 MWD burnup has been evaluated for the LEU (30/20) core. The value obtained is

$$\ell = 27.3 \ \mu\text{sec}$$

slightly greater than the beginning of core life value of 26.3 μsec.

The values of the prompt negative temperature coefficient for TRIGA LEU (30/20) fuel at 2000 MWD burnup have already been presented in Tables 4-6(a) and 4-6(b) and shown in Figure 4.14 compared with the values at beginning of life.

4.5.7 Reactivity Loss at Reactor Power

Reactivity Loss, PRNC

The prompt negative temperature coefficient of reactivity is active in all reactor operations for which the fuel temperature is elevated above ambient. Consequently, core reactivity is lost at any power above a few kilowatts (when fuel temperatures begin to rise). Calculations of k-effective have been made for reactor powers of 0.5, 1.0, and 1.4 MW, respectively. During the commissioning tests of the PRNC FLIP core, direct measurements were made to evaluate the reactivity loss at various reactor power levels up to 1.4 MW. These calculated and measured values of reactivity losses are tabulated in Table 4-7.

Table 4-7. Calculated and Measured Reactivity Loss, PRNC

P(MW)	$\rho = \Delta k/k\beta$	$\Delta\rho(\$)_{calc}$	$\Delta\rho(\$)_{meas}$
0	5.639		
0.5	4.432	1.21	1.20
1.0	4.118	1.52	1.84
1.4	3.928	1.71	2.39

As can be seen from a comparison of measured and calculated reactivity loss, the agreement is good at 0.5 MW, but for powers of 1.0 and 1.4 MW, the measured reactivity loss is greater than the calculated value. The magnitude of the reactivity losses is related directly to the calculated temperatures. In Table 4-8, the measured temperatures ($T_{0.3}$) agree well with the calculated T_{03} temperatures at the three power levels, 0.5, 1.0, and 1.4 MW. Thus, the lack of agreement for reactivity losses at the higher power levels, 1.0

and 1.4 MW, is unexpected. This lack of agreement could be attributed to a 30-60°C error in the actual average core temperature at 1.0 and 1.4 MW.

Table 4-8. Calculated and Measured Fuel Temperatures, PRNC

P(MW)	$\hat{T}_{meas}$ (°C)	T_{calc} (°C)		
		$\hat{T}$	$\hat{T}_{0.3}$	$\overline{T}_{avg\ core}$
0.5	221	275	266	202
1.0	342	362	344	234
1.4	421	449	423	252

Reactivity Loss, TAMU

Calculations of core reactivity were made for operating power levels up to 1.3 MW. From these calculated values of k_{eff}, the loss in reactivity has been computed and is shown in Figure 4.16. The reactivity loss at 1.0 MW is $1.69 (cold-hot reactivity swing). After burnup to 2000 MWD, the reactivity loss at 1.0 MW is $1.51.

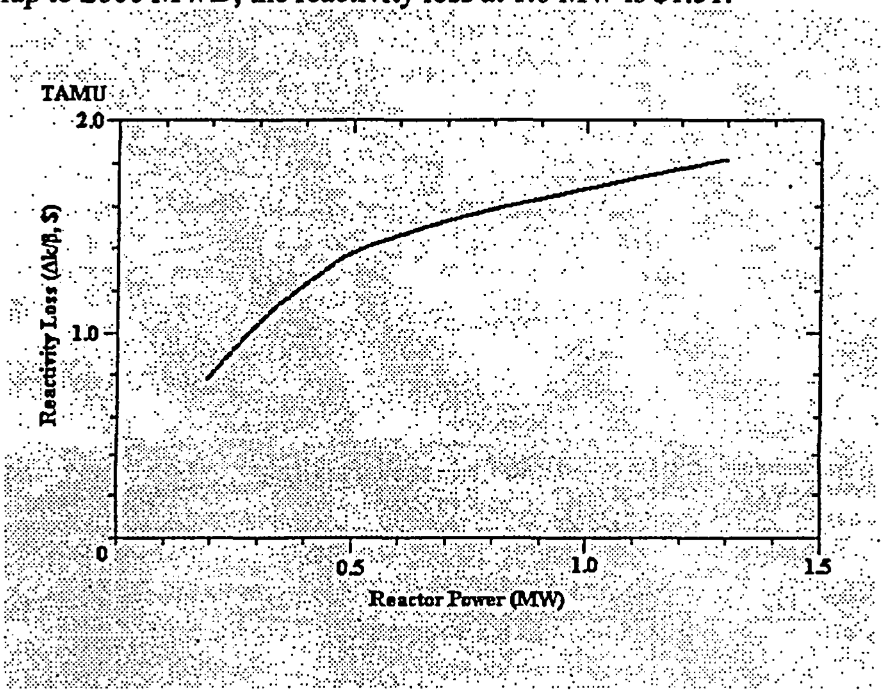


Figure 4.16. Calculated Reactivity Loss at Various Reactor Power Levels

4.5.8 Power Peaking; Temperature Peaking, PRNC Core

Power peaking in the core is analyzed on the basis of the following component values:

1. $\overline{P}_{rod} / \overline{P}_{core}$; rod power factor, the power generation in a fuel rod (element)

relative to the core averaged rod power generation.

2. $(\hat{P}/\overline{P})_{axial}$: axial peak-to-average power ratio within a fuel rod.

3. $(\hat{P}_{rod}/\overline{P}_{rod})_{radial}$; rod peaking factor, the peak-to-average power in a radial plane within a fuel rod.

Since maximum fuel temperature is the limiting operational parameter for the core, the peaking factor of greatest importance for steady-state operation is $\overline{P}_{rod}/\overline{P}_{core}$. The maximum value of this factor for the hottest rod, the hot-rod factor, [$(\overline{P}_{rod}/\overline{P}_{core})$ = hot-rod factor], determines the power generation in the hottest fuel element. When combined with the axial power distribution, the hot-rod factor is used in the thermal analysis for determination of the maximum fuel temperature. The radial power distribution within the element has only a small effect on the peak temperature but is also used in the steady-state thermal analysis.

The rod peaking factor $(\hat{P}_{rod}/\overline{P}_{rod})_{radial}$ is of importance in the transient analysis for calculating maximum fuel temperatures in the time range where the transfer is not yet significant. It is used in the safety analysis to calculate the peak fuel temperature under adiabatic conditions, where temperature distribution is the same as power distribution.

Peaking factors calculated for the PRNC TRIGA FLIP core are shown in Table 4-9. The axial power distribution shown in Figure 4.17 is relatively independent of fuel temperature or radial position in the core.

The fuel temperatures for selected reactor power levels have been calculated for the hottest and average fuel rods. These results are presented in Table 4-8 together with the experimentally measured fuel temperatures for the same reactor power levels.

It will be noted that the instrumented fuel element was in fact located in the hottest core location. Thus, the fuel temperature $\hat{T}_{meas}$ is the measured fuel temperature in the hottest fuel element. Since the sensing tip of the thermocouple was 0.30 inches from the axial centerline of the fuel element, the temperatures reported in Table 4-8 were calculated for the hottest radial position ($\hat{T}$) and for a position 0.30 inches from the center line ($\hat{T}_{03}$). Finally, the average core temperature ($\overline{T}_{avg\ core}$) was calculated.

Table 4-9. Power Peaking Factors - PRNC	
Type of Peaking	Beginning of Core Life (95 Fuel Rods and 1 Stainless Steel Dummy)
$(\overline{P}_{rod} / \overline{P}_{core})_{max}$	1.55
$(\hat{P} / \overline{P})_{axial}$	1.27
Average $(\hat{P}_{rod} / \overline{P}_{rod})_{radial}$	1.52
Peak $(\hat{P}_{rod} / \overline{P}_{rod})_{radial}$	1.967

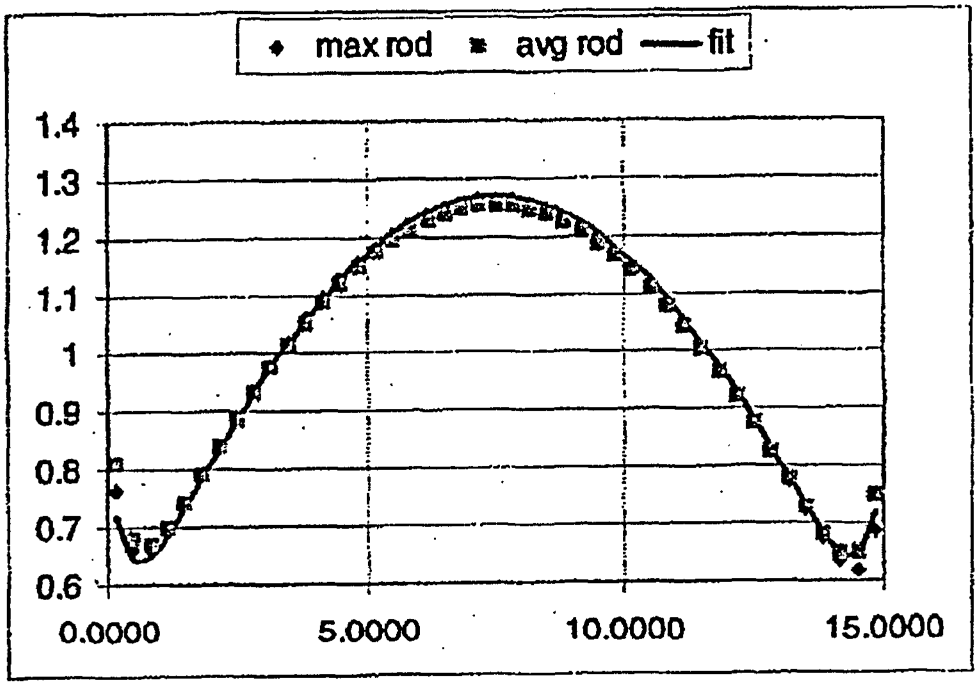


Figure 4.17 PRNC Axial Power Profile versus Distance from Bottom of Fueled Section (inches)

4.5.9 Power Peaking; Temperature Peaking, TAMU Core

The peaking factors calculated for the TAMU LEU (30/20) core at both BOL and EOL are shown in Table 4-10(a) and 4-10(b). The axial power distribution shown in Figure 4.18 is relatively independent of fuel temperatures or radial position in the core. Unlike PRNC, the peak power density does not occur in the fuel element with the maximum rod factor power. Therefore, it is necessary to know the rod factor in both the maximum power rod, $(\overline{P}_{rod}/\overline{P}_{core})_{max}$, and the peak power density rod, $(\overline{P}_{rod}/\overline{P}_{core})_{peak}$.

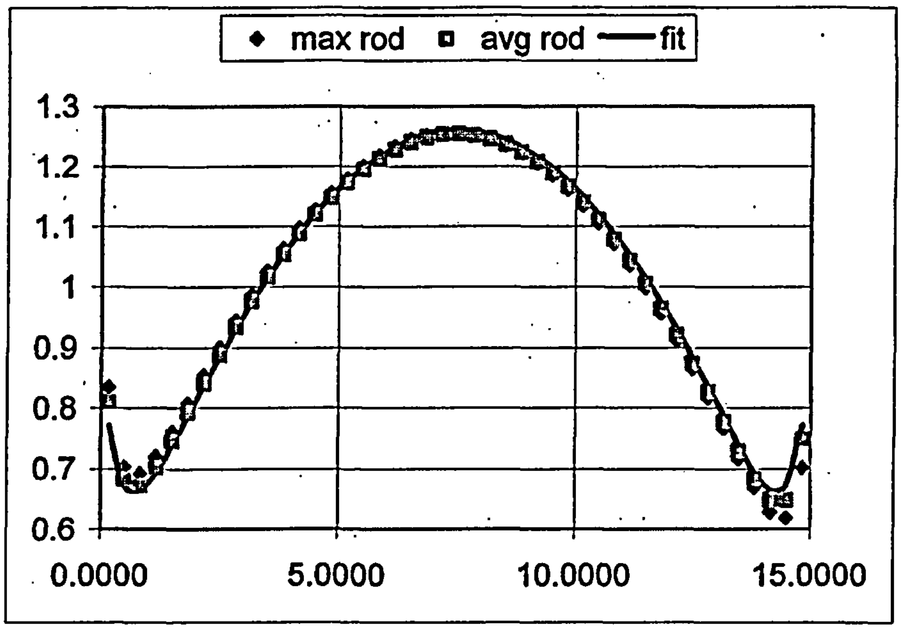


Figure 4.18 TAMU Axial Power Profile versus Distance from Bottom of Fueled Section (Inches)

TABLE 4-10(a) Power Peaking Factors – TAMU – BOL

Type of Peaking	Beginning of Core Life (90 Fuel Rods)
$(\overline{P}_{rod}/\overline{P}_{core})_{max}$	1.565
$(\hat{P}/\overline{P})_{axial}$	1.26
Average $(\hat{P}_{rod}/\overline{P}_{rod})_{radial}$	1.57
$(\overline{P}_{rod}/\overline{P}_{core})_{peak}$	1.446
Peak $(\hat{P}_{rod}/\overline{P}_{rod})_{radial}$	2.297

TABLE 4-10(b) Power Peaking Factors – TAMU – EOL	
Type of Peaking	End of Core Life (2000 MWD) (90 Fuel Rods)
$(\overline{P}_{rod} / \overline{P}_{core})_{max}$	1.511
$(\hat{P} / \overline{P})_{axial}$	1.26
Average $(\hat{P}_{rod} / \overline{P}_{rod})_{radial}$	1.530
$(\overline{P}_{rod} / \overline{P}_{core})_{peak}$	1.352
Peak $(\hat{P}_{rod} / \overline{P}_{rod})_{radial}$	2.200

The fuel temperatures for TAMU steady state operation have been calculated for the hottest, measured, and average fuel rods. These results are presented in Table 4-11. The value for $\hat{T}_{0.3}$ is the thermocouple temperature that is located 0.3 inch from the fuel centerline.

TABLE 4-11. Calculated Fuel Temperatures, TAMU

P (MW)	$\hat{T}$	$\hat{T}_{0.3}$	$\overline{T}_{core}$
0.5	289	260	205
1.0	373	329	237
1.3	440	384	249

For the TAMU reactor, it will be noted that the instrumented fuel element (IFE) is not located at the hottest core position. The IFEs are located at (5E4) and (6D4); the hottest fuel element is calculated to be (5D3) (See Figure 4.2 for location). Other positions can be chosen for the IFEs. The sensing tip of the thermocouple is 0.3 inch (7.62 mm) from the fuel axial center line, just outside the 0.25 in (6.35 mm) diameter zirconium rod positioned along the axial center of the fuel. The results reported in Table 4-11 give the peak fuel temperature $\hat{T}$ in the hottest fuel element, the computed temperature $\hat{T}_{0.3}$ in the IF3 that can be compared with future measured temperatures, and the average core temperature $\overline{T}$.

The average power per fuel element in the TAMU core operating at 1000 kW with 90 fuel elements is 11.1 kW/element. In the following, please refer to Figure 4.2. The fuel element immediately adjacent to the Transient rod (5D3) produces the greatest power (17.4 kW/element). An ideal location for one of the Instrumental Fuel Elements (IFE) would be this location (5D3); however, this location is within the cluster that has the transient rod. A three-rod locking plate on this cluster precludes placing an IFE within the cluster. The IFE is located at the next nearest location, 5E4, where the power generated is 15.4 kW/element.

The water in position 3D leads to significant power peaking, especially in fuel elements such as 4D3 and 4D4. An IFE in 4D4 would generate a slightly lower power (15.3 kW/element), but has a peak power density closer to the maximum peak power density in the whole core, even larger than in the fuel element immediately adjacent to the Transient rod (5D3). Consequently, a thermocouple element ($T_{0.3}$) in this location would give the highest measured fuel temperature during pulsing.

The core configuration in Figure 4.2 indicates that a second IFE is located at 6D4. In this location the power generated is 14.0 kW/element with a low power density.

4.5.10 Pulsing Operation, PRNC

Most of the 65 TRIGA reactors have pulsing capability. Thousands of TRIGA reactor pulses have been safely performed. A calculational procedure (TRIGA-BLOOST) based on a space-independent kinetics model (Ref. 8) has been developed for predicting pulse performance of TRIGA reactors.

The following simplified relationships are given to show qualitatively how the pulsing performance is influenced by the important reactor parameters:

$$\tau = \ell / \Delta k_p = \text{reactor period}$$

$$\overline{\Delta T} = \frac{2\Delta k_p}{\alpha} = E/C$$

$$\hat{P} = \frac{C(\Delta k_p)^2}{2\alpha\ell} = \text{peak pulsed power}$$

$$E = \frac{2C\Delta k_p}{\alpha} = \text{total energy release in prompt burst}$$

where

ℓ = prompt neutron life
α = prompt negative temperature coefficient
C = total heat capacity of the core available to the prompt pulse energy release
$\overline{\Delta T}$ = change in average core temperature produced by the prompt pulse

Δk_p = that portion of the step reactivity insertion which is above prompt critical

Water filled regions within the core promote flux peaking and result in increased power peaking and peak fuel temperatures, especially during a reactivity pulse. The PRNC core was a compact core with no in-core experimental regions that could be water filled. However, all five control rods had water-followed regions and thus constituted regions of enhanced power peaking. These regions were correctly modeled in the codes that are used. The instrumented fuel element gave the peak fuel temperature since it was positioned beside a control rod with its water follower.

The BLOOST pulsing performance results have been prepared for reactivity insertions of \$1.45, \$1.95, and \$2.30. Measured data from pulses performed during the commissioning tests are available for direct comparisons. These data are also presented in Table 4-12.

TABLE 4-12. Pulse Performance: Calculated and Measured, PRNC 95 TRIGA HEU (FLIP) Fuel			
Parameter	\$1.45 Pulse	\$1.95 Pulse	\$2.30 Pulse
Measured Data			
• $\hat{P}$ (MW)	308	1163	1954
• E $(MW-\text{sec})$*	11.5	17.4	20.5
• $\hat{T}_{0.3}$ (°C)	306	411	466
BLOOST-calc			
• $\hat{P}$ (MW)	436	1814	3262
• E $(MW-\text{sec})$	16.8	27.2	33.5
• $\hat{T}$ (°C)	488	766	909
• $\overline{T}_{\text{core}}$ (°C)	177	286	347
• $\hat{T}_{0.3}$ (°C)	327	515	618
*All measured values of E contain a small "tail contribution (until the 1.0 sec Scram) estimated at 25-33% of total value.			

The values for pulsing during the commissioning of the PRNC FLIP core were significantly smaller than those predicted in the PRNC SAR. These experimentally measured values are also smaller than the values currently calculated. It may be noted that the peak thermocouple reading ($\hat{T}_{0.3}$) in the hottest fuel rod has also been calculated for a distance of 0.3 inch from the fuel vertical centerline, same location as the sensing tip of the thermocouple in the hottest fuel rod.

The results for energy from the BLOOST-calculations are reported at a time of about 1 second after the pulse initiation (well after the peak pulsed power) and at about the time a control rod SCRAM was initiated. The pulse is completed about 0.3 seconds after pulse initiation at which time the peak fuel temperature is computed. Thereafter, the peak fuel temperature (at the outer surface of the fuel rods) decreases as energy flows both to the center of the fuel rod and to the cooling water. However, the average core temperature continues to rise as energy is accumulated from the "tail" of the pulse until at about 1 second, when all control rods were scrammed, shutting down the pulsed reactor. Account has been taken for the power peaking in the water filled region when the transient rod is pulsed. The hottest fuel rod is the instrumented fuel element and is adjacent to the transient rod, reflecting this power peaking.

4.5.11 Pulse Operation – TAMU – BOL

The TAMU reactor has had extensive experience with pulsing performance using fuel having a strong temperature dependent prompt negative temperature coefficient of reactivity (α). The new LEU (30/20) fuel also produces a similarly strong temperature

dependence of α. Comparing the curves for α for FLIP fuel (Figure 4.13) and for LEU (30/20) fuel (Figure 4.14), one notes similar temperature dependences; however, the magnitude of α is somewhat smaller for the LEU fuel. The BOL neutron lifetime is 26.3 μsec; the EOL neutron lifetime is 27.3 μsec.

Table 4-13 presents the beginning of core life pulsing parameters for the TAMU LEU (30/20) core. Results for the FLIP core are included for easy comparison. Figure 4.19 shows a plot of the pulsed fuel temperatures as a function of reactivity insertion for TAMU LEU (30/20) fuel at BOL. Results for reactivity insertions of $1.45, $1.95, $2.30, and $2.50 are shown. The peak fuel temperature in the core (4D3) is listed. Not only are measured temperature results ($\hat{T}_{03}$) shown for the planned instrumented fuel element in position 5E4, but also for positions 4D4 and 6D4. The position 4D4 was included since a fuel element in this location experiences the highest peak power density (though not the highest power per element) and hence the highest measured temperature $\hat{T}_{03}$. Since it will be shown in a later section that the peak pulsed fuel temperature will need to be limited to 830°C, the reactivity insertion ($2.10) to produce this temperature is indicated in Figure 4.19. Table 4-13 also shows that the maximum insertion of the transient rod must be limited to $2.95 in order to limit peak fuel temperatures to less than the 1150°C Safety Limit. An electro-mechanical interlock on the UP position of the transient rod and pulsing circuitry shall prevent accidental pulsing of more than $2.95 thereby preventing a pulse which could exceed the Safety Limit based on the analysis presented in Table 4-13.

At TAMU, it has been standard practice to place one of the two IFEs in 5E4 and a second one at 6D4. This analysis suggests that the IFE in 6D4 should be relocated to 4D4.

TABLE 4-13. Calculated Pulse Performance for 95 FLIP Fuel Elements and 90 LEU (30/20) Fuel Elements – BOL

Parameter	FLIP Fuel – 95 Elements			LEU (30/20) Fuel – 90 Elements					
	$1.45	$1.95	$2.30	$1.45	$1.95	$2.30	$2.50	$2.95	$3.21
$\hat{P}$ (MW)	436	1814	3262	227	1008	1873	2468	3434	3775
E (MW-sec)	16.8	27.2	33.5	11.3	19.8	25.5	28.2	29.6	35.5
$\hat{T}$ (°C) (4D3)	488	766	909	459	755	921	1006	1144	1182
$\bar{T}_{core}$ (°C)	177	286	347	155	264	330	365	422	438
$\hat{T}_{03}$ (°C) (5E4)	327	515	618	200	333	413	456	526	545

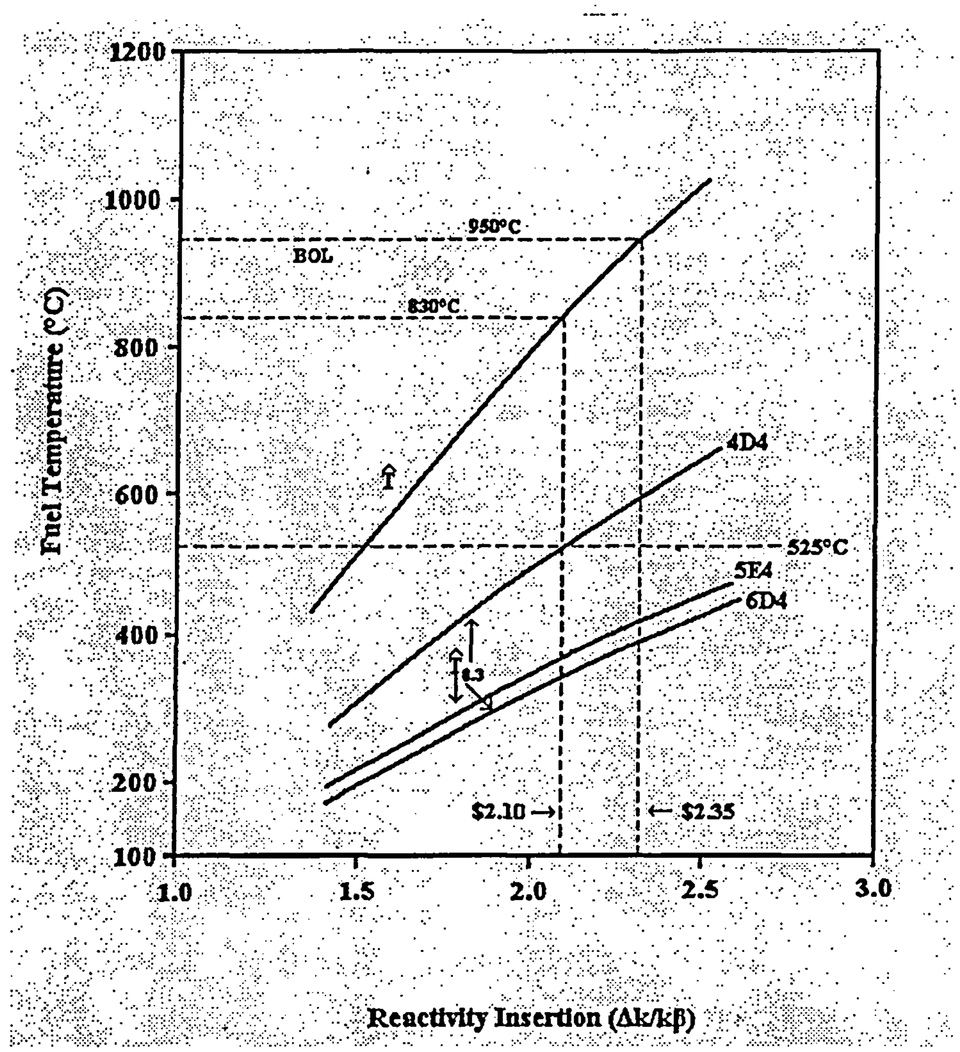


Figure 4.19 Peak Fuel Temperature ($\hat{T}$) and Typical IFE Measured Fuel Temperatures ($T_{0.3}$) as a Function of Reactivity Insertion at Beginning of Core Life, TAMU

4.5.12 Pulse Operation – TAMU – EOL

The BLOOST code has been used to calculate the pulsing performance of the LEU (30/20) core at 2000 MWD burnup. The procedure is the same as used above for BOL conditions. Results are shown in Table 4-14 for reactivity insertions from low power (300W) of $1.45, $1.95, $2.30, $2.50, $2.95, and $3.21. Results are presented for peak pulsed power, integrated energy, peak fuel temperature in hottest fuel rod, average reactor core temperature, and thermocouple measured temperature.

Figure 4.20 illustrates the dependence of fuel temperatures on reactivity insertions at 2000 MWD burnup. A particular object is to explore whether the reactivity insertion of $2.10 will give higher or lower peak fuel temperature compared to the results at start of core life. (See Figure 4.19 for beginning of life results.) A reactivity insertion of $2.10

at 2000 MWD burnup will give a slightly <u>lower</u> peak fuel temperature (i.e., 810°C contrasted to the beginning value of 830°C).

In view of the fact that the peak fuel temperatures ($\hat{T}$) at 2000 MWD are lower rather than higher than the initial peak fuel temperature, the Limiting Reactivity Insertion for pulse operation can safely remain at $2.10, as determined earlier.

Similarly, the maximum allowed accidental pulse of $2.95 which was shown previously in Table 4-13 to accomodate the Safety Limit of 1150°C is also shown in Table 4-14 to meet the Safety Limit. Therefore, the electro-mechanical interlock discussed in the previous section will prevent an accidental pulse from exceeding the Safety Limit both at BOL and EOL.

TABLE 4-14. Calculated Pulse Performance for 90 LEU (30/20) Fuel Elements for BOL and EOL – TAMU

Parameter	LEU (30/20) Fuel – 90 Elements – BOL				LEU (30/20) Fuel – 90 Elements – EOL					
	$1.45	$1.95	$2.30	$2.50	$1.45	$1.95	$2.30	$2.50	$2.95	$3.21
$\hat{P}$ (MW)	227	1008	1873	2468	230	1028	1915	2536	3716	4046
E $(MW-\text{sec})$	11.3	19.8	25.5	28.2	11.8	21.2	23.0	30.9	33.0	35.0
$\hat{T}$(°C) (6D1)	459	755	921	1006	442	733	901	988	1142	1188
$\overline{T}_{core}$ (°C)	155	264	330	365	162	278	350	388	458	478
$\hat{T}_{0.3}$(°C) (5E4)	200	333	413	456	208	354	444	492	577	602

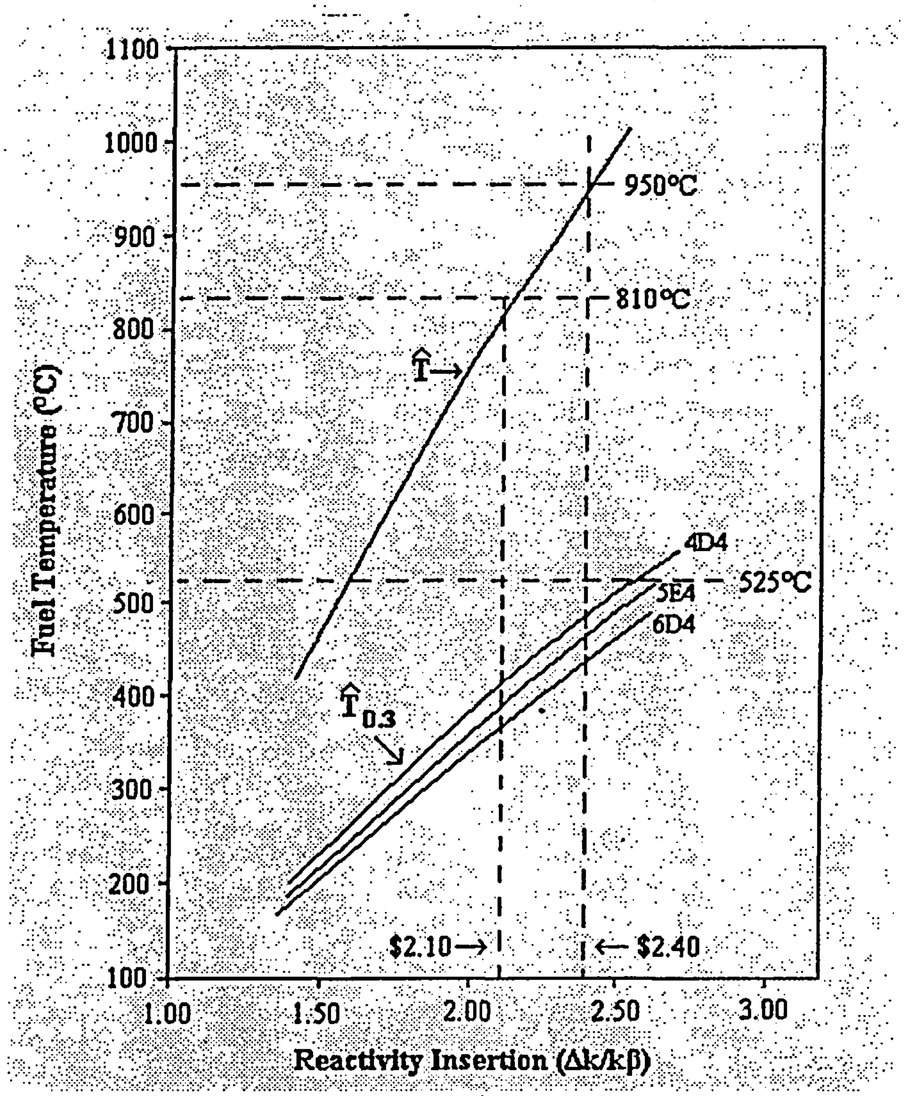


Figure 4.20 Peak Fuel Temperature ($\hat{T}$) and Typical IFE Measured Fuel Temperatures ($\hat{T}_{0.3}$) as a Function of Reactivity Insertion at 2000 MWD Burnup

4.6 Functional Design of the Reactivity Control System

The proposed HEU to LEU conversion of the TAMU core requires the addition of an electro-mechanical interlock on the adjustable UP limit for the transient rod. When permanently set for transient rod insertions no greater than $2.95 in PULSE MODE, this will prevent an accidental pulse from exceeding the Safety Limit.

No other changes in the reactivity control system are required.

4.7 Thermal-Hydraulic Analysis – PRNC

4.7.1 Analysis of Steady State Operation.

The following evaluation has been made for a TRIGA system operating with cooling from natural convection water flow around the fuel elements. In this study, the predicted steady state thermal-hydraulic performance of the PRNC TRIGA core was determined for the reactor operating at 1.4 MW and a water inlet temperature of 30°C. Although the PRNC TRIGA reactor was designed and operated at 2.0 MW, extensive measurements were made at 1.4 MW during the commissioning tests. These experimental data are used for the benchmark comparisons. An average powered fuel rod and a maximum powered fuel rod were analyzed. The STAT computer code (Ref. 9) was used to determine the natural convection flow rate, the coolant and clad axial temperature profiles, and the clad wall heat flux axial profile. The STAT code was also used to determine the clad wall maximum heat flux versus coolant inlet temperature for departure from nucleate boiling. The TAC2D thermal analysis code (Ref. 10) was used to determine the fuel average and fuel maximum temperatures.

4.7.2 STAT Code Analysis and Results.

STAT is a computer program for calculating the natural convection heat transfer and fluid flow in an array of heated cylinders. It calculates the natural convection flow through a vertical water coolant channel bounded by cylindrical heat sources. Output from the STAT code includes: channel flow rate, outlet velocity, temperature rise of the fluid along the channel, maximum heat flux and maximum clad temperature. The assumption is made that there is no cross flow between adjacent channels. Input to the program includes the following:

1) Size and spacing of the heat sources;

2) Axial heat source distribution;

3) Pressure head above the source;

4) Constants to be used in generic expressions for convective and sub-cooled boiling heat transfer coefficients and for the convective friction factor;

5) Inlet and exit pressure loss coefficients;

6) Inlet water temperature.

Analysis is performed on a single flow channel divided into axial segments. The natural convection system for the PRNC TRIGA was based on the 4-rod cluster of fuel elements. The representation used herein establishes one flow channel bounded by four fuel elements. The reactor geometry, power factors and heat transfer and friction data for the STAT input are given in Table 4-15.

TABLE 4-15. STAT Input for Reactor and Core Geometry and Heat Transfer, PRNC

Core and Reactor Geometry		
Unheated core length at inlet, mm		100
Unheated core length at outlet, mm		103
Length from top of unheated length to top of locking plate, mm		21
Distance from top of pool surface to top of locking plate, mm		7170
Heat Transfer and Friction Data		
Void detachment fraction		0.0
Sub-cooled boiling correlation coefficient		0.2494
Sub-cooled boiling correlation exponent		3.797
Convection correlation coefficient		0.023
Convection correlation exponent for the Reynolds number		0.8
Convection correlation exponent for the Prandtl number		0.4
Friction factor correlation coefficient		0.079
Friction factor correlation exponent		0.25
Inlet pressure loss coefficient		1.672
Exit pressure loss coefficient		0.6
Ambient pressure at elevation 750 m, MPa		0.092
Peaking Factors (Axial and Radial)		
Initial core	Axial Power Factor	1.27
	Hot Rod Factor	1.55

A STAT thermal hydraulic analysis was done for an average flow channel and a maximum powered channel. The analysis was conducted by considering the hydraulic characteristics of a typical flow channel represented by the geometric data given in Table 4-16.

TABLE 4-16. Hydraulic Flow Parameters For a Typical Flow Channel	
Flow area (mm^2/rod)	552.26
Wetted perimeter (mm/rod)	112.6
Hydraulic diameter (mm)	19.62
Fuel element heated length (mm)	381
Fuel element diameter (mm)	35.842
Fuel element surface area (mm^2)	4.29×10^4

The heat generation in the fuel element is distributed axially in a cosine distribution chopped at the end such that the peak-to-average ratio is 1.27 (Table 4-9). It is further given that there are 95 fuel elements in the initial core. The hot-rod power ratio is assumed to be 1.55 (Table 4-9) for initial core conditions.
The axial power profile for STAT is given by the analytical equation:

$$Q(z) = APF * \cos\left(\pi \frac{(z - ZL/2)}{Z_{ext}}\right) * \left(1 + G\,E * \exp\left[\frac{-AK * z}{L}\right]\right)$$

where Z_{ext}, GE, and AK are parameters to fit the shape of the axial profile determined by DIF3D having the following numerical values:

$Z_{ext} = 49.784$ cm
GE = 0.77
AK = 45.0

The driving force is supplied by the buoyancy of the heated water in the core. Countering this force are the contraction and expansion losses at the entrance and exits to the channel, and the acceleration and potential energy losses and fiction losses in the cooling channel itself. The pressure drops through the flow channel are dependent on the flow rate while the available static driving pressure is fixed for a known core height and ambient pressure.

The STAT code calculates both a convection and a boiling heat transfer coefficient and uses the larger of these two to determine the clad-to-fluid temperature difference. The boiling heat transfer correlation, the convection heat transfer correlation, and the friction factor equation as used by STAT are, respectively:

$$Q_{boiling} = 0.2494(T_w - T_{sa}t)^{3.797}$$

$$Nu = 0.023 \mathrm{Re}^{0.8} \mathrm{Pr}^{0.4}$$

$$f = \frac{0.079}{\mathrm{Re}^{0.25}}$$

The zero void detachment fraction given in Table 4-15 implies that none of the vapor bubbles that are generated due to subcooled boiling detach from the wall surface, enter the main coolant and subsequently increase the buoyancy of the main coolant stream.
A summary of the STAT results for PRNC is given in Table 4-17.

TABLE 4-17. Steady State Results for PRNC, 1.4 MW (Results are for One Flow Channel)

Initial Core (95 elements)	Average Rod	*Maximum Rod*
Channel natural convection mass flow rate, kg/sec	0.084	0.097
Channel total pressure drop, Pa	55.59	81.11
Exit coolant flow temperature, °C	71.86	86.50
Maximum wall heat flux, W/cm^2	43.49	67.41
Maximum flow velocity, cm/sec	15.57	18.14
Maximum clad temperature, °C	135.1	137.3
Exit clad temperature, °C	132.3	134.2

The STAT code also calculates the maximum nucleate boiling heat flux, that is, the heat flux at which there is a departure from nucleate boiling (DNB) and the transition to film boiling begins. This is also termed the critical heat flux. Two correlations are used to calculate this heat flux. The first, given by McAdams (Ref. 11), indicates that the critical heat flux is a function of the fluid velocity and the fluid only. The second correlation is due to Bernath (Ref. 12). It encompasses a wider range of variables over which the correlation was made and it takes into account the effect of different flow geometries. It generally gives a lower value for the critical heat flux. The lower critical heat flux from the correlations is used here for determining the minimum DNB ratio, that is, the minimum ratio of the local allowable heat flux to the actual heat flux. The minimum DNB ratio is given in Table 4-18.

The STAT code analysis has been run for the critical heat flux for the PRNC reactor operating at 1.4 MW. The data was obtained by first selecting an inlet temperature (30°C) and then systematically increasing the reactor power until STAT indicated a DNB ratio equal to one. The maximum power per fuel element for which the DNB ratio is 1 versus the core inlet water temperature is 44 kW/element. For a core with a rod peaking factor of 1.55, this maximum fuel element power corresponds to a maximum reactor power of 2.70 MW. Hence, the DNB ratio for the PRNC TRIGA at the stated conditions is 1.93. These values are for BOL core conditions. The minimum DNB ratio is listed in Table 4-18.

TABLE 4-18. TRIGA Thermal and Hydraulic Parameters for PRNC, 1.4 MW	
Parameter	Initial Core
Number of fuel elements	95
Diameter, mm (in.)	35.842 (1.411)
Length (heated), mm (in.)	381 (15.0)
Core flow area, mm^2 (ft^2)	52465 (0.565)
Core wetted perimeter, mm (ft.)	10697 (35.10)
Flow channel hydraulic diameter, mm (ft.)	19.62 (0.0644)
Core heat transfer surface, m^2 (ft^2)	4.08 (43.87)
Hot rod factor	1.55
Axial peaking factor	1.27
Hot spot peaking factor*	1.97
Inlet coolant temperature, °C (°F)	30 (86)
Coolant saturation temperature, °C (°F)	117.0 (242.6)
Exit coolant temperature (average), °C (°F)	71.86 (161.3)
Exit coolant temperature (maximum), °C (°F)	86.50 (187.7)
Coolant mass flow, kg/sec (lb/hr)	8.02 (63.626)
Average flow velocity, mm/sec (ft/sec)	154 (0.506)
Peak fuel temperature in average fuel element, °C (°F)	333.6 (632.4)
Maximum wall temperature in hottest element, °C (°F)	137.3 (279.2)
Peak fuel temperature in hottest fuel element, °C (°F)	444.2 (831.5)
Core average fuel temperature, °C (°F)	232.2 (449.9)
Average heat flux, W/cm^2 (BTU/hr-ft^2)	34.35 (108,898)
Maximum heat flux in hottest element, W/cm^2 (BTU/hr-ft^2)	67.41 (213,734)
Minimum DNB ratio	1.93

*The "hot spot peaking factor" is defined as the product of the rod power factor and the axial peak to average power ratio.

4.7.3 TAC2D Fuel Temperature Analysis and Results.

STAT does not calculate fuel temperatures. The TAC2D general purpose heat conduction code was used to calculate steady state maximum and average fuel temperatures for the average powered rod and the maximum powered rod (Ref. 10). A radial-axial (R,Z) two-dimensional model of the center zirconium rod (6.35 mm diameter), the fuel annulus, the fuel-to-clad gap, and the 0.5 mm thick stainless steel cladding of a single fuel pin was constructed. The model included only the active length of the fuel pin.

TAC2D is a code for calculating steady-state and transient temperatures in two-dimensional problems by the finite difference method. The configuration of the body to be analyzed is described in the rectangular, cylindrical, or circular coordinate system by orthogonal lines of constant coordinate called grid lines. The grid lines specify an array of nodal elements. Nodal points are defined as lying midway between the bounding grid lines of these elements. A finite difference equation is formulated for each nodal point in

terms of its heat capacity, heat generation and heat flow paths to neighboring nodal points.

The TAC2D code requires as input a geometric description of the problem and properties of the materials considered. The radial and axial power distributions in the fuel are also provided as input. The problem is defined in cylindrical R-Z geometry. The axial distribution of the clad wall temperature from the STAT code was imposed on the outside surface of the clad for the TAC2D model outer radial boundary. Alternatively, the axial distribution of the surface heat transfer coefficient and coolant temperature from the STAT code could be used to model the outer radial boundary with consistent results. The fuel-to-clad interface conductance assumes the fuel pin is sealed with air and has an initial 0.003 mm cold gap. Reported fuel temperatures do not account for fuel swelling over long periods of operation and are therefore conservative. Some gap closure occurs due to the relative expansion of the fuel and cladding at normal operating temperatures. As the reactor is operated over time and irradiation effects in the fuel induce swelling, the gap closes even further, tending to increase the gap conductance and in turn decrease the peak fuel temperature.

A summary of the TAC2D results for PRNC has been given in Table 4-8 for 0.5, 1.0, and 1.4 MW operation with 95 fuel elements.

4.8 Thermal Hydraulic Analysis – TAMU LEU

4.8.1 Analysis of Steady State Operation, TAMU.

The following evaluation has been made for the TRIGA LEU (30/20) fuel system with 4-rod configuration operating with cooling from natural convection water flow through 4-rod clusters of fuel. The steady state thermal-hydraulic performance of the TAMU LEU TRIGA core was determined for operation at 1.0 MW with a water inlet temperature of 30°C.

An average powered fuel rod and a maximum powered fuel rod were analyzed. The STAT computer code (Ref. 9) was used to determine the natural convection flow rate, the coolant and clad axial temperature profiles, and the clad wall heat flux axial profile. The STAT code was also used to determine the clad wall maximum heat flux versus coolant inlet temperature for departure from nucleate boiling. The TAC2D thermal analysis code (Ref. 10) was used to determine the fuel average and fuel maximum temperatures.

4.8.2 STAT Code Analysis and Results, TAMU

The STAT analysis was performed using the method outlined in Section 4.7. The reactor geometry, power factors and heat transfer and friction data for the STAT input are given in Table 4-19. The natural convection system for the TAMU TRIGA was based on the 4-rod cluster of fuel elements. The representation used herein establishes one flow channel bounded by four fuel elements.

A STAT thermal hydraulic analysis was done for an average flow channel and a maximum powered channel. The analysis was conducted by considering the hydraulic

characteristics and flow parameters of a typical flow channel represented by the geometric data given in Table 4-20.

The heat generation in the fuel element is distributed axially in a cosine distribution chopped at the end such that the peak-to-average ratio is 1.26 (Table 4-11(a), 4-11(b)). It is further given that there are 90 fuel elements in the initial core. The hot-rod power ratio is assumed to be 1.565 for initial core conditions.

The parameters Z_{ext}, GE, and AK for the axial power profile equation Q(z) (see Section 6.2.5.2) are:

$$Z_{ext} = 50.212 \text{ cm}$$
$$GE = 0.70$$
$$AK = 45.0$$

A summary of the STAT results for the 1.0 MW TAMU (30/20) is given in Table 4-21.

4.8.3 TAC2D Fuel Temperature Analysis and Results, TAMU

Using the methods given in Section 4.7.3, a TAC2D analysis was performed for the 1.0 MW TAMU LEU (30/20) core for operation with 90 fuel elements. See Table 4-11.

TABLE 4-19. STAT Input for Reactor and Core Geometry and Heat Transfer, TAMU

Core and Reactor Geometry		
Unheated core length at inlet, mm		100
Unheated core length at outlet, mm		103
Length from top of unheated length to top of locking plate, mm		21
Distance from top of pool surface to top of locking plate, mm		7170
Heat Transfer and Friction Data		
Void detachment fraction		0.0
Sub-cooled boiling correlation coefficient		0.2494
Sub-cooled boiling correlation exponent		3.797
Convection correlation coefficient		0.023
Convection correlation exponent for the Reynolds number		0.8
Convection correlation exponent for the Prandtl number		0.4
Friction factor correlation coefficient		0.079
Friction factor correlation exponent		0.25
Inlet pressure loss coefficient		1.672
Exit pressure loss coefficient		0.6
Ambient pressure at pool surface, MPa		0.092
Peaking Factors (Axial and Radial)		
Initial core	Axial Power Factor	1.26
	Hot Rod Factor	1.565

TABLE 4-20. Hydraulic Flow Parameters, TAMU	
Flow area (mm^2/element)	552.26
Wetted perimeter (mm/element)	112.6
Hydraulic diameter (mm)	19.62
Fuel element heated length (mm)	381
Fuel element diameter (mm)	35.842
Fuel element surface area (mm^2)	4.29×10^4

TABLE 4-21. Steady State Results for TAMU, 1.0 MW
(Results are for One Flow Channel)

Initial Core (90 elements)	Average Rod	Maximum Rod
Channel natural convection mass flow rate, kg/sec	0.077	0.089
Channel total pressure drop, Pa	44.15	63.87
Exit coolant flow temperature, °C	64.79	76.82
Maximum wall heat flux, W/cm^2	32.54	50.92
Maximum flow velocity, cm/sec	14.09	16.50
Maximum clad temperature, °C	133.8	135.9
Exit clad temperature, °C	131.4	133.2

4.8.4 Steady-State Analysis Results Summary, TAMU

Table 4-22 lists the pertinent heat transfer and hydraulic parameters for the 1.0 MW TAMU TRIGA reactor. Results are presented therein for an average channel and a maximum powered channel (hot channel) at initial core conditions. Also shown are the peak fuel temperatures in the hottest and average fuel element calculated with the TAC2D code.

The STAT code analysis has been run for the critical heat flux for the TAMU reactor operating at 1.0 MW. The data was obtained by first selecting an inlet temperature (30°C) and then systematically increasing the reactor power until STAT indicated a DNB ratio equal to one. The maximum power per fuel element for which the DNB ratio is 1 versus the core inlet water temperature is 42 kW/element. For a core with a rod peaking factor of 1.565, this maximum fuel element power corresponds to a maximum reactor power of 2.42 MW. Hence, the DNB ratio for the TAMU TRIGA at the stated conditions is 2.42. These values are for BOL core conditions. The minimum DNB ratio is listed in Table 4-22.

TABLE 4-22. TRIGA Thermal and Hydraulic Parameters for TAMU, 1.0 MW	
Parameter	Initial Core
Number of fuel elements	90
Diameter, mm (in.)	35.842 (1.411)
Length (heated), mm (in.)	381 (15.0)
Core flow area, mm^2 (ft^2)	49,703 (0.535)
Core wetted perimeter, mm (ft.)	10,134 (33.25)
Flow channel hydraulic diameter, mm (ft.)	19.62 (0.0644)
Core heat transfer surface, m^2 (ft^2)	3.86 (41.56)
Hot rod factor	1.565
Axial peaking factor	1.26
Hot spot peaking factor*	1.97
Inlet coolant temperature, °C (°F)	30 (86)
Coolant saturation temperature, °C (°F)	117.0 (242.6)
Exit coolant temperature (average), °C (°F)	64.79 (148.6)
Exit coolant temperature (maximum), °C (°F)	76.82 (170.3)
Coolant mass flow, kg/sec (lb/hr)	6.90 (54,731)
Average flow velocity, mm/sec (ft/sec)	140 (0.459)
Peak fuel temperature in average fuel element, °C (°F)	282.4 (540.4)
Maximum wall temperature in hottest element, °C (°F)	135.9 (276.6)
Peak fuel temperature in hottest fuel element, °C (°F)	368.1 (694.6)
Core average fuel temperature, °C (°F)	206.8 (404.2)
Average heat flux, W/cm^2 ($BTU/hr\text{-}ft^2$)	23.24 (73,690)
Maximum heat flux in hottest element, W/cm^2 ($BTU/hr\text{-}ft^2$)	50.92 (161,438)
Minimum DNB ratio	2.42
*The "hot spot peaking factor" is defined as the product of the rod power factor and the axial peak to average power ratio.	

4.9 Thermal Neutron Flux Values, TAMU LEU Core

4.9.1 Thermal Neutron Flux Values in LEU Core

The DIF3D code provides neutron flux values. Figure 4.21 presents a 3-D representation of the thermal neutron distribution throughout the core and into the surrounding water and graphite (including the coupling box).

Figure 4.22 shows the flux plot through the transient rod in a direction perpendicular to the face of the coupler box. In the region near the 24-cm position, the thermal neutrons peak over a 3-4 cm distance in the water adjacent to the core. Near the 70-cm position, the thermal neutrons peak over a significantly greater distance in the graphite of the coupler box.

Figure 4.23 presents a graphical representation of the neutron flux across the core through the transient rod in a direction parallel to the face of the coupler box. This plot starts in the water reflector/shield, crosses the reactor, and ends in the water reflector on the other side of the reactor core. It can be noted that the thermal neutron flux is relatively weak

inside of the fuel rods due to the erbium and large uranium loading, but rises substantially in regions of water or graphite.

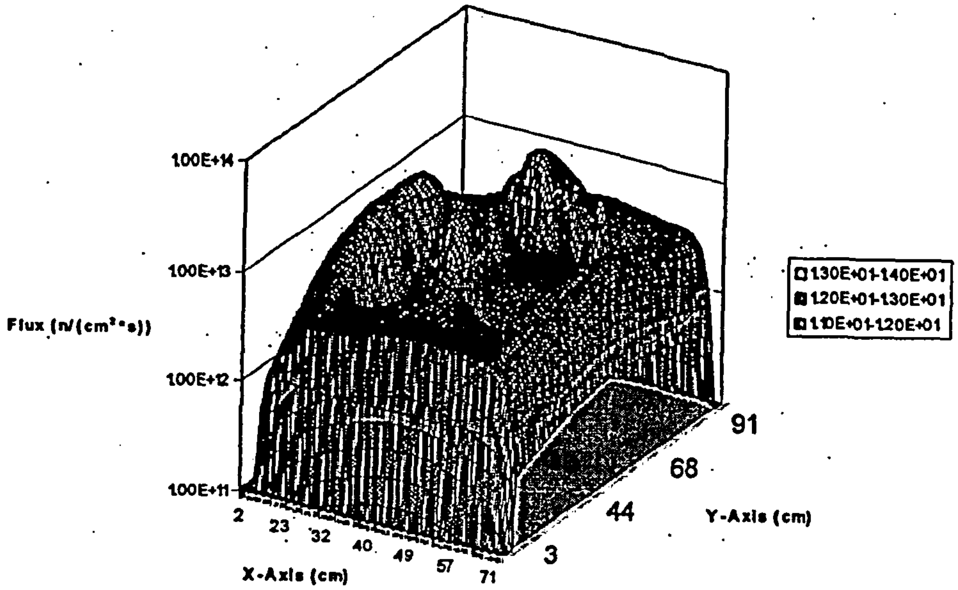


Figure 4.21 TAMU Initial Cycle: Thermal Neutron Flux (E<0.42 eV) at maximum axial power peaking position (76.7cm), BOL

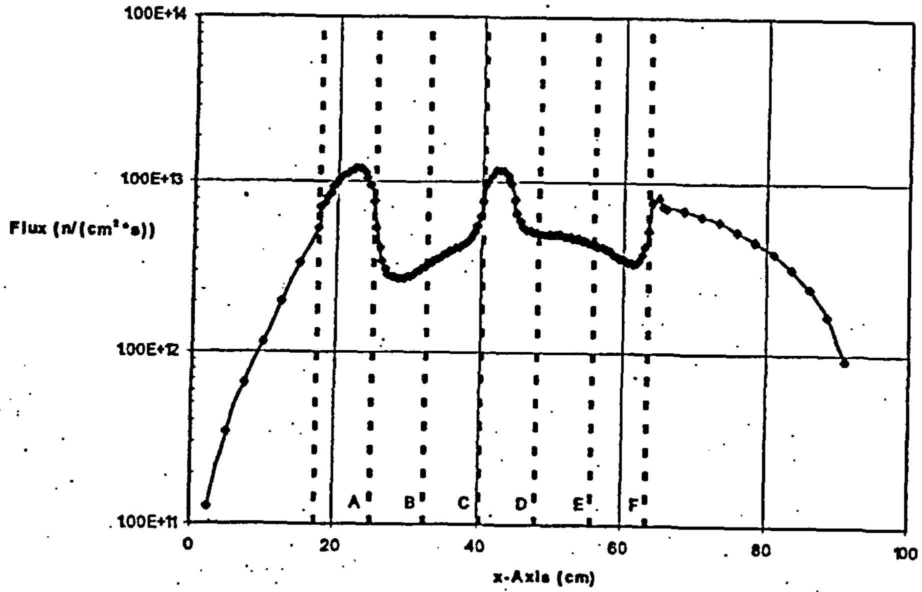


Figure 4.22 TAMU Initial Cycle: Thermal Neutron Flux (E<0.42 eV) through the Transient Rod at max axial power peaking position (76.7 cm), BOL

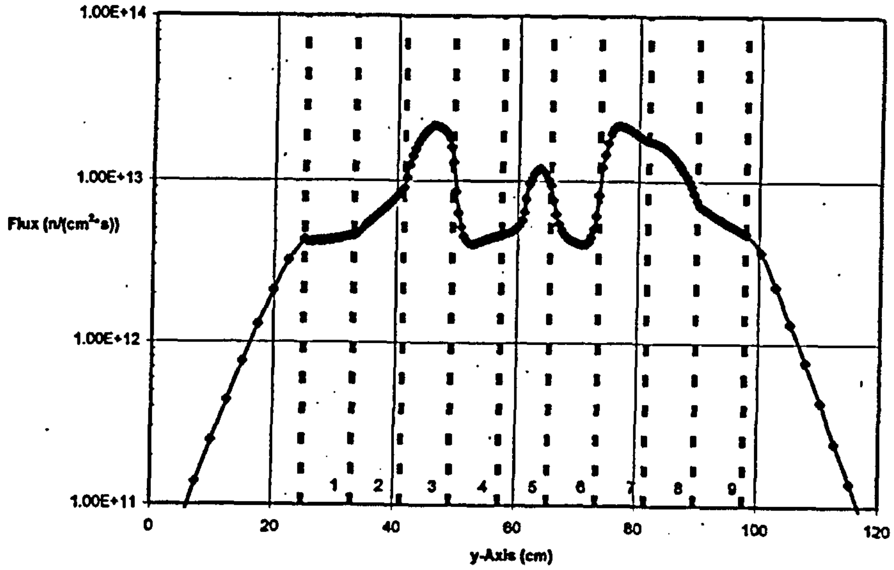


Figure 4.23 TAMU Initial Cycle: Thermal Neutron Flux (E<0.42 eV) through the Transient Rod at max axial power peaking position (76.7 cm), BOL

5. Reactor Coolant System

The HEU to LEU conversion does not require any changes to the reactor coolant system.

6. Engineering Safety Features

There will be no changes to the existing engineering design safety features, with the exception of the addition of an electro-mechanical switch that will prevent energizing the pulse circuit in the event the mechanical pulse stop is not installed.

7. Instrumentation and Control

The HEU to LEU conversion will include an addition to the instrumentation and control circuits in order to activate the electro-mechanical switch that will prevent energizing the pulse circuit in the event the mechanical pulse stop is not installed.. See Section 6 above.

8. Electrical Power System

The HEU to LEU conversion does not require any changes to the electrical power systems.

9. Auxiliary System

Existing procedure will be used for fuel storage.

10. Experimental Facility and Utilization

The HEU to LEU conversion does not require any changes to experimental facility and utilization of the NSCR.

11. Radiation Protection and Radioactive Waste Management

The HEU to LEU conversion does not require any changes to the radiation protection and radioactive waste management of NSCR facility.

12. Conduct of Operation

12.1 Organization and Staff Qualification
The HEU to LEU conversion does not require any changes to the organization and staff qualification of the NSCR personnel.

12.2 Procedures
The HEU to LEU conversion does not require any fundamental changes to the NSCR standard operating procedures with the exception of any references to HEU flip or standard fuel.

12.3 Operator Training and Re-qualification
The HEU to LEU conversion does not require any changes to the NSCR operator training and re-qualification program with the exception of updating training materials to describe new fuel type and accidental analysis.

12.4 Emergency Plan
The HEU to LEU conversion does not require any changes to the existing NSCR Emergency Plan.

12.5 Physical Security
The HEU to LEU conversion does not necessitate any changes to the existing security plan at the time of conversion. [redacted]

12.6 Reactor Reload Consideration
As already discussed in Section 4.5.6, the TAMU LEU (30/20) core is a "lifetime" core. It will have capability to operate at 1.0 MW on demand for 35 years following a weekly schedule of ≤70 MWHr. However, it is conceivable that one or more fresh instrumented fuel elements (IFE) may be acquired and installed in a heavily burned core. The concern for excessive power peaking is evaluated and discussed in the following.

To assess extreme power peaking that can occur when adding fresh fuel elements to a depeleted core of fuel elements, the following case was analyzed. Three fresh fuel elements were added to the TAMU core after 2000 MW-days of burnup. Because the IFEs after such a burnup may have failed and need to be replaced, the IFE locations were selected for placing fresh fuel elements. Specifically, fresh fuel was placed in locations 6D4, 5E4, and 4D4. The resulting rod power factors for these locations are 1.401, 1.502, and 1.546. All of these rod power factors are lower than the maximum rod power factor of 1.565 which occurs at BOL. Peak temperatures during steady state operation would

likewise be lower than peak temperatures at BOL. For pulsing transients, the peak temperature is dependent on the peak power factor ($\hat{P}_{rod}/\overline{P}_{core}$). For the three fresh fuel elements in 6D4, 5E4, and 4D4, the peak power factor is 2.777, 3.064 and 4.585 respectively. The peak power factor of 4.585 is higher than the peak power factor at BOL which means either that the maximum allowable reactivity insertion during a pulse would be lower or that a new IFE should not be placed in 4D4. The power peaking associated with fresh fuel reload would be lessened if heavily depleted fuel in the core center is replaced with lightly depleted fuel from the core periphery and additional fresh fuel elements are placed in the core periphery.

12.7 LEU Startup Plan

A detailed Startup Plan together with Acceptance Criteria is presented in Appendix A.1.

13. Safety and Accident Analysis

13.1 Safety Analysis

General Discussion and Summary

The safety of TRIGA fuel is due entirely to the design features. The safety features of a standard TRIGA fueled core are well known. Each of the LEU fuel types is designed to replace a standard fuel (8.5 wt % U, 20% enriched) element, as regards reactivity; that is, 100 fresh standard fuel elements in a compact configuration is intended to have about the same core excess reactivity as 100 of the more heavily loaded TRIGA fuel elements.

As part of the RERTR program, various tests were performed on high-uranium content, low-enriched TRIGA fuels and the test results submitted to the NRC. The NRC concluded in their Safety Evaluation Report that both the 20-20 and 30-20 uranium-zirconium hydride fuels "are generally acceptable for use in other licensed TRIGA reactors, with the provision that case-by-case analyses discuss individual reactor operating conditions in applications for authorization to use them" (Ref. 13).

In the present document, it is shown that one-for-one, fresh TRIGA FLIP fuel and fresh TRIGA LEU (30/20) fuel behave very similar as regards cold, clean critical. However, both these types of TRIGA fuel react strongly to in core water filled regions. In PRNC, all control rods had water followers that were accounted for in the pulsing model. In TAMU, all control rods near core center will have air or metal followers; however, two water filled regions representing future experiments generated high pulsed temperatures in certain in-core fuel rods.

13.2 Safety Limits

The safety of the operating TRIGA reactor system with LEU (30/20) fuel is related directly to the maximum temperature of the fuel and the continued availability of coolant. As demonstrated for all TRIGA fuel elements, the Safety Limit for water cooled fuel is taken conservatively as 1150°C. The Safety Limit for these fuel elements when air cooled is 950°C.

As analyzed in this report, all proposed reactor operations will involve low fuel temperatures with large margins of safety. The peak fuel temperature in steady state operation at 1.0 MW (1.3 MW momentarily for the purpose of testing power level scrams) is 373°C (440°C). Both of these temperatures have large margins of safety for 1150°C. In normal pulsing, the TAMU facility has chosen 830°C as the limit for the peak fuel temperature (with good margin of safety) and controls the reactivity insertion to maintain this limit. The limiting peak pulsing fuel temperature (830°C) has been chosen to address the problem of hydrogen migration resulting from long term high power operation.

The two power level scrams (125% of 1.0 MW) are used to assure a reactor power at a level that gives acceptable fuel temperatures as noted above.

The fuel temperature scram (1) (525°C) is maintained in all modes of operation. The high power level scrams (2) (125%) are effective in steady state mode of operation.

In addition to the protection provided by the several, redundant scrams, administrative procedures and written operating procedures contribute additional limits on operation to protect the reactor, the facility, and the public.

13.3 Evaluation of LSSS for TAMU LEU (30/20) Fuel

Steady State Mode

The value of the Limiting Safety System Setting (LSSS) is chosen to prevent the TRIGA Safety Limit (1150°C) from being reached in any mode of operation. The limiting safety system setting in an instrumented fuel element has been selected as 525°C. The location of the fuel cluster containing the instrumented fuel element shall be chosen to be as close as possible to the hottest fuel element in the core. (The hottest element in the core is (5D3) in Fig. 4.2 adjacent to the transient rod. In the present analysis, the instrumented fuel element (IFE) is located at (5E4). Other locations can be chosen for the IFE.) The LSSS temperature setting is smaller than the Safety Limit by an amount to account for several factors, including:

$$\left.\begin{array}{ll} \text{i.} & \text{Accuracy of temperature calibration} \\ \text{ii.} & \text{Precision of electronic readout/scram circuitry} \\ \text{iii.} & \text{Account taken of location of sensing tip of thermocouple} \\ & \text{0.3 inch from axial center line of IFE} \\ \text{iv.} & \text{Difference in peak temperature in IFE compared to that in} \\ & \text{the hottest fuel element} \end{array}\right\} \begin{array}{c}\text{Safety}\\ \text{Margin}\end{array}$$

The basis is essentially the same as appears in the current TAMU license and Technical Specifications. The basis for selecting 525°C as the limiting safety system temperature setting is the following. The limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated, preventing the safety limit from

being exceeded. A part of this margin is used to account for the difference between the maximum core temperature and measured temperatures resulting from the actual location of the thermocouple, 0.3 inches from the axial center line of the fuel element. If the instrumented fuel element were located in the hottest position in the core, the difference between the true and measured temperatures would be small. However, this fuel position is not available for technical reasons due to the core configuration with the 4-rod fuel clusters and to the location of the transient rod. The location of the instrumented elements is therefore restricted to the positions close to the central fuel element. Calculations indicate that, for this case, the true temperature at the hottest location in the core at 1.0 MW will differ from the measured temperature by about 13.4% (373/329 = 1.134) (See Table 4-11). Thus, for the steady state mode of operation, if the temperature in the thermocouple element were to reach the trip setting of 525°C, the true temperature at the hottest location in the core would be less than 600°C, providing a safety margin of at least 550°C for LEU (30/20) type elements. At a steady state reactor power of 1.3 MW (for momentary testing and surveillance measurements) the peak fuel temperature $\hat{T}$ in the hottest fuel in the TAMU LEU (30/20) core would only be at most 440°C. These resulting safety margins are ample to account for any remaining uncertainty in the accuracy of the fuel temperature measurements and any overshoot in reactor power resulting from a reactor transient during steady state mode operation.

Pulse Mode

In the pulse mode of operation, the same limiting safety system setting will apply. However, the temperature channel will have no effect on limiting peak powers generated because of the relatively long time constant (seconds) for the recorded temperature as compared with the width of the pulse (few milliseconds). In this mode, however, the temperature trip, if activated, will cause all scrammable rods to fall and will act to reduce the amount of energy generated in the entire pulse transient by cutting the "tail" of the energy transient even if the pulse rod remains stuck in the fully withdrawn position.

13.4 Maximum Allowable Pulsed Reactivity Insertion

In Sections 4.5.11 and 4.5.12, the pulsing performance of the LEU (30/20) core has been reviewed for both beginning of core life and at 2000 MWD burnup. It is demonstrated therein that the beginning of life limiting peak pulse fuel temperature of 830°C is produced by a reactivity insertion of $2.10. At a core burnup of 2000 MWD, a reactivity insertion of $2.10 will produce slightly less peak fuel temperature; namely, ~810°C.

In view of the small change in peak fuel temperatures with core burnup, a maximum allowable pulsed reactivity insertion of $2.10 is a reasonable choice.

13.5 Accident Analysis

Analysis Changes to DBA/MHA Event

In the current TAMU SAR, the Design Basis Accident (DBA) is defined as the loss of integrity of the fuel cladding for one fuel element and the simultaneous loss of pool water

resulting in fission product release. NUREG/CR-2387 (Ref. 14) suggests, and NRC accepts, that for a 1.0 MW TRIGA reactor, the DBA is the release in air of fission products from a single irradiated fuel element. The loss of pool water is typically treated separately as a Loss of Coolant Accident (LOCA). In the TAMU SAR, the simultaneous loss of coolant and rupture of a single fuel element in air is considered a DBA-MHA event.

In this section, the radiological impact of the loss of fission products from a single TRIGA fuel element is reviewed. To compare the relative abundance of fission products, the pertinent operating parameters are compared for a FLIP core and a LEU (30/20) core. For the LEU (30/20) core, the core average fuel temperature at 1.0 MW is 237°C (Table 4-11) compared to 234°C (Table 4-8) for a FLIP core. The major difference lies in the fission product inventory due to burnup. A TAMU 90-element FLIP core has a life of about 2350 MWD versus ~2000 MWD for the TAMU LEU (30/20) core. The energy burnup capability based on >50% of the U-235 has been evaluated as 77 MWD/fuel element for FLIP fuel (Ref. 13). For TAMU LEU (30/20) fuel, the energy burnup capability based on >50% of the U-235 (Ref 8-3) gives 57 MWD/fuel element. The evaluated dose resulting from the fission product release from a single FLIP fuel element was found acceptable by the NRC.

The hottest fuel element in the LEU core at the end of life will contain less activity than assumed in the DBA analysis for FLIP fuel. Thus, the radiological impact from a DBA event on the members of the public and the facility workers will be less than that evaluated for FLIP fuel.

In the revised TAMU SAR submitted to NRC in 2003, the thyroid dose to an individual worker who lingered in the Reactor Hall for one hour after rupture of a fuel element was listed as 49 Rem. It might be better to indicate that workers in the Reactor Hall when the fuel element ruptures would promptly leave this hall in less than an estimated 5 minutes (not 1 hour), thus receiving a thyroid dose of no more than 4.1 Rem ($\frac{5}{60}$ x 49).

In conclusion, the prior, accepted analysis of the fission product release in air from one TRIGA FLIP fuel element bounds the results expected from the LEU (30/20) core.

Analysis of Changes to LOCA Event

A detailed analysis (Ref. 15) has evaluated the safety of irradiated fuel elements in a loss-of-coolant accident (LOCA) for long term operation at 1.0 MW. With power levels no higher than 21 kW/element, air cooling is sufficient to prevent excessive fuel temperatures with essentially no delay between reactor scram and loss of coolant. For power densities up to 23 kW/element, a 15-minute delay is required between reactor scram and loss of coolant. TAMU has further refined the analysis (Ref. 16) to account for the fact that the 1.0 MW operation is limited to ≤70 MWHr/week. As a result, fuel elements with power densities up to 28 kW/element can be safely cooled by air with a 15-minute delay between reactor scram and complete loss of coolant.

For the LEU (30/20) core, operating at 1.0 MW, the largest power per element is

17.6 kW/element. This is well below the 21 kW/element noted above as not requiring any delay between reactor scram and complete loss of coolant. (It will be noted further that it is incredible to drain over 100,000 gallons of coolant water from the reactor tank in zero time).

Several of the assumptions in the Accident Analysis (Ref. 15) are tabulated in Table 13-1 for FLIP fuel and for the TAMU LEU (30/20) fuel. Each of the listed parameters for the LEU fuel will reduce the estimated pressure inside the fuel clad and increase the safety margin for the LOCA event.

TABLE 13-1. Comparison of Assumptions for FLIP (HEU) and LEU (30/20) Fuel for LOCA Event

Parameter	FLIP (HEU)	LEU (30/20)
Energy Burnup Capability (50% of the U-235)	77 MWD /element	57 MWD/element
H/Zr Ratio $\hat{T}$ (steady state)	1.7	≤1.65
Lifetime Operation	600°C	≤400°C
Air Back Fill	Continuous	≤70 MWHr/week
	Present in analysis at EOL	Not present at EOL*

*After a few hours operation at full power, oxides and nitrides deplete the air backfill.

In conclusion, the margin of safety during a LOCA event with the TAMU LEU (30/20) fuel is greater than that previously evaluated for the TAMU FLIP (HEU) fuel. Based on the previous analysis (Ref. 13), the peak power/fuel element of 17.6 kW per element lies comfortably below the TAMU limit established for a maximum weekly operation of 70 MWHr and a 15 minute delay time before complete loss of cooling water. In fact, for power/fuel element ratio up to 21 kW/element, all fuel can be cooled by air flow alone with no delay between reactor scram and complete loss of coolant water.

Accidental Pulsing from Full Power

BOL, Beginning of Core Life

The rapid insertion of a large amount of positive reactivity in the reactor operating at 1.0 MW is postulated. The method of inserting this reactivity is either by the ejection of an inserted transient rod or unplanned removal of a two-dollar experiment (an experiment that is required by Technical Specifications to be securely mounted in core). The Technical Specification limits the maximum insertion of a transient rod to that value which produces a maximum allowed fuel temperature no higher than the Safety Limit (1150°C). This reactivity is about $2.95 for the TAMU LEU (30/20) core. The full worth of the transient rod will be limited to reactivities that produce temperatures less than the Safety Limit (1150°C) by installing a mechanical stop which physically prevents the withdrawal of the transient rod at reactivities greater than $2.95. Furthermore, the reactor will not be operable in the Pulse Mode if the mechanical stop is not installed completing an electro-mechanical switch (interlock).

Pulsing from full reactor power (1.0 MW) would be clearly an accident. In addition to *the mechanical pulse stop safety interlock there is another safety interlock that prevent* application of air to the piston for any reactor power above 1 kW. In addition, administrative procedures prevent the reactor operator from switching the Mode Switch to PULSE during high power operation and pushing the PULSE FIRE button. For the operator to do so would clearly violate three standard operating procedures.

The sequence of events leading to the postulated transient rod ejection accident at BOL is the following:

The consequences of the above sequence of events are the following:

1. The reactor power increases from 1.0 MW to a peak pulsed power of about 827 MW.
2. The maximum fuel temperature (864°C) is reached immediately after the peak power.
3. The energy release is about 15.8 MW-sec in about 1.0 second when the maximum measured fuel temperature ($\hat{T}_{03} = 614°C$) is reached.
4. At peak fuel temperatures substantially below 1150°C, the strength of the clad maintains clad integrity so long as it remains water cooled.

This accident can be viewed in a number of ways. The transient rod reactivity is limited to about $2.95 by the pulse stop, not the full stroke value of $3.21. The equilibrium pressure of hydrogen over the fuel is not achieved during the abbreviated pulse. It was incorrectly assumed in the earlier analyses that the back-fill air left in the fuel during manufacture is still present at EOL, whereas in fact it has disappeared, forming oxides and nitrides during the first days of operation at full power. With the action of the dual

power scrams at 125%, the scram of the control rod bank starts even before peak pulsed power is attained.

The pulsing calculations from power have involved hand calculations since BLOOST cannot handle a pulse from power. BLOOST is a zero-dimensional, combined reactor kinetics-heat transfer code. It cannot handle the "inverted U" temperature distribution in fuel operating in steady state coupled with the "U shaped" power distribution in a pulsed fuel element. BLOOST calculates an average core temperature as a function of time.

Calculations were made to establish the average fuel temperature at the steady state starting power of 1.0 MW. A value of 237°C was determined for $\overline{T}_{core}$.

The BLOOST calculations indicate that the highest average fuel temperature in the pulsed core immediately after the transient pulse is 356°C. From three-dimensional diffusion theory calculations, the peak-to-average power ratio was determined for steady state operation to be 4.2. Although the highest temperatures occur at the center of the hottest fuel element during 1 MW steady state operation (373°C) and before the pulse, the maximum fuel temperature after pulsing occurs at the edge because of the large power peak.

The peak-to-average value at the edge of the hottest pulsed element is 4.2. Using these power ratios and considering the energy release during the transient superimposed on the energy density levels under steady state, coupled with the volumetric heat content of the fuel, a maximum fuel temperature of 864°C was obtained based on the average core temperatures computed by BLOOST.

An alternative method of producing the accidental pulse from full power [redacted] Although the Technical Specification requires [redacted] the removal time is typically assumed to be 0.3 second (considerably longer than the 0.1 second for the engineered transient rod drive). The much slower withdrawal time will result in activating the 125% power scrams at a lower portion of the reactivity insertion curve. This will result in lower peak power and lower peak fuel temperatures than those for a $2.95 transient rod. Thus, neither accident endangers the reactor.

To review, the accidental pulsing of the transient rod requires the following:

1. [redacted]
2. [redacted]

Note: Both must occur for the accident to occur.

[redacted]

1. [redacted]

Note: [redacted]

EOL Pulsing from Full Power at 2000 MWD Burnup

The BLOOST code was run with input appropriate for the LEU core at 2000 MWD. Pulsing results were obtained for a series of reactivity insertions under the same conditions as set forth in the above section. Table 13-2 lists these results along with those for the beginning of core life.

TABLE 13-2. Comparison of Results from Pulsing from 1.0 MW at Beginning of Life (BOL) and End of Life (EOL)

Parameter	BOL			EOL (2000 MWD)		
$\Delta k/k\beta$ ($)	2.10	2.95	3.21	2.10	2.95	3.21
$\hat{T}$ (°C)	740	864	866	740	888	889
$\overline{T}_{core}$ (°C)	356	423	423	370	454	456
$\hat{T}_{0.3}$ (°C)	555	614	614	576	654	654
MW-sec	10.9	15.8	15.9	12.3	18.8	18.8
$\hat{P}$ (MW)	471	827	830	511	983	986

In Table 8-2, the effects are evident for shifts in the power distribution with burnup for the core. The measured temperature ($\hat{T}_{0.3}$) increases from 555°C to 576°C with burnup. The peak temperature in the hottest fuel element rises slightly, from 564°C to 888°C; but even 888°C is well below the 1150°C Safety Limit.

The conclusion is that accidental pulsing from full power is not a hazard for the reactor, either at Beginning- or End-of 2000 MWD burnup.

14. REVISED TECHNICAL SPECIFICATIONS

The following is a revision as needed for Section 14 from the TAMU License No R-83 dated March 1983 including revisions through Amendment No. 15 (1999). In any changes were made in a particular section then all the technical specifications for that section are included. There are no changes necessary are proposed to 14.4 or 14.6.

14.1 Definitions

14.1.1 Abnormal Occurrence
An abnormal occurrence is an unscheduled incident or event that the Nuclear Regulatory Commission determines is significant from the standpoint of public health or safety.

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

14.1.3 Channel
A channel is the combination of sensors, lines, amplifiers and output devices, which are for measuring the value of a parameter.

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

14.1.3.2 Channel Calibration
A channel calibration is an adjustment of the channel such that its output corresponds, with acceptable accuracy, to known values of the parameter that the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and constitutes a channel test.

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

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

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

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

14.1.7 Experimental Facilities
Experimental facilities shall mean beam ports, including extension tubes with shields,

thermal columns with shields, vertical tubes, through tubes, in-core irradiation baskets, irradiation cell, pneumatic transfer systems and in-pool irradiation facilities.

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

14.1.9 FLIP Core
A FLIP core is an arrangement of TRIGA-FLIP fuel in a reactor grid plate.

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

14.1.11 Fuel Element
A fuel element is a single TRIGA fuel rod of LEU type.

14.1.12 Instrumented Element
An instrumented element is a special fuel element in which a sheathed chromel-alumel or equivalent thermocouple is embedded in the fuel near the horizontal center plane of the fuel element at a point approximately 0.3 inch from the center of the fuel body.

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

14.1.14 Limiting Safety System Setting
The limiting safety system setting is the setting for automatic protective devices related to those variables having significant safety functions.

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

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

14.1.17 Movable Experiment
A movable experiment is one for which it is intended that the entire experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

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

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

14.1.20 Operational Core – Steady State
A steady state operational core shall be an LEU core for which the core parameters of shutdown margin, fuel temperature and power calibration have been determined.

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

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

14.1.23 Reactivity Worth of an Experiment
The reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter the experiment position or configuration.

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

14.1.25 Reactor Operating
The reactor is operating whenever it is not secured.

14.1.26 Reactor Safety Systems
Reactor safety systems are those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action. Manual protective action is considered part of the reactor safety system.

14.1.27 Reactor Secured
A reactor is secured when:

1) It contains insufficient fissile material or moderator present in the reactor and adjacent experiments to attain criticality under optimum available conditions of moderation and reflection, or

2) The reactor console is secured, and
   a) No work is in progress involving core fuel, core reflector material, installed control rods, or control rod drives unless they are physically decoupled from the control rods, and
   b) No experiments in or near the reactor are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value of

one dollar.

14.1.28 Reactor Shutdown
The reactor is shut down if it is subcritical by at least one dollar with the reactor at ambient temperature, xenon-free and with the reactivity worth of all experiments included.

14.1.29 Reportable Occurrence
A reportable occurrence is any of the following that occurs during reactor operation:

1) Operation with actual safety system settings for required systems less conservative than the limiting safety-system settings specified in the Technical Specifications 14.2.2.

2) Operation in violation of limiting conditions for operation established in the technical specifications.

3) A reactor safety system component malfunction that renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns (Note: Where components or systems are provided in addition to those required by the technical specifications, the failure of the extra components or systems is not considered reportable provided that the minimum number of components or systems specified or required perform their intended reactor safety function.)

4) An unanticipated or uncontrolled change in reactivity greater than one dollar.

5) Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or containment boundary (excluding minor leaks) where applicable which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both.

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

14.1.30 Rod-Control
A control rod is a device fabricated from neutron absorbing material and/or fuel that is moved up or down to control the rate of a nuclear reaction. It may be coupled to its drive unit allowing it to perform a safety function (scram) when the coupling is disengaged.

14.1.31 Rod-Regulating
The regulating rod is a low worth control rod used primarily to maintain an intended power level that need not have scram capability and may have a fueled follower. Its

percent withdrawal may be varied manually or by the servo-controller.

14.1.32 Rod-Shim Safety
A shim-safety rod is a control rod having an electric motor drive and scram capabilities. It may have a fueled follower section.

14.1.33 Rod-Transient
The transient rod is a control rod with scram capabilities that is capable of providing rapid reactivity insertion to produce a pulse.

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

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

14.136 Scram Time
Scram time is the time measured from the instant a simulated signal reaches the value of the LSSS to the instant that the slowest scrammable control rod reaches its fully inserted position.

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

14.1.38 Shall, Should and May
The word "shall" is used to denote a requirement: the word "should" to denote a recommendation; and the word "may" to denote permission, neither a requirement nor a recommendation. In order to conform to this standard, the user shall conform to its requirements but not necessarily to its recommendations.

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

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

14.1.41 True Value
The true value is the actual value of a parameter.

14.1.42 Unscheduled Shutdown
An unscheduled shutdown is any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions that could adversely affect safe operation. It does not include shutdowns that occur during testing or check out operations.

14.2 Safety Limit and Limiting Safety System Setting

14.2.1 Safety Limit-Fuel Element Temperature

Applicability

This specification applies to the temperature of the reactor fuel.

Objective

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

Specifications

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

Bases

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

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

element cladding is water cooled.

14.2.2 Limiting Safety System Setting

Applicability

This specification applies to the scram settings that prevent the fuel element temperature from reaching the safety limit.

Objective

The objective is to prevent the fuel element temperature safety limits from being reached.

Specification

a) For steady state operation:

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

b) For pulsing operation:

> The limiting safety system setting shall be 975°F (525°C) as measured in an instrumented fuel element. The instrumented element shall be located adjacent to the central bundle with the exception of the corner positions. Pulsing is not allowed if this limiting safety system channel is not operable.

Basis

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

The temperature safety limit for LEU fuel is 2100°F (1150°C). Due to various errors in measuring temperature in the core, it is necessary to arrive at a Limiting Safety System Setting (LSSS) for the fuel element safety limit that takes into account these measurement errors. One category of error between the true temperature value and the measured temperature value is due to the accuracy of the fuel element channel and any overshoot in reactor power resulting from a reactor transient during steady state mode of operation. Although a lesser contributor to error, a minimum safety margin of 10% was applied on an absolute temperature basis. Adjusting the fuel temperature safety limit to degrees Kelvin, °K, and applying a 10% safety margin results in a safety limit reduction of 150°C. Applying this first margin of safety, the safety setting would be 1000°C for LEU. However, to arrive at the final LSSS it is necessary to allow for the difference between the measured temperature value and the peak core temperature, which is a function of the location of the thermocouple within the core. For example, if the thermocouple element were located in the hottest position in the core, the difference between the true and measured temperatures would be only a few degrees since the thermocouple junction is at the mid-plane of the element and close to the anticipated hot spot. However, at the

TAMU this core position is not available due to the location of the transient rod. For the TAMU the location of the instrumented elements is therefore restricted to the positions closest to the central element. Calculations indicate that, for this case, the true temperature at the hottest location in the core will differ from the measured temperature by no more than 40%. When applying this 40% worst case measurement scenario and considering the previously mentioned sources of error between the true and measured values, a final LSSS temperature of 975°F (525°C) is imposed on operation. Viewed on an absolute temperature scale, °K, this represents a 37% safety margin in the LEU safety limit.

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

The reactor high power level safety limit for TRIGA LEU fuel during steady state operation will be a measured power of 125% (1.25 MW) on either of the two power safety channels. The high power safety drawers are aligned with the linear power monitor during annual calibration for nominal 1.0 MW operation. These safety channels independently measure reactor power and have been a part of Texas A&M University reactor operation safety systems for over 30 years. During the years of 1 MW operation of the TAMU TRIGA reactor the LSSS temperature limit of 975°F (525°C) has never been reached although several scrams of the safety channels have been recorded indicating that the LSSS of 125% for the high power level safety channels is more conservative than the temperature setting.

14.3 Limiting Conditions for Operation

14.3.1 Reactor Core Parameters

14.3.1.1 Steady State Operation

Applicability

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

Objective

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

Specifications

The reactor power level shall not exceed 1.3 megawatts (MW) under any condition of operation. The normal steady state operating power level of the reactor shall be 1.0 MW. However, for purposes of testing and calibration, the reactor may be operated at higher power levels not to exceed 1.3 MW during the testing period.

Basis

Thermal and hydraulic calculations indicate the TRIGA fuel may be safely operated up to power levels of at least 2.0 MW with natural convection cooling.

14.3.1.2 Pulse Mode Operation

Applicability

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

Objective

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

Specification

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

Basis

TRIGA fuel is fabricated with a nominal hydrogen to zirconium ratio of 1.6 for LEU fuel. This yields delta phase zirconium hydride that has high creep strength and undergoes no phase changes at temperatures over 1000°C. However, after extensive steady state operation at 1 MW the hydrogen will redistribute due to migration from the central high temperature regions of the fuel to the cooler outer regions. When the fuel is pulsed, the instantaneous temperature distribution is such that the highest values occur at the surface of the element and the lowest values occur at the center. The higher temperatures in the outer regions occur in fuel with a hydrogen to zirconium ratio that has now substantially increased above the nominal value. This produces hydrogen gas pressures considerably in excess of the expected for $ZrH_{1.6}$. If the pulse insertion is such that the temperature of the fuel exceeds 874°C, then the pressure will be sufficient to cause expansion of microscopic holes in the fuel that grows with each pulse. The pulsing limit of 830°C is obtained by examining the equilibrium hydrogen pressure of zirconium hydride as a function of temperature. The decrease in temperature from 874°C to 830°C reduces hydrogen pressure by a factor of two, which is an acceptable safety factor. This phenomenon does not alter the safety limit since the total hydrogen in a fuel element does not change. Thus, the pressure exerted on the clad will not be significantly affected by the distribution of hydrogen within the element.

In practice the pulsing limit of 830°C will be translated to a reactivity insertion limit for the LEU core.

Initially, the pulse insertions shall be increased by small increments to a maximum of $2.00 to allow an extrapolation of peak temperatures, thereby establishing the maximum

allowed pulse insertion for the LEU core..

14.3.1.3 Shutdown Margin

Applicability

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

Objective

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

Specifications

The reactor shall not be operated unless the shutdown margin provided by control rods is greater than \$0.25 with:

a) The highest worth non-secured experiment in its most reactive state,

b) The highest worth control rod and the regulating rod (if not scrammable) fully withdrawn, and

c) The reactor in the cold condition without xenon.

Basis

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

14.3.1.4 Core Configuration Limitation

Applicability

This specification applies to a full LEU core.

Objective

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

Specifications

a) The TRIGA core assembly shall be LEU.

b) The reactor shall not be taken critical with a core lattice position water-filled except for positions on the periphery of the fuel region. Water filled holes in the

inner fuel region shall not be permitted.

c) The instrumented element, if present and serving as the Limiting Safety System, shall be located adjacent to the central bundle with the exception of the corner positions.

Bases

a) Reference: 14.2.2 Limiting Safety System Setting.

14.3.1.5 Maximum Excess Reactivity

Applicability

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

Objective

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

Specifications

The maximum reactivity in excess of cold, xenon-free critical shall not exceed 5.5% $\Delta k/k$ ($7.85).

Basis

Although maintaining a minimum shutdown margin at all times ensures that the reactor can he shut down, that specification does not address the total reactivity available within the core. This specification, although over-constraining the reactor system, helps ensure that the licensee's operational power densities, fuel temperatures and temperature peaks are maintained within the evaluated safety limits. The specified excess reactivity makes up for negative reactivity due to power coefficients samarium poisoning, xenon poisoning, experiments, and fuel depletion.

14.3.2 Reactor Control and Safety Systems

14.3.2.1 Reactor Control Systems

Applicability

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

Objective

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

Specifications

The reactor shall not be operated unless the measuring channels listed in the following table are operable.

Measuring Channel	*Min. No. Operable*	*Operating Mode*	
		S.S.	*Pulse*
High Power Level	2	X	
Fuel Element Temperature	1	X	X
Linear Power Level	1	X	
Log Power Level	1	X	
Integrated Pulse Power	1		X

Bases

Fuel temperature displayed at the control console gives continuous information on this parameter, which has a specified safety limit. The power level monitors assure that the reactor power level is adequately monitored for both steady state and pulsing modes of operation. Monitoring of the high power level channel is important since it is used to ensure the temperature safety limit is not reached, since the power level is related to the fuel temperature.

14.3.2.2 Reactor Safety Systems

Applicability

This specification applies to the reactor safety-system circuits.

Objective

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

Specifications

The reactor shall not be operated unless the safety circuits described in the following table are operable.

Safety Channel	*Number Operable*	*Function*	*Effective Mode*	
			S.S.	*Pulse*
Fuel Element Temperature	1	SCRAM @ LSSS (975°F)(525°C)	X	X
High Power Level	2	SCRAM @ 125%	X	
High Power Level Detector Power Supply	2	SCRAM on loss of supply voltage, or low power supply.	X	

Console Scram Button	1	SCRAM at operator's discretion.	X	X
Preset Timer	1	Transient rod scram time to be 15 seconds or less after pulse.		X
Log Power	1	Prevent pulsing above 1 kW		X
Log Power	1	Prevent withdrawal of shim safeties at <4 x 10^{-3} W (Low count interlock).	X	
Transient Rod position	1	Prevent application of air in steady state mode unless transient rod is fully inserted.	X	
Shim Safeties & Regulating Rod Position	1	Prevent withdrawal of shim safeties and regulating rod while in pulse mode.		X

<u>Bases</u>

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

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

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

The interlock to prevent pulsing at powers above 1 kW.

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

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

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

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

14.3.2.3 Scram Time

<u>Applicability</u>

This specification applies to the time required for the scrammable control rods to be fully inserted from the instant that the fuel temperature safety channel or the high power level safety channel variable reaches their respective Limiting Safety System Setting.

<u>Objective</u>

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

Specification

The scram time measured from the instant a simulated signal reaches the value of the LSSS to the instant that the slowest scrammable rod reaches its fully inserted position shall not exceed 1.2 seconds.

Basis

This specification assures that the reactor will be promptly shutdown when a scram signal is initiated. Experience and analysis have indicated that for the range of transients anticipated for a TRIGA reactor, the specified scram time is adequate to assure the safety of the reactor.

14.4 Limiting Conditions for Operation

No changes to this section or required or proposed.

14.5 Design Features

14.5.1 Reactor Fuel

Applicability

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

Objective

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

Specifications

a) TRIGA-LEU fuel

1) The individual unirradiated LEU fuel elements shall have the following characteristics:

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

3) Hydrogen-to-zirconium ratio (in the ZrH_x): nominal 1.6 H atoms to 1.0 Zr atoms with a maximum H to Zr ratio of 1.65.

4) Natural erbium content (homogeneously distributed): nominal 0.90 wt%. (See bases below for contract specifications.)

5) Cladding: 304 stainless steel, nominal 0.020 inch thick.

Bases

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

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

The maximum hydrogen-to-zirconium ratio of 1.65 could result in a maximum stress under accident conditions in the fuel element clad about a factor of two greater than for a hydrogen-to-zirconium ratio of 1.60. This increase in the clad stress during an accident would not exceed the rupture strength of the clad.

14.5.2 Reactor Core

Applicability

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

Objective

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

Specifications

a) The core shall be an arrangement of TRIGA uranium-zirconium hydride fuel-moderator 4-rod, 3-rod, and 2-rod clusters positioned in the reactor grid plate.

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

Bases

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

b) The core will be assembled in the reactor grid plate that is located in a pool of light water. Water in combination with graphite reflectors can be used for

neutron economy and the enhancement of experimental facility radiation requirements.

14.6 Administrative Controls

No changes to this section or required or proposed.

Appendix A1

A.1. LEU (30/20) Startup Plan

A.1.1 Initial Criticality

Based on practical experience derived from the criticality approach with several other TRIGA reactors, criticality is expected with a loading of 58-68 fresh, 30/20 fuel elements including the fuel follower control rods. TRIGA 30/20 fuel is defined as 30 w% U, 19.75% enriched in U-235.

The loading of fuel elements to obtain criticality will be accomplished using the standard inverse multiplication curve (1/M) approach. This is based on the fact that subcritical multiplication is given as

$$M = 1/(1-k)$$

from which one obtains

$$1/M = 1-k$$

where k ranges from 0 (no fuel) to 1 (at criticality). The experimental values for 1/M subcritical multiplication are given by the count rate with no fuel C_0 divided by C_n for loading step n. However, for the present 1/M application for approach to critical, the value C_0 can start at any convenient loading point. For the TRIGA application, C_0 is usually the count rate with fuel follower control rods (installed but withdrawn from the core) and instrumented fuel elements installed in the core together with the fuel in the 3-rod clusters that contain the FFCRs and instrumented fuel.

<u>Acceptance Criteria</u>: The 1/M criticality is expected with a fuel loading between 58 and 68 fuel elements.

A.1.2 Critical Mass and Criticality Conditions for the 30/20 LEU Core

Measurements Upon Attaining Criticality

The core excess reactivity shall be determined upon reaching criticality with all control rods (including fuel follower control rods) fully withdrawn, using the period method.

The estimated control rod reactivity worth for each control rod is obtained using the Rod Drop technique with either the Reactivity Computer or the classical rod drop negative period measurements with a stopwatch.

<u>Acceptance Criteria</u>: The Rod Drop reactivity worths for the five, scrammable control rods are expected to lie between $1.50 and $3.00.

A.1.3 Initial Control Rod Calibration Tests

The TAMU 1 MW TRIGA reactor with LEU 30/20 fuel is provided with six control rods: 4 fuel-followed control rods, one air-followed Transient rod and one water-followed Regulating rod.

The reactivity insertion procedure is used in conjunction with the Reactivity Computer to calibrate each control rod as a function of its withdrawal distance from the fully inserted position in the core. For this procedure, the available core excess reactivity must at least equal the reactivity worth of the most reactive control rod. Therefore, additional reactor fuel must be added to provide the required core excess reactivity.

Each control rod is calibrated starting from its fully inserted position. Each small positive reactivity insertion is indicated on the reactivity computer and is then counter balanced by an appropriate insertion of negative reactivity from the remaining control rods operating in a bank.

A differential and integral calibration curve is prepared for each control rod. Using the calibration curves for the control rods, determine a reliable value for the interim core excess reactivity with the control rods in a banked configuration.

Acceptance Criteria: The calibration curve results for control rod worth are expected to vary between a low value of ~\$1.00 and a high value of \$3.75, depending on the type of rod and location in the core.

A.1.4 Final Core Loading/Final Rod Calibrations

The required fuel loading for achieving full power operation can be installed. A total of 90 fuel elements will be loaded. While loading fuel, all but two control rods are full DOWN. If more than four fuel elements have been added, it is necessary to recalibrate individually all six control rods using the procedure already described above in Section A.1.3.

After the final core loading is complete, and before additional control rod calibrations, it is useful to establish an initial setting of the console reactor power using the temperature coefficient of reactivity, α. For a TRIGA reactor, this leads to a value of about "1 cent reactivity per kilowatt of power". This relationship holds within a factor of about two (2) for all TRIGA reactors with reactor power levels up to, and in excess of, 100 kW and can be used initially to make approximate power level settings.

Following a final recalibration of all control rods, the excess reactivity with the cold, clean fuel is determined for a full core loading.
Additional measurements will be performed to assure that the "stuck rod criteria" is met by the assembly of control rods (i.e., reactor shut down by at least 25 cents reactivity with the most reactive control rod fully removed from the core).

At this point, the "zero power" reactivity commissioning tests are complete and the reactor is ready for the power calorimetric power tests.

Acceptance Criteria:

- With 90 fuel elements and with the core positioned against the Coupler box, the excess reactivity is expected to be about $7.79, the computed value.
- With the same core and same location, the "shut down margin" with the most reactive rod stuck out of the core will be greater than $0.25.

A.1.5 Calorimetric Reactor Power Calibration

The calorimetric power calibration takes advantage of the fact that natural convection provides adequate cooling for a TRIGA core operating at power levels up to and including 2.0 MW.

In the so-called "slope" method of calibration, the rate of temperature rise will be determined for the reactor pool water [dT/dt (°C/hr)] while the reactor is operating at power P and the tank water is stirred. For the TAMU TRIGA reactor, the so-called Tank Constant (°C /MWh) is calculated from the water volume in the reactor tank. From this and the measured time rate of pool water temperature rise, the actual reactor power can be computed as

$$P(MW) = [dT/dt\ (°C/hr) / \text{Tank Constant}\ (°C\ /MWh)\ ]$$

The calorimetric power calibration with effective circulation of tank water is conducted in two steps. The first is conducted at low to intermediate power (~250 kW) to determine the initial, nearly correct power reading on all power channel detectors. The second power calibration will then be performed at an indicated power level of about 750 kW, close to the licensed reactor power of 1 MW.

With the power level P computed from the above formula, and with the reactor operating at this power, the detectors for the power measuring channels on the reactor console will be adjusted to assure that the console correctly indicates this power level.

Note: At this point, the low-to-intermediate power tests for commissioning have been completed. Tests at higher, and full, power can now be conducted.

Acceptance Criteria: After the final power calibration, all power channel indications will agree within 2% at full reactor power, 1.0 W.

A.1.6 Initial Approach to Full Power

Outline of Approach

The object of this test is to approach full power operation in carefully programmed steps, recording fuel temperatures, all power indications on each measuring channel, and all control rod positions together with calculated reactor core excess reactivity. Continue the stepwise power increase until the power level of 1.0 MW is reached.

It is important to complete the stepwise increase in power without any decreases in reactor power so that the expected increase in measured fuel temperatures can be quantified (i.e., future values of fuel temperatures at power levels below 1.0 MW will be slightly increased above the very first measured values). During the first operation at 1.0 MW, the hot fuel will expand, stretching the fuel cladding by a small, permanent amount. For the second, and subsequent approaches to full power (1 MW), the fuel must heat to a slightly higher temperature to cause expansion to the slightly larger cladding diameter. The two sets of measured fuel temperatures demonstrate this effect.

Repeat the stepwise increase in power starting from a low power (< 1 kW). Record the fuel temperatures at each of the same power levels used in the previous stepwise rise in power. Plot the two sets of fuel temperatures versus reactor power to demonstrate the "hysteresis" effect caused by peak fuel temperature.

Acceptance Criteria: At full power (1.0 MW), the reactivity loss is expected to lie in the range from $1.50 to $2.00, values that verify the presence of a large negative coefficient of reactivity.

Linearity Check on the Power Indication Channels

For several user applications of the research reactor, it is useful to be able to rely on the linearity of the power readout instrumentation on the reactor console. To establish the degree of linearity for this power instrumentation, a test is conducted using as a standard the well established linearity of the current in the fission detector with reactor power level (for currents above the dark current ~ 5×10^{-8} amp). This D.C. current (up to about 1.0 milliamp at 1.0 MW in steady state) provides an adequate reference against which to compare the console power indications over most of the important energy range.

Take data from low power (few hundred watts) up to 1.0 MW for each power measuring console channel. Prepare a log-log graph for each power channel showing the console power indication versus the D.C. return current from a fission counter detector. The straightness of the resulting line connecting the data points demonstrates the linearity of the console power measuring channels.

Acceptance Criteria: A log-log plot of the detector indications versus the D.C. return current in a fission counter shall be a nearly straight line over a power span from about 100 kW to 1.0 MW, thus demonstrating detector channel linearity.

Tests of 125% Power Scram

The power level Scram at 125% of 1.0 MW is an important component of the Safety System. Operation at about 1.0 MW with Scram at 1.25 MW assures an adequate margin of safety. Scram at 1.25 MW is sufficiently above the full power (1.0 MW) that normal operational variation around 1.0 MW is unlikely to accidentally activate the 125% scram point.

The object of the test is to assure that a power level of 125% of 1.0 MW will in fact scram the reactor. At this point in the Commissioning Program, all 125% scram checks have been performed electronically with low or zero reactor power.

Acceptance Criteria: The reactor shall scram reliably when a Safety power channel reaches an indicated 1.25 MW.

A.1.7 Pulsing Mode of Operation

Criteria for Determining Maximum Reactivity Insertion (Maximum Pulsed Energy Release)

Several considerations are at work in determining the maximum pulse power/reactivity insertion:

(i) Determine maximum reactivity insertion that produces maximum permitted fuel temperature in hottest fuel rod;
(ii) Determine if value of maximum reactivity insertion decreases as prompt negative coefficient of reactivity decreases with fuel burnup.
(iii) Determine reduced value of maximum reactivity insertion as long term steady state operation creates increased ratio of H/Zr in outer periphery of fuel rods.
(iv) If longest core lifetime (burnup) is desired, limit pulsed $\hat{T}$ in hottest fuel to a value no higher than $\hat{T}$ in hottest fuel in steady state mode of operation.
(v) Recognize the experimenters' desire for peak thermal neutron flux; hence, largest safe reactivity insertion.

The reactor operator/owner must balance the long term needs of the TAMU facility against the users' requirements as an aid in determining the maximum permitted reactivity insertion. The reactor operator/owner must also establish whether the peak pulsed fuel temperature in the hottest fuel rod will be restricted to values (1) no greater than the peak fuel temperature in steady state operation for longest core life, (2) up to the safe temperature limit set forth in the applicable SAR; or (3) somewhere in between these limits.

Pulse Calibration Procedures

- Install a high speed analog or digital recorder to record the peak power (nv) output from the pulsing channel, one or more fuel temperatures, and an accurate shape of the pulse. The use of the (nv) data will permit an accurate evaluation of the peak power; and the prompt reactor period, which, with the prompt neutron lifetime, can be used to determine the effective pulsed reactivity insertion.
  Provide separate calibration plots of peak power and fuel temperature.
- Perform a series of pulses starting at about $1.25 and increasing in 25 cent increments to the maximum reactivity determined from the considerations set

forth above. For each pulse, record the high speed data for $\hat{P}$, the initial pulsed reactor period deduced from the plot of data, and the time variation of the fuel temperature(s) in the hottest fuel element. For at least one large pulse, record the peak fuel temperature before allowing any rod to scram for several seconds after the pulse.

Note: If it turns out not to be possible to record the peak power simultaneously on two high speed recorder channels having different gains, it may be necessary to make at least two pulses at each reactivity insertion, one with gain set to give $\hat{P}$ and one with gain set to give proper period data early in the rise of the pulse. (See Acceptance Criteria in Section 10.5.7.3 below)

Pulsing Data Report

Calculate the effective reactivity insertion for each pulse from the measured prompt reactor period.

For the range of pulse insertions, plot $\hat{P}$ versus (reactivity insertion)^2; 1/period versus reactivity insertion; fuel temperature(s) versus reactivity insertion; and Integrated Energy (MW-sec) versus reactivity insertions.

Acceptance Criteria: Each of the plots of pulsing performance shall be consistent with a linear (straight line) dependence of either (Δk_P) or (Δk_P)^2, as appropriate.

Appendix A2

A.2. FLIP (HEU) and LEU (30/20) Fuel Storage

In principle, from the criticality point of view, TRIGA fuel of any type can be stored in the same facility. The reason is that all TRIGA LEU fuel and FLIP fuel elements were designed as a one-for-one replacement for a standard 8.5 wt % 20% enriched fuel element.

A.2.1 TAMU Fuel Storage Facilities

Figure A2.1 Fuel Storage [redacted]

A.2.2 Model For Storage Facilities

The approach to storage has been very conservative, with the aim to establish upper bounds for criticality.

Storage of LEU (30/20) [redacted]
$k_{eff} = 0.12994 \pm 0.00024$

Storage of LEU (30/20) [redacted]
$k_{eff} = 0.45526 \pm 0.00043$

[redacted]: As expected, the results for both FLIP and LEU (30/20) fuel gave very small values of k_{eff}, far below the limit of 0.8.

	FLIP	LEU
[redacted]	$k_{eff} = 0.45353 \pm 0.00043$	$k_{eff} = 0.46428 \pm 0.00045$
[redacted]	$k_{eff} = 0.45324 \pm 0.00047$	$k_{eff} = 0.46420 \pm 0.00045$

The computed k_{eff} values are the following:

	FLIP	LEU
[redacted]	$k_{eff} = 0.46040 \pm 0.00041$	$k_{eff} = 0.47105 \pm 0.00044$
	$k_{eff} = 0.45968 \pm 0.00046$	$k_{eff} = 0.47110 \pm 0.00043$

Conclusion: The storage of fresh LEU (30/20) fuel [redacted] is entirely safe, [redacted]
The storage of spent FLIP fuel and/or fresh LEU (30/20) fuel [redacted] is entirely safe.

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