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April 16, 2009

Mr. Alexander Adams
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Reference: Oregon State University TRIGA Reactor (OSTR)
Docket No. 50-243, License No. R-106

Subject: Oregon State University TRIGA® Reactor Post Conversion Reactor Startup Report

Mr. Adams:

Enclosed with this letter you will find one copy of the Reactor Startup Report. The format of the report follows the guidelines described in *Outline of Reactor Startup Report* (attachment 2 of the Conversion Order issued September 4, 2008).

If you have any questions, please call me at the number above. I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 4/16/09.

Sincerely,

Steve Reese
Director

Enclosure

cc: Document Control, NRC
John Cassady, OSU
Rich Holdren, OSU
Todd Palmer, OSU (w/o enclosure)
Todd Keller, OSU (w/o enclosure)
Craig Bassett, NRC

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Reactor Startup Report for the Oregon State TRIGA® Reactor Using Low Enrichment Uranium Fuel

Submitted by S. Todd Keller, Reactor Administrator

April 2009

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I. Introduction

On August 4, 2008, the Oregon State University TRIGA® Reactor (OSTR) was operated for the final time with Highly Enriched Uranium (HEU) fuel. The reactor was then shut down and all HEU fuel was removed from the reactor tank. The core was subsequently reloaded with Low Enriched Uranium (LEU) fuel. Initial criticality with LEU fuel was achieved at 1545 on October 7, 2008.

This report documents all significant activities that were performed to convert the OSTR from HEU to LEU fuel, and presents the results of post-conversion testing and evaluation. Section II contains a timeline and description of significant activities necessary for conversion. Section III describes restart activities and startup testing. In section IV, the results of the startup tests are presented. The results of startup testing are discussed in section V, as well as lessons learned during the conversion process. Although the conversion was a complex long term evolution, the process went smoothly, primarily due to professionalism of the staff and organization of the project managers. The few problems that did arise were dealt with efficiently. A discussion of problems and deviations encountered during conversion can be found in section V.

The OSTR is a TRIGA® Mark II reactor, manufactured by General Atomics. The reactor functions as a highly flexible research tool and has several in-core and ex-core irradiation facilities, as well as four beam lines. The reactor is licensed to operate up to 1100 KW in steady state mode and up to several thousand MW in pulse mode. The newly converted core is loaded with fuel containing low enriched uranium in a U/Zr/H/Er fuel matrix designed specifically for the OSTR. A more detailed description of the OSTR can be found in Appendix A, Facility Description.

Formal testing per the restart procedure was completed October 24, 2008. Additional characterization and flux mapping was performed during the month of November. The OSTR was released for general customer use on December 1, 2009.

II. Major Conversion Activities

Successful conversion of the OSTR was accomplished via coordination of major work efforts at OSU, General Atomics / CERCA, the Department of Energy (Argonne National Laboratory and Idaho National Laboratory) and the Nuclear Regulatory Commission. OSU was initially given approval to possess, but not use LEU fuel, and the new fuel was fabricated and delivered to OSU. An approved Safety Analysis Report was required by the NRC prior to issuing the conversion order. The conversion order was issued on September 4, 2008, and the conversion milestone was met on September 29, 2008. A chronological list of all major conversion activities is given in Table 1.

Completion of all activities listed in Table 1 was required in order to meet the conversion milestone. Some of these activities required several thousand man-hours of work. Others required design and fabrication of new tools and structures, or development of new procedures. Detailed descriptions of these activities can be found in the OSTR chronological logs and the Reactor Supervisors logbook. The complete LEU fuel receipt procedure can be found in Appendix B – Fuel Receipt Procedure.

Table 1, Major Conversion Activities

September 30, 2007	Conversion Safety Analysis Report (CSAR) neutronic analysis complete
October 31, 2007	CSAR steady state thermal hydraulic analysis complete
November 1, 2007	CSAR preliminary transient (pulse) thermal hydraulic analysis complete
November 6, 2007	CSAR submitted to NRC
April 4, 2008	Possession limit increase order received
April 24, 2008	Develop LEU fuel receipt procedure
May 7, 2008	Fabricate new LEU fuel storage facility
May 19, 2008	Receive LEU shipment (one of four)

Table 1 (continued), Major Conversion Activities

May 27, 2008	Receive LEU shipment (two of four)
June 20, 2008	Final set of answers to Request for Additional Information (RAI) questions submitted to NRC
June 23, 2008	Receive LEU shipment (three of four)
June 27, 2008	Receive LEU shipment (four of four)
June 30, 2008	Fabricate spent HEU fuel storage rack
July 2, 2008	Move LEU fuel from casks to storage facility
August 26, 2008	Develop restart procedure
August 28, 2008	Develop HEU offload process and fabricate required tools
September 4, 2008	NRC issues conversion order (21 day comment period begins)
September 9, 2008	All HEU removed from OSTR reactor tank
September 9, 2008	All old aluminum clad graphite reflectors removed from OSTR core
September 9, 2008	Obtain new fuel inspection / measurement tool
September 10, 2008	NRC issues HEU License renewal
September 23, 2008	Certified bridge crane for refueling operations
September 25, 2008	Obtain new control rod connector assemblies and extension rods
September 29, 2008	Receive new stainless steel clad graphite reflectors
September 29, 2008	Conversion milestone accomplished at 1:18 PM

III. Startup Testing

The conversion milestone was met on September 29 by placing one new LEU fuel element in the OSTR gridplate. At that time, the core was completely empty, except for the aluminum central thimble. After the milestone was met, the LEU fuel element was removed. Reactor startup then proceeded in a systematic manner as directed by the restart procedure. The complete LEU core restart sequence can be found in Appendix C – Restart Procedure.

The major steps of the restart sequence are:

- Verify that all administrative requirements for reactor operation have been implemented.
- Install a neutron source and graphite reflector elements.
- Install control rods.
- Install B-ring and C-ring fuel elements.
- Use a standard 1/M plot to monitor reactor state and predict when criticality will be achieved. Continue to add fuel batch-wise until criticality is achieved.
- Calibrate the control rods. Continue to add fuel, calibrating the control rods after each batch addition. Establish an operational core configuration.
- Evaluate the reactivity worth of representative fuel elements.
- Perform a full power calorimetric.
- Perform an extended full power run to 'season' fuel.
- Perform Pulse mode and Square Wave mode testing.
- Perform ICIT core and CLICIT core testing.
- Perform flux mapping.

The restart procedure was written to take the reactor from a fully defueled state and restore it to normal operation with LEU fuel. Guidance was also provided to ensure that all testing needed to verify compliance with acceptance criteria was performed. The acceptance criteria are specified in the CSAR, and are summarized in Table 2 for convenience. All of the acceptance criteria were met.

Table 2, LEU Core Startup Acceptance Criteria

Criterion	Specification
Initial Criticality	The minimum critical mass is expected with a fuel loading between 60 and 70 fuel elements.
Control Rod Worth (Drop Method)	The worth of the control rods is expected to be between \$1.50 and \$4.00, depending on the type of rod and its location in the core.
Control Rod Worth (Period Method)	For the operational core, the worth of the control rods is expected to be between \$1.50 and \$4.00, depending on the type of rod and its location in the core.
Shutdown Margin & Excess Reactivity	Excess reactivity and shutdown margin shall be calculated for an operational core in all allowed operating modes. Calculations shall be based on measurements taken on a cold, clean core. Excess reactivity and shutdown margin shall be within the allowed values of the Technical Specifications.
Power and Temperature Coefficient	The power and temperature coefficients shall be verified to be negative over all operating ranges. The power defect (the reactivity change between power sufficiently low that no fuel heating is occurring and full power) shall be between \$1.50 and \$3.00.
Reactor Power Calorimetric Calibration	After completion of the power calibration, including instrument adjustment, the linear, safety and percent power channels shall indicate between 99.0 and 101.0 percent at full reactor power of 1.0 MW.
SCRAM Tests	The Safety Channel SCRAM shall be activated at an indicated power no greater than 106 percent on the Safety channel. The Percent Power channel SCRAM shall be activated at an indicated power no greater than 106 percent on the Percent channel.
Pulse Mode Tests	The graph of peak temperature vs. prompt reactivity should show a linear dependence. The graph of integrated power vs. prompt reactivity should show a linear dependence. The graph of peak power vs. (prompt reactivity)^2 should show a linear dependence.

In addition to measurements and tests specified by the restart procedure, additional characterizations have been (or will be) performed on the new core. These include such things as alternate flux map verifications, self shielding evaluations, measurement of temperature profiles and additional reactivity measurements. Discussion of these additional evaluations is included in section V.

IV. Comparison of Predicted and Measured HEU and LEU Parameters

This section is divided into ten subsections, in accordance with guidance provided with the September 4, 2008 NRC conversion order. The eleventh subsection, *discussion of results*, is presented as a separate section, and combined with a discussion of lessons learned in Section V. Information pertinent to predicted and measured LEU core parameters is presented in each subsection. Predicted and measured HEU core parameters are included whenever such information is available and relevant.

A. Critical Mass

The same grid plates are used in the HEU and LEU cores, so control rod locations and allowed fuel and reflector positions are unchanged. Geometry of the fuel elements is essentially identical. Each fuel element has an active fuel length of 38.1 cm (15.0 in). Fuel element OD is nominally 3.75 cm (1.48 in). The fueled region has a 0.32 cm (0.125 in) radius zirconium pin at its center. The fueled region is reflected above and below by graphite.

HEU fuel contains 8.5 mass percent of 70% enriched uranium, therefore each kg of fuel meat contains (1000)(0.085)(0.70) = 59.5 grams of U-235. LEU fuel contains 30 mass percent of 19.75% enriched uranium, so each kg of LEU fuel meat contains (1000)(0.30)(0.1975) = 59.25 grams of U-235. Although HEU fuel contains slightly more U-235, it also contains more of the burnable poison erbium. The geometry of the HEU and LEU initial critical cores was also slightly different due to different placements of the fuel elements, and dissimilar arrangement of graphite reflectors. As a result, the HEU core achieved criticality with one less fuel element than the LEU core. Minimum critical data is shown in Table 3. The HEU initial critical configuration is shown in Figure 1, and the LEU initial critical configuration is shown in Figure 2.

Table 3, Minimum Critical Core

Fuel Type	Predicted #FE's	Measured #FE's
LEU	69	66
HEU	Not known	65

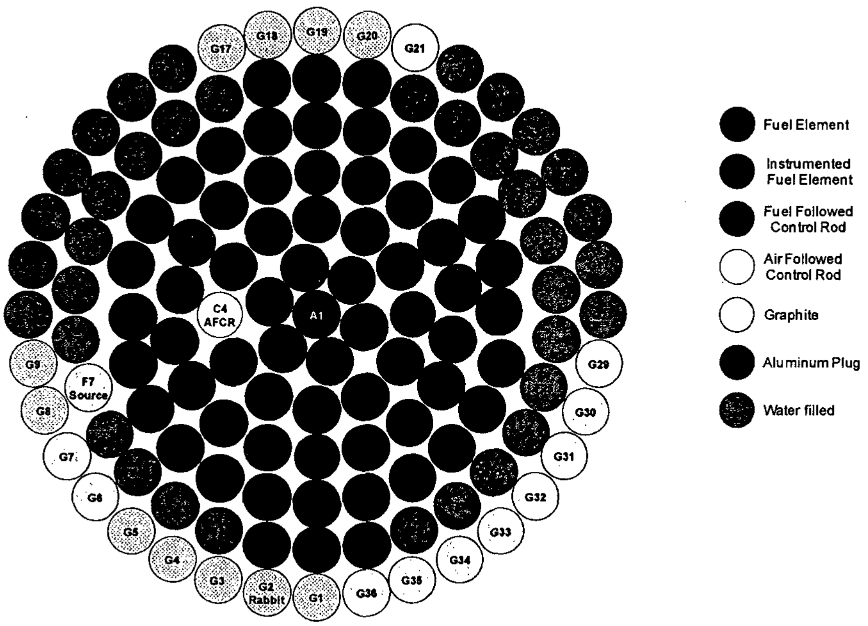

Figure 1, HEU Initial Critical Core Configuration

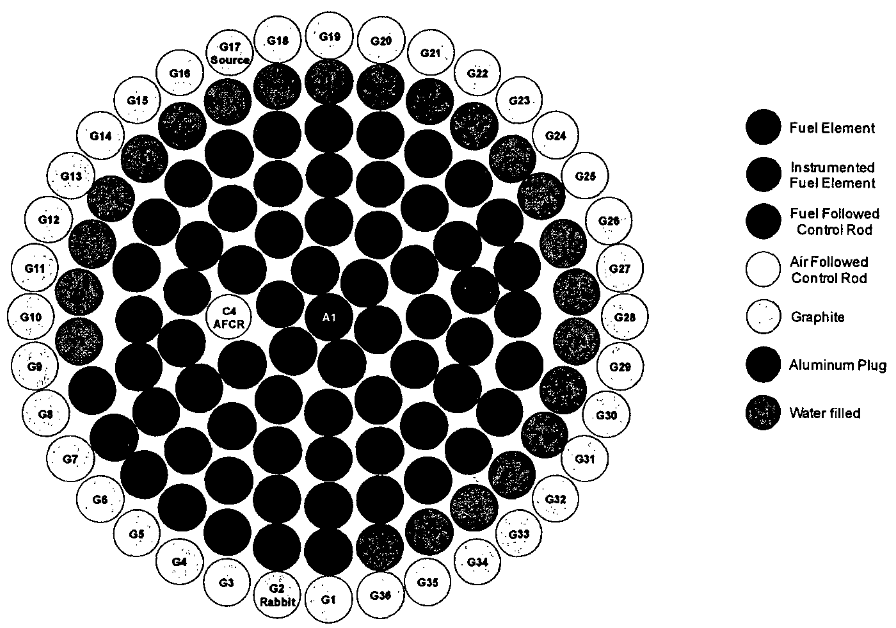

Figure 2, LEU Initial Critical Core Configuration

B. Excess (Operational) Reactivity

In the LEU core, core excess reactivity is limited by technical specifications to no more than \$7.55 in all modes of operation. The shutdown margin assumes, by definition, that 1) experiments are in their most reactive state, 2) the most reactive control rod is fully withdrawn and 3) the core is in its reference configuration. The shutdown margin limit is \$0.55 for the LEU core. The shutdown margin limit for the HEU core was \$0.57. Although there was no technical specification core excess reactivity limit for the HEU core, an implied limit did exist. The implied core excess limit was equal to the minimum (least subcritical) shutdown margin plus the worth of the three least reactive control rods. For the last HEU core, a typical value of maximum achievable core excess was

$$-\$0.57 + \$2.73 + \$2.56 + 2.78 = \$7.50$$

The value of core excess for the HEU initial critical core shown in Figure 1 was \$0.12. The value of core excess for the LEU initial critical core shown in Figure 2 was too small to accurately measure, but the value of core excess with one additional fuel element in position F-8 was \$0.48. These reactivity values were determined by measuring reactor period with all control rods fully withdrawn. Based on subsequent F-ring reactivity worth measurements of ~\$0.40 per element, the reactor was supercritical by ~\$0.08 in the LEU initial critical configuration.

The initial HEU operational core is shown in Figure 3. This core had a core excess of \$7.14 and a shutdown margin of \$0.88. The initial LEU operational core is shown in Figure 4. This core had a core excess of \$7.12 and a shutdown margin of \$1.24. All LEU core startup testing, except for measurements taken on ICIT or CLICIT cores, was performed with the core arranged in the NORMAL configuration as shown in Figure 2 or Figure 4.

As discussed in Section V, it is beyond the scope of the CSAR analyses to model a core undergoing simultaneous changes due to geometry (i.e. fuel movement) and number density (i.e. depletion). Thus a single, nominal core configuration was modeled throughout core life to characterize reactivity behavior. The reactivity behavior of theoretical HEU and LEU cores is shown in Figure 5. The reactivity bias of 0.45%Δk/k referred to by Figure 5 is a correction applied uniformly to all reactivity calculations used in the CSAR which accounts for the average difference between measured and modeled HEU reference cores.

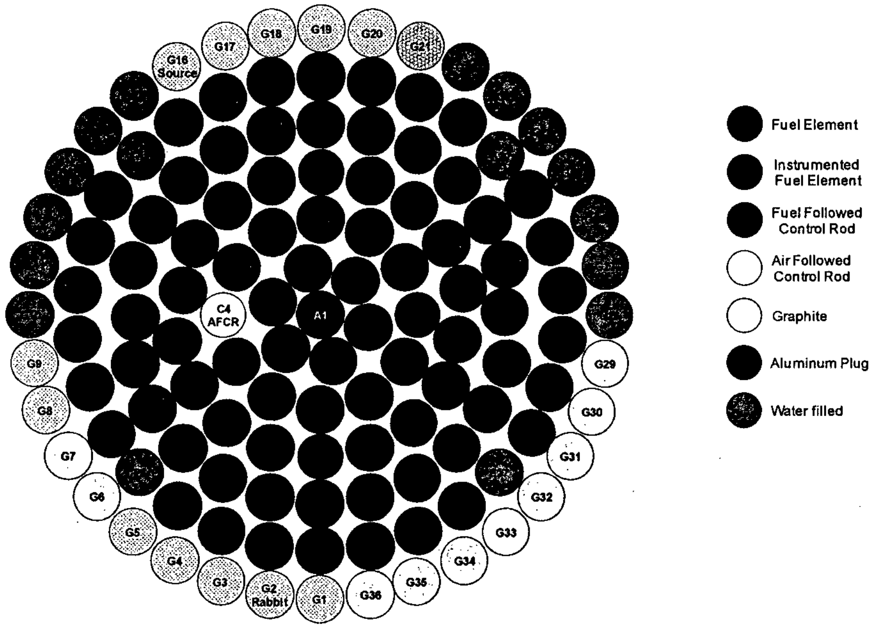

Figure 3, Initial HEU Full Power Core Configuration

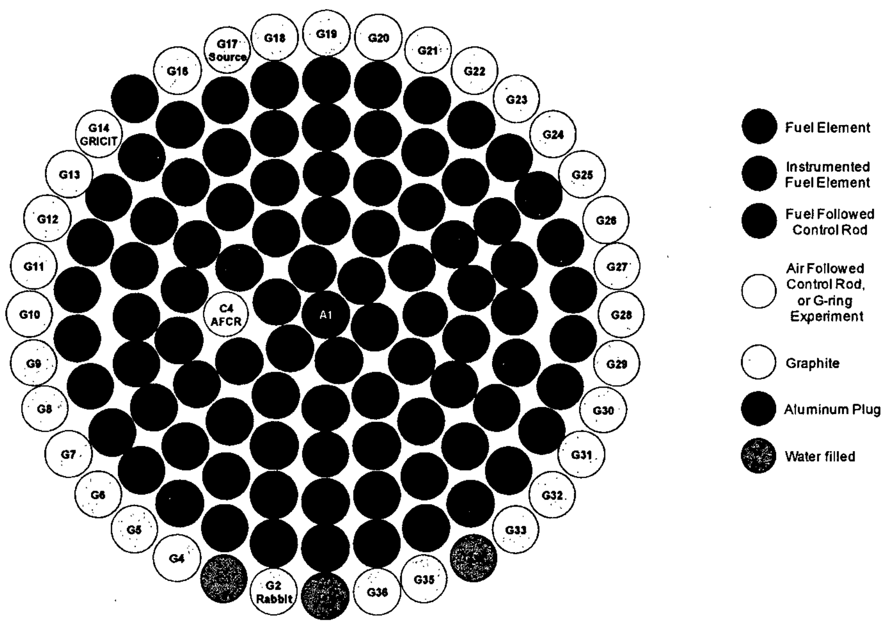

Figure 4, Initial LEU Full Power Core Configuration

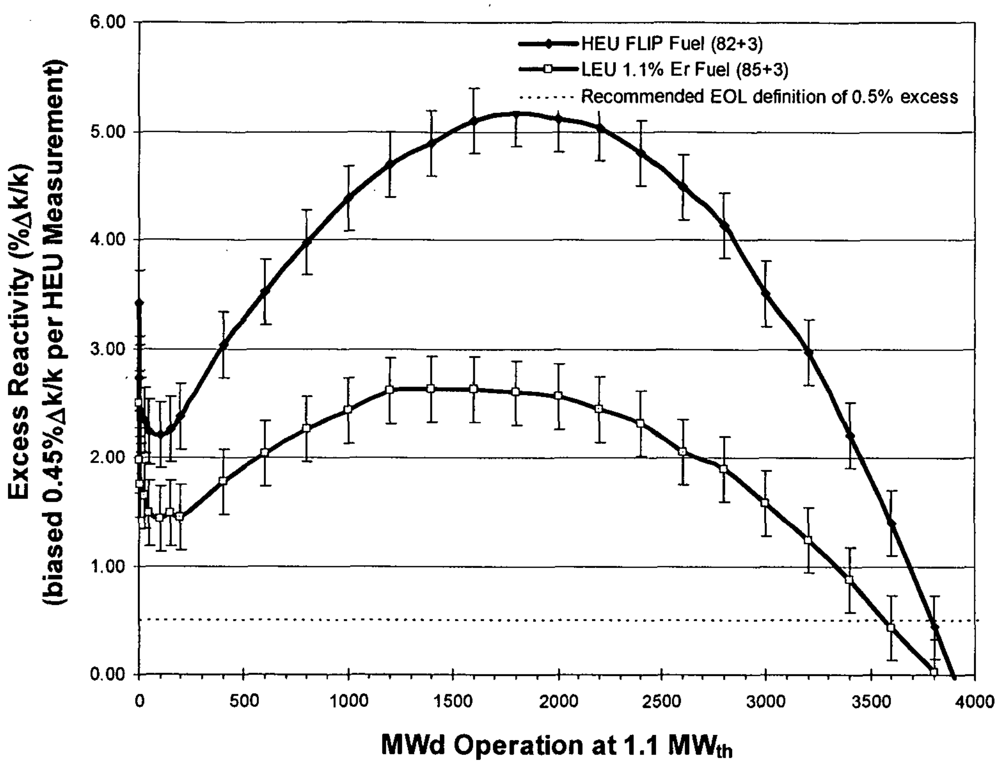

Figure 5, HEU and LEU Theoretical Core Reactivity Behavior

Note that it is not possible to measure the reactivity state of these theoretical cores since these exact configurations never actually existed. Behavior of the HEU core did track well with predicted behavior of a theoretical core, with reactivity changes due to burnout peaking at ~1700 MW-Days of operation.

C. Control Rod Calibrations

Predicted and measured rod worth curves for the HEU core at beginning-of-life (BOL) are shown in the CSAR. Results are summarized in Table 4. Predicted and measured values of total rod worth agree reasonably well, except for the transient rod where the predicted value is 26.6% greater than the measured value. The core model used for HEU BOL rod worth calculations was essentially identical to the operational core.

Table 4, Summary of HEU BOL Total Integrated Rod Worth

Control Rod	Measured Rod Worth [$]	MCNP5 Predicted Rod Worth [$]
Shim Rod	2.75	2.54 +/- 0.17
Safety Rod	2.94	3.01 +/- 0.17
Regulating Rod	3.71	3.72 +/- 0.20
Transient Rod	2.33	2.95 +/- 0.16
Sum of all Rods	11.73	12.22 +/- 0.35

The CSAR also includes predicted rod worth curves for the LEU core at BOL. Control rod worth in the LEU core was measured as soon as the operational core configuration was established. Measured and predicted values of integrated control rod worth are summarized in Table 5. The measured and predicted control rod calibration curves for each control rod are shown in Figure 6 though Figure 9. Note that the model used to predict LEU rod curves was similar to the initial operational core configuration, but not identical. Although accuracy of the worth measurement technique has not been well characterized, it has been shown to be less than 5% per measurement (OSU response to RAI question #11, 4/22/2008, rev 10). Thus a rod calibrated with eight 'pulls' would have total IRW measurement accuracy of +/- 14%.

Table 5, Summary of LEU BOL Total Integrated Rod Worth

Control Rod	Measured Rod Worth [$]	MCNP5 Predicted Rod Worth [$]
Shim Rod	2.76	2.55 +/- 0.16
Safety Rod	2.66	2.60 +/- 0.16
Regulating Rod	3.71	3.36 +/- 0.19
Transient Rod	2.86	2.86 +/- 0.15
Sum of all Rods	11.99	11.37 +/- 0.33

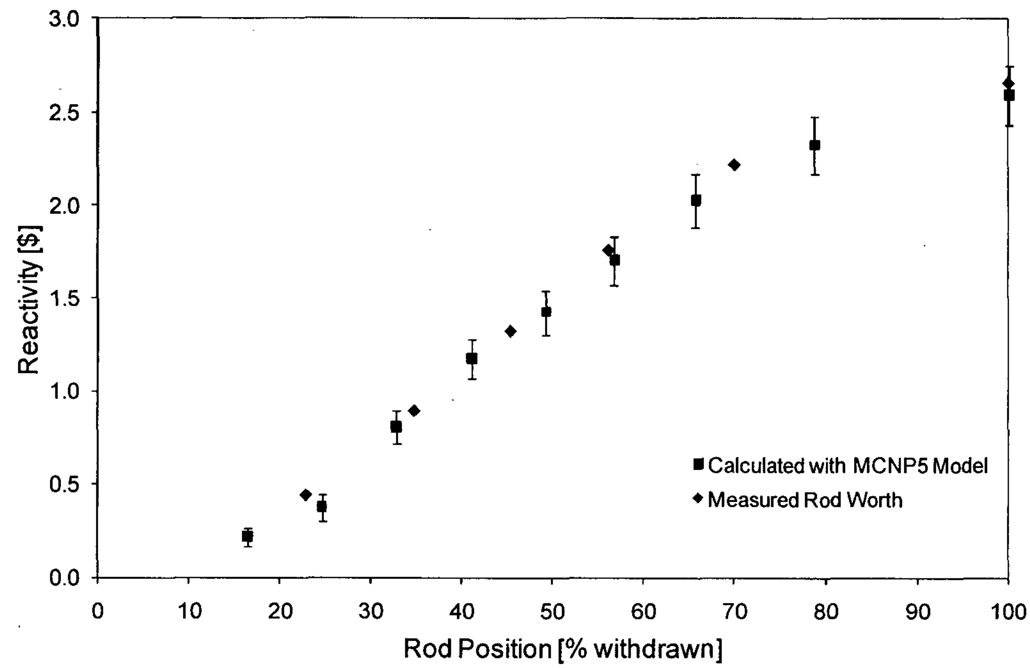

Figure 6, Safety Control Rod Calibration Curve

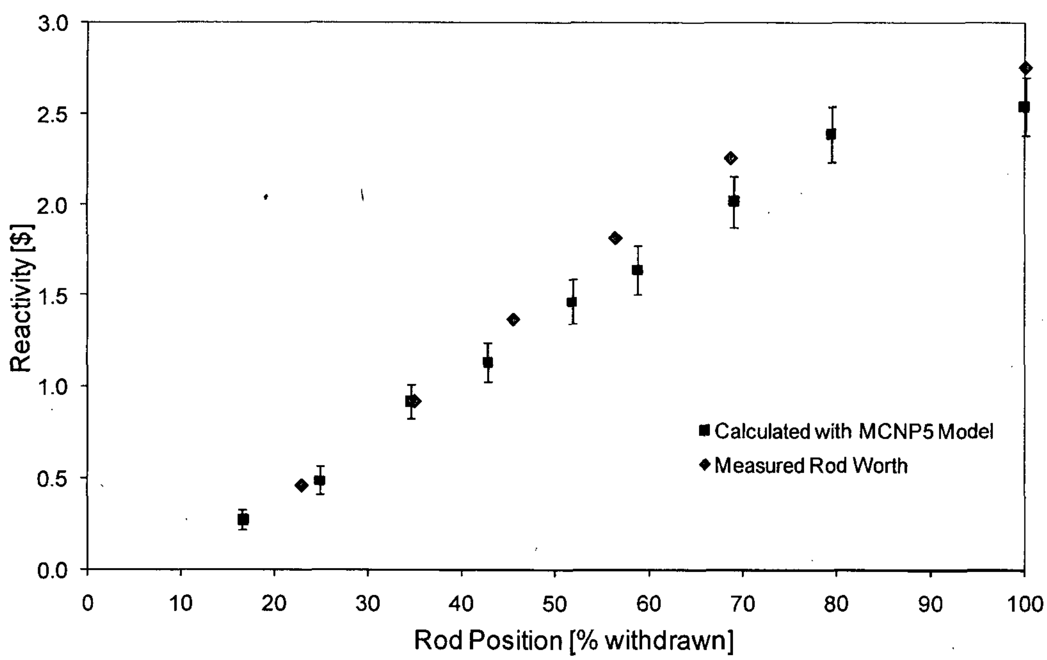

Figure 7, Shim Control Rod Calibration Curve

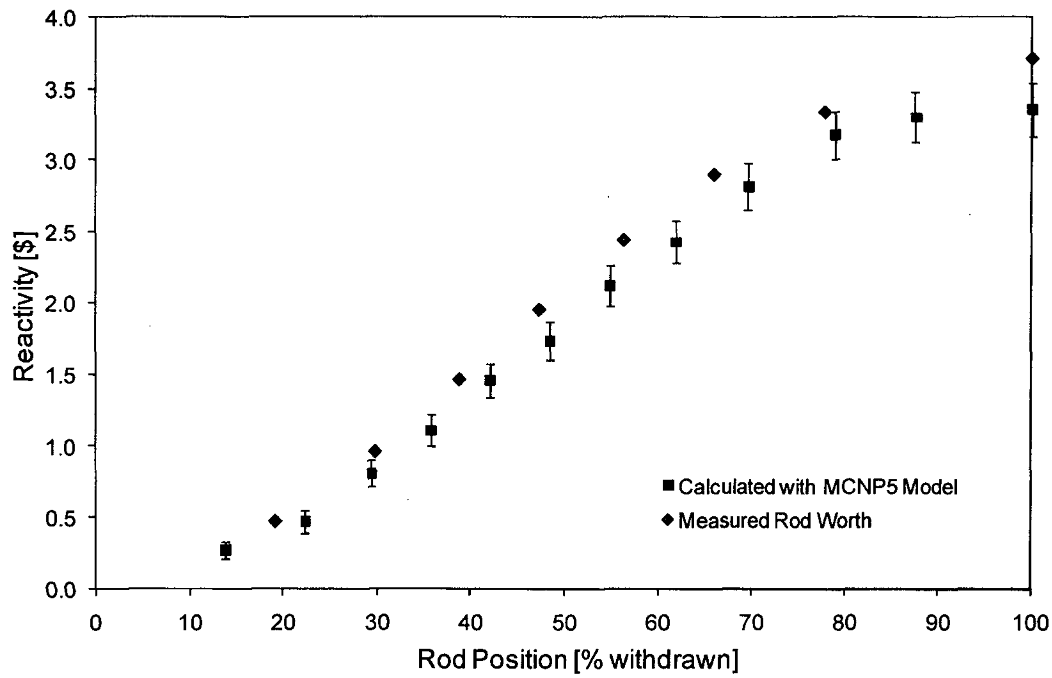

Figure 8, Regulating Control Rod Calibration Curve

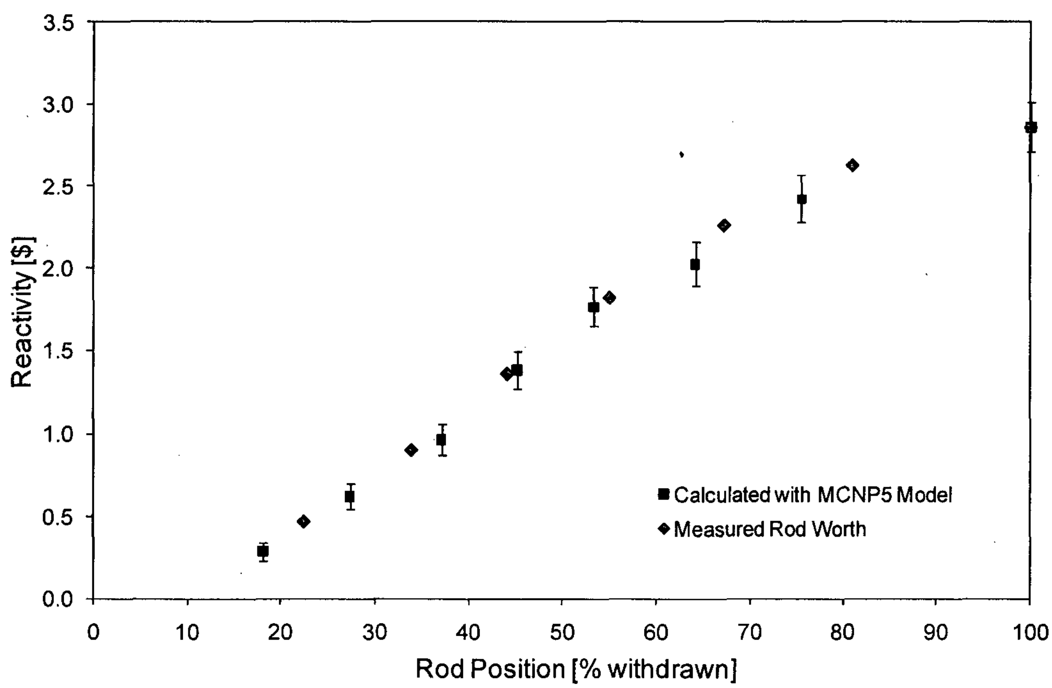

Figure 9, Transient Control Rod Calibration Curve

D. Reactor Power Calibration

Four power calibrations were performed in the NORMAL core configuration during restart testing. Calibrations in the ICIT and CLICIT core configurations were subsequently performed to verify that calibration settings established in the NORMAL configuration were suitable for ICIT and CLICIT mode operation. The calibration procedure consists of securing reactor cooling and measuring the rate of rise of pool temperature. Indicated power is then adjusted to match calculated thermal power by physically adjusting ion chamber and fission chamber position.

In accordance with the restart procedure, the first calibration was performed "at or near 500 KW" with a maximum IFE temperature not to exceed 200°C. Results of the calibration showed that average indicated power was 503 KW during the calibration, but calculated thermal power was only 248 KW. A second 50% power calibration was performed after instrument adjustment. During this calibration, indicated power was 505 KW and calculated thermal power was 508 KW. No instrument adjustments were made after the second 50% calibration.

A third calibration was performed at 90% (nominal) power. During the calibration, average indicated power was 902 KW while calculated thermal power was 908 KW. Adjustments were made, and then the fourth calibration was performed at 100% nominal power. During the final calibration, average indicated power was 975 KW while calculated thermal power was 974.5 KW. No adjustments were made.

During calibration in the ICIT core, average indicated power was 896 KW while calculated thermal power was 853 KW. During calculation in the CLICIT core, average indicated power was 995 KW while calculated thermal power was 917 KW. Since indicated power is higher than actual power in these core modes, the calibrations are considered to be conservative. No instrument adjustments were made.

Trends observed during LEU calorimetric calibrations were similar to HEU core behavior. The HEU core was typically calibrated in the NORMAL mode. This resulted in ICIT and CLICIT cores showing indicated power greater than true calculated power.

E. Shutdown Margin

As discussed in section IV.B, shutdown margin is inextricably tied to core excess reactivity. In the HEU core, there was no technical specification regarding core excess reactivity, and shutdown margin was managed by adjusting core fuel inventory as needed over core lifetime. The shutdown margin of the initial operational LEU core was measured at $1.24 with a corresponding core excess of $7.12. Shutdown margin and core excess will be managed in the same manner for the LEU core as the HEU core, although Figure 5 indicates that the reactivity changes due to burnout in the LEU core will be significantly smaller than those seen in the HEU core. Note that no prediction of shutdown margin or core excess was performed for the initial operational core (as configured), so no formal comparison between predicted and measured values can be made.

F. Neutron Flux Distribution

Neutron flux distribution in the HEU core was measured many times at many locations using many different techniques during HEU core life. The most recent measuring techniques involved the use of a cadmium covered gold foil to measure activation due to epithermal flux and an uncovered gold foil to measure activation due to total flux. Using results of bare and covered foil irradiations, the thermal flux could be derived. No theoretical flux calculations were performed since there is no accurate model of the partially depleted HEU core in existence.

LEU core flux distribution was measured using covered and uncovered foils as soon as practical. Fluxes in all major facilities were evaluated. Peak values of measured flux in the various facilities are shown in Table 6. Measured flux distributions for all facilities in the LEU core are shown in Figure 10 though Figure 14. Measured flux distributions in the HEU core ICIT facility are shown in Figure 10 for comparison. MCNP Calculations of predicted values of flux distributions in the LEU core have not been completed.

Table 6, Peak Fluxes in the HEU and LEU Cores

Facility	HEU Peak Thermal Flux [n/sec-cm^2]	HEU Peak Epi Flux [n/sec-cm^2]	LEU Peak Thermal Flux [n/sec-cm^2]	LEU Peak Epi Flux [n/sec-cm^2]
ICIT	1.1E13 +/- 7E11	9E11 +/- 8E10	5.5E12 +/- 3E11	1.0E12 +/- 1E11
CLICIT	~0	1.2E12 +/- 1E11	~0	1.3E12 +/- 1E11
GRICIT	7.2E12 +/- 4E11	4.3E11 +/- 2E10	3.4E12 +/- 2E11	3.3E11 +/- 2E10
Lazy Susan	3.0E12 +/- 2E11	1.2E11 +/- 7E9	2.3E12 +/- 2E11	9.6E10 +/- 1E10
Th. Column	8E10 +/- 1E10	~0	7E10+/- 9E9	~0
Rabbit	1.0E13+/-8E11	4.0E11 +/- 3E10	8.3E12 +/- 8E11	1.2E11 +/- 1E10

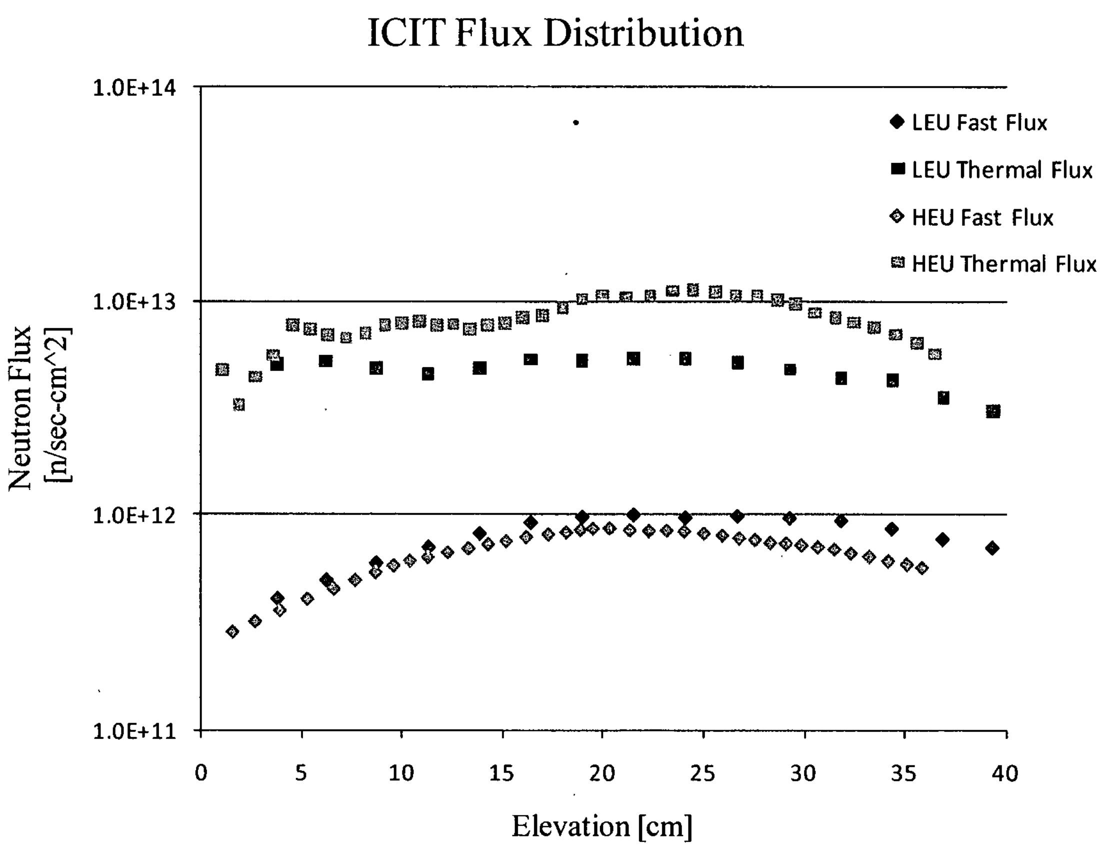

Figure 10, Comparison of ICIT flux distribution in the HEU and LEU Cores

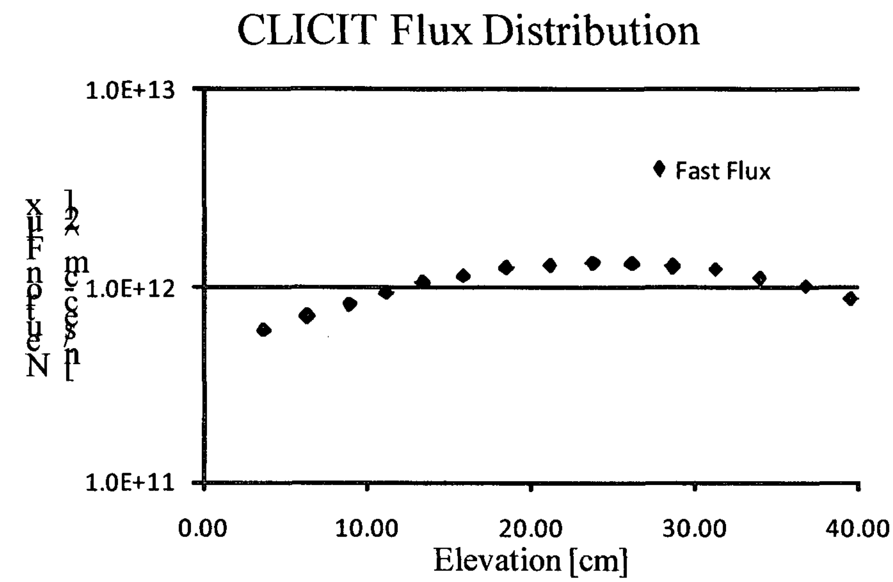

Figure 11, Measured flux in the LEU Core CLICIT Facility

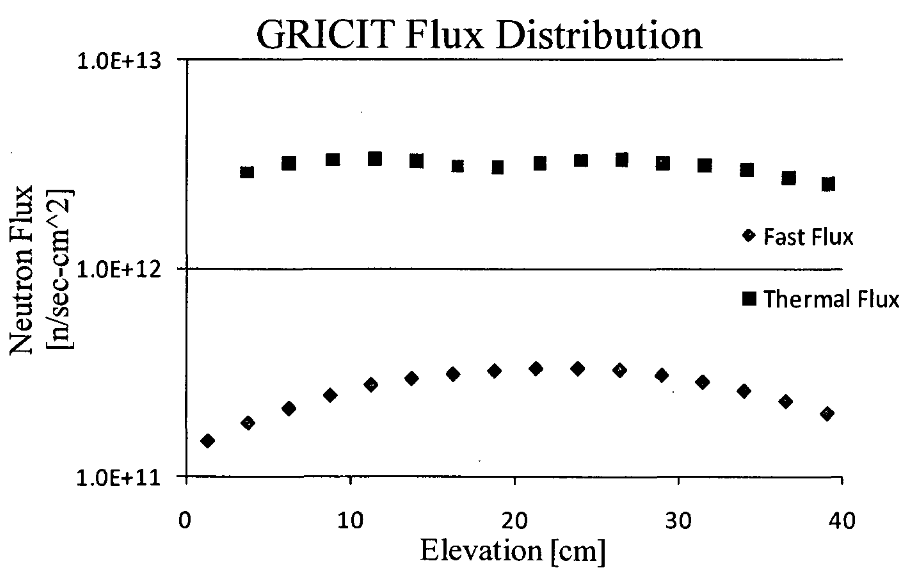

Figure 12, Measured flux in the LEU Core GRICIT Facility

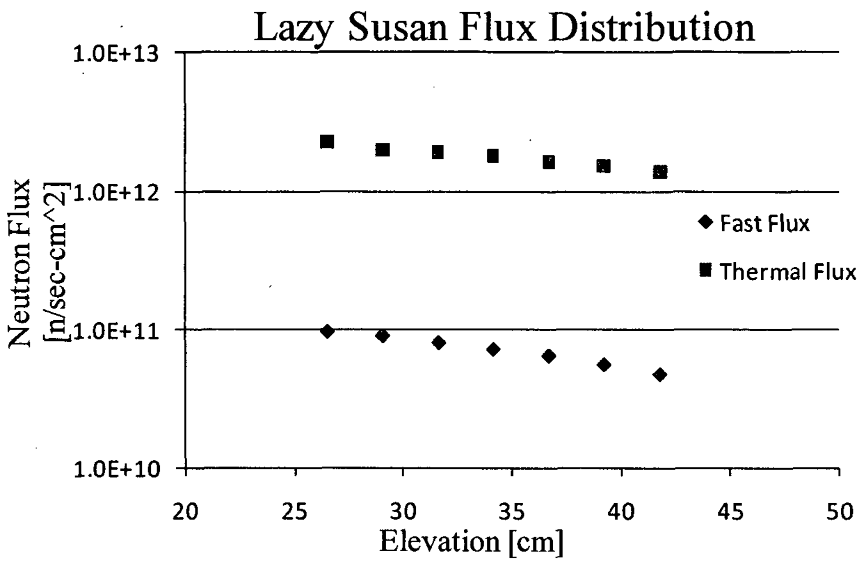

Figure 13, Measured flux in the LEU Core Lazy Susan Facility

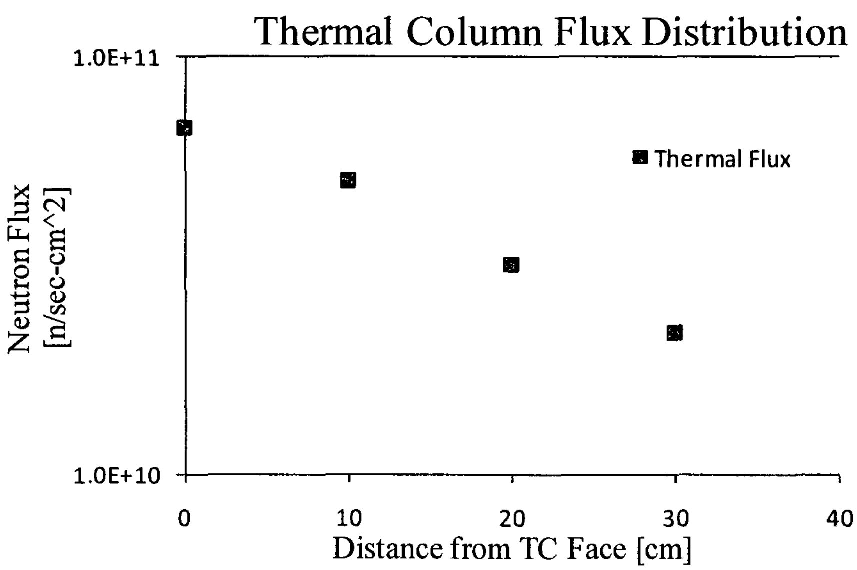

Figure 14, Measured flux in the LEU Core Thermal Column Facility

G. Reactor Physics Measurements

Various reactor physics measurements were taken on the LEU core during the initial startup or shortly thereafter. A summary of predicted and measured reactor parameters is shown in Table 7. Values in the "HEU predicted" column are taken from CSAR HEU middle of life (MOL) core calculations, and thus may not accurately predict actual HEU parameters since the HEU MOL model is not identical to the HEU core at the time the core was removed. Certain parameters in the "HEU measured" column have not been recently evaluated, but historic values are included for reference.

Measurements in addition to the items in Table 7 are planned or are in progress. These measurements include such things as temperature profiles, reactivity worth measurements of additional facilities and reactor response to non-steady-state reactivity perturbations.

Table 7, Summary of Reactor Physics Measurements

Parameter	HEU Predicted	HEU Measured	LEU Predicted	LEU Measured
Void Coefficient (core center)	-\$0.86/% void	-\$0.51/% void	-\$0.96/% void	-\$0.65/% void
Void Coefficient (core average)	-\$0.16/% void	Not Measured	-\$0.19/% void	Not Measured
Fuel Temperature Coefficient	-0.51¢/°C	-0.38¢/°C	-0.59¢/°C	-0.53¢/°C
Pool Temperature Coefficient	-0.57¢/°C	~ -0.25¢/°C	-0.72¢/°C	-0.40¢/°C
Effective Delayed Neutron Fraction	0.0078	None Recent (0.0070)	0.0076	0.0080
Neutron Generation Time	32.5 μsec	None Recent (53 μsec)	22.6 μsec	25.6 μsec[3]
β/ℓ	240[1]	262[2]	336[1]	312[2]
Power Defect	\$2.05	\$1.72	\$2.16	\$2.41

1. Predicted value derived from ratio of predicted β and ℓ.
2. Measured value obtained from pulse test data, see section IV.J.
3. Calculated from experimentally determined value of β and experimentally determined value of β/ℓ.

H. Initial LEU Core Loading

During loading of LEU fuel, the criticality state of the core was tracked using a standard 1/M plot. As expected, the criticality prediction became more accurate as the reactor became closer to critical. Using all data points, the predicted critical condition was 68.5 fuel elements. Using the last three data points, the predicted critical condition was 66.3 fuel elements. The reactor was actually taken slightly supercritical after the addition of the 66th fuel element. The critical configuration is shown in Figure 2. With all control rods fully withdrawn, the core containing 67 fuel elements had an excess reactivity of $0.48. With each subsequent 'batch' fuel addition, the reactor became closer to critical and the allowed number of fuel elements that could be added to the core between count rate measurements was decreased in accordance with the restart procedure.

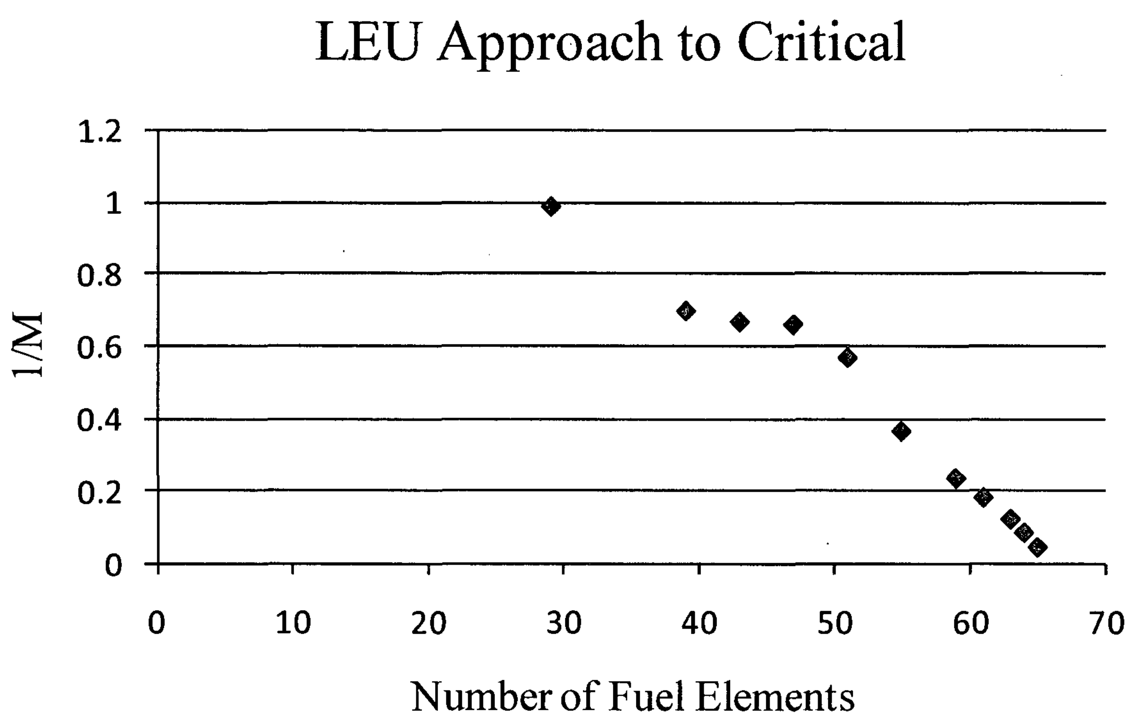

Figure 15, LEU 1/M Plot

I. Primary Coolant Measurements

Primary coolant total activity is monitored continuously. Samples are taken on a monthly basis and analyzed for individual isotope activity. During the conversion outage, activity levels (individual and total) were seen to decrease. After the restart, coolant activity levels returned to normal, pre-shutdown levels.

J. Pulse Test Results

Pulse testing was conducted in the initial operational LEU core by performing a series of pulses in \$0.10 increments, starting at \$1.10 and ending at \$2.25. These tests were performed to confirm the linearity of plots as required by the Pulse Mode Test acceptance criteria. Additionally, physics data such as the temperature coefficient of reactivity, and the ratio β/ℓ can be derived from pulse data. Pulse testing using \$0.25 increments in the ICIT and CLICIT cores was also performed. Results of pulse testing in the initial operational core are shown in Figure 16 through Figure 18.

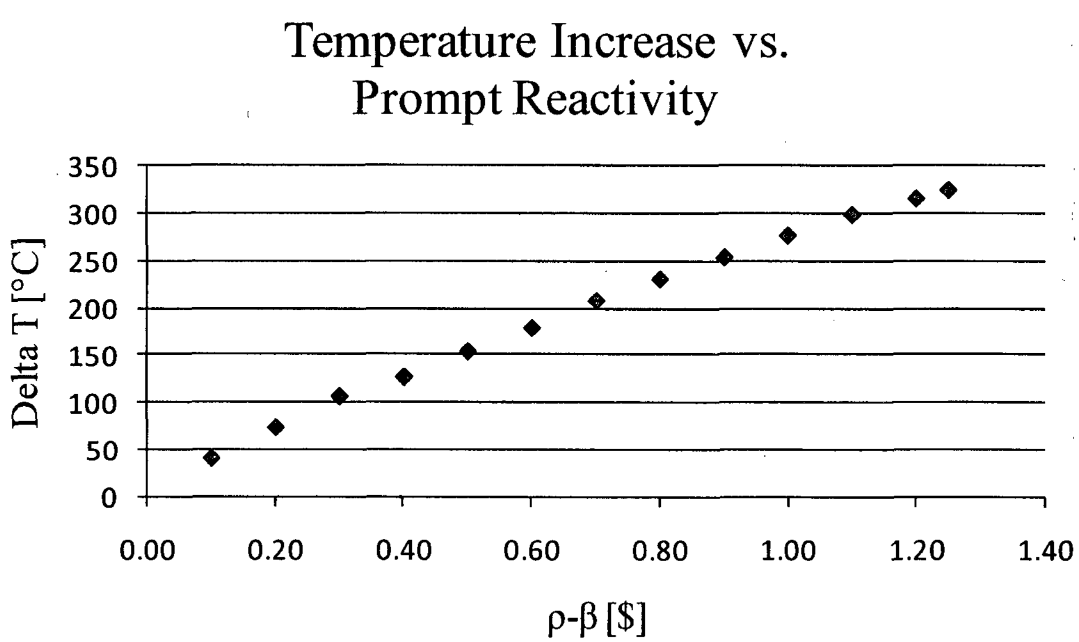

Figure 16, LEU Temperature vs. Prompt Reactivity

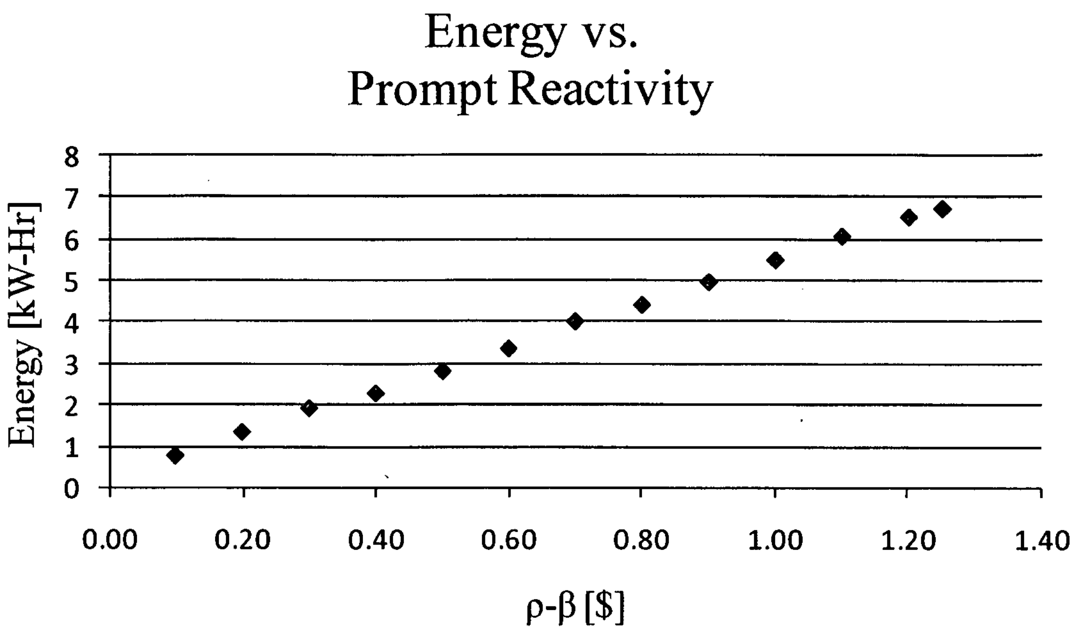

Figure 17, LEU Energy vs. Prompt Reactivity

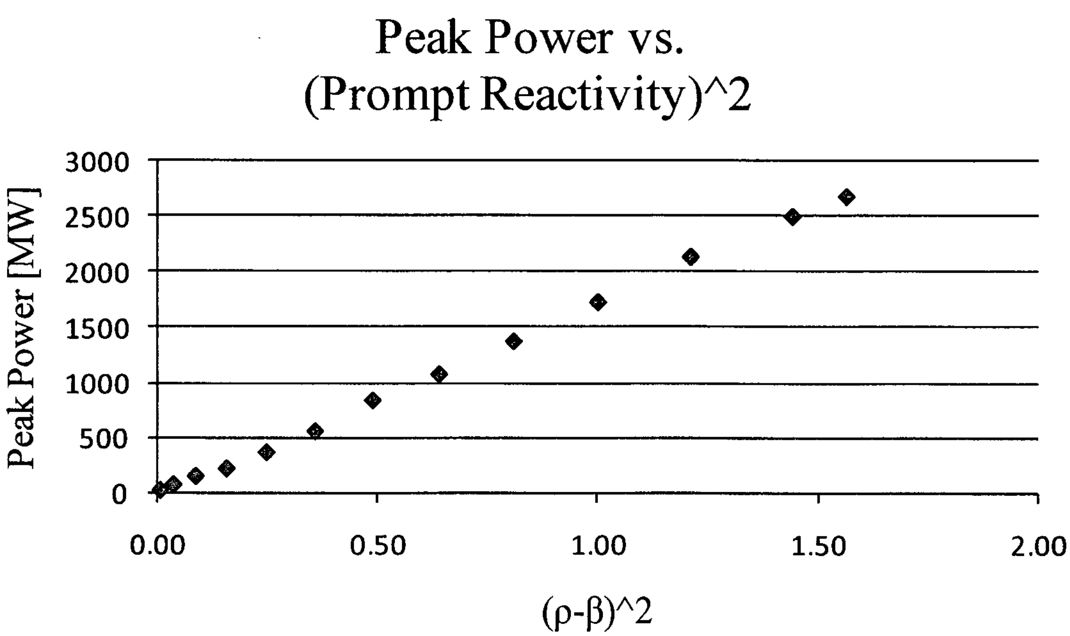

Figure 18, LEU Peak Power vs. (Prompt Reactivity)^2

V. Discussion of Results and Lessons Learned

In general, predicted results were in good agreement with measured results in both the HEU and the LEU cores. This is especially true of 'global' reactor parameters such as excess reactivity and delayed neutron fraction. When comparing measurement vs. prediction, it should be noted that the core configurations used for CSAR calculations were similar, but not identical to actual installed cores. The operating LEU BOL core has more fuel elements than the LEU BOL core modeled in the CSAR. The operating HEU MOL core had a different number of fuel elements than the HEU BOL core modeled in the CSAR, and the distribution of materials in each fuel element differed (perhaps significantly) due to inherent modeling limitations. Differences between predicted and measured values can be attributed principally to differences between modeled and operational cores.

A. Critical Mass

The LEU core satisfied the acceptance criterion for critical mass. The model predicted the reactor would be critical after the addition of the 69th fuel element. The reactor was actually taken critical after the addition of the 66th fuel element. The acceptance criterion was 60 – 70 fuel elements. Since the predicted criticality was so high in the acceptance band, the loading plan was biased to preferentially load low erbium elements in the core center, thus reducing the number of elements required for criticality. Also, the model assumed 1.1 mass percent of erbium in all elements while fuel fabrication records show that actual erbium loading (core average) was 1.07 mass percent. Since true erbium content was less than the value used in the model, and the core was preferentially loaded with lower erbium elements in the middle, it is not surprising that criticality was achieved with three fewer elements than predicted.

B. Excess (Operational) Reactivity

The LEU core satisfied the acceptance criterion for excess reactivity. The initial operational core was configured to optimize flux in the various OSTR irradiation facilities. As such, the initial core configuration was not identical to the core configuration modeled in the CSAR. The installed core also differed from the modeled core due to variations in fuel composition. For these reasons, no meaningful comparison could be made between the modeled core and the installed core regarding initial values of core excess reactivity.

Modeling a reactor that is undergoing simultaneous changes due to depletion and fuel movement is very challenging. The OSTR was modeled using a single core configuration over the entire core lifetime to simplify modeling tasks. Thermal hydraulic analyses were performed at representative times during core life (beginning, middle and end of core lifetime). Adjustments to neutronic and thermal-hydraulic models were made to compensate for changes in core geometry as needed for accuracy and conservatism of modeling. Details can be found in the CSAR.

When using a single configuration to model a TRIGA® core over its entire lifetime, changes in shutdown margin and core excess of several dollars are observed. Over the core lifetime, the HEU core was shown to demonstrate larger reactivity changes than the LEU core (due principally to higher initial erbium content). The comparative magnitude of reactivity changes over core lifetime is shown in Figure 5. In order to meet shutdown margin and core excess reactivity limits, it will be necessary to periodically add or remove fuel elements from the core. It is expected that such core manipulations will be needed less frequently in the LEU core than the HEU core.

C. Control Rod Calibrations

The LEU core satisfied the acceptance criteria for control rod worth measured by both the drop method and the period method. Control rod calibrations at the OSTR have traditionally been performed using the period method where the test rod is withdrawn in increments and the reactivity worth of each increment is calculated from the measured reactor period using the Inhour Equation. During initial stages of the restart, the core did not possess sufficient excess reactivity to allow the reactor to be critical with three control rods fully withdrawn and one rod fully inserted, as required to perform calibration by the period method. During these times, an alternate method was used to calibrate control rods. This rod drop calibration method is discussed in detail in Appendix C – Restart Procedure. Tests performed on the HEU core prior to shutdown showed that integrated control rod worth (IRW) calculated using the drop method were within +15 to -10 percent of the values calculated using the historically accepted period method, with an average deviation of -2 percent. This was deemed acceptable accuracy for estimating IRW, core excess and shutdown margin during the initial stages of the restart procedure. Data obtained during restart indicated that rod worth measurements taken in the LEU core using both techniques were consistent.

HEU BOL calculations of rod worth compared well with measured values of rod worth, except for the regulating rod where the measured value was 21% less than the predicted value. In the LEU core, the model correctly predicts the exact value of the regulating control rod, but the measured value of the transient control rod is 10% higher than the predicted value. The reason for the large deviation between predicted and measured worth in the HEU core regulating rod is discussed in the answers to the CSAR RAI questions. The deviation is shown to be within measurement and prediction error of the methods used. This reasoning applies to the LEU core as well. A comparison of HEU and LEU rod worth predictions and measurements is shown in Table 8.

Table 8, HEU and LEU Core Predicted and Measured BOL Control Rod Worth

Rod	HEU Core			LEU Core		
	Predicted	Measured	Deviation	Predicted	Measured	Deviation
Shim	2.54	2.75	+8%	2.55	2.76	+8%
Safety	3.01	2.94	-2%	2.60	2.66	+2%
Transient	3.72	3.71	-0%	3.36	3.71	+10%
Regulating	2.95	2.33	-21%	2.86	2.86	0%

D. Reactor Power Calibration

The LEU core satisfied the acceptance criteria for the reactor power calorimetric calibration. After the initial large adjustment at 50% nominal indicated power (~25% calculated thermal power), only small adjustments were needed at higher power levels. The large, initial adjustment can be attributed to the fact that in the HEU core, the fuel was shifted away from the fission chamber while in the LEU core, fuel is more geometrically centered and thus closer to the fission chamber, thus resulting in a significantly higher flux at the detector for a given power level.

E. Shutdown Margin

The LEU core satisfied the acceptance criterion for shutdown margin. Shutdown margin is controlled in the same manner as excess reactivity, by adding or removing elements from the core. Since the initial operational core differs from the core as modeled in the CSAR, no explicit comparison between predicted and measured values of shutdown margin can be made.

F. Neutron Flux Distribution

Fresh HEU fuel contains approximately the same amount of U-235 as fresh LEU fuel. It is estimated that since the HEU core was loaded, approximately 20% of the U-235 has been consumed, and a small amount of plutonium has been created (exact amounts of uranium depletion and plutonium breeding depend on fuel element history and spatial location within the fuel element). Since HEU fuel at MOL contains less fissile material than LEU fuel at BOL, neutron flux in the LEU BOL core should in general be less than neutron flux in the HEU MOL core. Flux in the LEU core is further 'diluted' due to the fact that there are more fuel elements in the operational LEU core than in the operational HEU core. As expected, thermal flux in the LEU core is lower than thermal flux in the HEU core.

At full power of 1.0 MW, the fission rate in either core would be essentially the same, and thus the number of fast neutrons produced per unit time would be unchanged between the two cores. As above, the increased number of fuel elements tends to dilute the fast flux in the new core. This is counterbalanced by the slight spectral hardening in the LEU core due to decreased amounts of ZrH moderator. The HEU fuel meat contains 90% ZrH by mass, while the LEU fuel meat contains 69% ZrH by mass. Fast flux in the new core is thus expected to be very similar to fast flux in the new core.

These trends are confirmed by the results shown in Table 6. A significant decrease in thermal flux is seen throughout the reactor. Fast flux shows a slight increase, except for the rabbit facility. The reduced fast flux in the rabbit in the LEU core is likely due to better neutron thermalization achieved by additional water holes adjacent to the rabbit.

It should be noted that the HEU MOL core configuration is significantly different from the HEU BOL configuration shown in Figure 3, thus making detailed, meaningful comparisons between measured HEU MOL and LEU BOL fluxes difficult. Furthermore, the large decrease in thermal flux shown by initial flux measurements may not be as severe as indicated since the measured results do not account for self shielding in the flux foils. Preliminary results reported by OSTR users indicate that initial estimates of flux measured in the LEU core tend to be low by ten to twenty percent. The degree of flux underprediction due to self shielding is also known to be about 15% in the flux measurement foils used by the OSTR. Rather than a 40% reduction in thermal flux, a 25% reduction is more likely.

G. Reactor Physics Measurements

Agreement between predicted and measured LEU core parameters was reasonably good. As stated elsewhere, the HEU MOL model is significantly different from the actual core, so meaningful comparison is difficult.

H. Initial LEU Core Loading

Criticality predictions obtained from the 1/M plot were accurate. The predicted initial critical core load of 69 fuel elements was also quite good, considering the preferential core loading and actual erbium content of the fuel, as fabricated.

I. Primary Coolant Measurements

Primary coolant activity during the conversion outage decreased as would be expected due to decay of activated materials, and continuous purification of primary water by circulation through the resin bed. After the reactor was restored to normal operation, gross activity and activity of individual isotopes returned to essentially the same levels seen before the outage.

J. Pulse Test Results

The LEU core satisfied the acceptance criterion for pulse mode testing. Each graph shows strong linearity. When a linear trend line is applied to each graph, the R^2 correlation factor is greater than 0.99 in each case. The experimentally determined value of β/ℓ is also in good agreement with the ratio of β and ℓ predicted by the LEU model.

K. Non-Routine Activities and Conditions

Refueling the OSTR is a very uncommon activity. The last core replacement occurred in 1976 when the original standard LEU fuel was removed and HEU Fuel Lifetime Improvement Program (FLIP) fuel was installed. Conversion activities of 2007/2008 were based, to some degree, on 1976 refueling activities. The OSTR also benefitted from experience gained during recent conversion activities at Texas A&M and the University of Florida. The conversion was completed without any major, unexpected difficulties, but there were a few non-routine conditions encountered during the process. These are discussed below in chronological order of discovery or occurrence.

1. Splitting and blistering of old aluminum clad reflectors. All aluminum clad reflector elements remaining in the core at the time of conversion were known to be swollen to the extent that they could no longer be removed through the upper gridplate. The scope of conversion work included removing all of the old reflector elements and loading new, stainless steel clad reflector elements. Upon removal, it was found that some of the old elements were severely cracked (Figure 19) and blistered (Figure 20). Some reflector elements were also found to have a greenish tint in their upper and lower sections. The mechanism that caused cracking, blistering and discoloration is not precisely known, but stress corrosion is suspected. The blistering is likely due to localized corrosion due to impurities in the aluminum. The pronounced crack in the clad of some of the reflectors can be attributed to the method of manufacture, where the clad was rolled and welded. The weld location would later be susceptible to stress cracking. No pattern due to core location or element orientation could be determined.

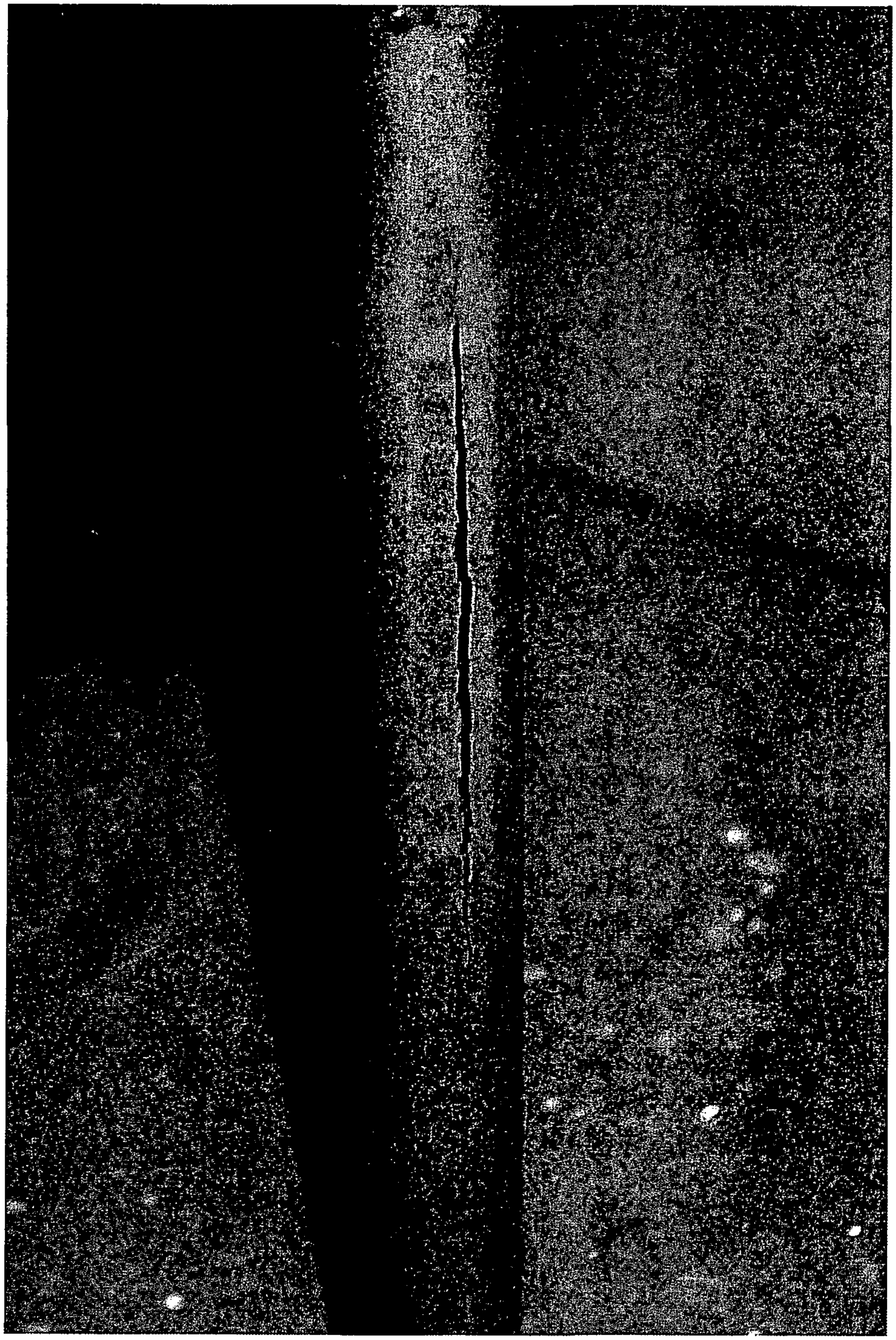

Figure 19, Cracked Reflector Element

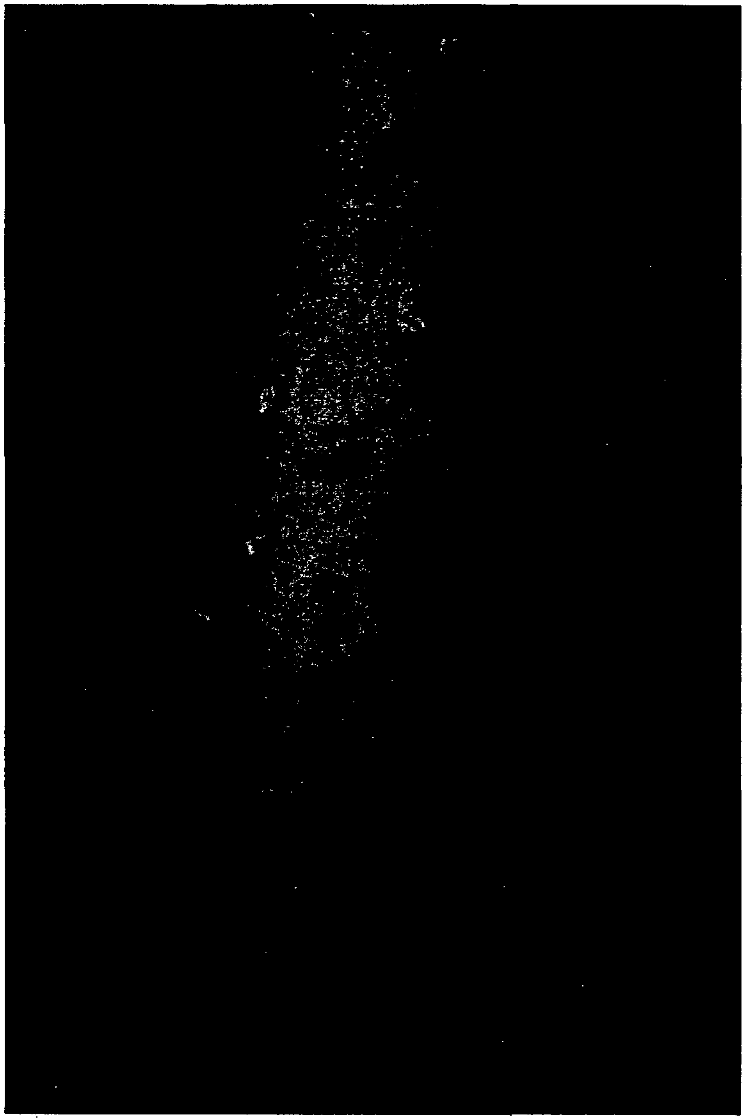

Figure 20, Blistered Reflector Element

2. Elevation change of fuel element in core lattice position D8. The OSTR lower gridplate has eight penetrations to accommodate control rods. The four unused positions are plugged by adapters which allow placement of fuel in these locations. The adapters have two indexing pins on the lower end and a socket which accepts fuel on the upper end. While loading fuel in position D8 (an adapter location), it was noticed that the element was elevated ~1 cm higher than adjacent fuel elements. Investigation revealed that the adapter indexing pins were not properly seated in the safety plate located below the lower gridplate. Once the adapter indexing pins were reseated, the fuel element was reloaded and observed to be at a normal elevation. Fuel elements in the other three adapter locations were verified to be at the proper elevation.

3. Missed Technical Specification surveillance during initial criticality. When the reactor is operating, an operability check of the console is required daily. On October 7, the OSTR was taken critical for the first time with LEU fuel. Fuel loading to achieve initial criticality had been planned for October 8, but the staff was ahead of schedule. The following day (October 8), it was determined that the required console operability checks had not been performed prior to initial criticality. This event was not a Technical Specification violation since it involved a missed surveillance, but not operation in violation of a Limiting Condition for Operation. Corrective actions consisted of a two day suspension of reactor operation to allow thorough review of the newly implemented LEU technical specifications, and addition of signature hold-points to the restart procedure.

4. Fluctuation of top thermocouple indication. During initial full power operation, one channel of temperature indication on the IFE was observed to periodically drop 100°C to 200°C for several minutes to several hours. The intermittently failing channel is the lowest reading of the three channels (the 'top' channel). The other two channels are operating normally, and reading in the expected range. Reactor safety is not affected. The OSTR staff has not placed the second LEU IFE in the core to verify that it operates normally. The bad channel on the installed IFE continues to operate erratically. Due to time scales involved, it is apparent that item #4 has no connection to item #5.

5. Power fluctuation during initial high power runs. During initial full power operation, reactor power was observed to dip five to ten percent and then recover within a few seconds. It is believed that these power fluctuations were due to dissolved air coming out of solution and forming gas bubbles in the core. The amount of air in solution in the primary water at the end of the outage would have been much higher than usual since there was no hot spot in the tank for over two months and hence no gas stripping. The dissolved gas theory is supported by two pieces of evidence.

   First, the magnitude of the dip and recovery was exacerbated when the reactor was operated in automatic mode. This was because when power dropped, the regulating rod would withdraw to compensate, thus briefly placing the reactor on a positive period. Once the gas bubble achieved sufficient buoyancy, it would exit the core and introduce additional positive reactivity. The timing of these events tended to cause larger power overshoots on recovery.

   Second, the power oscillations became less frequent as the reactor was operated at high power. This indicates that the abnormally high level of dissolved gas was gradually reduced over time. As water was circulated through the hot core, dissolved gasses were stripped out until normal levels were restored. This is further supported by visual observations. During initial power operation, large amounts of bubbly water could be seen exiting the core. After several hours of full power operation, power oscillations had essentially ceased, and the condition of the water leaving the core had returned to normal.

During initial operation, the reactor was operated principally in Steady State mode to minimize the amplitude of power oscillation. After several hours of full power operation, the reactor was returned to automatic mode. No further abnormal oscillations were seen. Although the core was preferentially loaded with low erbium elements in the core center, it is not believed that the power oscillations were due to localized boiling. If it were, the phenomenon would still be occurring.

6. Length adjustment of fission chamber support stalk. The initial power calibration indicated that nuclear instruments (NI's) were reading too high by a factor of ~2. The customary manner of adjusting NI's involves physically raising or lowering the detector as needed. In this case, it was necessary to raise the fission chamber support stalk several centimeters to achieve the desired reading. As a result, the top of the support stalk was high enough that it interfered with the closing of the reactor tank lids. To allow the fission chamber to operate at the proper elevation and also allow closing of the reactor tank lids, the fission chamber support stalk was shortened by ~15 cm.

L. Lessons Learned

There will be a formal 'lessons learned' meeting at General Atomics headquarters involving all parties during the month of April 2009. In advance of that meeting, the major lessons learned at the OSTR during conversion can be summarized as follows:

- Where possible, take advantage of opportunities to adapt and utilize materials, and especially procedures previously developed at the facility being converted or other previously converted facilities.
- The importance of having a well organized project manager cannot be overstated. The OSTR conversion process benefitted greatly from having an organized and experienced project manager.

- A Walk-through or dry-run should be performed whenever possible. Before the start of any major activity, take the time to briefly review the sequence of major activities, expected indications and actions to take in the event that things don't go as planned.

- When performing repetitive tasks such as receiving new fuel or offloading spent fuel, take the time to do things right the first time. This may extend the duration of the first repetition, but will make all the rest go more quickly and smoothly.

- At the OSTR, the single most important tool was found to be a small, durable high quality camera. Remote viewing equipment made many difficult jobs significantly easier. Some of the work would have been impossible to complete without remote viewing capabilities.

Appendix A – Facility Description

The OSTR is a Mark II TRIGA® reactor licensed for operation up to 1.1 MW. The reactor is situated in a water filled aluminum tank. The water provides cooling, shielding and neutron moderation. Primary water is circulated through the tank, although flow through the core is provided by natural circulation. A secondary water system cools the primary water and rejects heat to the atmosphere. The tank is surrounded by concrete which supports the tank and provides shielding. Cross section drawings of the reactor are shown in Figure 21 and Figure 22[1].

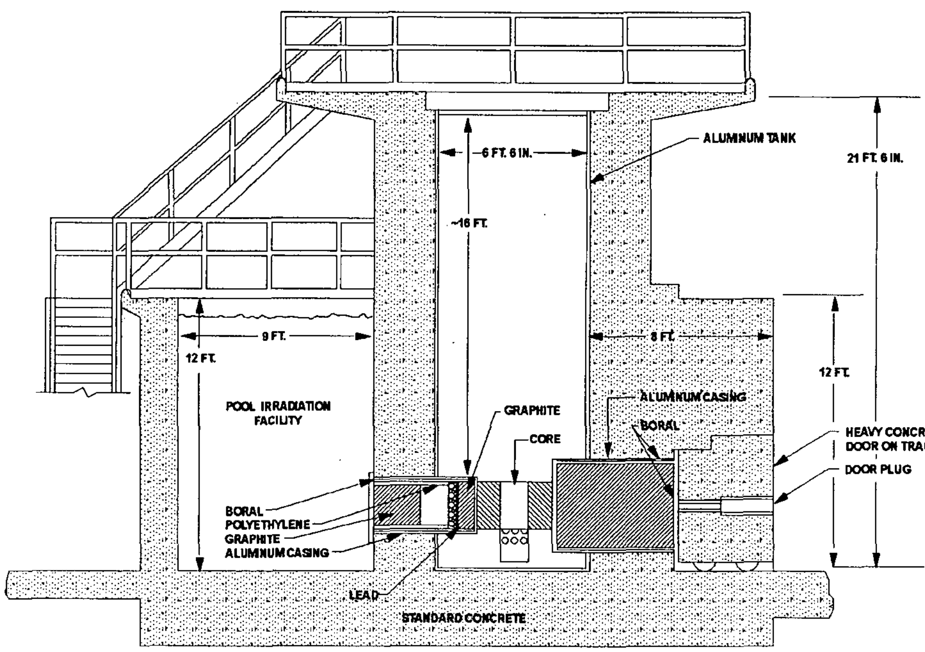

Figure 21, OSTR Vertical Section

[1] Figures and information in this section have been adapted from the Oregon State TRIGA Reactor Training Manual.

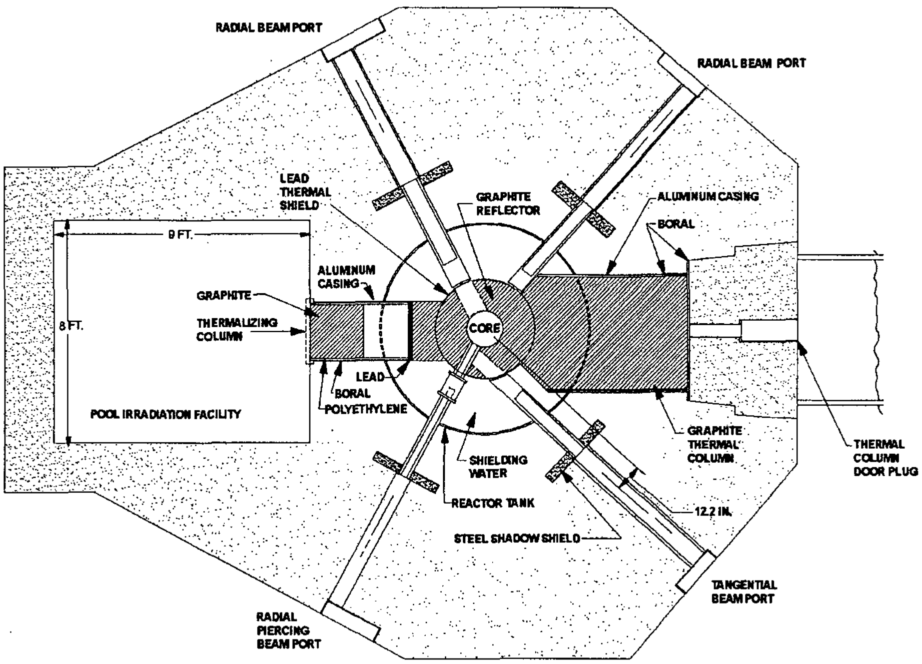

Figure 22, OSTR Horizontal Section

The core is surrounded by a graphite reflector designed to reduce neutron leakage. A 5.1 cm. blanket of lead surrounds the reflector and reduces the radiation load on the concrete bioshield. The reflector is attached to a square platform which is bolted to the bottom of the tank. The reflector supports the upper and lower grid plates which position and restrain all of the core components. As shown in Figure 22, the reflector is pierced by several beam ports. A detailed diagram of the core assembly is shown in Figure 23.

The grid plates provide 127 distinct core element positions. Positions are arranged in seven rings. The A-"ring" is the center position. The G-ring is the outer ring and has 36 positions. The grid plates are shown in Figure 24. Typical core loadings, and the grid plate locations themselves do not have rotational or mirror symmetry. Full core representation is thus required when modeling; half-core (or quarter-core or sixth-core, etc.) modeling is precluded.

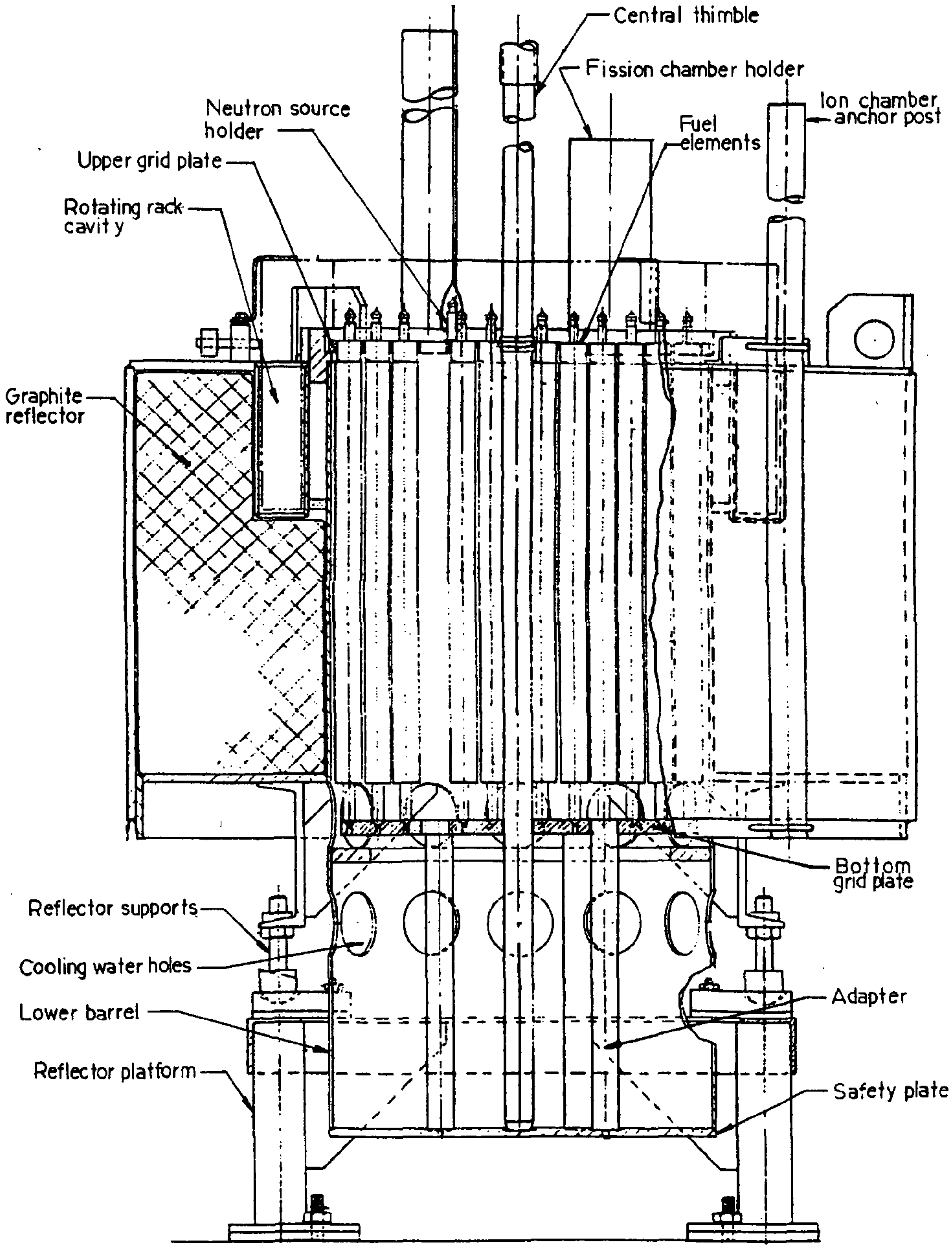

Figure 23, Core Assembly

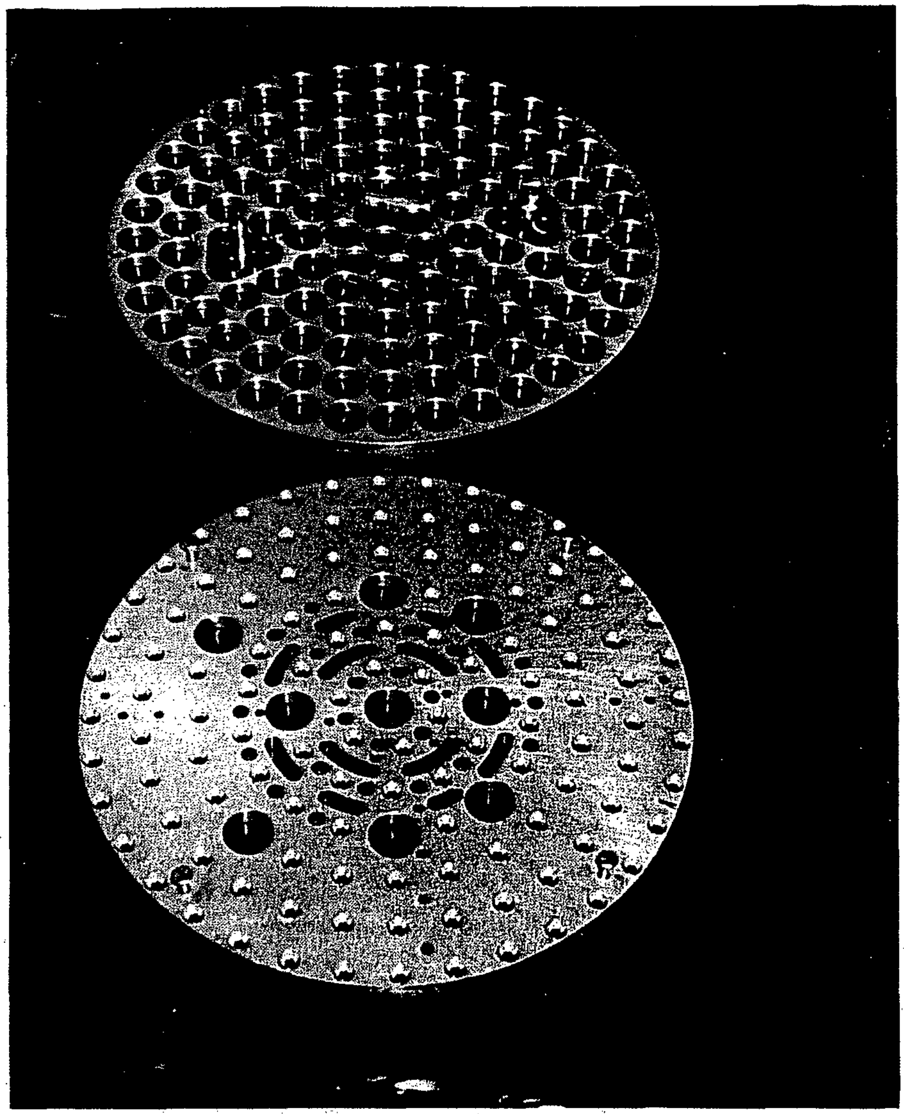

Figure 24, Upper and Lower Gridplates

Core positions may contain fuel elements, graphite reflector elements, control rods, thimbles, a neutron source, experiment holders or may be vacant (water filled). The nuclear instrumentation neutron detectors are located outside the core (adjacent to the reflector). One fuel element in the B-ring is equipped with temperature monitoring thermocouples. Core changes are performed on a routine basis as required to meet experimenter needs. The initial operation configuration as of February 2009 is shown in Figure 4.

Three control rods are fuel-followed, and the fourth is air-followed. The upper section of each control rod contains 25 weight percent B_4C dispersed in graphite. As the control rod is withdrawn from the core, the absorber section moves out and the follower is drawn up into the core. The fuel followed control rods (FFCR) are positioned with magnetically coupled electric drives. The air followed control rod (AFCR) is positioned with a pneumatic drive. A reactor pulse may be performed by ejecting the AFCR to a predetermined height, rapidly introducing up to $2.25 of positive reactivity.

The physical properties of the fuel are what make the reactor inherently safe and capable of pulsing. A team of scientists and engineers under Edward Teller was charged with designing a reactor "so safe that if it was started from its shutdown condition and all of its control rods instantaneously removed, it would settle down to a steady level of operation without melting any of its fuel or releasing fission products."[2] TRIGA® fuel takes advantage of the properties of zirconium hydride and the "warm neutron principle" (described below) to achieve the desired level of inherent safety. LEU fuel performance is further enhanced by the addition of the burnable poison erbium. The presence of erbium in new fuel allows a higher loading of U-235, and thus a longer fuel lifetime, as well as enhanced neutronic characteristics.

The hydrogen in the ZrH lattice acts as a bound oscillator which can gain or lose energy in quanta of $h\upsilon \approx 0.14$ eV. The ZrH matrix acts as a very efficient moderator as long as neutron energies are above 0.14 eV. Below this energy, neutrons in the fuel can only lose energy via the inefficient process of exciting acoustic Debye type modes. Most neutron thermalization below 0.14 eV in the TRIGA® core occurs in the water.

When fuel temperature increases (due to an increase in power or loss of coolant), the number of excited hydrogen oscillators in the fuel increases, and the probability that a thermal neutron in the fuel will gain energy also increases. The higher energy neutron will have a longer mean free path for collision, and thus a higher probability of escaping from the fuel matrix before being captured. An increase in fuel temperature thus results in a decrease in reactivity

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[2] Fouquet, D. M, Razvi, J., Whittemore, W. L., "TRIGA Research Reactors: A Pathway to the Peaceful Applications of Nuclear Energy," Nuclear News, vol. 46, no. 12, pp. 46-56 (November 2003)

due to 1) an increase in the fuel absorption disadvantage factor, 2) a decrease in U-238 resonance escape probability due to Doppler broadening and 3) an increase in core leakage.

The new LEU fuel is composed of 20% enriched uranium in a ZrH matrix which also contains 1.1 weight percent erbium. Erbium 167 has a large parasitic absorption cross section near 0.5 eV. When fuel temperature increases, the thermal neutron spectrum is hardened as discussed above, and a portion of the thermal neutron population is shifted into the region of the erbium absorption peak (see Figure 25). The presence of erbium thus augments the negative temperature coefficient. Since a major portion of neutron thermalization occurs in the fuel, this is regarded as a prompt feedback mechanism.

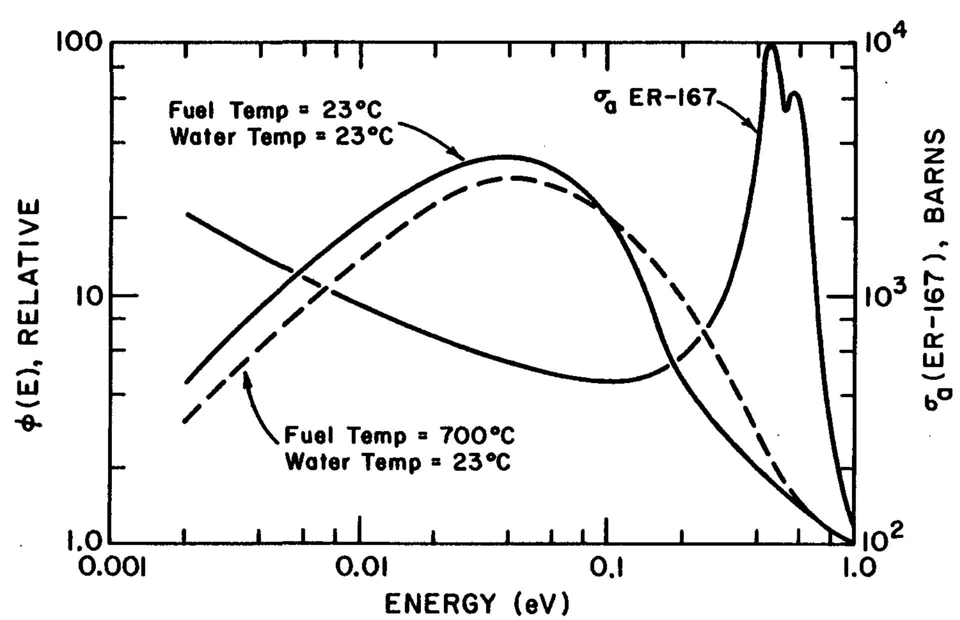

Figure 25, Thermal Neutron Spectra vs. Fuel Temperature

Appendix B – Fuel Receipt Procedure

OREGON STATE TRIGA REACTOR

OPERATING PROCEDURES

OSTROP 28

Procedure for Receipt of New Fuel

Reprinted: APRIL 2008

Revision No. 1

APPROVED BY: ______________________________

S. T. Keller

Reactor Administrator

Procedure for Receipt of New Fuel

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OSTROP 28

Procedure for Receipt of New Fuel

SCOPE

The OSTROP 28 procedure outlines the actions to be performed by the OSTR personnel during the receipt of unirradiated LEU TRIGA fuel.

I. INTRODUCTION

A. New LEU fuel is not expected to present a significant area radiation or contamination hazard. Each shipping cask will undergo a receipt survey in accordance with RCHPP 5, *Procedure for Receipt Radiation Surveys and Unpacking of Packages Containing Radioactive Material.* Contact radiation levels and contamination swipes of each fuel element will be performed as part of the receipt procedure.

B. New LEU fuel will be shipped to OSU in certified TN-BGC1 shipping casks. Personnel handling and opening casks must either be qualified or directly supervised by a person qualified to perform these operations. Qualification shall be certified in writing.

C. The TN-BGC1 shipping cask contains a TN-90 canister insert. Each insert can hold up to five TRIGA fuel elements or two instrumented TRIGA fuel elements.

D. For accountability and criticality control purposes, no more than ten LEU fuel elements shall be simultaneously unrestrained. A LEU fuel element is considered restrained if it is either fully inserted in the fuel storage rack (locked or unlocked) or if it is located inside a sealed TN-90 canister. A LEU fuel element is considered unrestrained if it does not meet either of the above two criteria.

E. All fuel located at the OSTR facility prior to receipt of the new LEU fuel shall be located in the OSTR pool, the bulk shield tank (BST) or a storage pit. No fuel located at the facility shall be moved while any new LEU fuel is unrestrained. Upon completion of this procedure, all new LEU fuel shall be stored in the dry storage rack. This procedure neither provides instruction for nor allows placement of new LEU fuel in the OSTR pool, the BST or a storage pit.

F. Security requirements for receipt of new LEU fuel are specified in the OSTR Physical Security Plan.

G. Individual unirradiated LEU fuel elements may be handled by hand. When handling this fuel, minimum protective equipment shall consist of a lab coat and gloves.

II. RECEIPT AND UNLOADING OF TN-BGC1 SHIPPING CASKS

A. Verify cask handler certification is current.

B. Verify the following equipment is staged or available:

- Two step ladders or other 4-ft-high working platform
- Crane or other device to lift the lids (approximately 88 lb)
- Rigging to lift lids (properly inspected, WLL 100 lb)
- Forklift for moving the containers between the truck and building
- Dolly, pallet jack, or transport cart for moving the shipping containers in the truck or building (minimum 873 lb capacity)
- Lock wire pliers and lock wire
- Alcohol (for cleaning O-rings and seating surfaces)
- Plastic bags and decontamination materials
- Blotter paper
- Radiological survey equipment

C. Verify shipping paperwork and container markings.

D. Transfer the first cask from the shipping vehicle to the reactor bay.

E. Perform a receipt survey on the cask per RCHPP 5, *Procedure for Receipt Radiation Surveys and Unpacking of Packages Containing Radioactive Material.*

F. Repeat steps D and E until all casks have been transferred from the shipping vehicle to the designated staging area in the reactor bay.

All fuel transferred from shipping vehicle to the OSTR bay:

____

Time / Date Signature (licensed SRO)

G. Verify that there are no more than five unrestrained fuel elements already present in the reactor bay and then open one cask and canister per the following steps:

1. Unlock the two snap locks on the shock absorbing cover, then turn the shock absorbing cover to free the two angled shanks.

WARNING: If the shock absorbing cover will be manually removed, two personnel will be required when removing the shock absorbing cover as it weighs more than 50 lbs. Care must also be taken if working on the ladders as the addition of the cover weight could cause an unbalanced condition.

2. If necessary, using the lifting A-frame and rigging, remove the shock absorbing cover [weight 40 kg (88.2 lb)] and place on raised supports.

3. Using a wrench, unscrew the tightening nut on the quick-connection cap, then remove the tightening nut and quick-connection cap.

4. Install the two tightening-ring tap-bolt handles into two diametrically opposed threaded holes on the tightening ring.

5. Place the plug pressure tool on the shipping container and secure it by screwing its knurled knobs (hand tight) into the four lateral threaded holes in the body.

6. Turn the thumb wheel on the hydraulic press tool to the fully clockwise (CW) position to close the bleed port.

7. Using the pump handle, increase pressure as read on the installed pressure gauge, until pressure indicates 300 BAR.

8. Loosen the tightening ring by rotating the two tap-bolt handles CCW until the bayonet ring can be completely unclamped.

9. Release the pressure on the plug by loosening (CCW direction) the thumb wheel on the pump body.

10. When pressure has been reduced to approximately zero, as indicated on the installed pressure gauge, remove the pressure tool.

11. Remove the bayonet ring/tightening-ring assembly and place on blotter paper.

12. If using rigging equipment to lift the plug, place the plug handling tool onto the shield plug and lock it in place.

13. Lift lid plug and place on blotter paper.

H. Transfer one fuel element from the cask to the inspection area. Remove all packing materials. Measure and record the following:

1. Fuel element serial number.

2. Highest radiation reading on contact and at one foot (amount and elevation of maximum reading).

3. Gross surface contamination level (amount).

Note: Length and bow measurements per step H.4 and H.5 may be performed at a later time and date, after all fuel has been received and stored, if desired.

4. Fuel standard length, deviation of fuel element length from the standard and the temperature of the element at the time that the length was measured (record to the nearest 0.001 inch or 0.001 cm and indicate units of measure).

5. Fuel element transverse bow (record as SAT or UNSAT).

I. Inspect the fuel element for any visible indications of damage including scratches, dents and discoloration (note that leakage most commonly occurs at weld sites). Record inspection results.

J. Transfer the fuel element from the inspection area to the storage rack. Record storage location.

K. Repeat steps H, I and J for each remaining fuel element in the shipping cask.

L. Continue opening and unloading casks per steps G, H, I, J and K until all fuel elements are stored in the dry storage rack.

M. Prepare TN-BGC1 containers for return.

Note: Empty casks may be prepared for return prior to completion of step L. Preparation of empty casks and loading of empty casks back onto the shipping vehicle may be performed in parallel with unloading/unpacking activities if authorized by the person in charge of the receipt evolution.

1. Perform Radiological Inspections of canisters/containers and record results on shipping paperwork.

2. Close canisters and containers per transport requirements.

   a. Replace the plug handling tool/shield plug assembly (weight 44.1 lb) on the shipping container.

   b. Install the bayonet ring/tightening ring assembly on the shield plug.

   c. Place the pressure tool on the shipping container and secure it by screwing its knurled knobs into the four lateral threaded holes in the body.

   d. Turn the thumb wheel on the hydraulic press tool to the fully CW position to close the bleed port.

   e. Using the pump handle, increase pressure as read on the installed pressure gauge, until pressure indicates 300 BAR.

   f. Clamp the bayonet ring completely and the tightening ring, by rotating the tap-bolt handles CW.

   g. Release the pressure on the plug by loosening (CCW direction) the thumb wheel on the pump body.

   h. When pressure has been reduced to approximately zero, as indicated on the installed pressure gauge, remove the pressure tool.

   i. Remove the two tightening-ring tap-bolt handles from the diametrically opposed threaded holes on the tightening ring.

3. Label containers as required per SHP requirements.

4. Transfer containers to transport vehicle.

N. Prepare transport vehicle for return to shipper.

Note: Shipping vehicle may be returned to the shipper without the empty casks if allowed by the shipping schedule.

1. Attach appropriate tags/placards to vehicle as required per SHP requirements.

2. Complete transportation forms.

FIGURE 1 - FUEL ELEMENT RECEIPT DATA

DATE:______________________ TIME:________________________

Fuel Element Serial Number:

Maximum radiation level (contact): ____________________(mr/hr)

____________________(elevation)

Maximum radiation level (1 foot): ____________________(mr/hr)

____________________(elevation)

Maximum surface contamination level: ____________________(DPM/smear)

Original length:

Standard________________________

Deviation_______________________

Temperature_____________________

Transverse bow: SAT / UNSAT

Visual inspection notes (use diagram next page if needed):

____

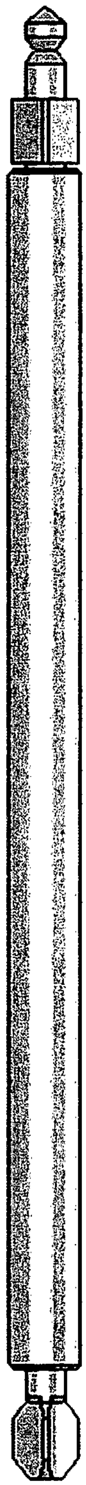
____

____

____

____

FIGURE 2 - FUEL ELEMENT DIAGRAM



Appendix C – Restart Procedure

OREGON STATE TRIGA REACTOR

OPERATING PROCEDURES

OSTROP 29

Reactor Re-Start with LEU 30/20 Fuel

Reprinted: 8/26/2008

Revision No. REV 0

APPROVED BY: ______________________

S. T. Keller

Reactor Administrator

Reactor Re-Start with LEU 30/20 Fuel

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OSTROP 29

Reactor Re-Start with LEU 30/20 Fuel

SCOPE

OSTROP 29 provides instructions for the initial startup and operation of the OSTR with new LEU 30/20 fuel. The procedure takes the OSTR from the fully defueled condition with no fueled elements in the reactor tank, through initial criticality and testing to routine full power operation.

I. INTRODUCTION

This procedure provides instruction for the initial startup and testing of the Oregon State TRIGA Reactor (OSTR). These instructions include prerequisites for startup, fuel element loading sequence, low power testing, high power testing and pulse mode testing. This procedure also provides instructions ensuring that all necessary instrument checks and calibrations required for reactor operation are performed.

The activities listed in this procedure incorporate minimum requirements specified in the Oregon State University (OSU) conversion SAR, Appendix A, Reload and Startup Guidelines (as amended). The acceptance criteria specified herein are taken from the OSU conversion SAR, Appendix B, Startup Acceptance Criteria (as amended).

During the re-start process, conditions may arise that require deviation from the written re-start procedure. These deviations shall be approved in advance by the Reactor Supervisor and the Reactor Administrator. These deviations will be documented by entries in the Reactor Supervisor log book and timely changes to this OSTROP in accordance with the requirements of OSTROP 6, Administrative and Personnel procedures.

If an acceptance criterion is not met, the reactor shall immediately be shut down. Reactor operation may be continued once the Reactor Supervisor and the Reactor Administrator agree on and implement (as appropriate) modifications to the reactor re-start procedure.

II. PREREQUISITES

All HEU items including fuel elements, instrumented fuel elements (IFE's), control rod followers and racked in-tank items have been removed from the reactor tank. The fission chamber, which contains HEU, is not to be removed.

________________________________ _______________

(Reactor Administrator) (Date)

All mechanical systems required for reactor operation, including control rod drives, a fuel handling tool, reactor water systems, gridplates, central thimble and CAMs and ARMs verified properly installed and/or operational.

(Reactor Supervisor) (Date)

The reactor Instrumentation and Control system required to support initial approach to criticality and low power operation verified operational.

(Scientific Instrument Technician) (Date)

All administrative requirements for reactor operation have been implemented.

(Reactor Administrator) (Date)

Neutron source installed.

(Reactor Supervisor) (Date)

Graphite reflector elements installed.

(Reactor Supervisor) (Date)

III. APPROACH TO CRITICAL AND INITIAL CONTROL ROD CALIBRATION

The criticality state of the new core shall be monitored using standard 1/M plots where M is the ratio of the count rate with n fuel elements in the core divided by the initial count rate. Startup count rate data shall be obtained using the installed fission chamber. The loading sequence is specified in Appendix B, Core Loading Sequence.

The initial countrate shall be obtained once the neutron source, three fuel-followed control rods and one air followed control rod are installed in the core and the B- and C-rings are fully loaded. This first initial countrate is expected to be very low. New "initial" countrates may be obtained and utilized for 1/M plots as directed by the Reactor Supervisor or Reactor Administrator.

Count rates for the 1/M plots shall be obtained with all four control rods fully withdrawn. Fueled components may only be added to the core when all four control rods are fully inserted. As fueled components are added to the core, maintain fuel movement records as required.

Note: Initial startup testing shall be performed with the core in the NORMAL configuration. The NORMAL configuration is expected to be the most limiting from the standpoint of shutdown margin. Additional testing and calibration shall be performed in the ICIT and CLICIT modes at the direction of the Reactor Supervisor once this procedure is completed for the NORMAL core configuration.

Fuel movements shall be performed in accordance with OSTROP 11, Fuel Element Handling Procedures. The licensed operator attending the console shall monitor flux levels during fuel movements as needed to detect significant changes in the effective multiplication factor.

Note: Steps III.A and III.B may be performed in either order.

A. Install Fuel Followed Control Rods (FFCRs) in positions C-10, D-1 and D-10.

B. Install Air Followed Control Rod (AFCR) in position C-4.

C. Verify satisfactory control rod motion and control rod scram times. It may be necessary to place the source close to the fission chamber to satisfy the technical specification minimum countrate interlock in order to withdraw control rods.

Note: Steps III.D and III.E may be performed in either order.

D. Install an IFE in position B-4.

E. Install fuel elements in the five remaining B-ring positions.

F. Install ten fuel elements in the vacant C-ring positions.

G. Withdraw all control rods, record the initial count rate and then insert all control rods.

H. Install ten fuel elements in positions D-2 through D-9 and positions E-6 and E-7. Obtain a countrate and update the 1/M plot.

I. Install ten fuel elements in positions D-11 through D-18 and positions E-18 and E-19. Obtain a countrate and update the 1/M plot.

J. Continue adding fuel to the E-ring, and then to the F-ring when the E-ring is full, per the loading sequence in Appendix B. When adding fuel, observe the following requirements:

- After filling the E-ring and before adding fuel to the F-ring, re-verify satisfactory control rod motion and control rod scram times.
- Add fuel elements in groups of no more than four fuel elements at a time until the 1/M plot indicates that the reactor is sub-critical by about ten fuel elements (with all rods withdrawn).
- Add fuel elements in groups of no more than two fuel elements at a time until the 1/M plot indicates that the reactor is sub-critical by about five fuel elements (with all rods withdrawn).
- Add fuel elements one at a time until the 1/M plot indicates that the reactor will be supercritical (with all rods withdrawn) after addition of the next fuel element.
- Obtain a count rate and update the 1/M plot after each group addition of fuel elements.

K. Add a fuel element and verify the reactor is supercritical with all control rods withdrawn. Determine the core excess with all control rods withdrawn by measuring reactor period with all rods withdrawn and using Appendix A, Reactivity Calculations, to determine the corresponding reactivity.

L. Perform a rough calculation of Integrated Rod Worth using the rod drop method as specified in Appendix A, Reactivity Calculations. Generate rough rod worth curves using the method specified in Appendix A.

M. Add fuel elements in groups of four or less. After the addition of each group, perform a rough calibration of the control rods using the rod drop method. Calculate estimated NRC shutdown margin (assuming the highest worth rod is fully withdrawn) and estimated core excess after the addition of each group. Continue loading the core until the estimated NRC shutdown margin is no less than $1.00.

N. Estimate core excess for the core configuration established in step III.M using critical rod height data and the rough rod worth curves generated above. Verify that core excess satisfies Technical Specification requirements.

O. Verify that the core configuration established in step III.M can be taken critical with the most reactive rod fully inserted.

IV. LOW AND INTERMEDIATE POWER TESTING

A. Calibrate all four control rods using the period method as specified in OSTROP 9, Control Rod Calibration Procedures. Calibration is required for only the NORMAL core configuration.

B. Measure the reactivity worth of a representative sample of fuel elements in the F-ring. Adjust the number of fuel elements in the F-Ring to achieve the operational core which has the desired excess reactivity and shutdown margin conditions.

C. If core configuration is changed from step IV.A, then re-calibrate all control rods using the period method. Re-calculate excess reactivity and shutdown margin.

D. Measure the reactivity worth of at least two elements per ring. Measurements taken in step IV.B may be used for F-ring reactivity measurements.

E. Calibrate the fuel element temperature measurement channel (if not already calibrated).

F. Perform channel tests of all safety channels and interlocks listed in the Technical Specifications which can be performed without operating at full power. OSTROP 15, Semi-Annual Surveillance and Maintenance Procedures may be used for guidance.

G. Perform a power increase to 500 kW in steps of 100 kW. Allow power to stabilize at each step. Record all flux channel indications and fuel temperature indications at each step.

H. Perform a power calibration at or near 500 kW per OSTROP 8, Reactor Power Calibration Procedures. This power calibration shall be performed at a maximum IFE temperature not to exceed 200°C.

V. HIGH POWER TESTING

A. Perform a power increase to 900 kW in steps of 100 kW. Allow power to stabilize at each step. Record all flux channel indications and fuel temperature indications at each step.

B. Perform a power calibration at or near 900 kW. Use available power and temperature indications to ensure that reactor power does not significantly exceed 900 kW.

C. Perform a power calibration at or near 1000 kW. After completion of the Power Calibration, including instrument adjustment, the linear, safety and percent power channels shall indicate between 99.0 and 101.0 percent at full reactor power of 1.0 MW.

D. Perform channel tests of the remainder of all safety channels and interlocks (those not completed in step IV.F) listed in the Technical Specifications.

E. Calculate the power coefficient of reactivity and fuel element temperature coefficient of reactivity.

F. Perform a continuous run of at least 12 hours at full power to observe and record operating characteristics. During this time, verify that a comprehensive radiation survey of the D-wing is performed.

G. After the initial extended run is complete, shut down the reactor. Restart the reactor when conditions will allow at least a six hour run. Repeat the stepwise increase to power as described in step IV.G, but increase power to 1000 kW in steps of 100 kW. Allow power to stabilize at each step. Record all flux channel indications and fuel temperature indications at each step.

VI. ADDITIONAL TESTING

A. Perform pulse mode operational tests. Pulse mode operation tests shall consist of, at a minimum, a sequence of pulses starting at about $1.10 and proceeding in $0.10 increments until the maximum pulse reactivity insertion limit is reached. Peak power, temperature and integrated power data shall be recorded from console instruments. Graphs of peak temperature vs. prompt reactivity, integrated power vs. prompt reactivity and peak power vs. (prompt reactivity)^2 shall be constructed.

B. Verify operability of square wave capability by performing several square waves to intermediate power levels.

C. Perform ICIT and CLICIT configuration testing as directed by the Reactor Supervisor.

D. Perform flux mapping of as many irradiation facilities as practical. When possible, flux mapping should determine both thermal and epithermal flux distributions.

APPENDIX A: REACTIVITY CALCULATIONS

AI. Period vs. Reactivity in the LEU BOL NORMAL core

In section 4.5.2 of the OSU Conversion SAR, prompt neutron lifetime is calculated to be 22.6 +/- 2.9 µsec for the LEU 30/20 BOL core. The total effective delayed neutron fraction is also calculated to be 0.0075. The Inhour Equation can be written

$$\rho(s) = \frac{sl}{sl+1} + \frac{1}{sl+1}\sum_{i=1}^{6}\frac{s\beta_i}{s+\lambda_i}$$

This equation has seven roots. In a core with positive reactivity, only one of the roots will be positive, and this root governs the long term (non-transient) kinetic behavior of the core. Steady state reactor period is equal to the inverse of the positive root. For fresh LEU fuel, the constants $\beta_1 - \beta_6$ and $\lambda_1 - \lambda_6$ are equal to the values for U-235 (with the β_i normalized to $\beta = 0.0075$). These values are shown in Table A-1, Inhour coefficients (new LEU Fuel). Figure A-1, reactivity vs. period (new LEU Fuel), was constructed by finding the positive root of the Inhour equation for a given positive reactivity using the constants from Table A-1.

Table A-1, Inhour coefficients (new LEU Fuel)

l = 22.6 µsec	
β_1 =2.85E-4	λ_1 = 1.27E-2 (sec^{-1})
β_2 = 1.60E-3	λ_2 =3.17E-2 (sec^{-1})
β_3 = 1.41E-3	λ_3 = 1.16E-1 (sec^{-1})
β_4 = 3.05E-3	λ_4 =3.11E-1 (sec^{-1})
β_5 =9.60E-4	λ_5 = 1.04E0 (sec^{-1})
β_6 = 1.95E-4	λ_6 =3.87E0 (sec^{-1})

To estimate the reactivity state of a supercritical core configuration, the reactor period shall be measured by any method approved by the Reactor Supervisor or Reactor Administrator. Figure A-1 can then be used to estimate core reactivity. In the event that an exceptionally long reactor period is measured (i.e. greater than 100 sec), it may be necessary to generate additional solutions using the Inhour equation.

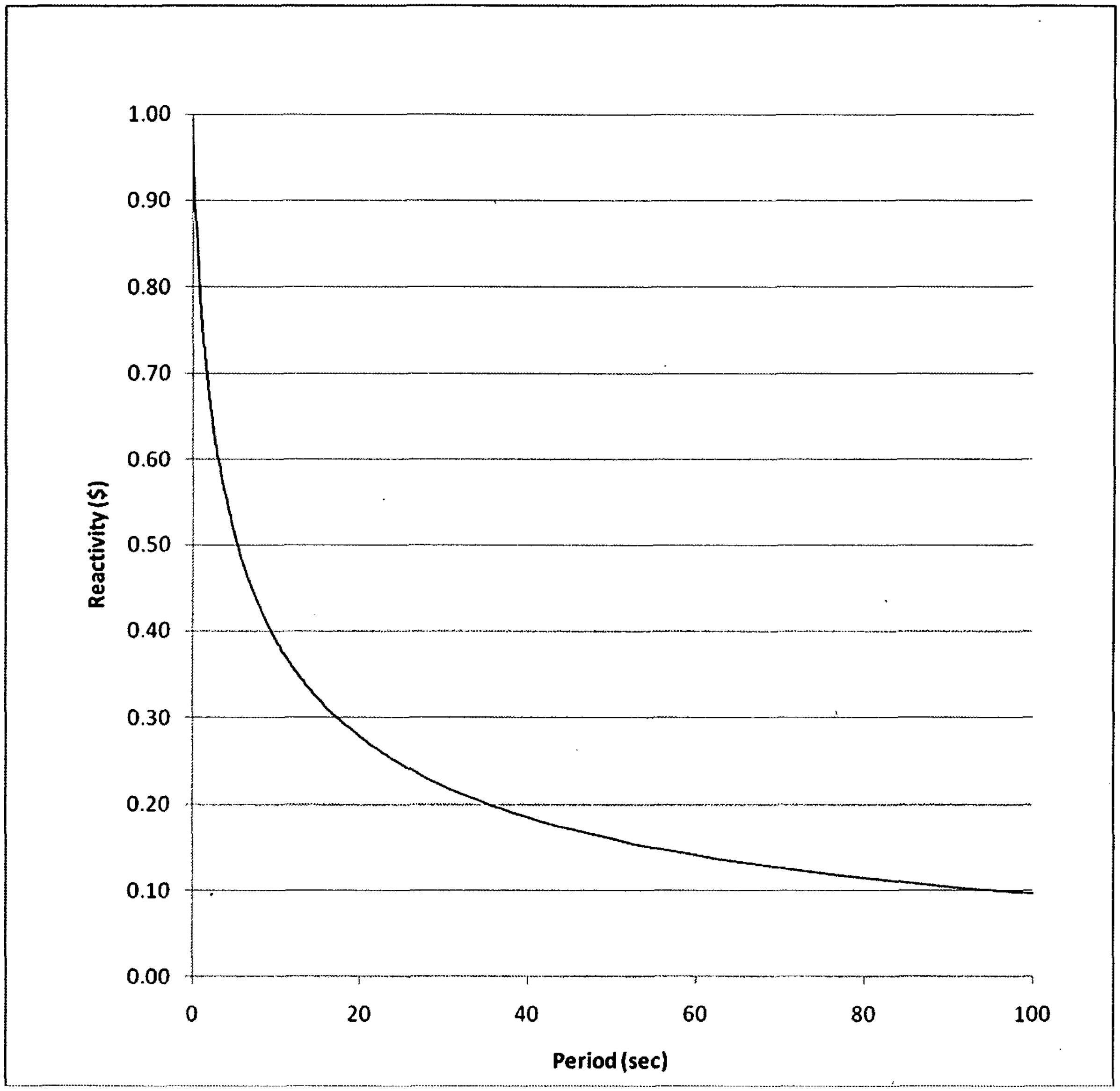

Figure A-1, reactivity vs. period (new LEU Fuel)

AII. Estimation of Integrated Rod Worth by the Rod Drop Method

When the period method cannot be used, integrated rod worth (IRW) may be calculated using Rod Drop methods. A so-called 'integral' method may be used to estimate the total IRW of a control rod, and then a typical rod worth curve shape can be fit to the IRW to produce a differential rod worth curve. These methods allow generation of best-estimate control rod calibration curves. These methods should not be used when it is possible to generate rod worth curves using the historically accepted period method of OSTROP 9.

It can be shown that following a rod drop from a critical configuration, the reactivity inserted can be calculated using the formula

$$\rho = \frac{-P_0(\Lambda^* + 1/\lambda)}{\int_0^{\infty} P(\tau)d\tau} \qquad (1)$$

where

ρ = reactivity (dollars)

P_0 = reactor power prior to the rod drop

$\Lambda^* = l/k\beta_{eff}$ = 3.06E-03 for the LEU beginning of life core

λ = one group delayed neutron precursor decay constant = 0.079 sec-1

t_0 = time at which the test rod is fully inserted

To estimate total IRW and generate a differential rod worth curve for a given control rod, proceed as follows:

- Configure the control room data computer to record the Log Power signal at a frequency of 100 samples per second for at least 6 minutes.
- Stabilize reactor power at 1000W with the test rod fully withdrawn.
- Commence data recording and immediately scram the test rod.
- Estimate IRW of the dropped rod using the equation (1). To evaluate the integral, use t_0 = time of initial power decrease plus rod drop time. Perform a numerical integration from t_0 to t_0 + 300 sec.
- Generate a best estimate control rod calibration curve using Table A-2, Rod Worth vs. Position. The position column refers to percentage of rod withdrawal. The worth column refers to fraction of total IRW.

Table A-2, Rod Worth vs. Position

Position	Worth	Position	Worth	Position	Worth	Position	Worth
1	0.002	26	0.223	51	0.586	76	0.890
2	0.005	27	0.237	52	0.600	77	0.899
3	0.009	28	0.250	53	0.615	78	0.908
4	0.013	29	0.264	54	0.629	79	0.916
5	0.018	30	0.278	55	0.643	80	0.923
6	0.024	31	0.292	56	0.657	81	0.931
7	0.029	32	0.306	57	0.671	82	0.938
8	0.036	33	0.320	58	0.684	83	0.945
9	0.043	34	0.335	59	0.698	84	0.951
10	0.050	35	0.349	60	0.711	85	0.957
11	0.058	36	0.364	61	0.724	86	0.963
12	0.067	37	0.379	62	0.737	87	0.968
13	0.076	38	0.394	63	0.749	88	0.973
14	0.085	39	0.408	64	0.762	89	0.977
15	0.095	40	0.423	65	0.774	90	0.982
16	0.105	41	0.438	66	0.786	91	0.985
17	0.115	42	0.453	67	0.797	92	0.989
18	0.126	43	0.468	68	0.809	93	0.991
19	0.137	44	0.483	69	0.820	94	0.994
20	0.149	45	0.498	70	0.831	95	0.996
21	0.160	46	0.513	71	0.842	96	0.998
22	0.172	47	0.527	72	0.852	97	0.999
23	0.185	48	0.542	73	0.862	98	1.000
24	0.197	49	0.557	74	0.872	99	1.000
25	0.210	50	0.572	75	0.881	100	1.000

APPENDIX B: CORE LOADING SEQUENCE

The following steps specify the core loading sequence. Each numbered step must be completed before the next step is commenced. Substeps may be completed in any order within the step. Two qualified operators shall initial for the completion of each step.

1. Establish initial configuration:

   - Install a neutron source in position G-17.
   - Verify the pneumatic rabbit is installed in position G-2.
   - Install graphite elements in all remaining G-Ring positions.

   Completed ______________ (Date) __________ (initial) __________ (initial)

2. Install control rods:

   - Install an air followed control rod in position C-4 (transient rod).
   - Install Fuel Followed Control Rod 11632 in position C-10 (regulating rod).
   - Install Fuel Followed Control Rod 11633 in position D-1 (safety rod).
   - Install Fuel Followed Control Rod 11634 in position D-10 (shim rod).

   Completed ______________ (Date) __________ (initial) __________ (initial)

3. Install fuel in the B-ring:
   - Install FE 11601 in position B-1.
   - Install FE 11600 in position B-2.
   - Install FE 11573 in position B-3.
   - Install IFE 11630 in position B-4.
   - Install FE 11617 in position B-5.
   - Install FE 11574 in position B-6.

   Completed ______________ (Date) __________ (initial) __________ (initial)

4. Install fuel in the C-ring:
   - Install FE 11619 in position C-1.
   - Install FE 11618 in position C-2.
   - Install FE 11616 in position C-3.
   - Install FE 11615 in position C-5.
   - Install FE 11562 in position C-6.

- Install FE 11561 in position C-7.
- Install FE 11560 in position C-8.
- Install FE 11572 in position C-9.
- Install FE 11570 in position C-11.
- Install FE 11599 in position C-12.

Completed ________________ ____________ ____________
(Date) (initial) (initial)

5. Install fuel in positions D-2 through D-9 and positions E-6 and E-7:
   - Install FE 11596 in position D-2.
   - Install FE 11571 in position D-3.
   - Install FE 11595 in position D-4.
   - Install FE 11597 in position D-5.
   - Install FE 11598 in position D-6.
   - Install FE 11594 in position D-7.
   - Install FE 11557 in position D-8.
   - Install FE 11559 in position D-9.
   - Install FE 11551 in position E-6.
   - Install FE 11549 in position E-7.

Completed ________________ ____________ ____________
(Date) (initial) (initial)

6. Install fuel in positions D-11 through D-18 and positions E-18 and E-19:
   - Install FE 11558 in position D-11.
   - Install FE 11613 in position D-12.
   - Install FE 11611 in position D-13.
   - Install FE 11612 in position D-14.
   - Install FE 11606 in position D-15.
   - Install FE 11608 in position D-16.
   - Install FE 11610 in position D-17.
   - Install FE 11604 in position D-18.
   - Install FE 11544 in position E-18.
   - Install FE 11582 in position E-19.

Completed ________________ ____________ ____________
(Date) (initial) (initial)

7. Install fuel element 11609 in position E-1.

Completed ________________ ____________ ____________
(Date) (initial) (initial)

8. Install fuel element 11555 in position E-2.

Completed ______________ ________ ________
(Date) (initial) (initial)

9. Install fuel element 11605 in position E-3.

Completed ______________ ________ ________
(Date) (initial) (initial)

10. Install fuel element 11603 in position E-4.

Completed ______________ ________ ________
(Date) (initial) (initial)

11. Install fuel element 11550 in position E-5.

Completed ______________ ________ ________
(Date) (initial) (initial)

12. Install fuel element 11548 in position E-8.

Completed ______________ ________ ________
(Date) (initial) (initial)

13. Install fuel element 11541 in position E-9.

Completed ______________ ________ ________
(Date) (initial) (initial)

14. Install fuel element 11543 in position E-10.

Completed ______________ ________ ________
(Date) (initial) (initial)

15. Install fuel element 11545 in position E-11.

Completed ______________ ________ ________
(Date) (initial) (initial)

16. Install fuel element 11576 in position E-12.

Completed ______________ __________ __________
(Date) (initial) (initial)

17. Install fuel element 11547 in position E-13.

Completed ______________ __________ __________
(Date) (initial) (initial)

18. Install fuel element 11583 in position E-14.

Completed ______________ __________ __________
(Date) (initial) (initial)

19. Install fuel element 11546 in position E-15.

Completed ______________ __________ __________
(Date) (initial) (initial)

20. Install fuel element 11581 in position E-16.

Completed ______________ __________ __________
(Date) (initial) (initial)

21. Install fuel element 11579 in position E-17.

Completed ______________ __________ __________
(Date) (initial) (initial)

22. Install fuel element 11542 in position E-20.

Completed ______________ __________ __________
(Date) (initial) (initial)

23. Install fuel element 11577 in position E-21.

Completed ____________ __________ __________
(Date) (initial) (initial)

24. Install fuel element 11580 in position E-22.

Completed ____________ __________ __________
(Date) (initial) (initial)

25. Install fuel element 11575 in position E-23.

Completed ____________ __________ __________
(Date) (initial) (initial)

26. Install fuel element 11540 in position E-24.

Completed ____________ __________ __________
(Date) (initial) (initial)

27. Install fuel element 11588 in position F-1.

Completed ____________ __________ __________
(Date) (initial) (initial)

28. Install fuel element 11586 in position F-2.

Completed ____________ __________ __________
(Date) (initial) (initial)

29. Install fuel element 11589 in position F-3.

Completed ____________ __________ __________
(Date) (initial) (initial)

30. Install fuel element 11591 in position F-4.

Completed ______________ ____________ ____________
(Date) (initial) (initial)

31. Install fuel element 11590 in position F-5.

Completed ______________ ____________ ____________
(Date) (initial) (initial)

32. Install fuel element 11587 in position F-6.

Completed ______________ ____________ ____________
(Date) (initial) (initial)

33. Install fuel element 11592 in position F-7.

Completed ______________ ____________ ____________
(Date) (initial) (initial)

34. Install fuel element 11593 in position F-8.

Completed ______________ ____________ ____________
(Date) (initial) (initial)

35. Install fuel element 11602 in position F-9.

Completed ______________ ____________ ____________
(Date) (initial) (initial)

36. Install fuel element 11614 in position F-10.

Completed ______________ ____________ ____________
(Date) (initial) (initial)

37. Install fuel element 11556 in position F-11.

Completed ________________ (Date) __________ (initial) __________ (initial)

38. Install fuel element 11564 in position F-12.

Completed ________________ (Date) __________ (initial) __________ (initial)

39. Install fuel element 11563 in position F-13.

Completed ________________ (Date) __________ (initial) __________ (initial)

40. Install fuel element 11584 in position F-14.

Completed ________________ (Date) __________ (initial) __________ (initial)

41. Install fuel element 11585 in position F-15.

Completed ________________ (Date) __________ (initial) __________ (initial)

42. Install fuel element 11578 in position F-16.

Completed ________________ (Date) __________ (initial) __________ (initial)

43. Install fuel element 11566 in position F-17.

Completed ________________ (Date) __________ (initial) __________ (initial)

44. Install fuel element 11568 in position F-18.

Completed ______________ ________ ________
(Date) (initial) (initial)

45. Install fuel element 11569 in position F-19.

Completed ______________ ________ ________
(Date) (initial) (initial)

46. Install fuel element 11553 in position F-20.

Completed ______________ ________ ________
(Date) (initial) (initial)

47. Install fuel element 11552 in position F-21.

Completed ______________ ________ ________
(Date) (initial) (initial)

48. Install fuel element 11565 in position F-22.

Completed ______________ ________ ________
(Date) (initial) (initial)

49. Install fuel element 11567 in position F-23.

Completed ______________ ________ ________
(Date) (initial) (initial)

50. Install fuel element 11554 in position F-24.

Completed ______________ ________ ________
(Date) (initial) (initial)

51. Install fuel element 11623 in position F-25.

Completed ______________ ____________ ____________
(Date) (initial) (initial)

52. Install fuel element 11621 in position F-26.

Completed ______________ ____________ ____________
(Date) (initial) (initial)

53. Install fuel element 11622 in position F-27.

Completed ______________ ____________ ____________
(Date) (initial) (initial)

54. Install fuel element 11620 in position F-28.

Completed ______________ ____________ ____________
(Date) (initial) (initial)

55. Install fuel element 11625 in position F-29.

Completed ______________ ____________ ____________
(Date) (initial) (initial)

56. Install fuel element 11627 in position F-30.

Completed ______________ ____________ ____________
(Date) (initial) (initial)