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GOING FROM HEU TO LEU: CONVERSION OF THE OREGON STATE TRIGA® REACTOR

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ABSTRACT

On September 29, 2008, the Oregon State TRIGA® reactor was converted from 70 percent enriched fuel to 20 percent enriched fuel. The conversion proceeded in accordance with the domestic reactor conversion subprogram of the Global Threat Reduction Initiative program. Major conversion activities are discussed in this paper. These activities consisted of: preparation of the Conversion Safety Analysis Report, receipt of new fuel, removal and storage of old fuel, LEU core loading and initial criticality, startup testing and shipment of spent HEU fuel. LEU core measurements indicate that fast and epithermal fluxes have increased slightly and thermal flux has decrease about 30 percent. Improved core geometry compensates for thermal flux reduction. Cooperation between government agencies, the fuel manufacturer and the licensee proved fundamental to the success of the project.

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. The first LEU fuel element was placed in the core on September 29, 2008. Initial criticality with LEU fuel was achieved at 1545 on October 7, 2008. The successful conversion represented the culmination of almost three years of intensive work by many people. Conversion of the OSTR was accomplished via coordination of work efforts at OSU, CERCA / General Atomics, the Department of Energy (Argonne National Laboratory and Idaho National Laboratory) and the Nuclear Regulatory Commission.

Major activities required to complete conversion of the OSTR are summarized in Table 1. Analyses for the Conversion Safety Analysis Report (CSAR) were performed using principally MCNP5[1] and RELAP5-3D[2]. These analyses are discussed briefly in the section titled *The Conversion Safety Analysis Report*. The remainder of the conversion activities are discussed in the section titled *Conversion of the OSTR*.

Table 1, Major Conversion Activities

September 2007	CSAR neutronic analysis complete
October 2007	CSAR steady state thermal hydraulic analysis complete
November 2007	CSAR preliminary transient (pulse) thermal hydraulic analysis complete
May 2008	Develop LEU fuel receipt procedure and fabricate storage facility
June 20, 2008	Final set of answers to Request for Additional Information (RAI) questions submitted to NRC
June 27, 2008	Receive LEU fuel
June 30, 2008	Receive spent HEU fuel storage rack
August 26, 2008	Restart procedure approved
September 4, 2008	NRC issues conversion order (21 day comment period begins)
September 9, 2008	All fuel, reflectors and experimental facilities removed from core
September 29, 2008	Conversion milestone accomplished
October 7, 2008	LEU Fuel Initial Criticality
December 1, 2008	Startup and Acceptance Testing Complete

The OSTR is a TRIGA® Mark II reactor which functions as a highly flexible research tool. It 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. Other than a four month conversion outage, the research capabilities of the OSTR have not been significantly impacted, although the expected lifetime of the core has been extended by at least 30 years.

THE CONVERSION SAFETY ANALYSIS REPORT

A high fidelity MCNP model of the OSTR core and surrounding structures was developed in 1997[3]. This model was revised and thoroughly reviewed for use in CSAR analyses. The model was used to calculate reactor parameters such as control rod worth, effective delayed neutron fraction, prompt neutron lifetime and prompt temperature reactivity coefficient. The model was also used to calculate power distribution throughout the core, and this information was used to establish peaking factors. RELAP was run using power distribution information generated from MCNP to analyze steady state thermal hydraulic conditions. RELAP was also used to analyze transient behavior of the reactor during pulse conditions.

The MCNP5 model and calculations

To demonstrate the capability of MCNP to accurately predict core neutronic parameters, the MCNP model was modified to simulate core conditions present during HEU core start-up testing. The initial critical core and the initial full power operational core were simulated. The calculational results of the model were shown to compare favorably with experimentally determined values. To establish model fidelity, 38 critical configurations were taken from initial HEU control rod calibration testing. These configurations were modeled with MCNP and found to have an average bias of +$0.45 +/- $0.10. All reported reactivity values are adjusted to reflect this bias[4].

At the OSTR, control rods are calibrated by incrementally withdrawing the test rod and establishing a positive stable period. The corresponding reactivity insertion can be calculated from the measured period. The rod calibration method was simulated using MCNP by creating a series of models with control rods at elevations established during control rod calibration testing. Calculated and measured rod worth was in good agreement. The value of core excess (core reactivity with all control rods withdrawn) was calculated to be $7.10 at beginning-of-life (BOL). The measured BOL value was $7.17.

In order to calculate core properties at times other than BOL, MCNP was linked with REBUS in order to perform depletion calculations[5]. Based upon the depletion analysis, middle-of-life (MOL) was determined to be at 1800 MWd which is the point that core excess reactivity reaches a maximum. End-of-life (EOL) for the HEU core was determined to be 3800 MWd which is the point that excess reactivity falls below $0.50. Once BOL, MOL and EOL timeframes had been established, parameters could be calculated for each core state. Parameters important to reactor safety were calculated and compared with measured values. All were found to be in good agreement.

The prompt-temperature coefficient associated with the HEU fuel, α_F, was calculated by varying the fuel temperature while leaving other core parameters fixed. The MCNP5 model was used to simulate the reactor with all rods out at 300, 400, 600, 800, and 1200 K. The results are shown in Figure 1. The prompt-temperature coefficient is observed to be a linear function over the given temperature range, and the magnitude of the calculated BOL and EOL prompt-temperature coefficients compare favorably with those of Simnad *et. al.*[6].

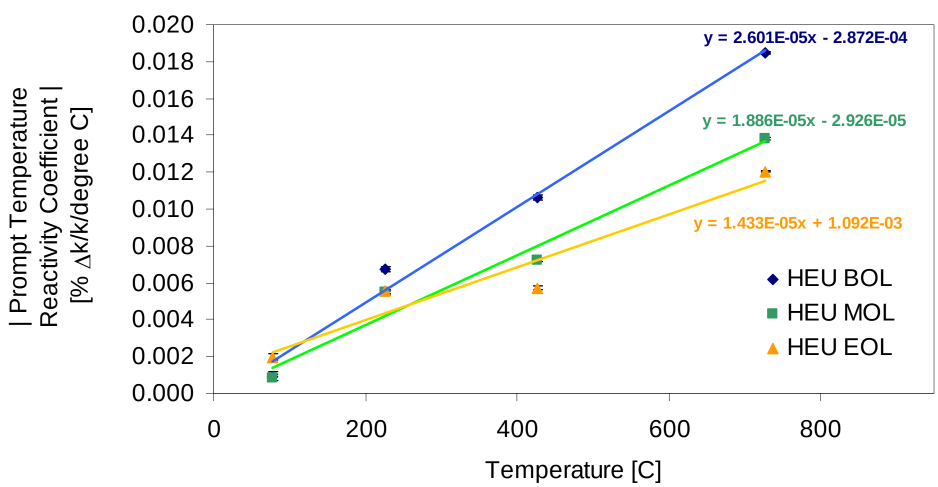

Figure 1, Magnitude of HEU Prompt Temperature Coefficient

The void coefficient of reactivity was also determined using the MCNP model. A core average value and a core center value were calculated. The calculated core center value was found to be in good agreement with measurements taken near the core center. A core average value has not been measured.

Power distributions within each rod throughout the core were calculated using MCNP. These distributions were used to calculate peak factors, and also the hot channel power used in RELAP. The following peak factors were calculated:

- Hot Channel Peak Factor = (maximum fuel rod power)/(core average fuel rod power)
- Hot channel Fuel Axial Peak Factor = (maximum axial power in the hot rod)/(average axial power in the hot rod)
- Hot Channel Fuel Radial Peak Factor = (maximum radial power in the hot rod)/(average radial power in the hot rod)

The 'Effective Peak Factor' is the product of these three individual peak factors.

This entire process was repeated to simulate the LEU core for CSAR analysis. A significant part of the startup and acceptance testing involved measurement of predicted LEU parameters. A summary of these results can be found in the OSTR Startup Report[7].

RELAP5-3D steady state calculations

RELAP was used to calculate steady state and transient thermal-hydraulic conditions in the OSTR. Steady state calculations were performed for the hot channel assuming the reactor was operating at maximum license power of 1.1 MW with a maximum license pool temperature of 49°C. Geometry and heat structure information was included in the input deck. To establish hot channel conditions, it was conservatively assumed that the maximum power channel had the minimum flow area and there was no cross flow between adjacent channels. The Bernath[8] correlation and the 2006 Groeneveld[9] critical heat flux tables were used to determine the Departure from Nucleate Boiling Ratio (DNBR). Steady state parameters predicted by the code include channel flow rate, axial fuel centerline temperature distribution, axial clad temperature distribution, axial bulk coolant temperature distribution and axial DNBR.

Behavior of the HEU core was modeled at BOL, MOL and EOL. The LEU core was also modeled for these states. The LEU core was also modeled in three different operating modes. A 'normal' core contains no central experimental facility. An In-Core-Irradiation-Tube (ICIT) core contains a hollow aluminum experimental facility near core center. A Cadmium-Lined-In-Core-Irradiation-Tube (CLICIT) core contains a cadmium lined hollow aluminum experimental facility near core center. Each configuration has different peaking factors at different times in core life. It was found that the ICIT core has the highest effective peak factor at all times in core life. The minimum departure from nucleate boiling ratio (DNBR) for any core was found to occur in the ICIT core at MOL. Using the most conservative correlation (Bernath), the minimum DNBR was calculated to be 2.06. Although the NRC has no established minimum DNBR limit for natural circulation research reactors, this value was deemed acceptable.

RELAP5-3D transient calculations

RELAP was used to calculate thermal hydraulic conditions in the OSTR under the transient conditions which occur during a reactor pulse. The behavior of the reactor kinetics module built into RELAP was validated by comparison against an OSU kinetics code and a General Atomics code known as BLOOST. RELAP was used to predict peak pulse power, total energy released during the pulse, peak temperature experienced during the pulse and temperature distribution within the hottest fuel element as a function of time during the pulse. A two channel RELAP model was used to simulate core transient behavior. One channel represents the core averaged parameters. The hot rod is represented as a single separate channel which accounts for radial, axial, and hot channel peak factors.

It was found that for a given pulse reactivity insertion, the maximum pulse (power, temperature and energy) occurs in the LEU ICIT core at middle-of-life. The suggested pulse fuel temperature limit is 830°C. By limiting pulse reactivity insertion to ≤ $2.30, maximum fuel temperature is limited to ≤ 819°C.

CONVERSION OF THE OSTR

Core design was an early part of the conversion process. General Atomics has historically provided '30/20' TRIGA® fuel which contains 30 weight percent of 19.75 percent enriched uranium in a zirconium hydride ceramic matrix. TRIGA® fuel may also contain a small amount of erbium which is used as a burnable poison and also contributes to the magnitude of the prompt temperature coefficient. The amount of erbium in 30/20 TRIGA® fuel is normally 0.9 weight percent, but OSU requested fuel containing 1.1 weight percent in order to prolong core life and to provide a larger core. A larger core was desired in order to enhance geometric coupling between the core and irradiation facilities which are located on all sides of the core. The manufacture of 1.1 weight percent erbium fuel proved problematic, but fabrication difficulties were overcome and a very high quality product was produced by CERCA / General Atomics and delivered to OSU.

LEU Fuel Receipt

The CSAR was submitted to the NRC in November of 2007. OSU was granted permission to possess, but not to use the LEU fuel in April of 2008. Fuel was sent from France in four shipments with all fuel delivered to OSU by the end of June 2008. OSU received a total of 90 standard fuel elements, four fuel followed control rods (FFCR's) and two instrumented fuel elements (IFE's).

To optimize manpower and meet TN-BGC1 cask availability requirements, fuel was unpacked in two batches. A certified cask handler provided by DOE performed all cask manipulations. Although the fuel underwent extensive quality control inspection at the manufacturing facility, a receipt inspection was performed by OSU prior to acceptance to validate the results of factory inspection and also to verify that no damage had occurred during shipping. The minor imperfections noted during the receipt inspection matched QC records from the factory. After inspection, each element was placed in a secure storage facility. The LEU fuel storage facility is shown in Figure 2.

Figure 2, LEU Fuel Storage Facility

HEU Core Offload

Final operation with HEU fuel occurred on August 4, 2008. The fuel was then allowed to cool for a month before it was transferred to the Bulk Shield Tank, a water filled storage pool adjacent to the reactor pool. Fuel elements were transferred one at a time using the Single Element Transfer Cask, with one complete trip taking 8-10 minutes. Dose rates on the side of the cask were ~1 Rem/Hr.

Removal of fuel elements was straightforward, but the FFCR's and the IFE's had to be shortened in order to fit into the NAC-LWT Cask shipping baskets. A hydraulic snipper was used for initial rough trimming to minimize extremity dose. After the FFCR's and IFE's had cooled an additional four months, they were further shortened to the required length using a reciprocating saw. Handling loops were attached to the IFE stalks using stainless steel hose clamps. Due to very tight spacing requirements in the shipping baskets, an alternate method was used with the FFCR's. To attach handling loops to the FFCR's, a hole was drilled in the top of the connector. A swaged loop of cable was then attached with a washer and a screw. This arrangement is shown in Figure 3.

Figure 3, Trimmed FFCR with Attached Handling Loop

Removal of the graphite reflector elements proved to be the most challenging aspect of defueling. The aluminum clad graphite reflectors had been in-core since 1967 and had undergone significant amounts of concentric swell and also some cracking, as can be seen in Figure 4. All reflectors in the core by 2008 had swollen to such a degree that they no longer passed through the upper gridplate. Removal of the upper gridplate was thus necessitated. Reflector elements were then removed from the core using manually operated snare tools, a tedious operation requiring large amounts of patience and manual dexterity.


Figure 4, Cracked Reflector Element

LEU Restart

The first LEU fuel element was installed in the core on September 29, 2009, thus meeting the conversion milestone. Reactor startup then proceeded in a systematic manner as directed by the restart procedure. New stainless steel clad reflectors were installed, followed by a neutron source and the four control rods.

A standard approach to criticality was performed by adding fuel in batches and using a 1/M plot to monitor reactor state and predict when criticality would be achieved. Control rods were calibrated as soon as a critical configuration was established. Control rod calibrations were repeated as more fuel was added to obtain an operational core configuration. Once the operational configuration was established, a calorimetric calibration was performed in order to verify accuracy of nuclear instrument readings. After the calorimetric calibration was completed, testing of the reactor in square wave and pulse mode was performed.

During pulse mode operation, a step change in reactivity greater than \$1.00 is inserted in order to produce a large magnitude, short duration transient. In square wave mode, a step change in reactivity less than \$1.00 is inserted to rapidly increase from low power to some intermediate steady state operating power. All initial core testing was done in the normal core configuration. All significant tests were repeated in the ICIT and CLICIT core configurations. The final step before releasing the reactor for general customer use was to perform flux mapping of all experiment facilities. Startup and acceptance testing were completed by the end of November 2008. As shown in Table 2 and Table 3, agreement between predicted and measured parameters was generally good.

Critical Configuration: MCNP modeling indicated that the core would be critical after the addition of the 69th fuel element. LEU fuel erbium content was specified as 1.1 weight percent, but erbium content per fuel element varied from 1.02 to 1.15 weight percent. Low erbium elements were preferentially loaded near core center, thus reducing the expected minimum critical number. Initial criticality was obtained with 66 fuel elements.

Control Rod calibration: Once the operational core was established, there was sufficient excess reactivity to use the standard control rod calibration method. Agreement between predicted and measured rod worth is shown in Table 2.

Table 2, Summary of LEU BOL Integrated Rod Worth

Control Rod	Measured Rod Worth [$]	MCNP5 Predicted Rod Worth [$]
Shim Rod	2.76 +/- 0.34	2.55 +/- 0.16
Safety Rod	2.66 +/- 0.33	2.60 +/- 0.16
Regulating Rod	3.71 +/- 0.52	3.36 +/- 0.19
Transient Rod	2.86 +/- 0.35	2.86 +/- 0.15
Sum of all Rods	11.99 +/- 0.78	11.37 +/- 0.33

Reactor Physics Parameters: Measured values of significant reactor parameters were in reasonable agreement with predicted values. Results are shown in Table 3.

Table 3, Summary of Reactor Physics Measurements

Parameter	LEU Predicted	LEU Measured
Fuel Temperature Coefficient [¢/ºC]	-0.59 +/- 0.08	-0.53 +/- 0.02
Pool Temperature Coefficient [¢/ºC]	-0.72 +/- 0.08	-0.40 +/- 0.04
Effective Delayed Neutron Fraction	0.0076 +/- 0.0001	0.0080 +/- 0.0004
Neutron Generation Time [μsec]	22.6 +/- 2.9	25.6 +/- 2.6
β/l [sec]	336 +/- 43	312 +/- 35
Power Defect [$]	2.16 +/- 0.25	2.41 +/- 0.10

Flux Distribution: At BOL, HEU fuel contains the same number density of U-235 as LEU fuel. After 30 years, the U-235 content of the average HEU fuel element has been reduced by 14%, and some Pu has been generated. At full power, the fission rate is the same in either core, so except for some spectral hardening due to the increased U-238 content and the resultant ZrH reduction, fast flux was expected to remain largely unchanged following conversion. Since there is more U-235 in BOL LEU fuel than MOL HEU fuel, thermal flux was expected to decrease in the new core. As shown in Table 4, these changes were exhibited in all experimental facilities except for the pneumatic rabbit where fast flux experienced a significant reduction. This can be attributed to additional water holes in the vicinity of the rabbit terminus which enhance thermalization.

Table 4, Peak Fluxes in the HEU MOL and LEU BOL 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

HEU Spent Fuel Shipment

The final step of the conversion process was removal of the HEU fuel from the facility. The Required Shipping Documents (RSD's) and shipment paperwork were prepared by OSU with the assistance of Secured Transportation Services (STS). Cask loading and shipping was performed primarily by NAC International. Fuel remained the responsibility of OSU until it was received by INL.

Spent HEU fuel elements were loaded into baskets which held 24 elements. Full baskets were placed in the inner shield and moved to the reactor bay exit using the facility crane. The inner shield was then placed in the outer shield staged outside the building. From there, the Dry Transfer System was used to move the basket to the NAC LWT cask. A total of four full baskets and one empty basket were placed in the NAC LWT cask and transferred to Idaho. The loading operation took a total of four days from arrival and setup to truck inspection and departure.

CONCLUSION AND LESSONS LEARNED

Despite the involvement of large amounts of manpower and infrastructure investment, the OSTR conversion proceeded smoothly with no significant setbacks or problems. This is largely due to the willingness of several organizations to work together and support each other's needs. During the conversion, 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. The other indispensable 'item' was the DOE project manager who oversaw the weekly status meetings, coordinated work efforts and made the impossible merely difficult. Although much effort was involved, the life of the facility was significantly extended.

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