WASHINGTON STATE UNIVERSITY REACTOR
LICENSE NO. R-76
DOCKET NO. 50-27

SAFETY ANALYSIS REPORT FOR THE HEU TO LEU CONVERSION OF THE WASHINGTON STATE UNIVERSITY REACTOR

REDACTED VERSION

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WASHINGTON STATE UNIVERSITY

Nuclear Radiation Center

AUG 15 2007

United States Nuclear Regulatory Commission
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Docket Number 50-27
Facility License Number R-76

The Washington State University Nuclear Radiation Center (WSU/NRC) operates a TRIGA-conversion reactor which utilizes a mixed HEU/LEU core. In accordance with Federal requirements, the WSU/NRC is cooperating with the United States Department of Energy (DOE), DOE contractors at the Argonne National Laboratory (ANL) and Idaho National Laboratory (INL), General Atomics (GA), and the Unites States Nuclear Regulatory Commission (NRC) to remove HEU fuel from the WSU/NRC, and to replace HEU fuel with LEU fuel in the WSU/NRC nuclear reactor.

This letter is being transmitted by Washington State University (WSU) to the NRC as part of the conversion process, and as an application by WSU to the NRC to amend Facility License Number R-76. Washington State University requests that the NRC issue two orders, one for increasing possession limits for uranium-235 (Possession Order), and a second order for conversion of the WSU reactor from HEU to LEU fuel (Conversion Order), in accordance with the following two requested modifications:

1. Application for amendment to License Number R-76, (Docket Number 50-27), to increase possession limits for uranium-235
2. Submittal of Safety Analysis Report for amendment to License R-76 authorizing changes to Technical Specifications for the conversion of the TRIGA nuclear reactor at the Washington State University Nuclear Radiation Center from HEU to LEU fuel.

Application for amendment to License R-76, (Docket Number 50-27), to increase possession limits for uranium-235

Washington State University is currently licensed to possess and use quantities of uranium-235 which are described in Amendment Number 10 to License Number R-76. Amendment Number 10 to License Number R-76 stipulates a possession limit of 25 kg of uranium-235 at various enrichments with the following specific categories of maximum limits:

PO Box 641300, Pullman, WA 99164-1300
509-335-8641 • Fax: 509-335-4433 • www.wsu.edu/~nrc

Inventory Limits of Uranium-235; Amendment 10 to License Number R-76

Maximum U-235 (kg)	% enrichment	Exempt status
10.0	< 20	Exempt 10 CFR 73.6 (a)
15.0	> 20	Exempt 10 CFR 73.6 (b)
4.90	> 20	Not exempt

The nuclear reactor facility at the WSU/NRC provided 11,521.45 user-hours of services for the reporting period of July 1, 2006 to June 30, 2007. Extensive use of the reactor was accomplished by providing irradiation services to multiple users and irradiating multiple samples simultaneously. As is evident by extensive use of the reactor, it is of great importance to the academic community at WSU and at other institutions to minimize the shut-down time that must necessarily result from refueling activities. Minimization of down-time may be accomplished by modification of the uranium-235 possession limits specified in Amendment 10 to License R-76. WSU requests an amendment to increase inventory limits of uranium-235 to permit receipt of fresh, unirradiated TRIGA fuel before shipment of spent HEU fuel from the WSU/NRC, and to accommodate the increased inventory of LEU that will necessarily result from the conversion process.

The WSU TRIGA Reactor Uranium Inventory Report for Period Ending March 31, 2007 describes the inventory of U-235 at the WSU/NRC. As part of refueling, the WSU/NRC is scheduled to receive additional U-235. Specific quantities describing fuel burnup status, enrichment, and storage location are available to the NRC on request.

Washington State University is currently authorized to possess twenty five (25) kilograms of uranium-235, at various enrichments. After the spent HEU fuel is shipped from the WSU/NRC, only LEU fuel with enrichment less than 20% will remain. The neutron (fission) detectors in the WSU reactor employ a small quantity of uranium enriched to greater than 20%; Washington State University requests that a small quantity (500 grams) of uranium-235 at enrichment greater than 20% be permitted for possession for use in nuclear detectors and for experimental research.

Washington State University requests the NRC to order the following inventory limit modification: that the inventory limit for uranium of enrichment less than 20% be amended to allow possession of 24.5 kilograms of uranium-235.

Washington State University requests that, after shipment of HEU fuel the inventory limit for uranium of enrichment greater than 20% be set at not less than 500 grams of uranium-235 for use in nuclear detectors and for experimental use.

Washington State University also requests that assembly of fresh, unirradiated, individual fuel elements into 3-element or 4-element fuel assemblies (also referred to as fuel bundles) be permitted under the Possession Order.

Submittal of Safety Analysis Report for License amendment authorizing changes to Technical Specifications for the conversion of the TRIGA nuclear reactor at the Washington State University Nuclear Radiation Center from HEU to LEU fuel.

Washington State University proposes changes to the Technical Specifications that are applicable to License Number R-76, as part of the process requirements for removal of HEU fuel from the reactor, and replacing the HEU with LEU fuel. The Safety Analysis Report (SAR) describing the proposed changes is attached to this letter. General Atomics provided the technical analysis for the SAR.

The attached SAR includes an appendix that describes the fuel storage facilities which will be employed to store the fresh, unirradiated fuel until such time that the NRC orders conversion of the WSU reactor. The description of the fuel storage facilities includes a criticality safety analysis.

I declare under penalty of perjury that the foregoing is true to the best of my knowledge.

Respectfully Submitted

Donald Wall

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Director
Nuclear Radiation Center
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Safety Analysis for the HEU to LEU Conversion of the Washington State University Reactor

TRD 070.01002 RGE 001 Rev. N/C

Final Report

Complete Final HEU/LEU Conversion Report

August 2007

Prepared By
TRIGA Reactors Division of General Atomics-ESI
San Diego, California

GENERAL ATOMICS
ELECTRONIC SYSTEMS INC.

TRIGA Reactors

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Summary
This report contains the results of the design, safety, and accident analyses performed for Washington State University (WSU) by General Atomics for the conversion of the WSU 1 MW TRIGA research reactor. Currently the WSU Mixed HEU Core 34A contains a mixture of TRIGA fuel elements which include highly enriched (70% U-235) fuel elements developed as part of the Fuel Life Improvement Program (FLIP) and low enriched (<20% U-235) standard fuel elements. The conversion will replace the FLIP/HEU fuel elements with specially designed TRIGA LEU fuel elements containing 30 weight percent uranium enriched to less than 20 % in U-235 (30/20 LEU). The 30/20 LEU fuel elements are specially designed for a one-to-one replacement of the FLIP HEU fuel elements. By replacing the FLIP HEU fuel elements, all HEU will have been removed from the core and eventually from the facility. This study investigates the performance and safety margins of the proposed converted LEU core under nominal and accident conditions. It also identifies any suggested changes to the WSU Final Safety Analysis Report and Technical Specifications, Ref. 1.

1.0 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 WSU 1 MW TRIGA reactor.

The HEU to LEU conversion requires a one-to-one swap for each of the TRIGA FLIP HEU fuel elements currently operating in WSU Core 34A by specially designed TRIGA LEU (30 /20) fuel elements. WSU Core 34A is also operating with low enriched standard TRIGA fuel element consisting of 8.5 weight percent uranium / 20 % enriched fuel (8.5/20) LEU. These latter fuel elements will not be replaced and will continue operating even after the replacement of the FLIP HEU fuel elements. In addition, the conversion will not require any changes to the remainder of the facility.

The proposed converted WSU Mixed LEU reactor core designated as Core 35A contains:

1. 49 – Fresh TRIGA (30/20 LEU) Fuel Elements whose fuel contains 30 weight percent uranium / < 20 % enriched in U-235. These elements will be referred as 30/20 LEU SFEs
2. 2 – Fresh TRIGA (30/20 LEU) Instrumented Fuel Elements (IFEs) whose fuel contains 30 weight percent uranium / 20 % enriched in U-235. These elements will be referred as 30/20 LEU IFEs
3. 68 – currently operating and partially burned TRIGA SFEs whose fuel contains 8.5 weight percent uranium / < 20 % enriched in U-235. These elements will be referred as 8.5/20 partially burned LEU SFEs
4. 1 – new Transient Pulsing Rod, this will be referred as Rod 3
5. 3 – currently operating control rod blades. These blades will be referred to as Shim blade

1, Shim blade 2, Shim blade 4

6. 1 – currently operating stainless steel regulating blade. This will be referred to as Servo blade 5.

Based on this core configuration, it is concluded that:

i) the shutdown margin meets the required limits;
ii) the reactivity coefficients remain essentially the same as for WSU Mixed HEU Core 34A;
iii) fuel integrity of the converted core 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 may require possible changes to the Technical Specifications as discussed in Section 14.

1.2 Summary and Conclusions of Principal Safety Considerations

The WSU Mixed LEU Core 35A meets all the safety requirements as specified in the 1979 Safety Analysis Report which is the current SAR in force. (Reference. 1)

1.3 Summary of Reactor Facility Changes

The replacement 30/20 LEU 4-rod and 3-rod fuel clusters have the same physical dimensions as the currently operating FLIP HEU 4-rod and 3-rod clusters. The clad for the 30/20 LEU fuel and the construction of the clusters are identical with those currently used with the FLIP HEU fuel. The 30/20 LEU fuel has been approved by the Nuclear Regulatory Commission (NRC) for use in non-power reactors. (Reference 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

The Texas A&M (TAMU) 1 MW TRIGA reactor has already been converted to TRIGA LEU (30/20) fuel. There are both similarities and differences between the WSU and TAMU reactors.

The similarities are primarily in the TRIGA SFEs and Clusters which are similar in configuration and design for both TAMU and WSU.

The differences are primarily in the core configuration. This includes:

- WSU will be a partial fuel replacement.
    - Fuel – For WSU, only the FLIP/HEU SFEs and IFEs will be replaced whereas TAMU was a full FLIP/HEU core conversion including the replacement of all SFEs, and IFEs.
    - Control – For WSU only the Transient Rod (TR) (Rod 3) will be replaced whereas for TAMU the entire control system consisting of the Fuel Followed Control Rods

(FFCR), TR, and Regulating Rod (RR) was replaced.

- The WSUs control system contains 3 control (shim) blades, a stainless steel regulating blade (servo blade) and a TR (water followed) whereas the TAMU control system has 4 standard TRIGA FFCRs, an RR and a TR (air followed).
- WSU has a uniform pitch between Fuel Clusters whereas the TAMU fuel cluster pitch is different in the two directions. Table 1 summarizes these similarities and differences between the two reactors.

Table 1 Comparison of WSU and TAMU Conversions

Component	TAMU	WSU
Conversion Core	Full Conversion: 30/20 LEU	Partial Conversion: 30/20 LEU & Partially Burned 8.5/20 LEU
Standard Fuel Element	Same	Same
Instrumented Fuel Element	Same	Same
Fuel Rod Clusters	Same	Same
Fuel Cluster Pitch	Variable in two directions	Uniform in the horizontal plane
Control Rods	4 TRIGA FFCRs	3 Control (Shim) blades
Regulating Rod	TRIGA RR – water followed	Stainless steel (Servo) blade
Transient Rod	TRIGA TR – air followed	TRIGA TR – water followed

2.0 Site Characteristics

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

3.0 Design of Structures, Systems, and Components

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

4.0 Reactor Description

4.1 Reactor Facility

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

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

Table 2 provides a comparison of the key design safety features of the HEU and LEU fuel clusters and a comparison of the key reactor and safety parameters that were calculated for each core. The results show that the WSU reactor facility can be operated as safely with the addition of the new LEU fuel clusters as with the present HEU fuel clusters.

The evaluation of WSU Mixed HEU Core No 34A provides an opportunity to benchmark the computational technique to be used for evaluating the converted WSU Mixed LEU Core 35A. The analyses produced operational parameters to be compared with the actual measured values from the operational data conducted by WSU Staff for Core 34A. The experimentally measured parameters included the reactivity for the fully loaded Core 34A which contains 51 FLIP HEU SFEs/IFEs and 68 partially-burned 8.5/20 LEU SFEs with full water reflection; the control rod calibration values; the reactivity loss and peak fuel temperatures as a function of reactor 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 was used to produce a plot of the prompt, negative temperature coefficient of reactivity (Δk/k-°C) versus reactor fuel temperature that can be compared with the value in the SAR (1979) for the same parameter.

The steady state parameters for the WSU Mixed LEU Core 35A were calculated using the same computational procedures adapted to the WSU Mixed HEU Core 34A configuration.

Table 2 Washington State University HEU – LEU Conversion Design Data

	Mixed HEU Core 34A		Mixed LEU Core 35 A	
FUEL PARAMETERS				
	FLIP HEU SFEs/IFEs	8.5/20 LEU SFEs	30/20 LEU SFEs/IFEs	8.5/20 LEU SFEs
Number of Fuel Rods	51	68	51	68
Fuel Type	UZrH	UZrH	UZrH	UZrH
Uranium Weight/Enrichment, %	8.5/70	8.5/20	30/20	8.5/20
Erbium, weight %	1.48	--	0.90	--
Zirconium Rod, OD, mm	6.35	6.35	6.35	6.35
Fuel meat, OD, mm	34.823	34.823	34.823	34.823
Fuel meat length, mm	381	381	381	381
Clad Material	304 SS	304 SS	304 SS	304 SS
Clad thickness, mm	0.508	0.508	0.508	0.508
REACTOR PARAMETERS				
Reactor Power				
Licensed Power, MW	1.0		1.0	
LCO Max Power, MW	1.3		1.3	
Max Fuel Temperature at 1 MW, °C	435		500	
Calculated Maximum Pulsing Operation with $\hat{T}$ limited to 830 °C, MW (a)	\$2.02		\$2.04 BOL \$2.20 EOL	
Cold Clean Excess Reactivity, Δk/kβ, \$	7.17		6.94	
Prompt Negative Temp. Coefficient of Reactivity -Δk/k-°C 23-1000°C	0.54×10^{-4} to 1.51×10^{-4}		0.60×10^{-4} to 1.27×10^{-4}	
Coolant Void Coefficient, Δk/k per 1% void	0.080%		0.135%	
Maximum Rod Power at 1 MW, kW/element	20.9		20.8	
Average Rod Power at 1 MW, kW/element	8.4		8.4	
Maximum Rod Power at 1.3 MW, kW/element	27.2		27.0	
Average Rod Power at 1.3 MW, kW/element	10.9		10.9	
Maximum Rod Power at DNB = 1.0, kW/element	51.7		52	
DNB Ratio at Operating Power	2.47		2.45	
Prompt Neutron Lifetime, μsec	30.7		28.2	
Effective Delayed Neutron Fraction	0.0076		0.0075	
Shutdown Margin, Δk/kβ (\$)with most reactive rod and Reg. Rod Stuck out	1.06		1.88	
SAFETY PARAMETERS				
Limiting Safety System Setting	500°C		500°C	
LCO Max Power, MW	1.3		1.3	
Minimum DNB ratio at 1.0 MW	2.47		2.45	
Minimum DNB ratio at 1.3 MW	1.90		1.89	
Calculated Maximum Positive Pulsed Reactivity Insertion to reach $\hat{T}$ =830°C, Δk/kβ (\$) (a)	\$2.02		\$2.04 BOL \$2.20 EOL	
Peak Pulsed Fuel Temperature, °C	830		830	

(a) Calculated maximum reactivity insertion for pulsing yields a conservative value, as shown by actual pulsing data.

4.2 Reactor Core

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

The WSU reactor is primarily a homogeneous, light water moderated and cooled, tank-type reactor fueled with a core containing a mixture of FLIP HEU SFEs and 8.5/20 LEU SFEs in either a 4-rod or 3-rod cluster configuration. The fuel clusters are supported by a grid box consisting of a cast aluminum grid plate suspended from the bridge by four-corner posts that form a suspension frame. The grid plate provides a 7 × 9 array of square holes for fuel clusters and two slots for control blades. The grid box accepts the 4-rod and 3-rod fuel clusters and the reflector elements. The reactor is supported from the top of the biological shield and is moveable along the central, long axis of the reactor tank. Figure 1 shows the WSU movable reactor core. At WSU, the most frequently used reactor locations are: (1) adjacent to the thermal column (BNCT filter) as shown in Figure 1, and (2) about 7 feet removed from the BNCT filter and water reflected. The arrangement for WSU Core 34A, a Mixed HEU core configuration is shown in Figure 2. It contains a mixture of 51 FLIP HEU SFEs and 68 – 8.5/20 LEU SFEs, 3 control blades, a servo blade, and a water-followed transient rod. All 5 control rods are supported from the bridge structure at the top of the biological shield.

Figure 1 Pool Structure at the Nuclear Radiation Center – Washington State University

Figure 2 WSU Mixed HEU Core 34A

4.2.1 Fuel Elements

The FLIP HEU, the partially burned 8.5/20 LEU, and the 30/20 LEU fuel elements have similar overall designs, i.e., they are all TRIGA fuel rods mounted in 4-rod or 3-rod, clusters.

Fuel Description

The geometries, materials, and fissile loadings of the current FLIP HEU SFEs, the partially-burned 8.5/20 LEU SFEs, and the 30/20 LEU SFEs are summarized in Table 3.

Figure 3 shows a 4-rod fuel cluster; Figure 4 shows a 3-rod cluster modified to accommodate the water followed transient rod. Figure 5 shows the nominal cluster and fuel element spacing.

The heat removal system for the WSU Mixed LEU Core 35A remains unchanged from that used with the Mixed FLIP HEU Core 34A. 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 30/20 LEU SFEs to be installed in the WSU core are contained within 4-rod and 3-rod fuel clusters exactly the same as for the present Mixed HEU core. Figures 6 and 7 show detailed illustrations of the fuel element and the instrumented (integrated thermocouple) fuel element. The fuel for the FLIP HEU, the partially burned 8.5/20, and the 30/20 LEU SFEs differs only in the alloy in the fuel sections of these illustrations; the dimensions are the same for these types of TRIGA fuel.

Table 3 Description of TRIGA HEU and LEU Fuel Elements

Design Data	TRIGA FLIP HEU	TRIGA 30/20 LEU	TRIGA 8.5/20 LEU
Number of Fuel Elements at Full Load	51	51	68
Fuel Type	U-ZrH (FLIP)	U-ZrH (30/20)	U-ZrH (8.5/20)
Enrichment, %	70	19.75	19.75
Uranium Density			
g/cm^3		2.14	0.5
wt-%	8.42	30	8.5
Number of Fuel Elements per Cluster	4	4	4
^{235}U per Fuel Bundle, g		597.26	156
^{235}U per Fuel Element, g		149.32	39
^{166}Er per Fuel Element, g	10.27	7.46	----------
^{167}Er per Fuel Element, g	7.09	5.15	----------
Erbium, wt-%	1.48	0.9	----------
Zirconium Rod Outer Diameter, mm	6.35	6.35	6.35
Fuel Meat Outer Diameter, mm	34.823	34.823	34.823
Fuel Meat Length, mm	381	381	381
Cladding Thickness, mm	0.508	0.508	0.508
Cladding Material	304 SS	304 SS	304 SS

Figure 3 Four Element Cluster

Figure 4 Three-Element Cluster with Guide Tube for Transient Rod

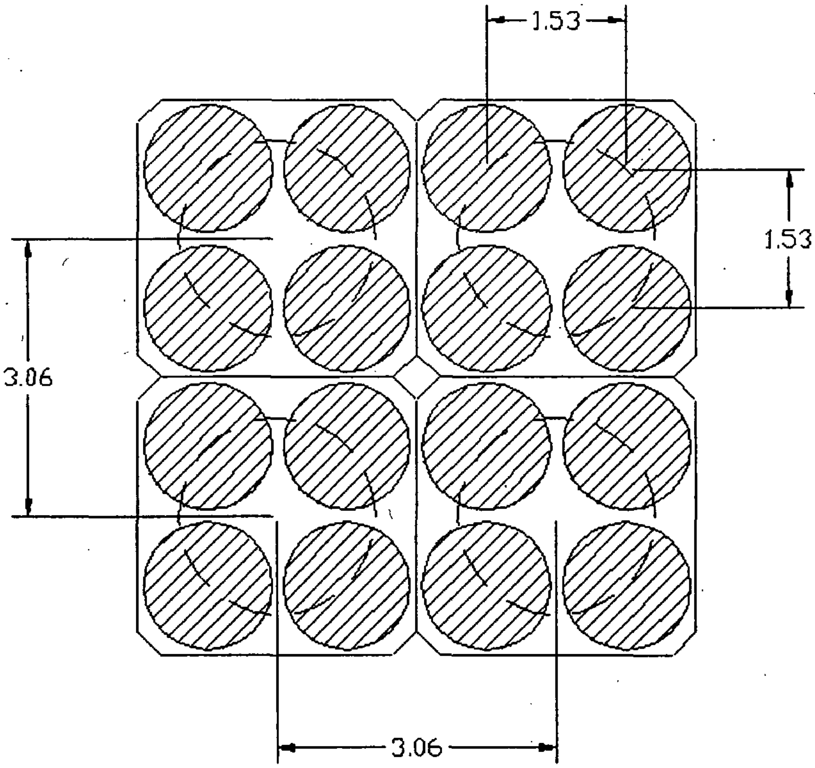

Figure 5 Nominal Fuel Element and Cluster Spacing in the WSU Core (dimensions are inches)

Figure 6 Detailed Drawing of Fuel Element

Figure 7 Instrumented Fuel Element

4.2.2 Control System

The three shims and the servo control elements of the WSU reactor are blade type elements as shown in Figure 8. The poison section of the safety blades is a boral sheet (35 wt.% B_4C, 65 wt.% Al) 40.5 inches long and 10.5 inches wide. The boral sheet is 3/8 inches thick and is clad with 1/8 inch aluminum. The regulating, servo blade is a stainless steel sheet about 11 inches wide and 40 inches long. Each blade is guided through its travel by a shroud, as shown in Figure 8. The shroud consists of two thin aluminum plates 38 inches high separated by aluminum spacers to provide a 3/4 inch control blade slot. Small flow holes are drilled at the bottom of the shroud to reduce the effects of viscous damping on the blade fall time. There is no plan to change either the WSU shim blades or the servo control blade.

The transient rod (TR3) is a standard TRIGA water-followed transient rod and will be replaced during the conversion, Figure 9.

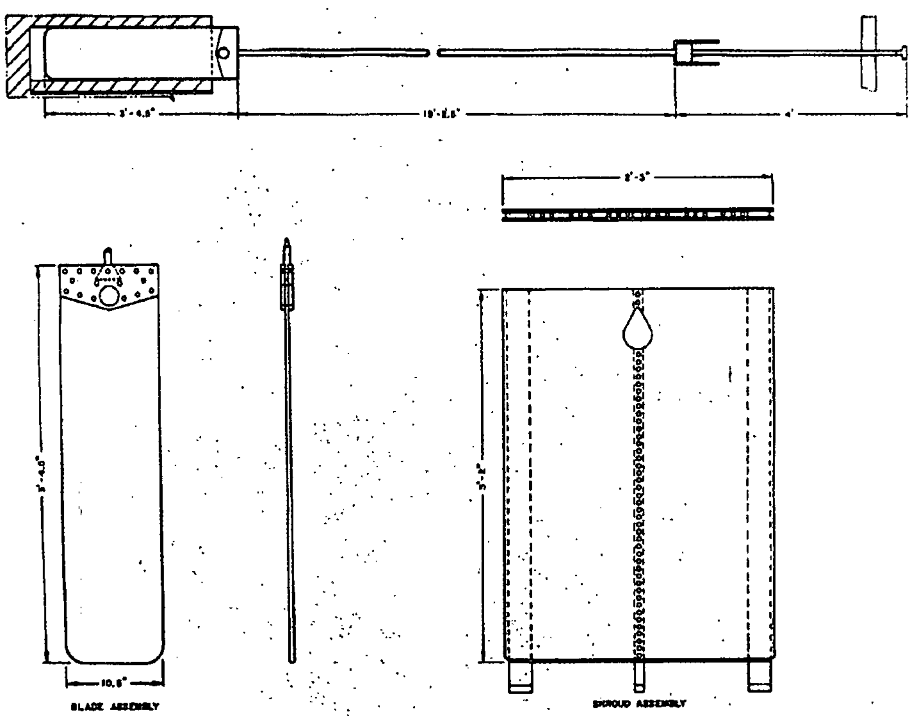

Figure 8 Control Blade

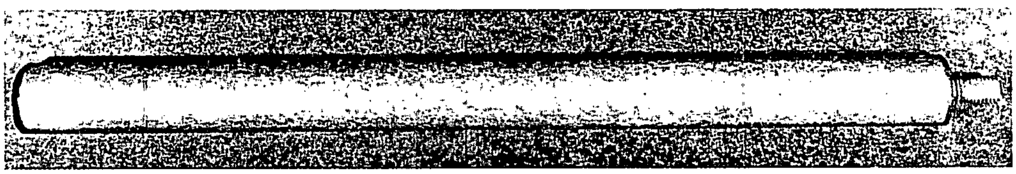

Figure 9 Standard TRIGA Water Followed Transient Rod

4.2.3 Neutron Reflector

There is no plan to change the reflector during the HEU to LEU conversion. The WSU reactor uses nuclear-grade graphite elements (as well as water) in the core as reflector, Figures 1 and 2 show the reflector regions.

4.2.4 Neutron Source and Holder

The proposed HEU to LEU conversion of the WSU core does not require any changes in the existing neutron source location, Fig. 4.2.

4.2.5 In-Core Experimental Facilities

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

4.2.6 Reactor Materials

The WSU conversion to LEU requires a change in the fuel element composition but no change in the fuel clad. Table 4 presents the material composition used in the computational models.

Table 4 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 (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% B_4C (transient rod)	B-10	0.020950	
	B-11	0.084310	
	C	0.026320	
Boral (35wt% B_4C 65 wt% Al)	B-10	0.008058	
	B-11	0.032233	
	C	0.010073	
	Al-27	0.038306	
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 Tank and Biological Shielding

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

4.4 Core Support Structure

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

4.5 Dynamic Design

4.5.1 Calculation Models; Nuclear Analysis Codes

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, is a multi-group code which solves the neutron diffusion equations with arbitrary group scattering. (References 3 and 4)

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

The BURP/DIF-3D module, (Reference 6), is used for the burnup calculations with the cross section data generated with GGC-5, (Reference 7).

MCNP5 Monte Carlo Code

This section discusses the MCNP5 models developed for these analyses and the benchmark calculations for the Mixed HEU Core 34A, and determines a reference critical Mixed LEU Core 35A.

Reactor calculations were performed in three dimensions for the full core loading of the WSU Mixed HEU Core 34A and for the initial criticality and the full core loading of the WSU Mixed LEU Core 35A 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.

To obtain the heavy metals and fission products densities in the partially-burned 8.5/20 LEU and FLIP HEU fuel rods in Core 34A the three dimensional depletion calculations were performed with BURP-DIF3D codes for each of FLIP and STD fuels. The FLIP fuels were burned 3000 days at 1 MW, and the STD fuels were burned 1500 days at 1 MW. The mass of U-235 and U-238 in Core 34A were used from the September 2005 WSU inventory report listed in Table 5.

The heavy metal masses and fission products were calculated for the WSU Mixed HEU Core 34A. To make the graphs easier to read the results are plotted as a function of burnup days along with polynomial fits for the nuclides in Figures 10 through 15

By using the U-235 mass in the inventory report and the U-235 polynomial fit the burnup days for each fuel rod were calculated. The obtained burnup days were used in the polynomial fits to calculate the heavy metals and fission products in the depleted fuel rods.

The nuclide densities used in the MCNP models are shown in Table 6 for the WSU FLIP HEU fuel meats, in Table 7 for the burned 8.5/20 LEU fuel meats, and Table 8 for the 30/20 LEU fuel meats.

The other materials beside the fuel used in the Mixed HEU and Mixed LEU MCNP models are listed in Table 4.

Table 5 Uranium inventories for the partially-burned 8.5/20 LEU (STD) and FLIP HEU fuel rods – WSU Mixed HEU Core 34A, as of September 30, 2005

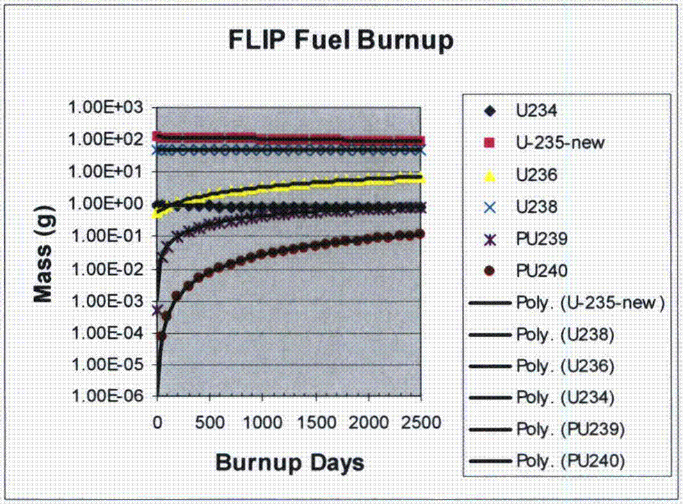

Figure 10 Masses for Typical Heavy Metals and Fission Products – Core 34A

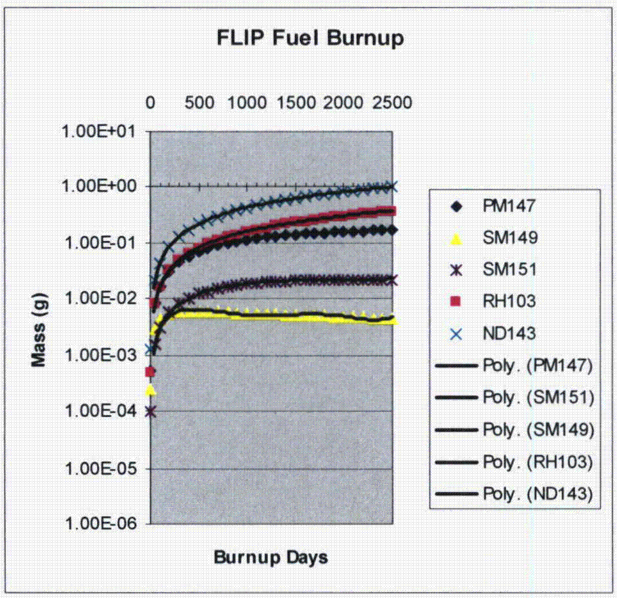

Figure 11 Masses for Typical Heavy Metals and Fission Products – Core 34A

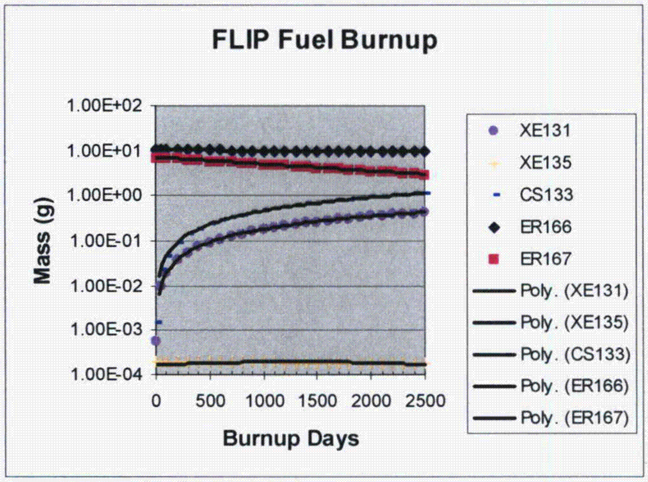

Figure 12 Masses for Typical Heavy Metals and Fission Products – Core 34A

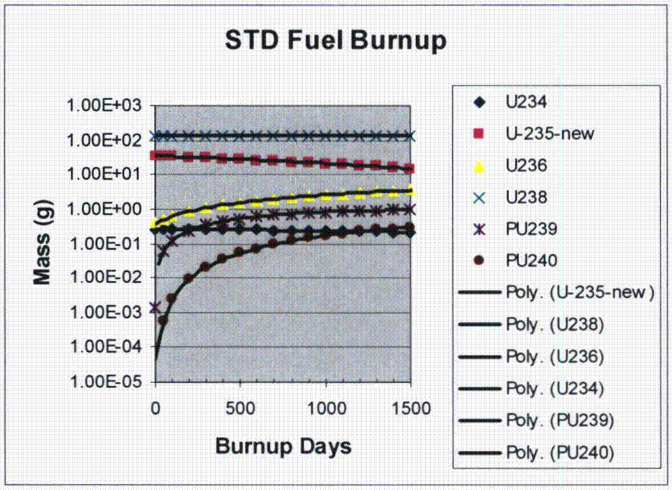

Figure 13 Masses for Typical Heavy Metals and Fission Products – Core 34A

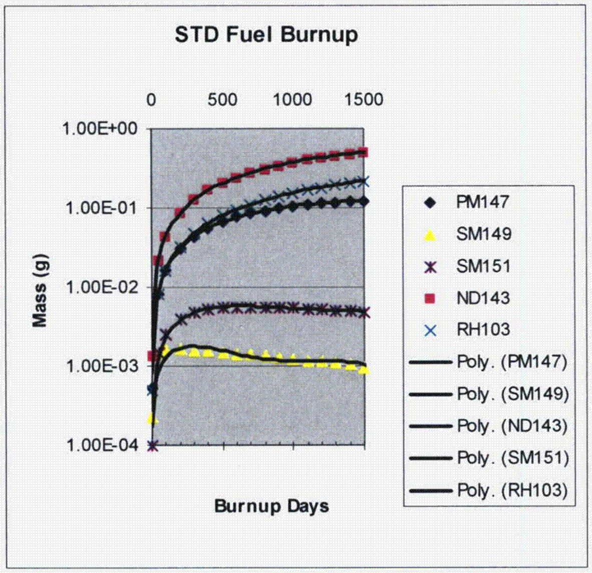

Figure 14 Masses for Typical Heavy Metals and Fission Products – Core 34A

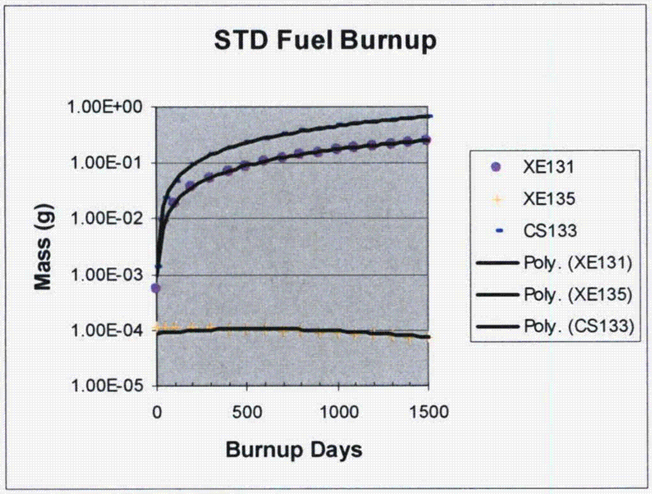

Figure 15 Masses for Typical Heavy Metals and Fission Products – Core 34A

Table 6 Nuclide densities for Burned FLIP/HEU fuel meat, MCNP Model for WSU Mixed HEU Core 34A

Nuclide	*C-3*	*C-4*	*C-5*	*C-6*	**D-2**
U-235	8.0007E-04	7.7473E-04	7.7462E-04	8.1192E-04	7.9579E-04
U-234	6.2968E-06	6.2201E-06	6.2197E-06	6.3325E-06	6.2838E-06
U-236	2.1441E-05	2.6122E-05	2.6144E-05	1.9233E-05	2.2236E-05
U-238	3.6591E-04	3.6499E-04	3.6498E-04	3.6634E-04	3.6576E-04
Pu-239	2.7087E-06	3.3299E-06	3.3327E-06	2.3983E-06	2.8177E-06
Pu-240	3.2430E-07	5.0162E-07	5.0250E-07	2.5159E-07	3.5226E-07
Rh-103	2.2324E-06	2.8306E-06	2.8333E-06	1.9489E-06	2.3342E-06
Xe-131	1.9948E-06	2.5322E-06	2.5347E-06	1.7405E-06	2.0861E-06
Cs-133	4.9636E-06	6.3355E-06	6.3419E-06	4.3200E-06	5.1957E-06
Nd-143	4.3222E-06	5.4824E-06	5.4877E-06	3.7725E-06	4.5196E-06
Pm-147	1.1871E-06	1.3923E-06	1.3932E-06	1.0747E-06	1.2250E-06
Sm-149	6.4244E-08	5.9000E-08	5.8988E-08	6.8133E-08	6.3001E-08
Sm-151	1.9122E-07	2.1646E-07	2.1655E-07	1.7590E-07	1.9618E-07
Er-166	1.0463E-04	1.0429E-04	1.0428E-04	1.0479E-04	1.0457E-04
Er-167	5.4872E-05	5.0590E-05	5.0571E-05	5.6972E-05	5.4129E-05
H	5.4593E-02	5.4593E-02	5.4593E-02	5.4593E-02	5.4593E-02
C	1.4961E-03	1.4961E-03	1.4961E-03	1.4961E-03	1.4961E-03
Zr	3.5299E-02	3.5299E-02	3.5299E-02	3.5299E-02	3.5299E-02
Hf	2.1179E-06	2.1179E-06	2.1179E-06	2.1179E-06	2.1179E-06
Total	9.2761E-02	9.2740E-02	9.2740E-02	9.2771E-02	9.2758E-02
Nuclide	D-3	D-4	D-5	D-6	E-3
U235	7.8678E-04	7.8611E-04	7.6130E-04	7.8202E-04	7.9388E-04
U-234	6.2566E-06	6.2546E-06	6.1793E-06	6.2422E-06	6.2781E-06
Pu-240	4.1389E-07	4.1858E-07	6.0600E-07	4.4788E-07	3.6504E-07
Rh-103	2.5476E-06	2.5632E-06	3.1435E-06	2.6597E-06	2.3796E-06
Xe-131	2.2777E-06	2.2918E-06	2.8139E-06	2.3786E-06	2.1269E-06
Cs-133	5.6840E-06	5.7200E-06	7.0613E-06	5.9417E-06	5.2994E-06
Nd-143	4.9334E-06	4.9638E-06	6.0895E-06	5.1510E-06	4.6077E-06
Pm-147	1.3004E-06	1.3057E-06	1.4838E-06	1.3379E-06	1.2415E-06
Sm-149	6.0824E-08	6.0690E-08	5.8426E-08	5.9949E-08	6.2486E-08
Sm-151	2.0566E-07	2.0631E-07	2.2621E-07	2.1018E-07	1.9830E-07
Er-166	1.0445E-04	1.0444E-04	1.0410E-04	1.0439E-04	1.0455E-04
Er-167	5.2590E-05	5.2478E-05	4.8434E-05	5.1792E-05	5.3799E-05
H	5.4593E-02	5.4593E-02	5.4593E-02	5.4593E-02	5.4593E-02
C	1.4961E-03	1.4961E-03	1.4961E-03	1.4961E-03	1.4961E-03
Zr	3.5299E-02	3.5299E-02	3.5299E-02	3.5299E-02	3.5299E-02
Hf	2.1179E-06	2.1179E-06	2.1179E-06	2.1179E-06	2.1179E-06
Total	9.2750E-02	9.2749E-02	9.2728E-02	9:2746E-02	9.2756E-02

Nuclide	E-4	E-5	E-6
U-234	6.2721E-06	6.2100E-06	6.2951E-06
U-236	2.2956E-05	2.6734E-05	2.1544E-05
U-238	3.6562E-04	3.6487E-04	3.6589E-04
Pu-239	2.9153E-06	3.4073E-06	2.7230E-06
Pu-240	3.7838E-07	5.2689E-07	3.2788E-07
Rh-103	2.4264E-06	2.9084E-06	2.2456E-06
Xe-131	2.1689E-06	2.6022E-06	2.0066E-06
Cs-133	5.4063E-06	6.5155E-06	4.9937E-06
Nd-143	4.6983E-06	5.6334E-06	4.3479E-06
Pm-147	1.2582E-06	1.4160E-06	1.1921E-06
Sm-149	6.1984E-08	5.8721E-08	6.4076E-08
Sm-151	2.0042E-07	2.1909E-07	1.9188E-07
Er-166	1.0452E-04	1.0424E-04	1.0462E-04
Er-167	5.3461E-05	5.0048E-05	5.4775E-05
H	5.4593E-02	5.4593E-02	5.4593E-02
C	1.4961E-03	1.4961E-03	1.4961E-03
Zr	3.5299E-02	3.5299E-02	3.5299E-02
Hf	2.1179E-06	2.1179E-06	2.1179E-06
Total	9.2754E-02	9.2737E-02	9.2761E-02

Table 7 Nuclide densities for Partially Burned 8.5/20 LEU fuel meat, MCNP Model for WSU Mixed HEU Core 34A and Mixed LEU Core 35A

Nuclide	*B-1*	*B-2*	*B-3*	*B-6*	**C-1**
U-235	2.0377E-04	2.1675E-04	2.0351E-04	2.0535E-04	2.1704E-04
U-234	1.8305E-06	1.8551E-06	1.8299E-06	1.8336E-06	1.8556E-06
U-236	1.1228E-05	9.1509E-06	1.1269E-05	1.0977E-05	9.1039E-06
U-238	1.0116E-03	1.0128E-03	1.0115E-03	1.0117E-03	1.0128E-03
Pu-239	3.7764E-06	2.9831E-06	3.7911E-06	3.6852E-06	2.9641E-06
Pu-240	3.5901E-07	2.1028E-07	3.6227E-07	3.3920E-07	2.0733E-07
Rh-103	1.3178E-06	9.8751E-07	1.3243E-06	1.2775E-06	9.8012E-07
Xe-131	1.1687E-06	8.7667E-07	1.1744E-06	1.1331E-06	8.7012E-07
Cs-133	2.9290E-06	2.1904E-06	2.9436E-06	2.8386E-06	2.1739E-06
Nd-143	2.4335E-06	1.8432E-06	2.4451E-06	2.3623E-06	1.8299E-06
Pm-147	7.6469E-07	6.0389E-07	7.6769E-07	7.4604E-07	6.0007E-07
Sm-149	1.7997E-08	1.9897E-08	1.7954E-08	1.8259E-08	1.9928E-08
Sm-151	6.3897E-08	5.9022E-08	6.3952E-08	6.3526E-08	5.8858E-08
H	5.4712E-02	5.4712E-02	5.4712E-02	5.4712E-02	5.4712E-02
C	1.4891E-03	1.4891E-03	1.4891E-03	1.4891E-03	1.4891E-03
Zr	3.5684E-02	3.5684E-02	3.5684E-02	3.5684E-02	3.5684E-02
Hf	2.1410E-06	2.1410E-06	2.1410E-06	2.1410E-06	2.1410E-06
Total	9.3128E-02	9.3138E-02	9.3128E-02	9.3130E-02	9.3138E-02
	C-2	C-7	D-1	D-7	E-1
U-235	1.9836E-04	2.0711E-04	2.0936E-04	2.0467E-04	2.1250E-04
U-234	1.8195E-06	1.8370E-06	1.8414E-06	1.8323E-06	1.8473E-06
U-236	1.2087E-05	1.0696E-05	1.0335E-05	1.1084E-05	9.8325E-06
U-238	1.0110E-03	1.0119E-03	1.0121E-03	1.0116E-03	1.0124E-03
Pu-239	4.0779E-06	3.5818E-06	3.4465E-06	3.7242E-06	3.2533E-06
Pu-240	4.3030E-07	3.1763E-07	2.9084E-07	3.4758E-07	2.5523E-07
Rh-103	1.4560E-06	1.2326E-06	1.1750E-06	1.2946E-06	1.0951E-06
Xe-131	1.2908E-06	1.0934E-06	1.0425E-06	1.1482E-06	9.7189E-07
Cs-133	3.2394E-06	2.7380E-06	2.6093E-06	2.8771E-06	2.4307E-06
Nd-143	2.6763E-06	2.2826E-06	2.1802E-06	2.3927E-06	2.0372E-06
Pm-147	8.2670E-07	7.2494E-07	6.9744E-07	7.5402E-07	6.5831E-07
Sm-149	1.7097E-08	1.8547E-08	1.8905E-08	1.8148E-08	1.9369E-08
Sm-151	6.4790E-08	6.3047E-08	6.2327E-08	6.3690E-08	6.1109E-08
H	5.4712E-02	5.4712E-02	5.4712E-02	5.4712E-02	5.4712E-02
C	1.4891E-03	1.4891E-03	1.4891E-03	1.4891E-03	1.4891E-03
Zr	3.5684E-02	3.5684E-02	3.5684E-02	3.5684E-02	3.5684E-02
Hf	2.1410E-06	2.1410E-06	2.1410E-06	2.1410E-06	2.1410E-06
Total	9.3125E-02	9.3131E-02	9.3132E-02	9.3129E-02	9.3135E-02
	E-2	E-7	F-1	F-2	F-3
U-235	1.9220E-04	2.0374E-04	2.1161E-04	2.0410E-04	1.9863E-04

U-234	1.8064E-06	1.8304E-06	1.8456E-06	1.8311E-06	1.8200E-06
U-236	1.3061E-05	1.1233E-05	9.9753E-06	1.1176E-05	1.2044E-05
U-238	1.0103E-03	1.0115E-03	1.0123E-03	1.0116E-03	1.0110E-03
Pu-239	4.4013E-06	3.7781E-06	3.3087E-06	3.7575E-06	4.0633E-06
Pu-240	5.1768E-07	3.5939E-07	2.6513E-07	3.5485E-07	4.2662E-07
Rh-103	1.6139E-06	1.3185E-06	1.1178E-06	1.3094E-06	1.4491E-06
Xe-131	1.4302E-06	1.1693E-06	9.9192E-07	1.1613E-06	1.2847E-06
Cs-133	3.5954E-06	2.9306E-06	2.4813E-06	2.9101E-06	3.2239E-06
Nd-143	2.9503E-06	2.4349E-06	2.0778E-06	2.4187E-06	2.6642E-06
Pm-147	8.9385E-07	7.6504E-07	6.6951E-07	7.6083E-07	8.2367E-07
Sm-149	1.6129E-08	1.7992E-08	1.9243E-08	1.8052E-08	1.7141E-08
Sm-151	6.5207E-08	6.3903E-08	6.1481E-08	6.3824E-08	6.4758E-08
H	5.4712E-02	5.4712E-02	5.4712E-02	5.4712E-02	5.4712E-02
C	1.4891E-03	1.4891E-03	1.4891E-03	1.4891E-03	1.4891E-03
Zr	3.5684E-02	3.5684E-02	3.5684E-02	3.5684E-02	3.5684E-02
Hf	2.1410E-06	2.1410E-06	2.1410E-06	2.1410E-06	2.1410E-06
Total	9.3120E-02	9.3128E-02	9.3134E-02	9.3129E-02	9.3125E-02
	F-6	F-7			
U-235	2.1347E-04	2.1105E-04			
U-234	1.8491E-06	1.8446E-06			
U-236	9.6778E-06	1.0065E-05			
U-238	1.0125E-03	1.0122E-03			
Pu-239	3.1928E-06	3.3434E-06			
Pu-240	2.4469E-07	2.7147E-07			
Rh-103	1.0707E-06	1.1321E-06			
Xe-131	9.5023E-07	1.0046E-06			
Cs-133	2.3760E-06	2.5133E-06			
Nd-143	1.9932E-06	2.1034E-06			
Pm-147	6.4611E-07	6.7654E-07			
Sm-149	1.9501E-08	1.9160E-08			
Sm-151	6.0681E-08	6.1705E-08			
H	5.4712E-02	5.4712E-02			
C	1.4891E-03	1.4891E-03			
Zr	3.5684E-02	3.5684E-02			
Hf	2.1410E-06	2.1410E-06			
Total	9.3135E-02	9.3134E-02			

Table 8 Nuclide densities for Fresh 30/20 LEU fuel meat, MCNP Model for WSU Mixed LEU Core 35A

Nuclide	Atomic Mass	Nuclide Density (atoms/b•cm)	Mass (g)
H	1.0079	0.04915763	28.86
C	12.011	0.00178701	12.50
Zr	91.224	0.03227955	1715.37
Er-166	165.93	0.00007717	7.46
Er-167	166.932	0.00005299	5.15
U-234	234.041	0.00000715	0.97
U-235	235.0439	0.00108821	149.00
U-236	236.0456	0.00000627	0.86
U-238	238.0508	0.00432194	599.33
Hf	178.49	1.93677E-06	0.20
Total		0.08877792	2519.51

Geometrical Models

The MCNP model for the full WSU Mixed HEU Core 34A consists of 51 FLIP/HEU SFEs and 68 partially burned 8.5/20 LEU SFEs, and each fuel cluster was averaged over 4 or 3 fuel rods. The nuclide densities for each fuel cluster are shown in Tables 5, 6, and 7. The full WSU Mixed LEU Core 35A core model consists of 51 – 30/20 LEU SFEs and 68 burned 8.5/20 SFEs and each fuel cluster was averaged over 4 or 3 fuel rods. The nuclide densities for each fuel cluster are shown in Tables 6, 7, and 8.

Approach-to-Critical-WSU Mixed HEU Core 34A

Since WSU Mixed HEU Core 34A evolved over a period of time there never was a critical configuration for this core. Therefore no MCNP model was developed for this benchmark case since there is no operational data for comparison.

Approach-to-Critical-WSU Mixed LEU Core 35A

A detailed MCNP model of the just critical WSU Mixed LEU Core 35A reactor was made which includes 47 – 30/20 LEU SFEs and 24 burned 8.5/20 SFEs, 3 blade type control rods, 1 stainless steel servo blade and 1 water-followed transient rod, 20 graphite blocks around the core, illustrated by Figure 16.

The critical case for the WSU Mixed LEU Core 35A was modeled to be in a position away from the thermal column (water reflected on the west side). This configuration was chosen since this is the most reactive arrangement. Figures 17 and 18 are the XY and XZ plots of the MCNP model of the Mixed LEU Core 35A cold critical case.

Figure 16 WSU Mixed LEU Core 35A – Cold Critical Core Configuration

WSU Mixed HEU Core 34A Reactor Model, Full Core (Away from Thermal Column)

A detailed MCNP model was developed for the water reflected WSU Mixed HEU Core 34A which includes 51 FLIP HEU SFEs and 68 burned 8.5/20 LEU SFEs, 3 blade type control rods, 1 stainless steel servo blade and 1 water-followed transient rod. In addition, a 2 inch thick lower cluster adapter, and a 5 inch thick aluminum grid plate below the fuel rods was modeled.

Figures 19 and 20 are the XY and XZ plots of the MCNP model of the full unrodded WSU Mixed HEU Core 34A with infinite water reflector. Figures 21 and 22 are the XY and XZ plots of the MCNP model of the full core for the WSU Mixed HEU Core 34A with all control rods inserted.

WSU Mixed LEU Core 35A Reactor Model, Full Core (Away from Thermal Column)

A detailed MCNP model was developed for the water reflected WSU Mixed LEU Core 35A which includes 51 – 30/20 LEU SFEs and 68 – burned 8.5/20 SFEs, 3 blade type control rods, 1 stainless steel servo blade and 1 water-followed transient rod. In addition, it was assumed that there were 20 graphite blocks around the core, 2 inches thick lower cluster adapter, and a 5 inches thick aluminum grid plate below the fuel rods.

Figures 23 and 24 are the XY and XZ plots of the MCNP model of the full unrodded core WSU Mixed LEU Core 35A. Figures 25 and 26 are the XY and XZ plots of the MCNP model of the full core WSU Mixed LEU Core 35A with all control blades inserted.

The configuration for the Mixed LEU Core 35A is shown in Figure 27.

Figure 17 WSU Mixed LEU Core 35A, Cold Critical Configuration
Radial Model for Monte Carlo (MCNP5) Transport Calculation

Figure 18 WSU Mixed LEU Core 35A, Cold Critical Configuration
Axial Model for Monte Carlo (MCNP5) Transport Calculations

Figure 19 WSU Mixed HEU Core 34A – Unrodded Full Core Configuration Radial Model for Monte Carlo (MCNP5) Transport Calculations

Figure 20 WSU Mixed HEU Core 34A – Unrodded Full Core Configuration Axial Model for Monte Carlo (MCNP5) Transport Calculations

Figure 21 WSU Mixed HEU Core 34A – All Rods Inserted Core Configuration Radial Model for Monte Carlo (MCNP5) Transport Calculations

Figure 22 WSU Mixed HEU Core 34A – All Rods Inserted Core Configuration Axial Model for Monte Carlo (MCNP5) Transport Calculation

Figure 23 WSU Mixed LEU Core 35A – Unrodded Full Core Configuration Radial Model for Monte Carlo (MCNP5) Transport Calculations

Figure 24 WSU Mixed LEU Core 35A – Unrodded Full Core Configuration Axial Model for Monte Carlo (MCNP5) Transport Calculations

Figure 25 WSU Mixed LEU Core 35A – All Rods Inserted Core Configuration Radial Model for Monte Carlo (MCNP5) Transport Calculations

Figure 26 WSU Mixed LEU Core 35A – All Rods Inserted Core Configuration Axial Model for Monte Carlo (MCNP5) Transport Calculations

Figure 27 WSU Mixed LEU Core 35A – General Configuration

Benchmark of Mixed HEU Core

The evaluation of WSU Mixed FLIP HEU Core 34A provided an opportunity to benchmark the computational techniques to be used for evaluating the WSU Mixed LEU Core 35A.

Approach-to-Critical – WSU Mixed HEU Core 34A

Since WSU Mixed HEU Core 34A evolved over a period of time there never was a critical configuration for this core. Therefore no MCNP model was developed for this benchmark case since there is no data for comparison.

Full Unrodded Core Loading – WSU Mixed HEU Core 34A

The full core loading in the water reflected WSU Mixed HEU Core 34A contains 51 – FLIP HEU SFEs and 68 partially burned 8.5/20 LEU SFEs. The analysis for this condition included the fission product and actinides inventories calculated for each fuel cluster of Core 34A based on U-235 depletion. The MCNP calculation gave an unrodded k_{eff} value with one sigma uncertainty:

$$k_{eff} = 1.05038 \pm 0.00016$$

This is equivalent to a reactivity of \$6.31 (β_{eff} = 0.0076, see section 4.4.5). The experimentally determined value was \$6.65, based on measured control rods worth.

Full Core Loading, All Control Rods Inserted – WSU Mixed HEU Core 34A

The full core loading in the water reflected WSU Mixed HEU Core 34A contains 51 – FLIP/HEU SFEs and 68 partially burned 8.5/20 LEU SFEs. The MCNP calculation with all control rods inserted gave a k_{eff} value with one sigma uncertainty:

$$k_{eff} = 0.95010 \pm 0.00016$$

This is equivalent to a reactivity shutdown of -\$6.91. The control system consisting of 3 control blades, a regulating servo blade and a transient rod has a calculated worth of \$13.22. It may be useful to note that the total worth of the experimentally determined individual five control rods was \$12.53.

4.5.2 Critical Core Configuration; Excess Reactivity

The number of fuel rods in the WSU Mixed HEU Core 34A is 51 – FLIP/HEU SFEs and 68 partially burned 8.5/20 LEU SFEs. The proposed core loading for the WSU Mixed LEU Core 35A will be 51 – 30/20 LEU SFEs and 68 burned 8.5/20 LEU SFEs.

Full Unrodded Core Loading – WSU Mixed HEU Core 34A

The full core loading for the WSU Mixed HEU Core 34A contains 51 – FLIP/HEU SFEs and 68 partially burned 8.5/20 LEU SFEs. The MCNP calculation gave an unrodded k_{eff} value with one sigma uncertainty:

$$k_{eff} = 1.05038 \pm 0.00016$$

This is equivalent to a reactivity of \$6.31. The experimentally determined measured value was \$6.65.

Full, Unrodded Core Loading – WSU Mixed LEU Core 35A

The full core loading in the WSU Mixed LEU Core 35A contains 51 – 30/20 LEU SFEs and 68 partially burned 8.5/20 LEU SFEs. The MCNP calculation gives an unrodded k_{eff} value with one sigma uncertainty of:

$$k_{eff} = 1.05019 \pm 0.00017$$

This corresponds to a core reactivity of \$6.37 ($\beta_{eff} = 0.0075$, see section 4.5.5).

4.5.3 Worth of Control Rods

Full Core Loading, All Control Rods Inserted – WSU Mixed HEU Core 34A

The full core loading for the water reflected WSU Mixed HEU Core 34A contains 51 – FLIP/HEU SFEs and 68 partially burned 8.5/20 LEU SFEs. The MCNP calculation with all control rods inserted gave a k_{eff} value with one sigma uncertainty:

$$k_{eff} = 0.95010 \pm 0.00016$$

This is equivalent to a reactivity shutdown of -$6.91. The five control rods have a calculated worth of $13.22.

The individual control rod/blade's worths (calculated and measured) are given in Table 9. The discrepancy between the sum of the calculated individual rod/blade worths, as shown in Table 9 ($10.68), and the calculated worth with all rods/blades out and all rods/blades in ($13.22) is due to the shadowing effects of adjacent control rods/blades when worths are calculated individually.

Table 9 WSU Mixed HEU Core 34A – Control Rod Worth

	Calculated MCNP	Measured
Blade 1 (Shim)	$ 1.32 ± 0.03	$ 1.68
Blade 2 (Shim)	$ 2.89 ± 0.03	$ 3.56
Transient Rod 3	$ 3.22 ± 0.03	$ 3.11
Blade 4 (Shim)	$ 2.86 ± 0.03	$ 3.99
Blade 5 (Servo)	$ 0.40 ± 0.03	$ 0.19
Total	$ 10.68 ± 0.07	$ 12.53

Full Core Loading, All Control Rods Inserted – WSU Mixed LEU Core 35A
The full core loading in the WSU Mixed LEU Core 35A reactor contains 51 – 30/20 LEU SFEs and 68 partially burned 8.5/20 LEU SFEs. The MCNP calculation with all control rods inserted gives a k_{eff} value with one sigma uncertainty:

$$k_{eff} = 0.94717 \pm 0.00017$$

This is equivalent to reactivity shutdown of -$7.44. The calculated six control rods have a combined reactivity worth of $10.97.

The individual calculated control rod/blade's worths are given in Table 10.

Table 10 WSU Mixed LEU Core 35A – Control Rod Worth

	Calculated MCNP
Blade 1 (Shim)	$ 1.34 ± 0.03
Blade 2 (Shim)	$ 2.99 ± 0.03
Transient Rod 3	$ 3.19 ± 0.03
Blade 4 (Shim)	$ 3.02 ± 0.03
Blade 5 (Shim)	$ 0.43 ± 0.03
Total	$ 10.97 ± 0.07

4.5.4 Shutdown Margin for Mixed HEU and Mixed LEU Cores

Shutdown Margin, WSU Mixed HEU Core 34A

As stated in the 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 and the regulating rod (if not scrammable) 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 control rod worths are shown in Table 9. The transient rod is calculated to be most reactive rod at $3.22. Actual measurements have shown that shim blade 4 is the highest worth rod with a worth of $3.99. The calculated shutdown margin (the core excess of $6.31 minus the total worths of blades 1, 2 and 3) is -$0.76 and the measured shutdown margin (the core excess of $6.65 minus the total worths of blades 1, 2 and transient rod) is -$1.70. Both this calculated and measured give adequate shutdown margin for Core 34A.

Shutdown Margin, WSU Mixed LEU Core 35A

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 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.

The MCNP 5 code was run for the WSU Mixed LEU Core 35A with the most reactive rod plus the non-scrammable regulating servo blade up out of the core and the three shim blades inserted in the core. The individual control rod's worth for Core 35A are given in Table 10. The transient rod is calculated to be the most reactive rod at $3.19. The calculated shutdown margin (the core excess of $6.37 minus the total worths of blades 1, 2 and 3) is -$0.98 which gives an adequate shutdown margin for WSU Mixed LEU Core 35A.

4.5.5 Additional Core Physics Parameters for Mixed HEU and Mixed LEU Cores

Effective Delayed Neutron Fraction, βeff – WSU Mixed HEU Core 34A

The effective delayed neutron fraction, β_{eff}, was derived from Monte Carlo calculations of the WSU Mixed HEU Core 34A with all control rods out.

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

$\beta_{eff} = 1 - [K_p / K_t]$

Where: K_p = core reactivity using prompt fission spectrum
K_t = core reactivity using prompt and delayed fission spectrum

The values of K_p and K_t calculated using MCNP are:
$K_p = 1.04244 \pm 0.00016$
$K_t = 1.05038 \pm 0.00016$

Using these values the result for WSU Mixed HEU Core 34A is:

$$\beta_{eff} = 0.0076 \pm 0.0002$$

Effective Delayed Neutron Fraction, βeff – WSU Mixed LEU Core 35A

The effective delayed neutron fraction, β_{eff}, for the WSU Mixed LEU Core 35A is calculated exactly as for the WSU Mixed HEU Core 34A above but with the updated WSU Mixed LEU Core 35A input parameters.

The values of K_p and K_t calculated using MCNP are:
$K_p = 1.04225 \pm 0.00017$
$K_t = 1.05019 \pm 0.00017$

The result for the Mixed TRIGA LEU fuel is:

$$\beta eff = 0.0075 \pm 0.0002$$

Prompt Neutron Life (ℓ) – WSU Mixed HEU Core 34A

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:

$$\ell = 30.7\ \mu\text{sec}$$

Prompt Neutron Life (ℓ) – WSU Mixed LEU Core 35A

Using the same 1/v absorber method described above for the WSU Mixed HEU Core 34A, the prompt neutron life (ℓ) has been evaluated for the WSU Mixed LEU TRIGA fuel in Core 35A.

The result for the prompt neutron life (ℓ) in the unrodded WSU Mixed LEU Core 35A is:

$$\ell = 28.2\ \mu\text{sec}$$

Prompt Negative Temperature Coefficient of Reactivity, α-WSU Mixed HEU Core 34A

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 11 were produced by DIF3D for the listed core temperatures.

Figure 28 is a histogram plot of the computed values of α as a function of reactor temperature.

Table 11 Reactivity Change with Temperature – WSU Mixed HEU Core 34A

Average Core Temperature °C	**k_{eff}**	**Δ k_{eff}**	**$\frac{k_a - k_b}{k_a k_b}$**	**$\alpha_{a,b}$ (Δk/k per °C)**
23	1.04196			
		0.01023	0.009516	5.376E-05
200	1.03173			
		0.00659	0.006231	7.788E-05
280	1.02514			
		0.01181	0.011369	9.474E-05
400	1.01333			
		0.0369	0.037294	1.243E-04
700	0.97643			
		0.04137	0.045311	1.510E-04
1000	0.93506			

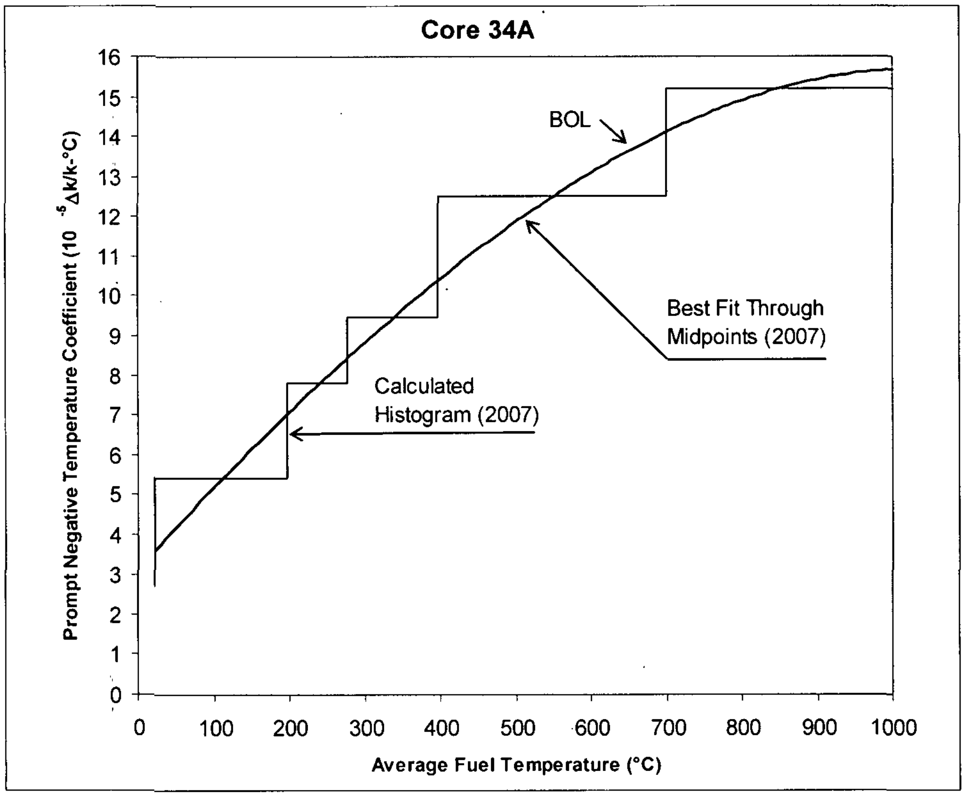

Figure 28 Prompt Negative Temperature Coefficient vs. Fuel Temperature – WSU Mixed HEU Core 34A.

Prompt Negative Temperature Coefficient of Reactivity, α – WSU Mixed LEU Core 35A

Following the procedure set forth in the section for the WSU Mixed HEU Core 34A, the calculated results from DIF3D for the WSU Mixed LEU Core 35A are listed in Table 12 for Beginning-of-Life (BOL) and in Table 13 for End-of-Life (EOL).

Figure 29 is a histogram plot of the computed values for α in Tables 12 and 13 as a function of core temperature for both BOL and EOL.

In Figure 29, it can be seen that the prompt negative temperature coefficient (α) for WSU Mixed LEU Core 35A has only a modest decrease in values at 1000 MWD burnup (EOL) (e.g., 12.5 x 10^{-5} to 10.5 x 10^{-5} Δk/k-°C at 700-1000°C).

Table 12 Reactivity Change with Temperature, WSU Mixed LEU Core 35A, (BOL)

Average Core Temperature ° C	**k_{eff}**	**Δk_{eff}**	**$\frac{k_a - k_b}{k_a k_b}$**	**$\alpha_{a,b}$ – ($\Delta k/k$ - °C)**
23	1.03863			
		0.0113	0.010590	5.983E-05
200	1.02733			
		0.00646	0.006160	7.699E-05
280	1.02087			
		0.01101	0.010680	8.900E-05
400	1.00986			
		0.03237	0.032792	1.093E-04
700	0.97749			
		0.03506	0.038058	1.269E-04
1000	0.94243			

Table 13 Reactivity Change with Temperature, WSU Mixed LEU Core 35A, (EOL)

Average Core Temperature ° C	**k_{eff}**	**Δk_{eff}**	**$\frac{k_a - k_b}{k_a k_b}$**	**$\alpha_{a,b}$ – ($\Delta k/k$ - °C)**
23	1.00916			
		0.01018	0.010098	5.705E-05
200	0.99898			
		0.00556	0.005603	7.003E-05
280	0.99342			
		0.00923	0.009440	7.867E-05
400	0.98419			
		0.02626	0.027854	9.285E-05
700	0.95793			
		0.02841	0.031906	1.064E-04
1000	0.92952			

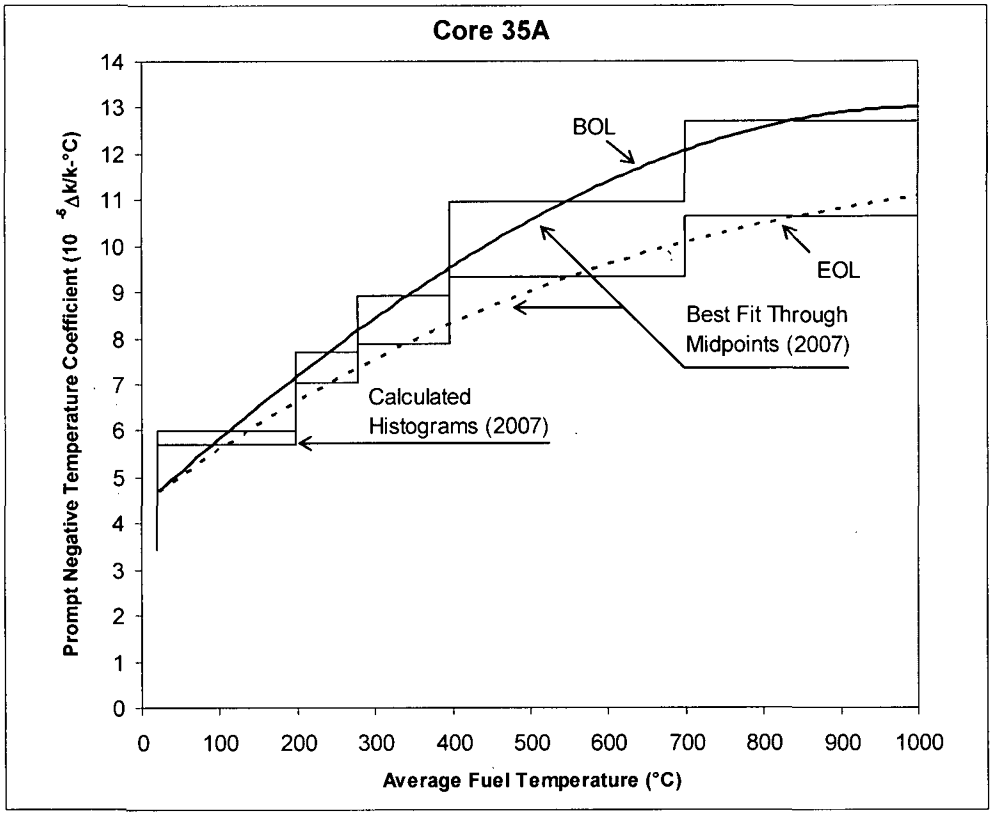

Figure 29 Prompt Negative Temperature Coefficient – WSU Mixed LEU Core 35A Fuel, Beginning of Life (BOL) and End of Life (EOL)

Void Coefficient – WSU Mixed HEU Core 34A

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 WSU Mixed HEU Core 34A reactor, the calculated void coefficient is about 0.080% 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 placing a dry, experimental tube, with a void volume of 205 cc in the 38.1 cm core fueled height, in the central region of the core (replacing a fuel rod) and then being accidentally flooded with water. The calculated loss in core reactivity would be about \$ 0.10 ± 0.03. 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 \$ 0.10. The insertion of \$ 0.10 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 – WSU Mixed LEU Core 35A

The void coefficient for the WSU Mixed LEU Core 35A is 0.135% $\Delta k/k$ per 1% water void. Using the same example from above in placing a dry, experimental tube, with a void volume of 205 cc in the 38.1 cm core fueled height, in the central region of the core (replacing a fuel rod) and then being accidentally flooded with water, the prompt gain in reactivity is about \$ 0.18 ± 0.03. 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 – WSU Mixed LEU Core 35A

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 280°C, the closest nuclear data available.

Figure 30 shows the results from design calculations for core excess reactivity as a function of burnup. 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 30 gives a lifetime of the initial core (with no fuel shuffling) of about 1000 MWD at 1.0 MW, full equilibrium xenon poisoning, and about \$ 0.40 reactivity left for burnup or experiments. For comparison, a burnup curve for WSU Mixed HEU Core 34A was calculated and is also shown in Figure 30. The WSU Mixed HEU Core 34A burnup curve gives a lifetime of the initial core (with no fuel shuffling) of about 1000 MWD at 1.0 MW, full equilibrium xenon poisoning, and about \$0.67 reactivity remaining.

The data on burnup at the 3-day interval indicates a xenon equilibrium poison value of about \$1.38. 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 \$1.59. 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.

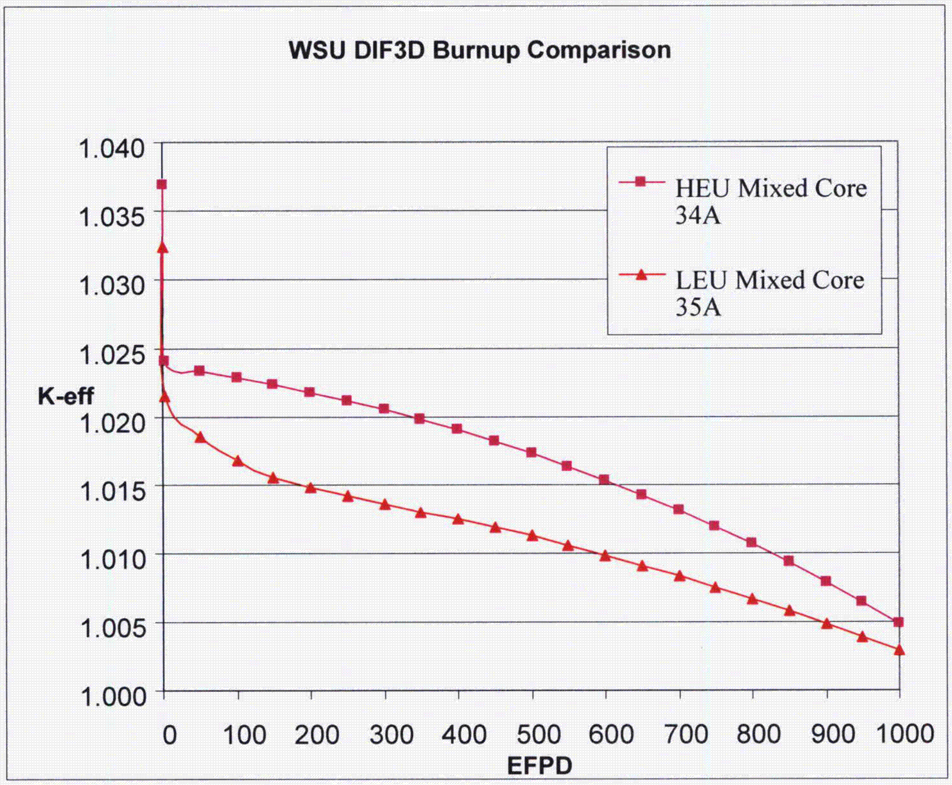

Figure 30 Reactivity versus Burnup for 1 MW TRIGA HEU Mixed and LEU Mixed (30/20) Cores

Reactor Parameters at 1000 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 1000 MWD burnup. The value obtained is

$$\beta = 0.0073 \pm 0.0002,$$

which is only slightly less than the beginning of core life value of 0.0075.

Similarly, using the procedure set forth above for the prompt neutron life, the prompt neutron life (ℓ) at 1000 MWD burnup has been evaluated for the LEU (30/20) core. The value obtained is

$$\ell = 27.0\ \mu\text{sec}$$

which is slightly less than the beginning of core life value of 28.2 μsec.

The values of the prompt negative temperature coefficient for TRIGA LEU (30/20) fuel at 1000 MWD burnup have already been presented in Tables 12 and 13 and shown in Figure 29 compared with the values at beginning of life.

4.5.7 Reactivity Loss at Reactor Power

Reactivity Loss, WSU Mixed HEU Core 34A

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 1.3 MW, respectively. During the operation of WSU Mixed HEU Core 34A, some measurements were made at 1.0 MW to evaluate the reactivity losses. The calculated and measured values of reactivity losses are tabulated in Table 14.

Table 14 Calculated and Measured Reactivity Loss, WSU Mixed HEU Core 34A

P(MW)	(\$)	$\Delta\rho(\$)_{calc}$	$\Delta\rho(\$)_{meas}$
0	5.299		
0.5	4.431	0.87	
1.0	3.845	1.45	2.20
1.3	3.716	1.58	

As can be seen from a comparison of measured and calculated reactivity loss, the measured reactivity loss at 1.0 MW is greater than the calculated value. The magnitude of the reactivity losses are related directly to the calculated temperatures. In Table 26, the measured temperatures ($T_{0.3}$) agree well with the calculated $T_{0.3}$ temperatures at 1.0 MW. Thus, the lack of agreement for reactivity losses could be attributed to a larger gap in low power elements resulting in higher average core temperature. If the core average temperature is 290°C instead of 221°C, Table 26, then the reactivity loss is calculated to be $2.18.

Reactivity Loss, WSU Mixed LEU Core 35A

Calculations of core reactivity were made for operating power levels up to 1.3 MW. These calculated values of the loss in reactivity are tabulated in Table 15. The reactivity loss at 1.0 MW is $2.52 (cold-hot reactivity swing). After burnup to 1000 MWD, the reactivity loss at 1.0 MW is predicted to drop to $1.60 partly due to a lower core average temperature at EOL.

Table 15 Calculated Reactivity Loss, WSU Mixed LEU Core 35A

P(MW)	($)	$\Delta\rho$($)
0	4.959	0
0.5	3.129	1.83
1.0	2.436	2.52
1.3	2.194	2.76

4.5.8 Power Peaking; WSU Mixed HEU Core 34A

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 (RPF) – The power generation in a fuel rod (element) relative to the core averaged rod power generation.

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

3. $(\hat{P}_{rod} / \overline{P}_{rod})_{radial}$; Intra Rod peaking factor (Intra Rod), 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 the RPF. The maximum value of this factor for the hottest rod, the 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 Intra Rod peaking factor is of importance in the transient analysis for calculating maximum fuel temperatures in the time range where the heat 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 WSU Mixed HEU Core 34A are shown in Table 16 and the axial power distribution are shown in Figure 31. Values are shown for the Hot Rod (D4NE), Average Rod, and the two IFE fuel Elements located in positions D6NW and C4NW.

Fuel temperatures for selected reactor power levels have also been calculated for the hottest and average fuel rods. These results are presented in Table 26 together with the experimentally measured fuel temperature for 1.0MW.

It will be noted that the instrumented fuel elements are located in core locations D6NWand C4NW, these locations are not the hottest core location. Since the sensing tip of the thermocouple is 0.30 inches from the axial centerline of the fuel element, the temperatures reported in Table 26 are calculated for the hottest radial position ($\hat{T}$) and for a position 0.30 inches from the center line ($\hat{T}_{0.3}$). Finally, the average core temperature ($\overline{T}_{avg\ core}$) was calculated.

Table 16 Power Peaking Factors – WSU Mixed HEU Core 34A

Current Operating Condition – Rods at Critical Positions						
	Cold Critical - 23°C			Hot Critical - 280°C		
	RPF	APF	Intra-Rod	RPF	APF	Intra-Rod
Hot Rod	2.56	1.27	2.01	2.49	1.29	1.99
Ave Rod	1.00	1.27	1.50	1.00	1.33	1.50
IFE 1 – D6NW	1.77	1.27	0.85	1.69	1.28	0.85
IFE 2 – C4NW	1.39	1.13	0.85	1.51	1.44	0.85

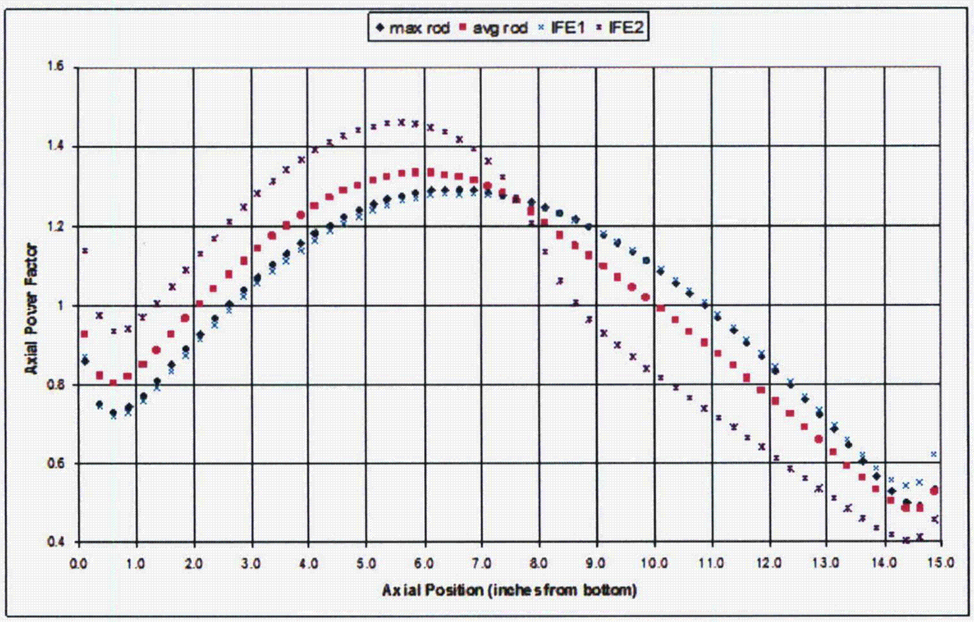

Figure 31 WSU Mixed HEU Core 34A – Axial Power Profile versus Distance from Bottom of Fueled Section

4.5.9 Power Peaking; WSU Mixed LEU Core 35A

The peaking factors calculated for the WSU Mixed LEU Core 35A at both BOL and EOL are shown in Table 17 and 18. The axial power distributions are shown in Figure 32.

Table 17 Power Peaking Factors – WSU Mixed LEU Core 35A - BOL

Beginning of Life _ Rods at Critical Positions						
	Cold Critical - 23°C			**Hot Critical - 280°C**		
	RPF	APF	Intra-Rod	RPF	APF	Intra-Rod
Hot Rod	2.56	1.27	1.35	2.47	1.29	1.19
Ave Rod	1.00	1.29	1.55	1.00	1.33	1.55
IFE 1 – D6NW	1.73	1.26	0.51	1.64	1.27	0.45
IFE 2 – C4NW	1.43	1.17	0.51	1.56	1.44	0.45

Table 18 Power Peaking Factors – WSU Mixed LEU Core 35A – All rods out, EOL

End of Life – All Rods Out			
	Hot Critical - 280°C		
	RPF	APF	Intra-Rod
Hot Rod	2.33	1.27	1.29
Ave Rod	1.00	1.25	1.52
IFE 1 – D6NW	1.55	1.26	0.49
IFE 2 – C4NW	1.78	1.24	0.49

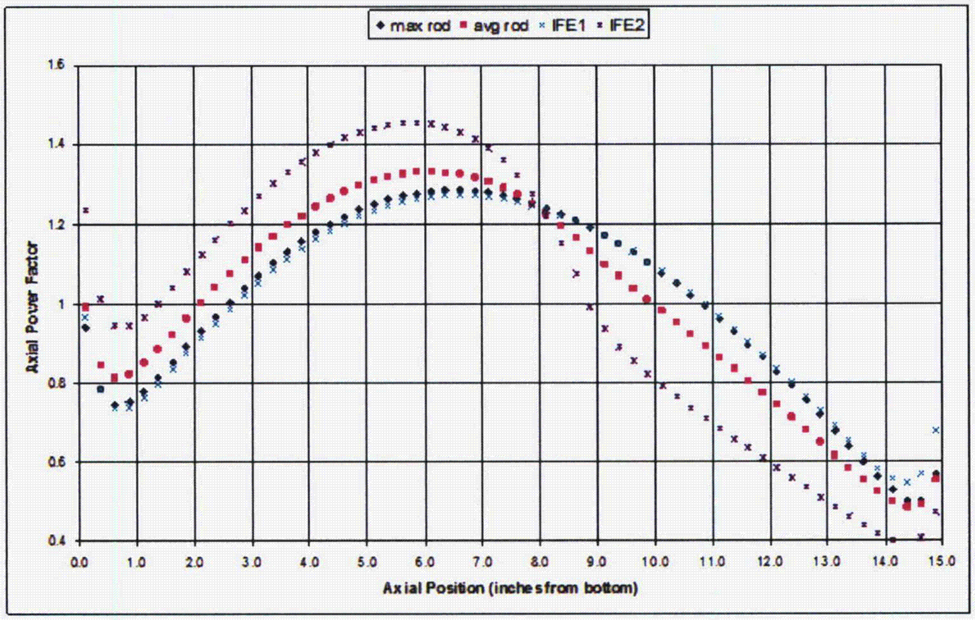

Figure 32 WSU Mixed LEU Core 35A – Axial Power Profile versus Distance from Bottom of Fueled Section

4.5.10 Pulsing Operation, WSU Mixed HEU Core 34A

Most of the 65 TRIGA reactors have pulsing capability. Thousands of TRIGA reactor pulses have been safely performed. A calculation 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 WSU Mixed HEU Core 34A is a compact core with no in-core experimental regions that could be water filled. However, the transient rod (Rod 3) is water-followed and thus constitutes a region of enhanced power peaking. This region is correctly modeled in the codes that are used.

The BLOOST pulsing performance results have been prepared for the currently operating WSU Mixed HEU Core 34A and are shown in Table 19 for reactivity insertions of \$1.50, \$1.75, \$2.00, \$2.30, \$2.50. Also shown in Table 19 is the peak fuel temperature in the core, fuel element (D4NE), and where available, the measured temperature results ($\hat{T}_{0.3}$) for the instrumented fuel elements in positions D6NW and C4NW are also shown.

Figure 33 shows a plot of the pulsed fuel temperatures as a function of reactivity insertion calculated using an assumed fuel to cladding gap of 0.2mils. Since the peak pulsed fuel temperature will be limited to 830°C during a pulse, the reactivity insertion (\$2.02) to produce this temperature is also indicated in Figure 33.

Measured data from pulses performed at various times during the operation of Core 34A are included. These data are also presented in Table 19. Table 20 illustrates the Calculated Current Pulse Performance for WSU Mixed HEU Core 34A, along with the Calculated Pulse Performance for Core 35A, at BOL. Numerous test pulses have been performed on the WSU Mixed HEU core 34A, while also recording relevant data parameters as functions of reactivity added, including peak temperatures for the two IFE's (D6NW and C4NW), energy release, and peak pulse power. Table 19A provides pulsing data for the WSU Mixed core. There are six \$2.00 pulses, and three \$2.15 pulses included in the table. Calculations performed at WSU have indicated that the reactor may be safely pulsed to \$2.20 with core configuration 34A. The data provided in Table 19A include results of direct measurements of energy release, as indicated on the Nuclear Power Pulse Channel, NPP-1000. The functionality of the NPP-1000 channel is confirmed by comparison with the Nuclear Multi-Range Linear channel, NMP-1000, during routine operations. Comparison of the actual measured energy release with the modeled values shows that the calculated values consistently over-predict the pulse energy release. As a result, the BLOOST code output for core 34A is very conservative. Since the same code and procedures were used to develop the model for core 35A, it is reasonable to conclude that the model for core 35A is also very conservative.

Initially, WSU will limit pulsing in core 35A to \$2.02, as described in this SAR. WSU will conduct a series of test pulses with reactivity values less than \$2.02, measuring the pulsing parameters, including peak power and total energy release, to experimentally determine a safe pulsing limit for core 35A.

The pulse is completed about 0.3 seconds after pulse initiation at which time the peak fuel temperature is computed. The energy results 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 is initiated. 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, in core position D4NE, is adjacent to the transient rod, reflecting this power peaking.

Table 19 Pulse performance: Measured and Calculated, WSU Mixed HEU Core 34A

Parameter	Pulse				
	\$1.50	**\$1.75**	**\$2.00**	**\$2.30**	**\$ 2.50**
Measured Data (a)					
$\hat{P}$ (MW)	240	440	1030		
E(MW – sec)	16	20	25	N/A	N/A
$\hat{T}_{0.3}$ (°C)					
D6NW	279	310	344		
C4NW	254	281	313		
BLOOST-calculation					
$\hat{P}$ (MW)	649	1321	2206	3537	4580
E(MW – sec)	19	26	32	40	45
$\hat{T}$ (°C) (D4NE)	558	701	820	954	1030
$\bar{T}$ core (°C)	201	252	300	356	387
$\hat{T}_{03}$ (°C)					
D6NW	260	313	358	405	436
C4NW	201	241	276	316	341

(a) Pulse data taken from test pulses performed on November 21, 2005

Table 20A Historical Pulsing Data for WSU Core 34A

Deleted: 19

Pulse number	Date	Reactivity added	$\hat{T}_{0.3}$ D6NW	$\hat{T}_{0.3}$ C4NW	Peak Power (MW)	Energy (MW•s)
1040	11/21/2005	1.25	242	227	60	11.5
1041	11/21/2005	2.00	332	317	1200	24
1043	11/21/2005	1.50	279	254	-----	16
1044	11/21/2005	1.50	279	254	240	16
1045	11/21/2005	1.75	310	281	440	20
1046	11/21/2005	2.00	344	313	1030	25
1047	12/5/2005	1.25	241	217	120	12.4
1048	5/31/2006	1.25	246	223	120	11.6
1049	5/31/2006	1.75	305	279	500	17.2
1050	5/31/2006	2.00	378	309	1000	20
1051	11/6/2006	1.25	259	230	160	12
1052	11/6/2006	1.50	292	252	260	13
1053	11/6/2006	1.75	319	286	480	17.2
1054	11/6/2006	2.00	355	317	----	21
1055	11/6/2006	2.15	375	334	1420	25
1056	11/6/2006	2.15	382	340	1420	24
1057	12/14/2006	1.75	317	281	1200	----
1058	1/29/2007	1.25	261	228	190	11.9
1059	1/29/2007	2.15	376	335	1420	24.8
1060	1/29/2007	2.00	355	315	1000	22
1061	1/29/2007	0.75	92	78	0	0
1062	1/29/2007	1.01	218	188	399	7.8
1063	1/29/2007	1.03	222	193	399	8.3
1064	5/14/2007	1.25	256	-----	160	11
1065	5/14/2007	2.00	354	316	1000	21.5
1066	5/14/2007	1.50	292	259	220	14.3
1067	5/14/2007	1.75	320	284	440	16.8
1068	5/14/2007	1.25	257	229	----	10.5
1069	5/14/2007	1.50	290	257	210	14.2
1070	8/7/2007	1.50	294	260	300	15

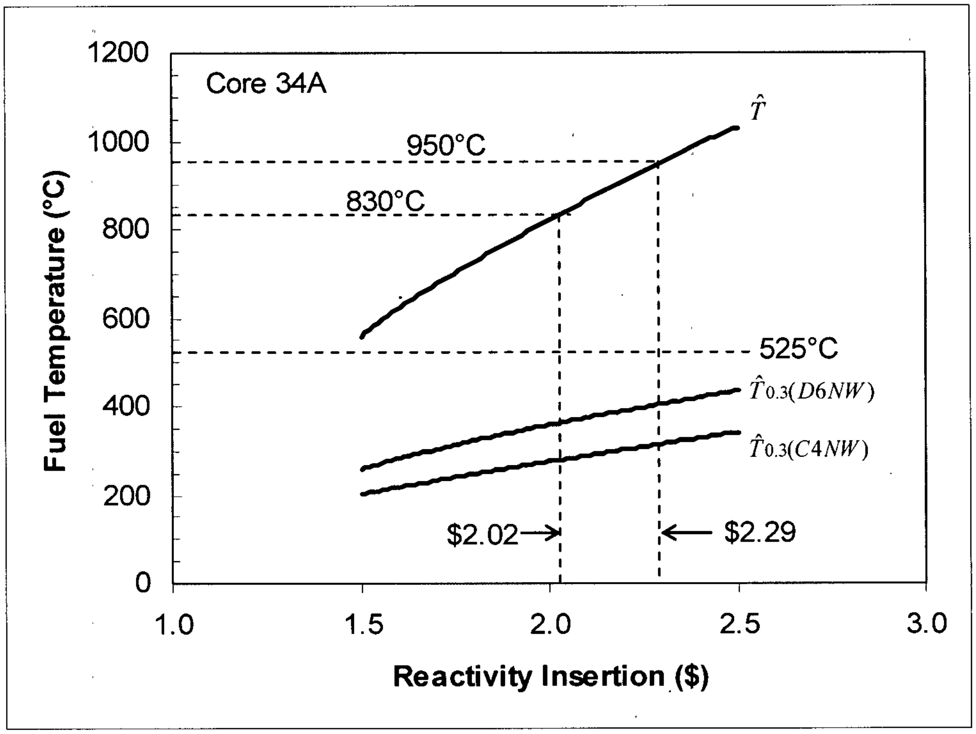

Figure 33 WSU Mixed HEU Core 34A – Pulse Performance at Current - Operating Conditions.

4.5.11 Pulse Operation – WSU Core Mixed LEU Core 35 A – BOL

The WSU 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 28, and for LEU (30/20) fuel, Figure 29, one notes similar temperature dependences; however, the magnitude of α is somewhat smaller for the LEU fuel. The BOL neutron lifetime is 28.2 µsec; the EOL neutron lifetime is 27.0 µsec.

Table 19A presents the beginning of core life pulsing parameters for the WSU Mixed LEU Core 35A. Results for the WSU Mixed HEU Core 34A are included for easy comparison. Results for reactivity insertions of $1.50, $1.75, $ 2.00, $2.30, $2.50, $2.75 and $3.19 are shown. The peak fuel temperature in the core (D4NE) is listed. Calculated temperature results ($\hat{T}_{0.3}$) are shown for the instrumented fuel elements in positions D6NW and C4NW. Figure 34 shows a plot of the pulsed fuel temperatures as a function of reactivity insertion for the WSU Mixed LEU Core 35A fuel at BOL.

Since the peak pulsed fuel temperature will be limited to 830°C during a pulse, the reactivity insertion ($2.04) to produce this temperature is also indicated in Figure 34. Table 20 also shows that the maximum insertion of the transient-rod reactivity required to reach the fuel temperature Safety Limit of 1150°C is slightly larger than $2.75.

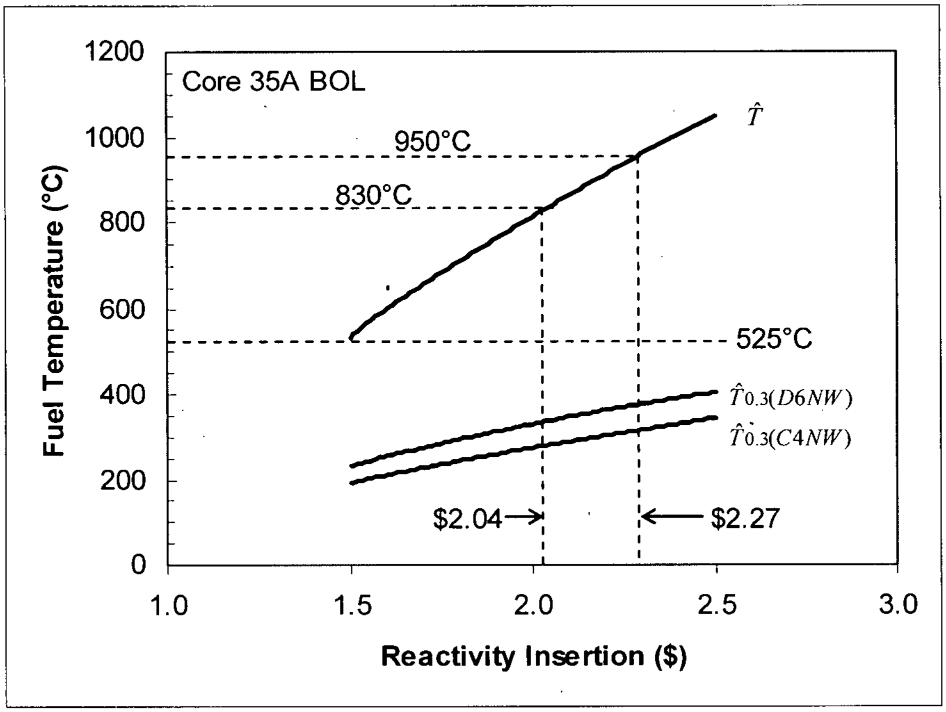

Figure 34 Pulse Performance at Beginning of Core Life, WSU Mixed LEU Core 35A

Table 21 Calculated Pulse Performance for WSU Mixed HEU Core 34A, Current Performance, and WSU Mixed Core 35A, BOL

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Parameter	WSU Mixed HEU Core 34A – Current					WSU Mixed LEU Core 35A – BOL						
	$1.50	$1.75	$2.00	$2.30	$2.50	$1.50	$1.75	$2.00	$2.30	$2.50	$2.75	$3.19
$\hat{P}$ (MW)	649	1321	2206	3537	4580	546	1143	1969	3222	4210	5506	6686
E (MW-sec)	19	26	32	40	45	16	22	29	36	41	46	52
$\hat{T}$ (°C) (D4NE)	558	701	820	954	1030	533	683	812	961	1046	1132	1234
$\overline{T}_{core}$ (°C)	201	252	300	356	387	174	227	276	334	367	401	442
$\hat{T}_{0.3}$ (°C)												
D6NW	260	313	358	405	436	232	285	331	379	405	437	479
C4NW	201	241	276	316	341	192	236	275	318	345	375	403

Table 22 Calculated Pulse Performance for WSU Mixed LEU Core 35A, BOL, and WSU Mixed Core 35A, EOL

Deleted: 21

Parameter	WSU Mixed LEU Core 35A – BOL							WSU Mixed LEU Core 35A - EOL						
	$1.50	$1.75	$2.00	$2.30	$2.50	$2.75	$3.19	$1.50	$1.75	$2.00	$2.30	$2.50	$3.00	$3.19
$\hat{P}$ (MW)	546	1143	1969	3222	4210	5506	6686	506	1099	1914	3172	4202	6373	6816
E (MW-sec)	16	22	29	36	41	46	52	16	22	29	36	42	52	55
$\hat{T}$ (°C) (D4NE)	533	683	812	961	1046	1132	1234	465	608	733	878	961	1119	1154
$\overline{T}_{core}$ (°C)	174	227	276	334	367	401	442	168	224	275	337	373	441	455
$\hat{T}_{0.3}$ (°C)														
D6NW	232	285	331	379	405	437	479	206	256	301	350	379	430	444
C4NW	192	236	275	318	345	375	403	227	281	330	381	408	473	489

4.5.12 Pulse Operation – WSU Mixed LEU Core 35A – EOL

The BLOOST code has been used to calculate the pulsing performance of the WSU Mixed LEU Core 34A at EOL, ~1000 MWD burnup. The procedure is the same as used above for BOL conditions. Results are shown in Table 21 for reactivity insertions of $1.50, $1.75, $2.00, $2.30, and $2.50, $3.00 and $3.19. Results are presented for peak pulsed power, integrated energy, peak fuel temperature in hottest fuel rod, average reactor core temperature, and predicted thermocouple temperatures.

Figure 35 illustrates the dependence of fuel temperatures on reactivity insertions at 1000 MWD burnup. As can be seen from Figure 35 a reactivity insertion of $2.20 is required at the EOL conditions of ~1000 MWD burnup to reach the limiting temperature condition of 830° C. In view of the fact that the peak fuel temperatures ($\hat{T}$) at 1000 MWD are lower rather than higher than the initial peak fuel temperature, the Limiting Reactivity Insertion for pulse operation can safely remain at $2.04, as determined earlier.

Similarly, the maximum possible accidental pulse of $3.19 (calculated worth at transient-rod) is required to just reach the Safety Limit of 1150°C. This is slightly higher than the $2.75 insertion required at BOL conditions as indicated in Table 21.

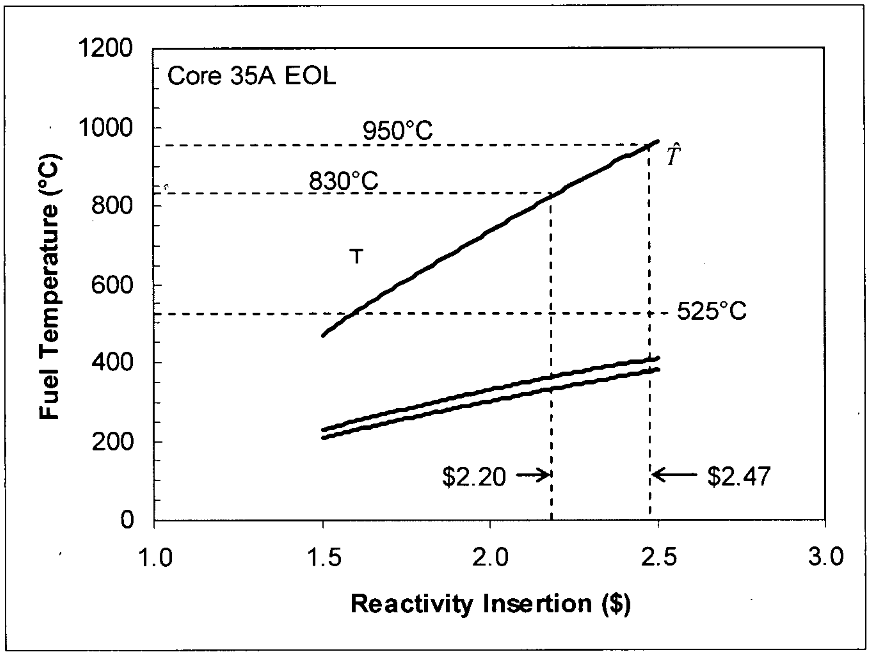

Figure 35 Pulse Performance at 1000 MWD Burnup – WSU Mixed LEU Core 35A

4.6 Functional Design of the Reactivity Control System

No changes in the reactivity control system are required.

4.7 Thermal-Hydraulic Analysis – WSU Core 34A

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 WSU Mixed HEU Core 34A, at current operating conditions, was determined for the reactor operating at 1.0 MW and a water inlet temperature of 30°C. Operational data are used for the benchmark comparisons. An average powered fuel rod and a maximum powered fuel rod were analyzed. The RELAP5 computer code, (Reference 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 reactor power at which critical heat flux is predicted to occur was calculated with the aid of the RELAP5 code. The TAC2D thermal analysis code, (Reference 10), was used to determine the fuel average and fuel maximum temperatures.

4.7.2 RELAP5 Code Analysis and Results.

RELAP5 is a computer program for calculating thermal hydraulics in nuclear reactors. In this application it is used to calculate the natural convection flow through a vertical water coolant channel bounded by cylindrical heat sources. Output from the RELAP5 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) Pool height above the core;

4) Inlet and exit pressure loss coefficients;

5) Inlet water temperature.

Analysis is performed on two flow channels divided into axial segments that represent a single fuel element and the entire core. The natural convection system for the WSU 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 hydraulic data for the RELAP5 input are given in Table 22.

Table 23 RELAP5 Input for Reactor and Core Geometry and Heat Transfer, WSU

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Core and Reactor Geometry	
Unheated core length at inlet, mm	100
Unheated core length at outlet, mm	124
Distance from top of pool surface to top of core, mm	6540
Hydraulic Data	
Inlet pressure loss coefficient	2.02
Exit pressure loss coefficient	1.38
Ambient pressure at pool surface, MPa	0.101

A RELAP5 thermal hydraulic analysis was done for an average flow channel, a maximum powered channel and the two IFE channels. The analysis was conducted by considering the hydraulic characteristics of a typical flow channel represented by the geometric data given in Table 23. The thermal and hydraulic parameters for the WSU Mixed HEU Core 34A are given in Table 24.

Table 24 Hydraulic Flow Parameters For a Flow Channel

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Flow area (mm^2/rod)	501.5
Wetted perimeter (mm/rod)	112.6
Hydraulic diameter (mm)	17.82
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 piece-wise fashion to represent the curves in Figure 31. There are 119 fuel elements in the initial core, 51 FLIP HEU SFEs and 68 partially burned 8.5/20 LEU SFEs. The hot-rod power ratio is 2.490, Table 16, for hot core conditions.

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 friction 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.

A summary of the RELAP5 results for WSU is given in Table 25

Table 25 TRIGA Thermal and Hydraulic Parameters for WSU Mixed HEU Core 34A, 1.0 MW

Deleted: 24

Parameter	Initial Core
Number of fuel elements	119
Diameter, mm (in.)	35.842 (1.411)
Length (heated), mm (in.)	381 (15.0)
Core flow area, mm^2 (ft^2)	59677 (0.64236)
Core wetted perimeter, mm (ft.)	13,399 (43.96)
Flow channel hydraulic diameter, mm (ft.)	17.82 (0.05845)
Core heat transfer surface, m^2 (ft^2)	5.105 (54.95)
Hot rod factor	2.490
Axial peaking factor	1.290
Inlet coolant temperature, °C (°F)	30 (86)
Coolant saturation temperature, °C (°F)	114 (238)
Exit coolant temperature (average), °C (°F)	61.21 (142.2)
Exit coolant temperature (maximum), °C (°F)	84.31 (183.8)
Coolant mass flow, kg/sec (lb/hr)	7.67 (60,836)
Average flow velocity, mm/sec (ft/sec)	130 (0.425)
Peak fuel temperature in average fuel element, °C (°F)	296 (564)
Maximum wall temperature in hottest element, °C (°F)	142.7 (288.9)
Peak fuel temperature in hottest fuel element, °C (°F)	435 (815)
Core average fuel temperature, °C (°F)	221 (429)
Average heat flux, W/cm^2 ($BTU/hr\text{-}ft^2$)	19.59 (62,112)
Maximum heat flux in hottest element, W/cm^2 ($BTU/hr\text{-}ft^2$)	62.75 (198,954)
Minimum DNB ratio	2.47

Table 26 Steady State Results for WSU Core 34A, 1.0 MW

Deleted: 25

Initial Core (119 elements – 51 FLIP HEU SFEs, 68 Burned 8.5/20 LEU SFEs)	**Average Rod**	**Maximum Rod**
Channel natural convection mass flow rate, kg/sec	0.0644	0.0921
Exit coolant flow temperature, °C	61.21	84.31
Maximum wall heat flux, W/cm^2	25.92	62.75
Maximum flow velocity, cm/sec	13.07	18.95
Maximum clad temperature, °C	130.1	142.7
Exit clad temperature, °C	121.1	130.6

The RELAP5 code also calculates the critical heat flux, that is, the heat flux at which there is a departure from nucleate boiling (DNB) and the transition to film boiling begins. The correlation used in RELAP5 Mod. 3.2 to calculate this heat flux is the Groeneveld 1986 Correlation. (Reference 11) A second correlation historically used by TRIGA reactors is due to Bernath. (Reference 12) Bernath gives a lower value for the critical heat flux compared to the 1986 Groeneveld correlation used in RELAP5 Mod. 3.2. The lower critical heat flux from the Bernath correlation 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 RELAP5 code analysis has been run for the critical heat flux for the WSU Core 34A reactor operating at 1.0 MW at benchmark conditions. The data was obtained using an inlet temperature of 30°C and systematically increasing the reactor power until RELAP5 indicated a DNB ratio equal to one based on the Bernath correlation. The maximum power per fuel element for which the DNB ratio is 1, is 51.7 kW/element. For a core with a rod peaking factor of 2.490, this maximum fuel element power corresponds to a maximum reactor power of 2.47 MW (51.7 kW per rod/2.490 × 119 rods = 2.47 MW). Hence, the DNB ratio for the WSU Core 34A at the stated conditions is 2.47. The minimum DNB ratio is 1.9 at a power level of 1.3MW.

4.7.3 TAC2D Fuel Temperature Analysis and Results.

The TAC2D general purpose heat conduction code was used to calculate steady state maximum and average fuel temperatures for the average powered rod, the maximum powered rod, and the two IFEs. 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 surface heat transfer coefficient and coolant temperature from the RELAP5 code is used to model the outer radial boundary. The fuel-to-clad interface conductance assumes the fuel pin is sealed with air. In order to account for gap closure due to fuel swelling, the cold gap was assumed to be 0.2 mils throughout the core. Based on the comparison of calculated versus measured reactivity loss in Section 4.5.7, it appears that the average core gap is slightly larger than 0.2 mils. Some additional gap closure occurs due to the relative expansion of the fuel and cladding at normal operating temperatures.

A summary of the TAC2D results for WSU Core 34A has been given in Table 26 for 0.5, 1.0, and 1.3 MW operations with 119 fuel elements.

Table 27 Calculated and Measured Fuel Temperatures, WSU Mixed HEU Core 34A

Deleted: 26

P(MW)	$\hat{T}_{meas}$ (°C)	T_{calc} (°C)			
		$\hat{T}$	$\hat{T}_{0.3}$		$\overline{T}_{avg\ core}$
			D6NW	C4NW	
0.5		332	255	258	156
1.0	302-304	435	318	321	221
1.3		520	371	375	234

4.8 Thermal Hydraulic Analysis – WSU Mixed LEU Core 35A

4.8.1 Analysis of Steady State Operation – WSU Mixed LEU Core 35A.

The following evaluation has been made for the TRIGA 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 WSU Core 35A 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 RELAP5 computer code 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 RELAP5 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 was used to determine the fuel average and fuel maximum temperatures.

4.8.2 RELAP5 Code Analysis and Results – WSU Mixed LEU Core 35A

The RELAP5 analysis was performed using the method outlined in Section 4.7. The reactor geometry and hydraulic data for the RELAP5 input are given in Table 22. The natural convection system for the WSU 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 RELAP5 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 23. The thermal and hydraulic parameters for the WSU Mixed LEU Core 35A are given in Table 27.

The heat generation in the fuel element is distributed axially in a piece-wise fashion to represent the curves in Figure 32. It is further given that there are 119 fuel elements in the initial core, 51 – 30/20 LEU SFEs AND 68 partially burned 8.5/20 SFEs. The hot-rod power ratio is assumed to be 2.474 for hot core conditions.

A summary of the RELAP5 results for the 1.0 MW WSU Core 35A is given in Table 28.

Table 28 TRIGA Thermal and Hydraulic Parameters for WSU Mixed LEU Core 35A, 1.0 MW

Deleted: 27

Parameter	Initial Core
Number of fuel elements	119
Diameter, mm (in.)	35.842 (1.411)
Length (heated), mm (in.)	381 (15.0)
Core flow area, mm^2 (ft^2)	59,677 (0.64236)
Core wetted perimeter, mm (ft.)	13,399 (43.96)
Flow channel hydraulic diameter, mm (ft.)	17.82 (0.05845)
Core heat transfer surface, m^2 (ft^2)	5.105 (54.95)
Hot rod factor	2.474
Axial peaking factor	1.286
Inlet coolant temperature, °C (°F)	30 (86)
Coolant saturation temperature, °C (°F)	114 (238)
Exit coolant temperature (average), °C (°F)	61.22 (142.2)
Exit coolant temperature (maximum), °C (°F)	84.06 (183.3)
Coolant mass flow, kg/sec (lb/hr)	7.67 (60,858)
Average flow velocity, mm/sec (ft/sec)	130 (0.425)
Peak fuel temperature in average fuel element, °C (°F)	399 (751)
Maximum wall temperature in hottest element, °C (°F)	142.6 (288.6)
Peak fuel temperature in hottest fuel element, °C (°F)	500 (932)
Core average fuel temperature, °C (°F)	303.7 (578.7)
Average heat flux, W/cm^2 (BTU/hr-ft^2)	19.59 (62,112)
Maximum heat flux in hottest element, W/cm^2 (BTU/hr-ft^2)	62.12 (196,970)
Minimum DNB ratio	2.45

Table 29 Steady State Results for WSU Core 35A, 1.0 MW

Deleted: 28

Initial Core (119 elements)	**Average Rod**	**Maximum Rod**
Channel natural convection mass flow rate, kg/sec	0.0644	0.0919
Exit coolant flow temperature, °C	61.22	84.06
Maximum wall heat flux, W/cm^2	25.93	62.12
Maximum flow velocity, cm/sec	13.08	18.90
Maximum clad temperature, °C	129.7	142.6
Exit clad temperature, °C	121.6	130.9

4.8.3 TAC2D Fuel Temperature Analysis and Results – WSU Mixed LEU Core 35A

Using the methods given in Section 4.7.3, a TAC2D analysis was performed for the 1.0 MW WSU Core 35A for operation with 119 fuel elements.

The fuel temperatures for WSU Mixed LEU Core 35A steady state operation have been calculated for the hottest, measured, and average fuel rods. These results are presented in Table 29. The value for $\hat{T}_{0.3}$ is the thermocouple temperature that is located 0.3 inch from the fuel centerline.

Table 30 Calculated Fuel Temperatures, WSU Mixed LEU Core 35A

Deleted: 29

P (MW)	$\hat{T}$ (°C)	$\hat{T}_{0.3}$ (°C)		$\overline{T}_{core}$ (°C)
		D6NW	**C4NW**	
0.5	407	350	361	242
1.0	500	427	440	304
1.3	541	457	469	327

It will be noted that neither of the instrumented fuel elements are located at the hottest core position. The IFEs are located at (D6NW) and (C4NW); the hottest fuel element is calculated to be (D4NE), Figure 27 which is adjacent to the transient rod. The guide tube for the transient rod has an OD of 1.485 in. which is slightly larger than the fuel element OD of 1.411 in. and as a result reduces the flow area of the maximum powered fuel rod by about 3.5%. In calculations done for the Texas A&M 1MW HEU to LEU conversion this reduction in flow area resulted in about a 3.3% decrease in the MDNBR; similar results are expected for WSU. 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 29 give the peak fuel temperature $\hat{T}$ in the hottest fuel element, the computed temperature $\hat{T}_{0.3}$ in the IFEs that can be compared with future measured temperatures, and the average core temperature $\overline{T}_{core}$.

The manufactured radial gap between the fuel and cladding is limited to a maximum of 2 mils and the average radial gap is limited to less than 1.75 mils. The TAC2D analysis for the maximum and IFE temperatures is based on the maximum cold gap of 2 mils and the average temperature is based on the maximum average gap of 1.75 mils. As the TRIGA core operates, offset swelling of the fuel tends to close the gap as was observed in the conversion of the Texas A&M TRIGA reactor. The effect of a closing gap on fuel temperatures is presented in Figure 36.

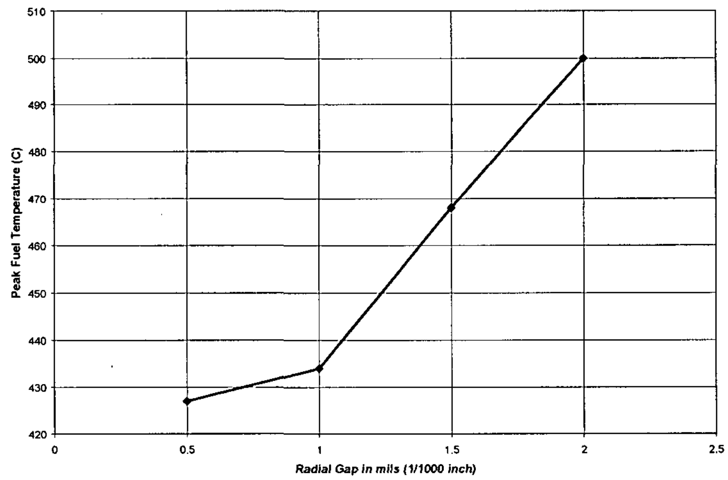

Figure 36 Peak Fuel Temperature as a function of Manufactured Radial Gap

The average power per fuel element in the WSU Mixed LEU Core 35A operating at 1000 kW is 8.4 kW/element. The highest powered fuel element is located immediately adjacent to the transient rod in location D4NE and produces a power of 20.8 kW/element. An ideal location for one of the Instrumental Fuel Elements (IFE) would be this location. The current configuration for the WSU Mixed LEU Core 35A has retained the IFEs in the same location as for WSU Mixed HEU Core 34A. These are core location D6NW with a power of 10.7 kW/element, and core location C4NW with a power of 12.1 kW/element.

4.8.4 Steady-State Analysis Results Summary – WSU Mixed LEU Core 35A

Table 27 lists the pertinent heat transfer and hydraulic parameters for the 1.0 MW WSU Mixed LEU Core 35A. 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 RELAP5 code analysis has been run for the critical heat flux for the WSU Mixed LEU Core 35A operating at 1.0 MW at BOL conditions. The data was obtained by using an inlet temperature of 30°C and then systematically increasing the reactor power until RELAP5 indicated a DNB ratio equal to one based on the Bernath Correlation. The maximum power per fuel element for which the DNB ratio is 1, is 52 kW/element. For a core with a rod peaking factor of 2.474, this maximum fuel element power corresponds to a maximum reactor power of 2.45 MW. Hence, the DNB ratio for the WSU Core 35A at the stated conditions is 2.45. The minimum DNB ratio is 1.88 at a power level of 1.3MW.

4.9 Thermal Neutron Flux Values, WSU Mixed LEU Core 35A

4.9.1 Thermal Neutron Flux Values in LEU Core

The DIF3D code provides neutron flux values. Figure 37 presents a 3-D representation of the thermal neutron distribution throughout the core and into the surrounding water and graphite.

Figure 38 shows the flux plot through the transient rod in a direction perpendicular to the face of the thermal column/BNCT Filter box. In the region between 5cm and 35 cm the flux through the partially burned LEU 8.5/20 fuel with the peak in the water of the control blades is seen. In the region between 35 cm and 135 cm the flux through the fresh 30/20 LEU fuel (flux depressed) is seen with the water peak in the transient rod. Finally in the region between 135 cm and 155 cm the flux through the opposite region of the partially-burned 8.5/20 LEU fuel with the peak in the area of the control blades is seen.

Figure 39 presents a graphical representation of the neutron flux across the core through the transient rod in a direction parallel to the face of the thermal column/BNCT Filter 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. In this orientation the flux depression in the fresh 30/20 LEU fuel is clearly evident in the central region of the plot with the peaking in the water hole for the transient rod. The depression in the fresh 30/20 LEU fuel is caused by the erbium and large uranium loading as it was evident in the Texas A&M post conversion flux measurements.

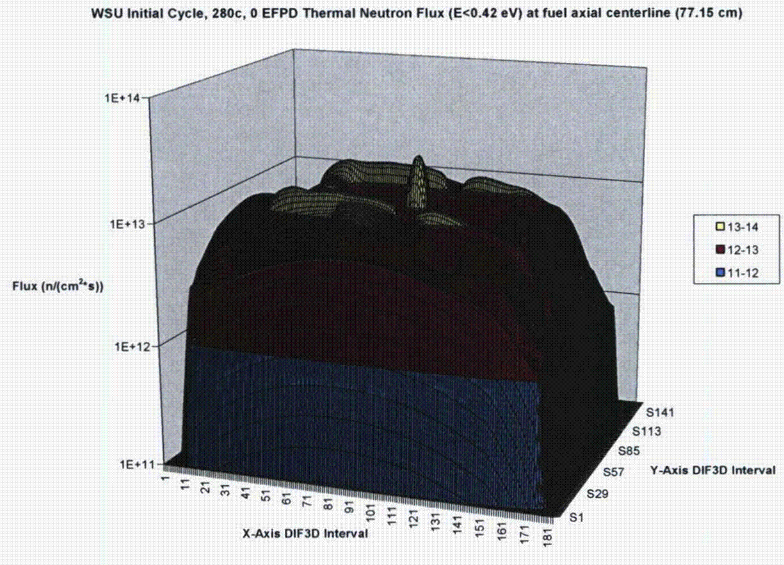

Figure 37 Thermal Neutron Flux (E<0.42 eV) at fuel axial centerline, BOL, WSU Mixed LEU Core 35A

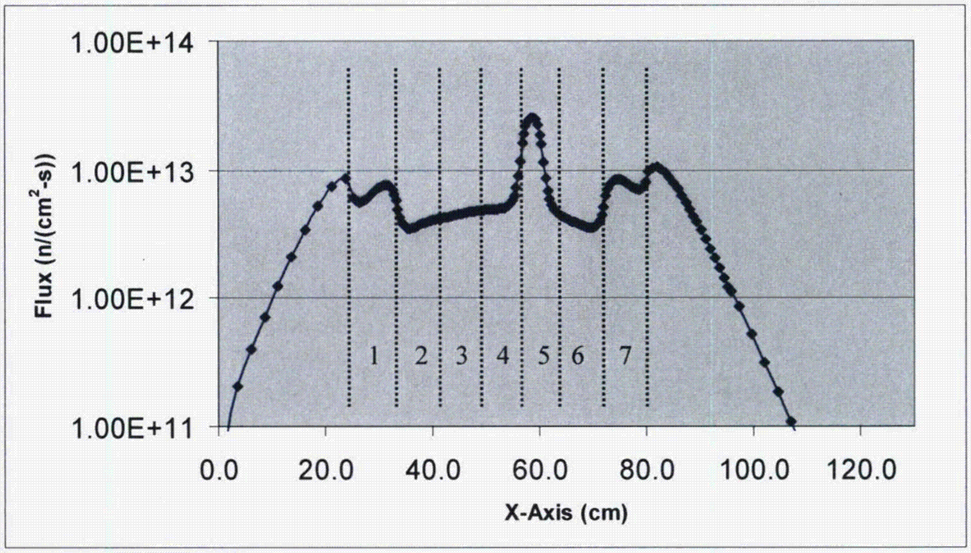

Figure 38 Thermal Neutron Flux (E<0.42 eV) Across the Core, West to East, through the Transient Rod at fuel centerline, BOL, WSU Mixed LEU Core 35A

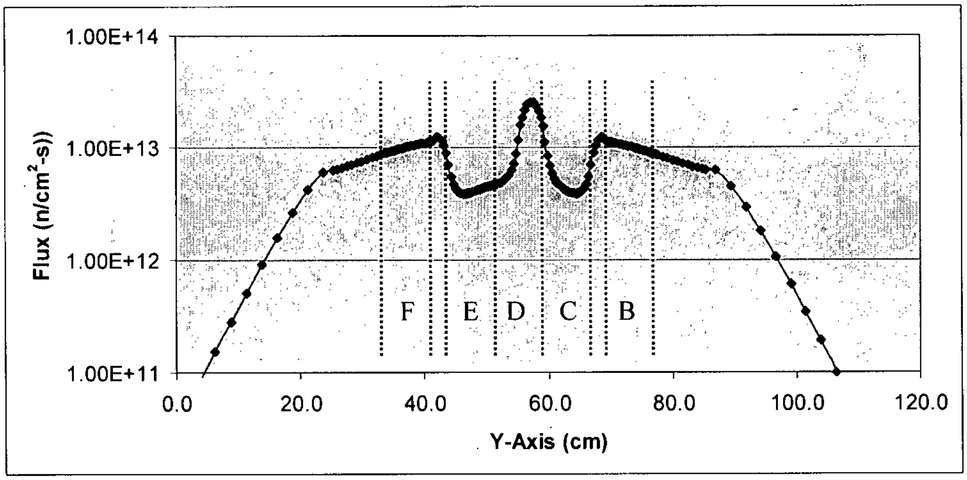

Figure 39 Thermal Neutron Flux (E<0.42 eV) Across the Core, South to North, through the Transient Rod at fuel axial centerline, BOL, WSU Mixed LEU Core 35A

5.0 Reactor Coolant System

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

6.0 Engineering Safety Features

The HEU to LEU conversion does not require any changes to the engineered safety features.

7.0 Instrumentation and Control

The HEU to LEU conversion does not require any changes to the instrumentation and control system.

8.0 Electrical Power System

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

9.0 Auxiliary System

Existing procedure will be used for fuel storage. Criticality aspects of the existing storage arrangements have been verified and are included in Appendix B of this document.

10.0 Experimental Facility and Utilization

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

11.0 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 UFTR facility.

12.0 Conduct of Operation

To be supplied by licensee.

12.1 Organization and Staff Qualification

The HEU to LEU conversion does not require any changes to the organization and staff qualification of the WSU Reactor Personnel

12.2 Procedures

The HEU to LEU conversion does not require any fundamental changes to the WSU reactor 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 WSU reactor 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 WSU reactor 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. However, WSU anticipates that there will be a need to make changes, for a period of time, when the HEU is removed from the core, stored, and until it is suitable for shipping. It is anticipated that changes to the plan will need to be made to comply with 10CFR73.60 and 10CFR73.67(d) during the time the fuel is no longer self-protected until the time that it is ultimately shipped. These proposed changes will be submitted by WSU under separate cover and withheld from public disclosure.

12.6 Reactor Reload Consideration
The WSU Mixed LEU Core 35A will have capability to operate at 1.0 MW on demand for about 13 years following a weekly schedule of $\leq$35 MWHr. During the life of Core 35A it is conceivable that at some fuel may be introduced in the partially burned core. The following are most likely scenarios for the introduction of the new fuel:

1. Some of the partially burned 8.5/20 fuel becomes fully burned and must be replaced
2. A fresh IFE is introduced because the thermocouples in one of the in-core IFEs have failed
3. A fresh 30/20 LEU SFEs is introduced

Case 1 above is straightforward because the power factors in the region of the core where the partially burned 8.5/20 fuel is located are very low. Therefore introducing another partially burned 8.5/20 SFE from the irradiated fuel already on hand at WSU into the same core region will not lead to peaking factors that exceed values considered in this analysis.

Cases 2 and 3 on the other hand could lead to possible higher peaking factor especially if the fresh 30/20 IFE/SFE is located near the transient rod water hole. This will be most likely during pulsing operation. Therefore, depending on the configuration of the core at the time that the fresh fuel is added, core locations 6C, 6E, or 2D should be considered for introducing any new fuel. These locations have very low peaking factors both at the beginning and the end of life.

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

13.0 Safety and Accident Analysis

13.1 Safety Analysis

13.1.1 General Discussion and Summary

The safety of TRIGA fuel is due entirely to its design features. The safety features of a standard TRIGA fueled core are well known. Each of the 30/20 LEU fuel elements is designed to replace a FLIP HEU (8.5 wt % U, 70% enriched) element, with regards to reactivity; that is, 100 fresh 30/20 LEU fuel elements in a compact configuration is intended to have about the same core excess reactivity as 100 of the more heavily loaded FLIP 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, (Reference 2) 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".

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. For the WSU analysis the transient rod is water followed and was properly modeled both for the steady state and pulsing operations.

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 is 500°C. In normal pulsing, 830°C has been chosen as the limit for the peak fuel temperature and administrative controls are in place to maintain the limit on prompt reactivity insertions. 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. Both of these temperatures have large margins of safety to 1150°C.

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

The fuel temperature scram (1) (500°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 WSU LEU (30/20) Fuel

13.3.1 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 500 °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 (D4NE), Figure 4.2. In the present analysis, the instrumented fuel elements (IFE) are located in core location D6NW and C4NW. (Other locations can be chosen for the IFE.) The LSSS temperature setting is smaller than the Safety Limit temperature to account for several factors, including:

i. Accuracy of temperature calibration ii. Precision of electronic readout/scram circuitry iii. Account taken of location of sensing tip of thermocouple 0.3 inch from axial center line of IFE iv. Difference in peak temperature in IFE compared to that in the hottest fuel element	Safety Margin

The basis for selecting 500°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 peak core temperature of 950°C in LEU fuel is the criteria established to provide a minimum safety margin of 200°C for all modes of operation. 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, as explained above, the IFEs have been maintained in core locations D6NW and C4NW. 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 17.1 % (500/427 = 1.1709) for the IFE in core location D6NW and 13.6 % (500/440 = 1.1363) for the IFE in core location C4NW, (See Table 29). Thus, for the steady state mode of operation, if the temperature in the thermocouple element were to reach the trip setting of 500°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 peak fuel temperature $\hat{T}$ in the hottest fuel in the WSU LEU (30/20) core would be at most 520°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.

13.3.2 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 1000 MWD, a reactivity insertion of $2.04 will produce a peak fuel temperature; namely, ~830°C.

In view of the small change in peak fuel temperatures with core burnup, a maximum allowable pulsed reactivity insertion of $2.00 is a reasonable choice.

13.5 Accident Analysis

13.5.1 Analysis Changes to DBA/MHA Event

The Maximum Hypothetical Accident (MHA) for a TRIGA reactor is defined as the loss of the integrity of the fuel cladding of one fuel rod in air. This MHA definition is consistent with the Design Basis Accident (DBA) endorsed by the NRC. NUREG/CR-2387, (Reference 13), 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 evaluated dose resulting from the fission product release from a single FLIP fuel element was found acceptable by the NRC. The loss of pool water is typically treated separately as a Loss of Coolant Accident (LOCA). The MHA for the WSU TRIGA reactor was originally analyzed in the Safety Analysis for converting the WSU TRIGA reactor to FLIP fuel of May 1974. (References 14 and 15) This analysis was revised in the 2002 SAR that was submitted to support WSUs application to renew its operating license. The revised analysis used the same basic data as used in the previous analysis but the radiological effects are calculated using more recent analysis methods and guidelines published by the Federal Government. (Reference 16)

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. The major parameter important to radiological impact is the fission product inventory due to burnup and power density. The WSU mixed HEU Core 34A has a life of about 1000 MWD and is essentially the same as the WSU mixed LEU Core 35A as discussed in Section 4.5.6. The energy burn up capability based on >50% of the U-235 has been evaluated as 77 MWD/fuel element for FLIP fuel. (Reference 13) For WSU LEU (30/20) fuel, the energy burnup capability based on >50% of the U-235, Ref 13 gives 57 MWD/fuel element.

The hottest 30/20 fuel element in the LEU core at the end of life will contain less activity than a FLIP element of similar burn up. Prior WSU SARs based the fission product inventory of the hottest fuel element using a power density of 30 kW per fuel rod and an infinite irradiation time. The 30 kW per fuel rod is much greater than the maximum predicted power density of 20.8 kW for WSU Mixed LEU Core 35A. The WSU SAR inventory was derived from the basic data of Pecking and King, (Reference17), along with the documented fact that only gaseous fission products escape when the cladding of a TRIGA fuel rod ruptures. Table 30 compares the WSU SAR inventory with an ORIGEN calculation for 30/20 fuel using similar assumptions. In performing the ORIGEN calculations, maximum inventories occurred after 200 MWD of burn up. This burn up maximized buildup while minimizing the effect of fissile material depletion. Thus, the radiological impact from an MHA event on the members of the public and the facility workers will be essentially the same as that evaluated for FLIP fuel.

The WSU 2002 SAR submittal concluded that the consequences of the MHA show that the only significant worst case radiation exposure is the thyroid dose to a person in the pool room. The conditions necessary to produce this exposure are the failure of the cladding of one fuel rod along with a complete loss of pool water. The maximum possible radiation exposure to an individual outside the facility under the postulated conditions is minimal. The exposures are significantly below the generally acceptable accident results for non-power reactors of not more than 5 rem whole body and 30 rem thyroid for occupational exposure and not more than 0.5 rem whole body and 3 rem thyroid for members of the general public. In addition, the calculated accident exposures are well below the maximum values established in 20.1201 for occupational exposure and 20.1301 for public exposure. Thus, no realistic hazard to the staff at the reactor as well as the general public would result from the MHA. These conclusions are equally applicable to the WSU Mixed LEU Core 35A due to the similarity in fission product inventories.

Table 31 Gaseous Fission Products in a Single TRIGA Fuel Rod

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ISOTOPE	Saturated Inventory (Ci)	
	SAR 2002	GA 2006
Br-82	40	1
83	137	135
84	253	263
85	330	309
87	780	504
Total Br	1540	1211
I-130m	260	4
131	734	734
132	1115	1104
133	1672	1704
134	2027	1961
135	1546	1596
136	785	818
Total I	8139	7919
Kr-83m	137	135
85m	330	309
85	67	20
87	634	631
88	912	894
89	1115	1157
Total Kr	3195	3145
Xe-131m	7	7
133m	40	49
133	1672	1661
135m	457	296
135	1621	1136
137	1545	1554
138	1166	1607
Total Xe	6508	6309

13.5.2 Analysis of Changes to LOCA Event

A LOCA analysis was performed as part of the 1979 Safety Analysis for Conversion to FLIP Fuel Conversion. (Reference 1) The analysis concluded that TRIGA fuel rods operating with power densities up to 22.3 kW/rod for standard fuel and 23.5 kW/rod for FLIP fuel would not fail in the event of a loss of coolant accident.

The conditions for this analysis included:

1. Operation of the reactor for essentially and infinite length of time
2. Sudden and complete loss of coolant (pool water)
3. The loss of water will shut down the reactor, however, the decay heat from the fission products will continue to produce heat in the fuel elements
4. The fuel clad temperature must be maintained below the point where a cladding failure would occur.

The analysis showed that the maximum temperature that TRIGA fuel can tolerate without damage to the cladding and subsequent release of fission products is

1. 900°C for ***standard fuel with H/Zr =1.7***
2. 940°C for ***FLIP fuel with H/Zr =1.6***

The current design for the TRIGA 30/20 LEU fuel has an average H/Zr = 1.6 with a range of 1.57 to 1.65. Therefore, its temperature capability would be closer to the 940°C for the FLIP fuel that was analyzed in the 1979 SAR. Finally the maximum power density for the 30/20 LEU fuel calculated as part of this HEU/LEU conversion is 22.9 kW/rod which is within the bounds of the above analysis. Therefore, there are no changes to the LOCA analyses that are required for the conversion.

13.5.3 Accidental Pulsing from Full Power

BOL, Beginning of Core Life LEU Core

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 also limits the maximum insertion of a transient rod to about that value which produces a maximum allowed fuel temperature no higher than the Safety Limit (1150°C). This reactivity is about $2.75 for the (30/20) core. Since the full worth of the transient rod is calculated to be $3.19, which could produce temperatures greater than the Safety Limit (1150°C), accident analysis was performed assuming the full $3.19 value of the transient rod.

Pulsing from full reactor power (1.0 MW) would be clearly an accident. On the one hand, an administrative control prevents 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 clearly violates the two Safety Procedures.

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

1. The reactor is in the steady-state mode, operating at 1.0 MW. The two redundant power scrams are set at 125% full power. The fuel temperature scram is set at 500 °C.
2. The transient rod is in its full down position, with a negative reactivity of about $3.19.
3. The operator turns the Mode Switch to PULSE.
4. The interlock that prevents energizing the pulse circuit when the reactor power is greater than 1 kilowatt fails.
5. The operator fires the transient rod, which reaches the maximum UP position in less than 0.1 second (the time for a full stroke travel).
6. When the reactor power rises from 100% power to the Power Scram set point of 125%, the reactor scrams with three control blades falling to their maximum inserted position in less than 2 seconds (the Technical Specification scram time for a full stroke).

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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 1614 MW.
2. The maximum fuel temperature (997 °C) is reached immediately after the peak power.
3. The energy release is about 25.2 MW-sec in about 1.0 second when the maximum measured fuel temperature ($\hat{T}_{0.3}$ = 807 °C) is reached.
4. At peak fuel temperatures 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 at the full stroke value of $3.19. 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 using Nordheim - Fuch model 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 306°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 492°C. From three-dimensional diffusion theory calculations, the peak-to-average power ratio was determined for steady state operation to be 3.8. Although the highest temperatures occur at the center of the hottest fuel element during 1 MW steady state operation (500°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 3.8. 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 997°C was obtained based on the average core temperatures computed by BLOOST.

An alternative method of producing the accidental pulse from full power is to remove an installed two-dollar experiment. Although the Technical Specification requires such an experiment to be securely locked in position in the core, somehow, the experiment is loosened and yanked up out of the core. Because of its mass, 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 $3.19 transient rod. Thus, neither accident endangers the reactor.

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

1. Failure of the 1 kW interlock that prevents air from being applied to the transient rod piston for reactor power above 1 kW.

2. Failure of reactor operator to follow written procedures.

Note: Both must occur for the accident to occur.

The accidental removal of a two-dollar experiment requires the following:

1. Deliberate violation of written procedures as operator unfastens the two-dollar experiment and proceeds to yank it out of the core.

Note: Only one gross failure must occur for this accident to occur – no interlocks need to fail.

EOL Pulsing from Full Power at 1000 MWD Burnup

The BLOOST code was run with input appropriate for the LEU core at 1000 MWD. Pulsing results were obtained for a series of reactivity insertions under the same conditions as set forth in the above section. Table 31 lists these results along with those for the beginning of core life.

Table 32 Pulsing from 1.0 MW, Beginning of Life (BOL) and End of Life (EOL)

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Parameter	**BOL**			**EOL (1000 MWD)**		
Δk/kβ ($)	2.00	2.75	3.19	2.00	2.75	3.19
$\hat{T}$ (°C)	670	925	997	650	930	1003
$\overline{T}_{core}$ (°C)	384	470	492	308	417	443
$\hat{T}_{0.3}$ (°C) –D6NW	638	762	798	798	637	675
$\hat{T}_{0.3}$ (°C) –C4NW	638	769	807	538	690	731
MW-sec	12.4	22.1	25.2	12.9	24.6	28.2
$\hat{P}$ (MW)	388	1239	1614	371	1350	1767

In Table 31, the effects are evident for shifts in the power distribution with burnup for the core. The measured temperature ($\hat{T}_{0.3}$) at $2.00 pulse decreases from 638°C to 498°C with burnup. The peak temperature in the hottest fuel element rises slightly, from 997°C to 1003°C; but even 1003°C is 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 1000 MWD burnup.

14.0 Revised Technical Specifications

The following is a revision, as needed, for Section 14 from the WSU License No. R-76 dated May 1979 including revisions through Amendment No. 8.

Only those Specifications are included below that required a change related to the conversion to LEU fuel from the mixed cores with FLIP and standard fuel. Therefore, the following listing is incomplete for the WSU facility, for which most of the Technical Specification requirements remain unchanged.

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 (Reactor Power <1kW and a stable period) 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 Washington State University 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 sub-critical 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 sub-critical 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

<u>Applicability</u>

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 1150°C (2100°F) 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 1150°C (2100°F) 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:

   1) The limiting safety system setting shall be 500°C (930°F) as measured in an instrumented fuel element. The instrumented element shall be located in the region of the core containing the fresh 30/20 LEU SFEs.

b) For pulsing operation:

   The limiting safety system setting shall be 500°C (930°F) as measured in an instrumented fuel element. The instrumented element shall be located in the region of the core containing the fresh 30/20 LEU SFEs. 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 1150°C (2100°F). 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 also 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 elements within the core. For example, if one of the thermocouple elements were located in the hottest position in the core, location D4NE, 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 WSU the IFEs are located in core locations D6NW and C4NW. Calculations indicate that, for these cases, the true temperature at the hottest location in the core will differ from the measured temperatures by no more than 16%. When applying this 16% worst case measurement scenario and considering the previously mentioned sources of error between the true and measured values, a limit temperature of about 850°C (1562°F) is obtained. Finally a margin of 350°C (660°F) was imposed in setting the LSSS temperature at 500°C (930°F).

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.

14.3 Limiting Conditions for Operation

14.3.1 Reactor Core Parameters

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.

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.

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 830°C (1526°F). 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 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 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.

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.

Core Configuration Limitation

Applicability

This specification applies to a full core assembly composed of a mixture of 8.5/20 LEU and 30/20 LEU fuel elements.

Objective

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

Specifications

a) The TRIGA core assembly shall be a mixture of LEU fuel elements.
   1) A minimum of 51 – 30/20 LEU SFEs located in the central region of the core
   2) A region of 8.5/20 SFEs surrounding the central region of higher loaded SFEs
b) The instrumented element, if present and serving as the Limiting Safety System shall be located in the 30/20 LEU region.

c) 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.

Bases

The limitation on the allowable core configuration is based on the results of the HEU/LEU conversion analysis and peaking rod factors. The limitation on power peaking ensures that the fuel temperature safety limit will not be exceeded in a mixed core.

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 5.6% $\Delta k/k$ ($ 8.00).

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

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.

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 (500°C)(930°F)	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 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

Bases

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 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.

Scram Time

Applicability

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

Objective

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

Specification

The scram time from the instant that a safety system setting is exceeded to the instant that the slowest scrammable control rod reaches its fully inserted position shall not exceed 2 seconds. For purposes of this section, the above specification shall be considered to be satisfied when the sum of the response time of the slowest responding safety channel, plus the fall time of the slowest scammable control rod, is less than or equal to 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.

14.5 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 (30/20) LEU fuel

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

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

3) Hydrogen-to-zirconium ratio (in the ZrH_x): nominal 1.6 H atoms to 1.0 Zr atoms.

4) Natural erbium content (homogeneously distributed): nominal 0.90 wt%.

5) Cladding: 304 stainless steel, nominal 0.5 mm thick.

b) TRIGA (8.5/20) LEU Fuel

1) Uranium content: 8.5wt % uranium enriched to a nominal 19.75% Uranium-235.

2) Zirconium hydride atom ratio; nominal 1.6 hydrogen to zirconium, ZrH_x.

3) Cladding: 304 stainless steel, nominal 0.5 mm 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 and 3-rod clusters positioned in the reactor grid plate.

b) The TRIGA core assembly may be composed of a combination of 8.5/20 LEU and 30/20 fuel elements provided that the 30/20 LEU fuel elements are located in a contiguous block in the central region of the core.

c) The reactor fueled with a mixture of LEU fuel shall not be operated with a core lattice position vacant in the 30/20 fuel region. Water holes in the 30/20 region shall be limited to single-rod holes. Vacant lattice positions in the core fuel region shall be occupied with fixtures that will prevent the installation of a fuel bundle.

d) 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. Mixed cores of Standard fuel have been tested by General Atomics. Calculations of mixed cores in the WSU reactor, have shown that such cores may be safely operated.

In mixed cores, it is necessary to arrange the 30/20 LEU Fuel elements in a contiguous, central region of the core to control flux peaking and power generation peak values in individual elements.

Vacant core lattice positions in the Standard fuel region will contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core. Vacant core positions are not permitted in the 30/20 LEU fuel region.

Appendix A

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. TRIGA 30/20 fuel is defined as 30 wt.% 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 the instrumented fuel elements installed in the core together with the fuel in the 3-rod clusters that contain the transient rod, but the transient rod withdrawn from the core.

Acceptance Criteria: The 1/M criticality is expected with a fuel loading as follows:

Partially Burned 8.5/20 SFEs between 24 – 32.
Fresh 30/20 SFEs between 43 – 47.

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 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.

Acceptance Criteria: The Rod Drop reactivity worths for the scrammable control rods are expected to lie between \$ 0.40 and \$3.20.

A.1.3 Initial Control Rod Calibration Tests

The WSU 1 MW TRIGA reactor with Mixed LEU core contains three (3) shim control blades, one (1) Servo control blade and one (1) water-followed transient 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.

<u>Acceptance Criteria:</u> The calibration curve results for control rod worth are expected to vary between a low value of ~\$0.50 and a high value of \$3.50, 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 119 fuel elements, 51 – Fresh 30/20 LEU SFEs and 68 Burned 8.5/20 LEU SFEs 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 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 calorimetric power tests.

Acceptance Criteria:

- With 119 fuel elements the excess reactivity is expected to be about \$ 6.37, 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 WSU 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 $ 2.25 to $ 2.75, 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 $\sim 5 \times 10^{-8}$ 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 WSU 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.

<u>Acceptance Criteria</u>: 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 B

B.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.

B.2.1 WSU Fuel Storage Facilities

Figure 40 shows the MCNP model for the dry storage of fresh, unirradiated fuel. There are five dry tubes (4 inches diameter) made of aluminum. Each tube can store two fuel elements, and they are spaced 3 inches apart.

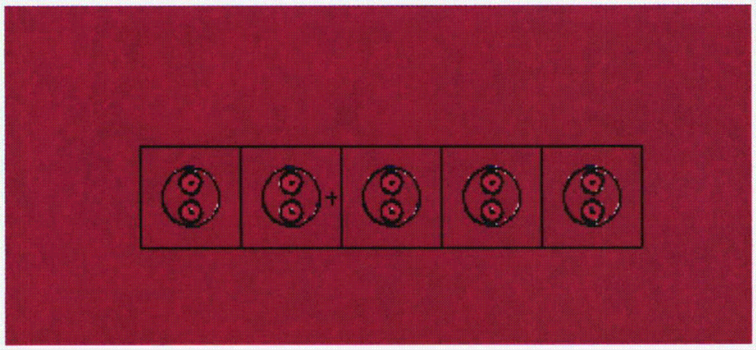

Figure 40 MCNP X-Y Plot for the Dry Fuel Storage Tubes

Figure 41 illustrates the in-tank wet racks storage facility. This consists of four large storage racks free standing on the pool floor, and the racks are spaced 8 inches apart. Each rack has nine pits, and each pit can store 4-rods cluster.

Currently, the in-tank storage has free space to store 32, 4-rod fuel clusters. The wet racks have 0.75 inch spacing between adjacent clusters, and 8.75 inches spacing along the width.

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

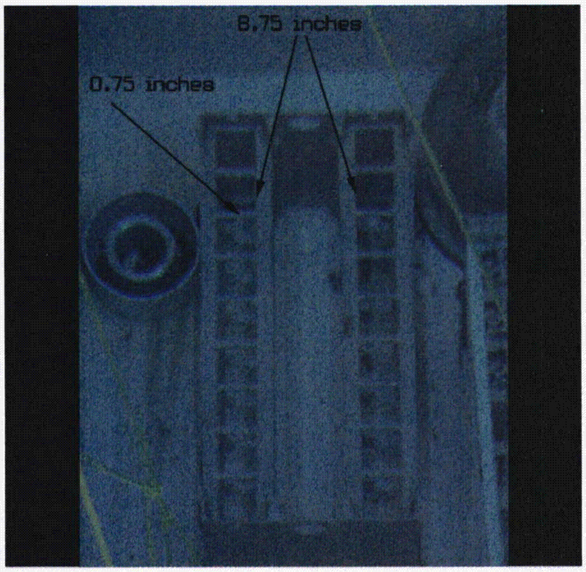

Figure 41 In-Tank Fuel Storage Arrangement

For the dry, fresh fuel storage, the MCNP model included the aluminum tubes with two fuel elements inside. The MCNP program was run twice, once with air in the storage room, and once with the storage room flooded.

Storage of LEU (30/20) in air	Storage of LEU (30/20) flooded
$k_{eff} = 0.03279 \pm 0.00006$	$k_{eff} = 0.37572 \pm 0.00028$

The MCNP model for the in-tank storage was similarly conservative. For this 4x9 stand alone floor storage facility. In the MCNP model 4x9 racks were modeled, and each pit was filled with the four fresh FLIP or four fresh 30/20 LEU fuels. 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
4x9, water reflected	$k_{eff} = 0.63456 \pm 0.00035$	$k_{eff} = 0.64886 \pm 0.00035$

Conclusion: The storage of fresh LEU (30/20) fuel in the fresh fuel locker is entirely safe, even if the locker were to flood.
The storage of spent FLIP fuel and/or fresh LEU (30/20) fuel in the wet storage facilities is entirely safe.

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