September 4, 2008

EA-08-251

Dr. Steven R. Reese, Director
Radiation Center
Oregon State University
100 Radiation Center
Corvallis, OR 97331-5903

SUBJECT: ISSUANCE OF ORDER MODIFYING LICENSE NO. R-106 TO CONVERT FROM HIGH- TO LOW-ENRICHED URANIUM FUEL (AMENDMENT NO. 22) – OREGON STATE UNIVERSITY TRIGA REACTOR (TAC NO. MD7360)

Dear Dr. Reese:

The U.S. Nuclear Regulatory Commission (NRC) is issuing the enclosed Order, as Amendment No. 22 to Facility Operating License No. R-106, which authorizes the conversion of the Oregon State University TRIGA Reactor from high-enriched uranium fuel to low-enriched uranium (LEU) fuel. This Order modifies the license, including the technical specifications, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.64. This regulation requires that non-power reactor licensees, such as the Oregon State University, convert to LEU fuel under certain conditions which Oregon State University now meets. The Order is being issued in accordance with 10 CFR 50.64(c)(3) and in response to your submittal of November 6, 2007, as supplemented on February 11, and June 20, 2008. The Order also contains an outline of a reactor startup report that you are required to provide to the NRC within 6 months following the return of the converted reactor to normal operation.

The Order becomes effective on the later date of either the day of receipt of an adequate number and type of LEU fuel elements that are necessary to operate the facility as specified in your submittal and supplements, or 20 days after the date of its publication in the *Federal Register*, provided there are no requests for a hearing.

Although this Order is not subject to the requirements of the Paperwork Reduction Act, there is nonetheless a clearance from the Office of Management and Budget (OMB), OMB approval number 3150-0012, that covers the information collections contained in the Order.

Copies of replacement pages for the facility operating license and technical specifications, and of the NRC staff safety evaluation for the conversion to LEU fuel are also enclosed. The Order is being sent to the *Federal Register* for publication.

Sincerely,

/RA/

Alexander Adams, Jr., Senior Project Manager
Research and Test Reactors Branch A
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Docket No. 50-243

Enclosures:
1. Order
2. Replacement Pages for License
3. Replacement Pages for Technical Specifications
4. Safety Evaluation

cc w/enclosures: See next page

Copies of replacement pages for the facility operating license and technical specifications, and of the NRC staff safety evaluation for the conversion to LEU fuel are also enclosed. The Order is being sent to the *Federal Register* for publication.

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Docket No. 50-243

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4. Safety Evaluation

cc w/enclosures: See next page

DISTRIBUTION:

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ADAMS Accession No.:ML082390775

OFFICE	DPR/LA	PRTA/PM	OGC	OE
NAME	EBarnhill eeb	AAdams aa	SUttal su	NHilton nh
DATE	8/28/08	8/29/08	9/3/08	9/2/08
OFFICE	PRTA/BC	DPR/D	NRR/D	PRTA/PM
NAME	DCollins dsc	MCase mc	JWiggins for ELeeds	AAdams aa
DATE	9/3/08	9/3/08	9/4/08	9/4/08

OFFICIAL RECORD COPY

Oregon State University Docket No. 50-243

cc:

Mayor of the City of Corvallis
Corvallis, OR 97331

Mr. Ken Niles
Oregon Office of Energy
625 Marion Street, N.E.
Salem, OR 97310

Dr. John Cassady, Vice President
for Research
Oregon State University
Administrative Services Bldg., Room A-312
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Mr. Todd Keller
Reactor Administrator
Oregon State University
Radiation Center, A-100
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Dr. Todd Palmer, Chairman
Reactor Operations Committee
Oregon State University
Radiation Center, A-100
Corvallis, OR 97331-5904

Test, Research, and Training
Reactor Newsletter
University of Florida
202 Nuclear Sciences Center
Gainesville, FL 32611

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
OREGON STATE UNIVERSITY	)	Docket No. 50-243
	)	EA-08-251
(Oregon State University TRIGA Reactor)	)	

ORDER MODIFYING FACILITY OPERATING LICENSE NO. R-106

I.

Oregon State University (the licensee) is the holder of Facility Operating License No. R-106 (the license), issued by the U.S. Nuclear Regulatory Commission (NRC). The NRC plans to renew the license on September 10, 2008. The license authorizes operation of the Oregon State University TRIGA Reactor (the facility) at a power level up to 1100 kilowatts thermal and in the pulse mode, with reactivity insertions not to exceed $2.55, and to receive, possess, and use special nuclear material associated with facility operation. The facility is a research reactor located on the campus of Oregon State University, in the city of Corvallis, Benton County, Oregon. The mailing address is Radiation Center, Oregon State University, 100 Radiation Center, Corvallis, Oregon 97331-5903.

II.

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.64, limits the use of high-enriched uranium (HEU) fuel in domestic non-power reactors (research and test reactors) (see 51 FR 6514). The regulation, which became effective on March 27, 1986, requires that if Federal Government funding for conversion-related costs is available, each licensee of a non-power reactor authorized to use HEU fuel shall replace it with low-enriched uranium (LEU) fuel acceptable to the Commission unless the Commission has determined that the reactor has a unique purpose. The Commission's stated purpose for these requirements was to reduce, to the maximum extent possible, the use of HEU fuel in order to reduce the risk of theft and diversion of HEU fuel used in non-power reactors.

Paragraphs 50.64(b)(2)(i) and (ii) require that a licensee of a non-power reactor (1) not acquire more HEU fuel if LEU fuel that is acceptable to the Commission for that reactor is available when the licensee proposes to acquire HEU fuel, and (2) replace all HEU fuel in its possession with available LEU fuel acceptable to the Commission for that reactor in accordance with a schedule determined pursuant to 10 CFR 50.64(c)(2).

Paragraph 50.64(c)(2)(i) requires, among other things, that each licensee of a non-power reactor authorized to possess and to use HEU fuel develop and submit to the Director of the Office of Nuclear Reactor Regulation (the Director) by March 27, 1987, and at 12-month intervals thereafter, a written proposal for meeting the requirements of the rule. The licensee shall include in its proposal a certification that Federal Government funding for conversion is available through the U.S. Department of Energy or other appropriate Federal agency. The proposal should also provide a schedule for conversion, based upon the availability of replacement fuel acceptable to the Commission for that reactor and upon consideration of other factors such as the availability of shipping casks, implementation of arrangements for available financial support, and reactor usage.

Paragraph 50.64(c)(2)(iii) requires the licensee to include in the proposal, to the extent required to effect conversion, all necessary changes to the license, the facility, and licensee procedures. This paragraph also requires the licensee to submit supporting safety analyses in time to meet the conversion schedule.

Paragraph 50.64(c)(2)(iii) also requires the Director to review the licensee proposal, to confirm the status of Federal Government funding, and to determine a final schedule, if the licensee has submitted a schedule for conversion.

Section 50.64(c)(3) requires the Director to review the supporting safety analyses and to issue an appropriate enforcement order directing both the conversion and, to the extent consistent with the protection of public health and safety, any necessary changes to the license, the facility, and licensee procedures. In the *Federal Register* notice of the final rule (51 FR 6514), the Commission explained that in most, if not all cases, the enforcement order would be an order to modify the license under 10 CFR 2.204 (now 10 CFR 2.202).

Any person, other than the licensee, whose interest may be affected by this proceeding and who desires to participate as a party must file a written request for hearing or petition for leave to intervene meeting the requirements of 10 CFR 2.309, “Hearing Requests, Petitions to Intervene, Requirements for Standing, and Contentions.”

III.

The U.S. Nuclear Regulatory Commission (NRC) maintains the Agencywide Documents Access and Management System (ADAMS), which provides text and image files of the NRC's public documents. On November 6, 2007, the licensee submitted its conversion proposal (ADAMS Accession No. ML080420546), which was supplemented on February 11, and June 20, 2008 (ADAMS Accession Nos. ML080730057 and ML082350345), including its proposed modifications and supporting safety analyses. HEU fuel elements are to be replaced with LEU fuel elements. The reactor core contains fuel elements of the TRIGA design, with the fuel consisting of uranium-zirconium hydride with 30 weight percent uranium. These fuel elements contain the uranium-235 isotope at an enrichment of less than 20 percent. The NRC staff reviewed the licensee's proposal and the requirements of 10 CFR 50.64 and has determined that public health and safety and common defense and security require the licensee to convert the facility from the use of HEU to LEU fuel in accordance with the attachments to this Order and the schedule included herein. The attachments to this Order specify the changes to the license conditions and technical specifications that are needed to amend the facility license and contain an outline of a reactor startup report to be submitted to NRC within 6 months following return of the converted reactor to normal operation.

IV.

Accordingly, pursuant to Sections 51, 53, 57, 101, 104, 161b, 161i, and 161o of the Atomic Energy Act of 1954, as amended, and to Commission regulations in 10 CFR 2.202 and 10 CFR 50.64, IT IS HEREBY ORDERED THAT:

Facility Operating License No. R-106 is modified by amending the license conditions and technical specifications as stated in the attachments to this Order (Attachment 1: MODIFICATIONS TO FACILITY OPERATING LICENSE NO. R-106; Attachment 2: OUTLINE OF REACTOR STARTUP REPORT). The Order becomes effective on the later date of either (1) the day the licensee receives an adequate number and type of LEU fuel elements to operate the facility as specified in the licensee proposal dated November 6, 2007 (ADAMS Accession No. ML080420546), as supplemented on February 11, and June 20, 2008 (ADAMS Accession Nos. ML080730057 and ML082350345), or (2) 20 days after the date of publication of this Order in the Federal Register.

V.

Pursuant to 10 CFR 2.202, any person(s) whose interest may be affected by this proceeding, other than the licensee, and who wishes to participate as a party in the proceeding must file a written request within 20 days after the date of publication of this Order, setting forth with particularity the manner in which this Order adversely affects his or her interest and addressing the criteria set forth in 10 CFR 2.309. If a hearing is held, the issue to be considered at such hearing shall be whether this Order should be sustained.

A request for a hearing must be filed in accordance with the NRC E-Filing rule, which became effective on October 15, 2007. The NRC issued the E-filing final rule on August 28, 2007 (72 FR 49139), and codified it in pertinent part at 10 CFR Part 2, “Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders,” Subpart B. The E-Filing process requires participants to submit and serve documents over the Internet or, in some cases, to mail copies on electronic optical storage media. Participants may not submit paper copies of their filings unless they seek a waiver in accordance with the procedures described below.

To comply with the procedural requirements associated with E-Filing, at least 10 days before to the filing deadline, the requestor must contact the Office of the Secretary by email at hearingdocket@nrc.gov, or by calling (301) 415-1677, to request (1) a digital identification (ID) certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any NRC proceeding in which it is participating, and/or (2) creation of an electronic docket for the proceeding (even in instances when the requestor (or its counsel or representative) already holds an NRC-issued digital ID certificate). Each requestor will need to download the Workplace Forms Viewer™ to access the Electronic Information Exchange (EIE), a component of the E-Filing system. The Workplace Forms Viewer™ is free and is available at http://www.nrc.gov/site-help/e-submittals/install-viewer.html. Information about applying for a digital ID certificate also is available on the NRC's public Web site at http://www.nrc.gov/site-help/e-submittals/apply-certificates.html.

Once a requestor has obtained a digital ID certificate, had a docket created, and downloaded the EIE viewer, he or she can then submit a request for a hearing through EIE. Submissions should be in portable document format (PDF) in accordance with NRC guidance available on the NRC public Web site at http://www.nrc.gov/site-help/e-submittals.html. A filing is considered complete at the time the filer submits its document through EIE. To be timely, electronic filings must be submitted to the EIE system no later than 11:59 p.m. eastern time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an email notice confirming receipt of the document. The EIE system also distributes an email notice that provides access to the document to the NRC Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the document on those participants separately. Therefore, any others who wish to participate in the proceeding (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request is filed so that they may obtain access to the document via the E-Filing system.

A person filing electronically may seek assistance through the "Contact Us" link located on the NRC Web site at http://www.nrc.gov/site-help/e-submittals.html or by calling the NRC technical help line, which is available between 8:30 a.m. and 4:15 p.m., eastern time, Monday through Friday. The help line number is (800) 397-4209 or, locally, (301) 415-4737.

Participants who believe that they have good cause for not submitting documents electronically must file a motion, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format. Such filings must be submitted by (1) first class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemaking and Adjudications Staff; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service.

Documents submitted in adjudicatory proceedings will appear in the NRC's electronic hearing docket at http://ehd.nrc.gov/EHD_Proceeding/home.asp, unless excluded pursuant to an order of the Commission, an Atomic Safety and Licensing Board, or a Presiding Officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers, in their filings. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a fair use application, participants are requested not to include copyrighted materials in their works.

If a hearing is requested and the request is granted by the Commission, the NRC will issue an order designating the time and place of the hearing.

In the absence of any request for hearing, the provisions as specified in Section IV shall be final twenty (20) days after the date of publication of this Order in the *Federal Register*.

In accordance with 10 CFR 51.10(d) this Order is not subject to Section 102(2) of the National Environmental Policy Act, as amended. The NRC staff notes, however, that with respect to environmental impacts associated with the changes imposed by this Order as described in the safety evaluation, the changes would, if imposed by other than an order, meet the definition of a categorical exclusion in accordance with 10 CFR 51.22(c)(9). Thus, pursuant to either 10 CFR 51.10(d) or 51.22(c)(9), no environmental assessment or environmental impact statement is required.

Detailed guidance which the NRC uses to review applications from research reactor licensees appears in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," February 1996, which can be obtained from the Commission's Public Document Room (PDR). The public may also access NUREG-1537 through the NRC's Public Electronic Reading Room on the Internet at http://www.nrc.gov/reading-rm/adams.html under ADAMS Accession Nos. ML0412430055 for part one and ML042430048 for part two.

For further information see the application from the licensee dated November 6, 2007 (ADAMS Accession No. ML080420546), as supplemented on February 11, and June 20, 2008 (ADAMS Accession Nos. ML080730057 and ML082350345), the NRC staff's requests for additional information (ADAMS Accession Nos. ML080090308 and ML081050294), and the cover letter to the licensee and the staff's safety evaluation dated September 4, 2008, (ADAMS Accession No. ML082390775). On April 4, 2008, the NRC staff issued an Order to the licensee to allow receipt and possession of the special nuclear material needed for the conversion (ADAMS Accession No. ML080730395). These documents are available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Public Electronic Reading Room at http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who have problems accessing the documents in ADAMS should contact the NRC PDR reference staff by telephone at (800) 397-4209 or (301) 415-4737 or by e-mail to pdr@nrc.gov.

Dated this 4th day of September 2008.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

James T. Wiggins, Deputy Director
Office of Nuclear Reactor Regulation

Attachments:
1. Modifications to Facility Operating
   License No. R-106
2. Outline of Reactor Startup Report

MODIFICATIONS TO FACILITY OPERATING LICENSE NO. R-106

A. License Conditions Revised by This Order

2.B.(2) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material,"

a. to receive, possess and use, in connection with operation of the facility, up to 16.30 kilograms of contained uranium-235 enriched to less than 20 percent in the form of TRIGA reactor fuel;

b. to receive, possess and use, in connection with operation of the facility, up to 100 grams of contained uranium-235 of any enrichment in the form of fission chambers and flux foils;

c. to receive, possess, but not use, up to 656 grams of uranium-235 enriched to less than 20 percent in the form of the core from the AGN-201 reactor;

d. to receive, possess, use, but not separate, in connection with operation of the facility, such special nuclear material as may be produced by operation of the facility; and

e. to possess, but not use, up to 12.83 kilograms of contained uranium-235 at equal to or greater than 20 percent enrichment in the form of TRIGA fuel until the existing inventory of this fuel is removed from the facility.

2.C.(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 22, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

OUTLINE OF REACTOR STARTUP REPORT

Within six months following the return of the converted reactor to normal operation, submit the following information to the NRC. Information on the HEU core should be presented to the extent it exists.

1. Critical mass

   Measurement with HEU
   Measurement with LEU
   Comparisons with calculations for LEU and if available, HEU

2. Excess (operational) reactivity

   Measurement with HEU
   Measurement with LEU
   Comparisons with calculations for LEU and if available, HEU

3. Control rod calibrations

   Measurement of HEU and LEU rod worths and comparisons with calculations for LEU and if available, HEU

4. Reactor power calibration

   Methods and measurements that ensure operation within the license limit and comparison between HEU and LEU nuclear instrumentation set points, detector positions and detector output.

5. Shutdown margin

   Measurement with HEU
   Measurement with LEU
   Comparisons with calculations for LEU and if available, HEU

6. Thermal neutron flux distributions

   Measurements of the core and measured experimental facilities (to the extent available) with HEU and LEU and comparisons with calculations for LEU and if available, HEU.

7. Reactor physics measurements

   Results of determination of LEU effective delayed neutron fraction, temperature coefficient, and void coefficient to the extent that measurements are possible and comparison with calculations and available HEU core measurements.

8. Initial LEU core loading

   Measurements made during initial loading of the LEU fuel, presenting subcritical multiplication measurements, predictions of multiplication for next fuel additions, and prediction and verification of final criticality conditions.

9. Primary coolant measurements

   Results of any primary coolant water sample measurements for fission product activity taken during the first 30 days of LEU operation.

10. Results of any test pulses performed and comparison with calculations and available HEU core measurements.

11. Discussion of results

    Discussion of the comparison of the various results including an explanation of any significant differences that could affect normal operation and accident analyses.

FACILITY OPERATING LICENSE NO. R-106

DOCKET NO. 50-243

REPLACEMENT PAGES FOR LICENSE

Replace the following pages of the License with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove	Insert
2	2
3	3
	4

G. The issuance of this renewed license will not be inimical to the common defense and security or to the health and safety of the public;

H. The issuance of this renewed license is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession and use of the byproduct and special nuclear materials as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30 and 70.

2. Renewed Facility License No. R-106 is hereby amended in its entirety to read as follows:

A. The license applies to the TRIGA nuclear research reactor (hereafter, the facility) owned by the Oregon State University. The facility is located in Corvallis, Oregon, and is described in the licensee's application for renewal of the license dated October 5, 2004, as supplemented.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the Oregon State University:

(1) Pursuant to Section 104c of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility as a utilization facility at the designated location in Corvallis, Benton County, Oregon in accordance with the procedures and limitations described in the application and set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material,"

a. to receive, possess and use, in connection with operation of the facility, up to 16.30 kilograms of contained uranium-235 enriched to less than 20 percent in the form of TRIGA reactor fuel;

b. to receive, possess and use, in connection with operation of the facility, up to 100 grams of contained uranium-235 of any enrichment in the form of fission chambers and flux foils;

c. to receive, possess, but not use, up to 656 grams of uranium-235 enriched to less than 20 percent in the form of the core from the AGN-201 reactor;

d. to receive, possess, use, but not separate, in connection with operation of the facility, such special nuclear material as may be produced by operation of the facility; and

e. to possess, but not use, up to 12.83 kilograms of contained uranium-235 at equal to or greater than 20 percent enrichment in the form of TRIGA fuel until the existing inventory of this fuel is removed from the facility.

(3) Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," to receive, possess and use in connection with operation of the facility:

a. up to a 7-curie sealed polonium-210 beryllium source which may be used for reactor startup;

b. up to a 3-curie sealed americium-241 beryllium neutron source which may be used for reactor startup; and

c. such byproduct material as may be produced by the operation of the facility. Byproduct material cannot be separated except for byproduct material produced in experiments.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Parts 20, 30, 50, 51, 55, 70, and 73; is subject to all applicable provisions of the Act, and to the rules, regulations and orders of the Commission now, or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady-state power levels not in excess of 1.1 megawatts (thermal), and in the pulse mode, with reactivity insertions not to exceed $2.55.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 22, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Physical Security Plan

The licensee shall maintain and fully implement all provisions of its NRC-approved physical security plan, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). The approved security plan consists of documents withheld from public disclosure pursuant to 10 CFR 73.21, entitled “Oregon State University TRIGA Reactor Physical Security Plan,” as revised.

D. This license is effective as of the date of issuance and shall expire twenty years from its date of issuance.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

/RA/

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Enclosure:
Appendix A Technical Specifications

Date of Issuance: September 10, 2008

FACILITY OPERATING LICENSE NO. R-106

DOCKET NO. 50-243

REPLACEMENT PAGES FOR TECHNICAL SPECIFICATIONS

Replace the following pages of Appendix A, Technical Specifications, with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
8	8
10	10
11	11
14	14
16	16
20	20
21	21
31	31
32	32

2.2 Limiting Safety System Setting

Applicability. This specification applies to the scram settings which prevent the safety limit from being reached.

Objective. The objective is to prevent the safety limits from being reached.

Specifications The limiting safety system setting shall be equal to or less than 510°C (950°F) as measured in an instrumented fuel element. The instrumented fuel element shall be located in the B-ring.

Basis. During steady state operation, maximum temperatures are predicted to occur in the LEU MOL ICIT core. Linear extrapolation of temperature and power from Table 4□ 27 of section 4.7.8 indicates that an IFE power of 23.1 kW will produce an indicated temperature of 510ºC in the IFE at the midplane thermocouple location. Of the nine LEU cores analyzed in the SAR, the highest ratio of maximum to minimum power for elements in the B□ ring is 1.121, so if the IFE is generating 23.1 kW, the maximum power in any B□ ring element would be limited to 23.1 x 1.121 = 25.9 kW. Figures 4□ 59, 4□ 64 and 4□ 69 indicate that at 25.9 kW, the maximum temperature anywhere in the hot channel fuel element will be less than 600ºC.

Additional analysis has shown that if a B□ ring element in the LEU MOL ICIT or LEU EOL ICIT core is replaced with fresh fuel, the highest ratio of maximum to minimum power in the B□ ring is 1.378. In these cores, if the IFE is generating 23.1 kW, the maximum power in any B□ ring element would be limited to 23.1 x 1.378 = 31.8 kW. Figures 4□ 59, 4□ 64 and 4□ 69 indicate that at 31.8 kW, the maximum temperature anywhere in the hot channel fuel element will be less than 700ºC.

3.1.3 Core Excess Reactivity

Applicability. This specification applies to the reactivity condition of the reactor and the reactivity worths of control rods and experiments. It applies for all modes of operation.

Objective. The objective is to assure that the reactor can be shut down at all times and to assure that the fuel temperature safety limit shall not be exceeded.

Specifications. The maximum available excess reactivity based on the reference core condition shall not exceed $7.55.

Basis. An excess reactivity limit of $7.55 allows flexibility to operate the reactor in all core modes (NORMAL, ICIT and CLICIT) without the need to add or remove fuel elements when changing between operating modes. The NORMAL core is the most reactive core. If operating a NORMAL core with the minimum shutdown margin of $0.55 and typical control rod worths of $2.70 (Safety), $2.70 (Shim), $2.70 (Regulating) and $4.00 (Transient) (section 4.2.2, Control Rods), the calculated NRC core excess would be -$0.55+$2.70+$2.70+$2.70 = $7.55. The shutdown margin calculation assumes a) irradiation facilities and experiments in place and the total worth of all non-secured experiments in their most reactive state, b) the most reactive control rod fully-withdrawn and c) the reactor in the reference core condition. Activities such as changing out of the NORMAL core, moving away from the reference state or adding negative worth experiments will make core excess more negative and shutdown margin less positive. The only activity which could result in requiring fuel movement to meet shutdown margin and core excess limits would be the unusual activity of adding an experiment with large positive reactivity worth.

3.1.4 Pulse Mode Operation

Applicability. This specification applies to the energy generated in the reactor as a result of a pulse insertion of reactivity.

Objective. The objective is to assure that the fuel temperature safety limit shall not be exceeded.

Specifications. The reactivity to be inserted for pulse operation shall be determined and limited by a mechanical block and electrical interlock on the transient rod, such that the maximum fuel element temperature shall not exceed 830°C.

Basis. The basis for the temperature limit given can be found in GA□ C26017, Pulsing Temperature Limit for TRIGA® LEU Fuel. The fuel temperature rise during a pulse transient has been estimated by RELAP□ 5□ 3D using non□ adiabatic models. The core analyzed had the highest accumulative peaking factor for any core configuration, was analyzed at the middle of life because that presents the highest core reactivity during the fuel lifetime and looked at the rod with the predicted maximum temperature in that core. These models predict pulse characteristics for operation of operational cores and should be accepted with confidence, relying also on information concerning prompt neutron lifetime and prompt temperature coefficient of reactivity. The reactivity value calculated to produce a temperature of 830°C is $2.33. Therefore limiting reactivity insertions to a maximum of $2.30 will ensure that fuel temperature will not exceed 830°C.

3.1.5 This section intentionally left blank.

3.1.6 Fuel Parameters

Applicability. This specification applies to all fuel elements.

Objective. The objective is to maintain integrity of the fuel element cladding.

Specifications. The reactor shall not operate with damaged fuel elements, except for the purpose of locating damaged fuel elements. A fuel element shall be considered damaged and must be removed from the core if:

a. The transverse bend exceeds 0.0625 inches over the length of the cladding;

b. Its length exceeds its original length by 0.125 inches;

c. A cladding defect exists as indicated by release of fission products; or

d. Visual inspection identifies bulges, gross pitting, or corrosion.

Basis. Gross failure or obvious visual deterioration of the fuel is sufficient to warrant declaration of the fuel as damaged. The elongation and bend limits are the values found acceptable to the USNRC (NUREG-1537).

3.2 Reactor Control And Safety System

3.2.1 Control Rods

Applicability. This specification applies to the function of the control rods.

Objective. The objective is to determine that the control rods are operable.

Specification. The reactor shall not be operated unless the control rods are operable. Control rods shall not be considered operable if:

Specifications. The reactor shall not be operated unless the minimum number of safety channels described in Table 2 and interlocks described in Table 3 are operable.

Table 2 - Minimum Reactor Safety Channels

Safety Channel	Function	Effective Mode		
		S.-S.	Pulse	S.-W.
Fuel Element Temperature	SCRAM @ 510°C	1	-	1
Power Level	SCRAM @ 1.1 MW(t) or less	2	-	2
Console Scram Button	SCRAM	1	-	1
Preset Timer	Transient rod SCRAM @ ≤ 15 sec after a pulse	-	1	-
High Voltage	SCRAM @ ≥25% of nominal operating voltage	1	1	1

Table 3 - Minimum Interlocks

Interlock	Function	Effective Mode		
		S.-S.	Pulse	S.-W.
Wide-Range Log Power Level Channel	Prevents control rod withdrawal @ less than 2 cps	1	-	-
Transient Rod Cylinder	Prevents application of air unless fully-inserted	1	-	-
1 kW Pulse Interlock	Prevents pulsing above 1 kW	-	1	-
Shim, Safety, and Regulating Rod Drive Circuit	Prevents simultaneous manual withdrawal of two rods	1	-	1
Shim, Safety, and Regulating Rod Drive Circuit	Prevents movement of any rod except transient rod	-	1	-
Transient Rod Cylinder Position	Prevent pulse insertion of reactivity greater than that which would produce a maximum fuel element temperature of 830°C .	-	1	1

insufficient to produce meaningful instrumentation response. If the operator were to insert reactivity under this condition, the period could quickly become very short and result in an inadvertent power excursion. A neutron source is added to the core to create sufficient instrument response that the operator can recognize and respond to changing conditions.

Transient Rod Cylinder Interlock: This interlock prevents the application of air to the transient rod unless the rod is fully inserted. This will prevent the operator from pulsing the reactor in steady-state mode.

1 kW Pulse Interlock: The 1 kW permissive interlock is designed to prevent pulsing when wide range log power is above 1 kW. Analysis of pulsing at full power shows that if the initial temperature was higher, the resulting peak temperature will be lower. However, there has not been an experiment to look at the relationship between heat generated within the fuel at power (i.e., > 1 kW) and heat generated on the surface of the fuel during a pulse. Therefore, this interlock prevents the reactor from pulsing at power levels which produce measurably significant increases in fuel temperature.

Shim, Safety and Regulating Rod Drive Circuit: The single rod withdrawal interlock prevents the operator from removing multiple control rods simultaneously such that reactivity insertions from control rod manipulation are done in a controlled manner. The analysis in SAR 13.2.2.2.2 and 13.2.2.2.3 show that the reactivity insertion due to the removal rate of the most reactive rod or all the control rods simultaneously is still well below the reactivity which would produce a fuel element temperature of 830°C.

Shim, Safety and Regulating Rod Drive Circuit: In pulse mode, it is necessary to limit the reactivity below that which would produce a fuel element temperature of 830°C. This interlock ensures that all pulse reactivity is due to only the transient rod while in pulse mode. Otherwise, any control rod removal in pulse mode would add to the inserted reactivity of the transient rod and create an opportunity for exceeding the reactivity insertion limit.

Transient Rod Cylinder Position Interlock: The transient rod cylinder interlock shall limit reactivity insertions below that which would produce a fuel element temperature of 830°C. This interlock limits transient rod reactivity insertions below this value. Furthermore, this interlock is designed such that if the electrical (i.e., limit switch) portion fails, a mechanical (i.e., metal bracket) will still keep the reactivity insertion below the criterion.

3.7.2 Effluents

Applicability. This specification applies to the release rate of ^{41}Ar.

Objective. The objective is to ensure that the concentration of the ^{41}Ar in the unrestricted areas shall be below the applicable effluent concentration value in 10 CFR 20.

Specifications. The annual average concentration of ^{41}Ar discharged into the unrestricted area shall not exceed 4×10^{-6} μCi/ml at the point of discharge.

Basis. If ^{41}Ar is continuously discharged at 2.5×10^{-6} μCi/ml (i.e., the concentration produced when the nitrogen purge of the rotating rack is off, all valves on the argon manifold are open, and all beam port valves are open), measurements and calculations show that ^{41}Ar released to the unrestricted areas under the worst-case weather conditions would result in an annual TEDE of 5 mrem (SAR 11.1.1.1.1). This is only 50% of the applicable limit of 10 mrem (Regulatory Guide 4.20). Therefore, an emission of 4×10^{-6} μCi/ml would correspond to an annual TEDE of 8 mrem which is still 20% below the applicable limit.

3.8 Limitations on Experiments

3.8.1 Reactivity Limits

Applicability. This specification applies to experiments installed in the reactor and its irradiation facilities.

Objective. The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications. The reactor shall not be operated unless the following conditions governing experiments exist:

a. The absolute value of the reactivity worth of any single unsecured experiment shall be less than $0.50; and

b. The sum of the absolute values of the reactivity worths of all experiments shall be less than $2.30.

Basis. The reactivity limit of $0.50 for movable experiments is designed to prevent an inadvertent pulse from occurring and maintain a value below the shutdown margin. Movable experiments are by their very nature experiments in a position were it is possible for a sample to be inserted or removed from the core while critical. That being said, the value is clearly less than the limit on pulsing.

The reactivity worth limit for all experiments is designed to prevent an inadvertent pulse from exceeding the recommended limit on pulsing. This limit applies to movable, unsecured and secured experiments. A maximum reactivity insertion of $2.30 will ensure that fuel temperature will not exceed 830°C.

3.8.2 Materials

Applicability. This specification applies to experiments installed in the reactor and its irradiation facilities.

Objective. The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications. The reactor shall not be operated unless the following conditions governing experiments exist:

a. Explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, in quantities greater than 25 milligrams shall not be irradiated in the reactor or irradiation facilities. Explosive materials in quantities less than 25 milligrams may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than half the design pressure of the container.; and

b. Experiments containing corrosive materials shall be doubly encapsulated. The failure of an encapsulation of material that could damage the reactor shall result in removal of the sample and physical inspection of potentially damaged components.

Basis. This specification is intended to prevent damage to reactor components resulting from failure of an experiment involving explosive materials. Operation of the reactor with the reactor fuel or structure potential damages is prohibited to avoid potential release of fission products.

3.8.3 Failures and Malfunctions

Applicability. This specification applies to experiments installed in the reactor and its irradiation facilities.

a. The shim, safety, and regulating control rods shall have scram capability and contain borated graphite, B_4C powder or boron, with its compounds in solid form as a poison, in aluminum or stainless steel cladding. These rods may incorporate fueled followers which have the same characteristics as the fuel region in which they are used.

b. The transient control rod shall have scram capability and contain borated graphite or boron, with its compounds in a solid form as a poison in an aluminum or stainless steel cladding. The transient rod shall have an adjustable upper limit to allow a variation of reactivity insertions. This rod may incorporate an aluminum- or air-follower.

Basis. The poison requirements for the control rods are satisfied by using neutron absorbing borated graphite, B_4C powder or boron as its compounds. These materials must be contained in a suitable clad material such as aluminum or stainless steel to ensure mechanical stability during movement and to isolate the poison from the tank water environment. Control rods that are fuel-followed provide additional reactivity to the core and increase the worth of the control rod. The use of fueled-followers in the fueled region has the additional advantage of reducing flux peaking in the water-filled regions vacated by the withdrawal of the control rods. Scram capabilities are provided for rapid insertion of the control rods which is the primary safety feature of the reactor. The transient control rod is designed for rapid withdrawal from the reactor core which results in a reactor pulse. The nuclear behavior of the air- or aluminum-follower, which may be incorporated into the transient rod, is similar to a void. A voided-follower may be required in certain core loadings to reduce flux peaking values (SAR 4.2.2).

5.3.3 Reactor Fuel

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

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

Specifications. TRIGA® Fuel Elements

The individual unirradiated fuel elements shall have the following characteristics:

1. Uranium content: nominal 30 wt% enriched to less than 20% in 235U;

2. Hydrogen-to-zirconium atom ratio (in the ZrH_x): between 1.5 and 1.65;

3. Natural erbium content (homogeneously distributed): nominal 1.1 wt%;

4. Cladding: 304 stainless steel, nominal 0.020 inches thick; and

5. Identification: top pieces of fuel elements will have characteristic markings to allow visual identification of elements.

Basis. Material analysis of OSTR 30/20 fuel shows that the maximum weight percent of uranium in any fuel element is less than 30.5 percent, and the maximum enrichment of any fuel element is less than 20.0 percent. The minimum erbium content of any fuel element is greater than 1.0 percent. The hydrogen to zirconium ratio for all fuel elements is between 1.55 and 1.65. An element loaded with the maximum U-235 content and minimum erbium content would result in an increase in power density of no more than 2.4% over an element with nominal uranium and erbium loading. An increase in the local power density of 2.4% reduces the safety margin by, at most, 4%. The maximum hydrogen-to-zirconium ratio of 1.65 could result in a maximum stress under accident conditions in the fuel element cladding of about a factor of two greater than the value resulting from a hydrogen-to-zirconium ratio of 1.60. However, this increase in the cladding stress during an accident would not exceed the rupture strength of the cladding (SAR 4.2.1).

5.4 Fuel Storage

Applicability. This specification applies to the storage of reactor fuel at times when it is not in the reactor core.

Objective. The objective is to assure that fuel which is being stored shall not become critical and shall not reach an unsafe temperature.

Specifications.

a. All fuel elements shall be stored in a geometrical array where the k-effective is less than 0.9 for all conditions of moderation.

b. Irradiated fuel elements and fuel devices shall be stored in an array which will permit sufficient natural convection cooling by water or air such that the temperature of the fuel element or fueled device will not exceed the safety limit.

Basis. The limits imposed are conservative and assure safe storage (NUREG-1537).

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING CONVERSION ORDER TO CONVERT FROM

HIGH-ENRICHED TO LOW-ENRICHED URANIUM FUEL

FACILITY OPERATING LICENSE NO. R-106

OREGON STATE UNIVERSITY TRIGA REACTOR

DOCKET NO. 50-243

1.0 INTRODUCTION

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.64 requires licensees of research and test reactors to convert from the use of high-enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel, unless specifically exempted. Oregon State University (OSU or the licensee) has proposed to convert the fuel in the OSU TRIGA Reactor (OSTR) from HEU to LEU. In a letter dated November 6, 2007, as supplemented on February 11, and June 20, 2008, the licensee submitted its proposal for conversion requesting approval of the fuel conversion and of changes to its Technical Specifications. To support this action the licensee submitted a conversion Safety Analysis Report (SAR) on which the HEU to LEU conversion and the Technical Specification changes are based. This Safety Evaluation Report provides the results of the NRC staff's evaluation of the licensee's conversion proposal. The evaluation was carried out according to the guidance found in NUREG-1537[1].

On April 4, 2008, the NRC issued an Order to the licensee for the receipt and possession of the special nuclear material needed for the conversion. This was to allow the licensee to receive the LEU from the manufacturer in France before the certification of the shipping containers used to ship the fuel expired in June 2008.

2.0 EVALUATION

2.1 Summary of Reactor Facility Changes

The OSTR is a TRIGA reactor, built by General Atomics (GA), and similar in design to many others operating in the U.S. and abroad. The reactor normally operates at a maximum thermal power level of 1.0 MW(t) although the technical specifications (TS) allow operation up to 1.1 MW(t). The reactor uses natural convection for cooling. It is presently fueled with highly enriched TRIGA fuel lifetime improvement program (FLIP) fuel. The HEU to LEU conversion

1 "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Standard Review Plan and Acceptance Criteria," NUREG 1537, Part 2, U.S. Nuclear Regulatory Commission, February 1996.

only requires changes in the fuel type, core configuration, and technical specifications for core operating limits; it does not require any changes to the remainder of the facility.

The aluminum clad graphite reflector assemblies will be replaced as part of conversion with stainless-steel clad graphite reflector elements. This is being done because many of the aluminum clad reflector assemblies are exhibiting characteristics of concentric swelling which makes removing them through the top grid plate difficult. The impact of this change on the operation or safety of the LEU core was reviewed by the staff and determined to be inconsequential.

While not part of conversion, the annular reflector assembly will also be replaced at the same time that the reactor is converted. The conversion process puts the reactor into a favorable configuration for the annular reflector assembly replacement.

2.2 Comparison with Similar Facilities Already Converted

A TRIGA reactor at Texas A&M University has converted using the same general design LEU fuel elements proposed for the OSU conversion. In addition, the TRIGA Mark F reactor at GA in San Diego was operating with a core partially made up of high density LEU conversion fuel when it was permanently shut down. There have been no performance issues in the use of this fuel in these reactors.

2.3 Fuel and Core Design

The physical size or cladding of the fuel elements will not change in the conversion from HEU to LEU. Only the content of the fuel alloy will change. The HEU FLIP fuel elements are designed to have an enrichment of 70 percent uranium-235, nominal uranium content of 8.5 weight percent (w/o), erbium poison content of 1.6 w/o, and zirconium hydride (ZrH_x) as the balance of the fuel, where the value for x is approximately 1.6. The fresh LEU fuel elements have a nominal enrichment of 19.75 percent uranium-235, nominal uranium content of up to 30 w/o, erbium poison content of 1.1 w/o, and ZrH_x as the balance of the fuel, where x is approximately in the range of 1.5-1.65. The change in fuel density means the LEU fuel has a uranium mass that is 4.2 times the mass in an HEU element. The additional mass is uranium-238. The main fuel characteristics are shown in Table 1.

Generic behavior of LEU TRIGA fuel with relatively high (30 w/o) uranium content has been previously approved for use in research and test reactors by the Nuclear Regulatory Commission (NUREG-1282)[2]. The licensee submitted this application because it still needs to present a safety analysis justifying the use of the LEU fuel in the OSTR.

The w/o or amount of erbium poison is reduced from the HEU core to the LEU core. The erbium poison reduces the reactivity present during core operation. This reduction lowers the power per element in order to allow the use of more fuel elements in the core. Consistent with Table 1 in NUREG-1282, the proposed erbium poison content of 1.1 w/o is within the erbium poison content range of 0.0 to 1.8 w/o approved for use.

The w/o or amount of ZrH_x is reduced from the HEU core to the LEU core, meaning there is less hydrogen in the fuel. The hydrogen is one of the main reasons for the very large negative fuel

2 "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors," NUREG-1282, U.S. Nuclear Regulatory Commission, August 1987.

temperature coefficient of reactivity. The reduced hydrogen content changes the value of the fuel temperature coefficient of reactivity, α_F, which is discussed in Section 2.3.3, "Dynamic Parameters."

Calculations indicate that there will be 84 standard fuel elements in the LEU core compared to the present 81. The LEU core will also have an instrumented fuel element (IFE), which is a fuel element containing thermocouples to measure fuel temperature. The 21 aluminum clad graphite reflector elements will be replaced with 24 stainless steel clad graphite reflector elements.

The initial LEU core is expected to go critical with 69 fuel elements loaded. If actual erbium content is less than 1.1 w/o, the minimum critical core may contain fewer elements. Note that this analysis assumes that the F-ring will be loaded sequentially, starting from the F-1 position, with each loaded fuel element next to the previously loaded element.

The licensee uses three basic cores, one with a fuel element in the B-1 position (the "Normal" core); an In-Core Irradiation Tube (ICIT) in the B-1 position (the ICIT core), or a Cadmium-Lined In-Core Irradiation Tube (CLICIT) in the B-1 position (the CLICIT core).

There are a total of four control rods in the reactor. All are standard control rods for TRIGA reactors. There are three control rods with fuel followers and a void-followed transient rod. The HEU to LEU conversion will not require replacement or modification of these control rods except for the replacement of the HEU fuel followers with LEU fuel followers.

The staff has reviewed the proposed fuel and core design of the LEU reactor. The staff concludes that the conversion from HEU to LEU fuel will not impact the overall basic design of the core and its control. Therefore, the staff finds the fuel and core design acceptable.

Table 1. Fuel Characteristics for OSTR HEU FLIP and LEU 30/20 Fuel

Fuel Type	HEU FLIP	LEU 30/20
Uranium content [mass %]	8.5	30
U-235 enrichment [mass % U]	70	19.75
Erbium content [mass %]	1.6	1.1
Fuel alloy inner diameter [mm]	6.35	6.35
Fuel alloy outer diameter [mm]	36.449	36.449
Fuel alloy length [mm]	381	381
Cladding material	Type 304 SS	Type 304 SS
Cladding thickness [mm]	0.508	0.508
Cladding outer diameter [mm]	37.465	37.465

2.4 Nuclear Design

2.4.1 Calculational Methodology

In order to carry out the OSTR neutronic analysis, a Monte Carlo N-Particle (MCNP) model was developed by the licensee using MCNP5. MCNP5-REBUS, an Argonne National Laboratory code developed to analyze reactor fuel cycles, was used for burnup calculations. RELAP5-3D, an Idaho National Laboratory code developed for multi-dimensional hydrodynamic and multi-dimensional reactor kinetics modeling, was used for pulsing calculations.

MCNP5 was benchmarked for the OSTR by comparing calculated results to measured values of k_{eff} for a fresh HEU core with control rods at different positions. A comparison of calculations with 38 measured critical statepoints with control rods at different positions was done. This benchmark work established a bias of approximately +0.0045 Δk/k +/- 0.0010 Δk/k for the calculated values when compared to the measured values of k_{eff}. This is an acceptably small error. The bias was used to correct all predictive calculations for the LEU core.

The excess reactivity was calculated and compared to measured values. The calculated excess reactivity of the FLIP HEU core at beginning-of-life (BOL) was $7.10 which compares favorably with the measured value of $7.17. Calculations of integral control rod worths also compared favorably with measured values for all four control rods. Differential control rod worths are more difficult to calculate; nevertheless, those calculations show reasonable agreement with measured values.

Using MCNP5-REBUS, the licensee defined middle-of-life (MOL) and end-of-life (EOL) for both the HEU and LEU cores, assuming operation at a constant power of 1.1 MW(t). MOL is when excess reactivity is at a maximum and EOL is when excess reactivity is 0.5% Δk/k. The values for the HEU core are 1800 MWd for MOL and 3800 MWd for EOL. The values for the LEU core are 1600 MWd for MOL and 3600 MWd for EOL. These values were then used in the analysis of all significant parameters.

Because the licensee used documented codes that are well-accepted and have been validated against data for TRIGA reactors, including the OSTR, the staff concludes that the calculational methodology used by the licensee is acceptable.

2.4.2 Excess Reactivity, Control Rod Worth, and Shutdown Margin

The calculated excess reactivity for the LEU core is $5.48 at BOL and $6.51 at MOL. This satisfies the TS 3.1.3, “Core Excess Reactivity,” limit of $7.55. The measured excess reactivity for the HEU core is $7.17 at BOL. The LEU core values are considerably lower than for the HEU core but are expected to be sufficient to permit all modes of operation while minimizing the need to alter the core configuration to accommodate the reactivity swing due to the depletion of erbium. The staff concludes that TS 3.1.3 will continue to be met after conversion.

The control rod worths were calculated using the MCNP5 model for both the HEU and LEU cores. The worth of the shim rod is calculated to be $2.54 for the HEU core and $2.55 for the LEU core. The worth of the safety rod is calculated to be $3.01 for the HEU core and $2.60 for the LEU core, a 16 percent decrease. The worth of the regulating rod is calculated to be reduced from $3.72 for the HEU core to $3.36 for the LEU core, an 11 percent decrease. The worth of the transient rod is calculated to be reduced from $2.95 for the HEU core to $2.86 for the LEU core. Given the reduction in excess reactivity of the core, the staff concludes that the worths of the control rods in the LEU core are acceptable.

The shutdown margin was calculated for both the HEU and LEU cores using the MCNP5 model by fully inserting three rods with the remaining rod fully withdrawn. TS 3.1.2, “Shutdown Margin,” states that this margin should be at least $0.55. Calculations indicate that the initial LEU core will comply with this requirement except during MOL with the regulating rod fully withdrawn. In order to account for this, the licensee indicated they will manage shutdown margin and excess reactivity by adding or removing fuel elements to ensure that the core meets the required shutdown margin. The staff evaluated the licensee's experience accommodating for the reactivity swing experienced with the HEU FLIP core during its life. Adjustments are required to be made to the FLIP core (which up to this point in the FLIP core life has consisted of the removal of fuel elements) to meet TS requirements and the licensee has been able to make the required adjustments. Therefore, the staff concludes that the ability to shut down the reactor will be maintained after conversion.

2.4.3 Dynamic Parameters

The prompt neutron lifetime, l_p, changes slightly as a result of the conversion from HEU to LEU fuel. The neutron lifetime was calculated with the 1/v absorber method, a standard technique where a small amount of boron is uniformly distributed throughout the reactor. The neutron lifetime is correlated to the associated changes in core reactivity and the value extrapolated to zero boron concentration is the lifetime used. The value of l_p for the HEU core was calculated to be 18.7 μs at BOL, 32.5 μs at MOL, and 37.1 μs at EOL. The value of l_p for the LEU core was calculated to be 22.6 μs at BOL, 19.0 μs at MOL, and 30.7 μs at EOL. The staff concludes that the changes in l_p are reasonable and that conversion is not expected to significantly change the basic behavior of the core.

The effective delayed neutron fraction, β_{eff}, was calculated for the HEU and LEU cores using the MCNP5 model with first only the prompt fission spectrum and then with the prompt and delayed fission spectrum. The value of β_{eff} for the HEU core was calculated to be 0.0076 at BOL, 0.0078 at MOL, and 0.0073 at EOL. The value of β_{eff} for the LEU core was calculated to be 0.0076 at BOL, 0.0073 at MOL, and 0.0075 at EOL. The changes in the calculated value of β_{eff} between the HEU and LEU core throughout core life are smaller than the uncertainty of the calculations. Taking into account the error in these numbers, it is acceptable to use 0.0075 for all the LEU analysis as was done by the licensee. These values of β_{eff} are consistent with those found for other research reactors. Therefore, the staff concludes that the changes in β_{eff} are acceptable and that conversion is not expected to significantly change the basic behavior of the core.

The void coefficient of reactivity was calculated using the MCNP5 model for both the HEU and LEU cores. The calculated void coefficients in the center of the cores are -\$0.86/% void for the HEU core and -\$0.96/% void for the LEU core. The calculated average void coefficients of the cores are -\$0.16/% void for the HEU core and -\$0.19/% void for the LEU core. The staff concludes that this change is not significant and will not significantly change the basic behavior of the core.

The moderator coefficient of reactivity, α_m, was calculated for the HEU and LEU cores using the MCNP5 model by varying the moderator temperature. Using a temperature range of 20 ^oC to 60 ^oC, the calculated value of α_m is -0.57 ¢/^oC for the HEU core and -0.72 ¢/^oC for the LEU core. The staff concludes that this change is not significant and will not significantly change the basic behavior of the core.

One of the inherent safety features of the TRIGA fuel design is the strong negative fuel temperature coefficient of reactivity, α_F. Because TRIGA reactors were designed for pulsing, TRIGA fuels could be subject up to \$5.00 of prompt positive reactivity insertion without damaging the fuel. The prompt feedback turns around power excursions before fuel damage due to overheating can occur. One of the reasons is the fuel contains a considerable amount of zirconium hydride, ZrH_x. The hydrogen in the fuel thermalizes the neutrons, and as the fuel heats up, the thermalization is reduced. Doppler broadening of erbium resonances also contributes to the negative feedback.

With conversion to LEU fuel, the amount of uranium is increased from approximately 8.5 w/o (HEU) to 30 w/o (LEU) with no change in the physical size of the fuel. Therefore, the amount of zirconium hydride is reduced from approximately 91.5 w/o (HEU) to 70 w/o (LEU). It is expected that this, along with the increase in the amount of uranium-238 in the fuel, will impact the fuel temperature coefficient.

The value of α_F was calculated for the HEU and LEU cores at BOL, MOL, and EOL as a function of temperature. At 77 ^oC, the calculated value of α_F is -9.31×10^{-4} %Δk/k/^{o}C for the HEU core at BOL and -12.3×10^{-4} %Δk/k/^{o}C for the LEU core at BOL. This increase in magnitude is expected due to the added uranium-238 in the LEU fuel. However, when pulsing, it is the zirconium hydride that dominates the value of α_F. At 727 ^oC, the calculated value of α_F is -120×10^{-4} %Δk/k/^{o}C for the HEU core at EOL and -84.6×10^{-4} %Δk/k/^{o}C for the LEU core at EOL. Although the value of the fuel temperature coefficient decreases in the LEU core during pulsing, TS limitations on pulsing will continue to protect the integrity of the fuel.

Combining the fuel temperature and moderator temperature coefficient results to determine the power defect in going from zero to full power permits an estimation of a power coefficient of reactivity. The value for the HEU core is -0.21 ¢/kW and for the LEU core, -0.22 ¢/kW.

General Atomics has recommended that temperature of the fuel should not exceed 830 °C during pulse mode operation because the hydrogen gas can build up pressure that could cause fuel damage (note that the safety limit, which prevents fuel failure, is not changed). The performance of the pulse mode operation is analyzed with the RELAP5-3D computer code using the point kinetics function. Calculations show a temperature of 830 °C will be reached with a pulse reactivity insertion of $2.33 for the LEU core. Assuming the most limiting core configuration, the LEU ICIT core at MOL, it is calculated that a reactivity insertion of $2.30 will result in a maximum fuel temperature of 819 °C.

Proposed TS 3.1.4, “Limiting Conditions for Operation – Reactor Core Parameters, Pulse Mode Operation,” for the LEU core limits fuel temperature during pulsing to 830 °C. There are electrical and mechanical interlocks on the transient rod that are set to limit the maximum pulse reactivity insertion to limit fuel temperature to 830 °C.

During start-up testing, the licensee will follow established procedures to perform pulse mode operation tests for the LEU core using a sequence of pulses, starting with a small pulse ($1.10) followed by gradual increases (+$0.10) in reactivity. The maximum pulse reactivity insertion to limit the fuel temperature to 830 °C will then be determined by the licensee.

The staff concludes that the LEU fuel temperature coefficient and the proposed TSs are acceptable. (See also the discussion of pulsing with respect to the thermal-hydraulics (Section 2.5) and TSs (Section 2.9)).

2.4.4 Power Peaking

The power peaking and hot rod thermal power were analyzed using the MCNP5 model for both the HEU and LEU cores. The licensee also analyzed power peaking for three different LEU core configurations, considering changes to the OSTR in-core irradiation facilities. BOL, MOL, and EOL are considered. It is assumed that the hottest spot occurs in the rod with the highest power at the axial position with the highest power density. Hence, the power peaking factor used in the thermal-hydraulic analysis is the product of the hot rod factor times the axial peaking factor where the axial peaking factor is obtained for the hot rod (see Table 2 below). Twenty axial segments and 20 radial segments are used to obtain sufficient detail in the calculation of the axial and radial power distribution in the hot rod. The impact of conversion (see Table 2 below) is that the calculated maximum peaking factors increased from 1.72 for the HEU core at BOL and MOL to 1.82 for the LEU core at MOL with an ICIT. The calculated maximum hot rod thermal power increased from 18.37 kW for the HEU core at MOL to 18.52 kW for the LEU core at MOL.

2.4.5 Conclusions

For the key neutronic characteristics of the OSTR core (i.e., excess reactivity, control rod worths, shutdown margin, reactivity coefficients, and other dynamic parameters) the conversion from HEU to LEU fuel will not cause any significant changes. All changes have been calculated using established methods and are taken into account in the safety analysis and proposed revisions to the TSs. The staff concludes that the changes in nuclear design due to conversion are acceptable.

2.5 Thermal-Hydraulic Design

The OSTR conversion SAR presents the thermal-hydraulic analysis for the HEU and LEU cores that considered: cooling the elements by natural circulation, steady-state operation, pulsing operation, and different times in core life.

2.5.1 Calculational Methodology

In order to carry out the steady state thermal-hydraulic analysis, the calculations were performed using RELAP5-3D. RELAP5-3D was benchmarked for the OSTR by comparing calculated results and current operational data.

RELAP5-3D was used to determine the natural convection flow rate, fuel centerline temperature profile, clad temperature profile, axial temperature profile, and radial fuel temperature distribution. RELAP5-3D also calculated coolant flow rate as a function of rod power and aided in calculating the power of the hottest rod at which critical heat flux is predicted to occur.

The parameters that were provided as inputs for RELAP5-3D include: inlet coolant temperature, system pressure at the top of the core, radial and axial heat source distribution, spacing of heat source nodes, and inlet and exit pressure loss coefficients. The parameters that were produced as outputs from RELAP5-3D include: channel flow rate, axial fuel centerline temperature distribution, axial clad temperature distribution, axial bulk coolant temperature distribution, and axial departure from nucleate boiling ratio (DNBR).

Because the licensee used a documented code that is well-accepted and has been validated at other reactors, including the OSTR, the staff concludes that the calculational methodology used by the licensee is acceptable.

2.5.2 Results of Thermal-Hydraulic Analysis

The steady-state analysis was done at the maximum licensed power of 1.1 MW(t) (the nominal operating power is 1.0 MW(t) with the high power SCRAM normally set at 1.06 MW(t)) and a water inlet temperature of 49 °C (the maximum pool temperature per TS 3.3 b). Results of the steady-state analysis include fuel and coolant temperatures and the minimum departure-from-nucleate-boiling ratio (MDNBR). For the pulse analysis, calculated peak fuel temperature and pulse peak power are presented.

The steady-state analysis considered the maximum power fuel element and the coolant subchannel associated with this single fuel rod. The driving force for the core flow is supplied by the column of water surrounding the core. A natural circulation flow rate is established to balance the driving head against the core entrance and exit pressure losses, and frictional, acceleration and hydrostatic head losses in the core flow channel. The inlet and outlet pressure loss coefficients were calculated by considering a series of abrupt contractions and expansions due to flow channel geometry changes.

The calculation of these local loss coefficients carries uncertainties. The effect of uncertainties in these local loss coefficients on the thermal analysis of the OSTR was evaluated by doubling the inlet and outlet loss coefficients. This resulted in lowering the critical heat flux (CHF) ratio (CHFR) by approximately 5 to 8 percent when the CHF was calculated at the reactor design power. It is observed that there is a larger variation of the CHFR among the different correlations that are commonly used in thermal analysis (at least 50 percent) than there is with the uncertainties in the local loss coefficients. Thus the evaluation of the CHFR is not very sensitive to the exact value of the inlet and outlet pressure loss coefficients.

The thermal-hydraulic calculations conservatively ignored cross flow between adjacent channels. The conservatism was demonstrated by setting up three hydraulic models: a single-channel model (single hot rod), a two-channel model (one hot rod and one average rod), and an eight-channel model (one hot rod and seven average rods). For the two multi-rod models, cross flow was incorporated through junctions connecting adjacent coolant channels at each axial nodal location. CHFR values were calculated for the three models using the Bernath correlation and the 2006 Groeneveld AECL look-up tables. Among the three models, the single-channel model (no cross flow) produced the most conservative result (the smallest CHFR). In the multi-channel models, coolant was diverted to the hot channel resulting in an increase in the minimum CHFR. The differences in the minimum CHFR among the three models were small (less than 1 percent for both the Bernath correlation and the 2006 Groeneveld look-up tables).

Given the inlet water temperature (49 °C), system pressure (absolute pressure at the top of the core), local pressure loss coefficients and the axial and radial power distribution for the bounding fuel element, RELAP5 calculated the natural circulation flow rate; and along the axial length of the flow channel, the coolant temperature, wall heat flux, the clad temperature and the peak fuel temperature. Geometric parameters for the flow channel and the fuel element are given in Table 4-18 of the conversion SAR and the parameters are identical for both the HEU and LEU cores. The flow channels in the OSTR are triangular, square or irregular, depending on core location. The RELAP5 calculation was based on the subchannel flow area for fuel elements in the B Ring which has the smallest rod-to-rod pitch and thus the smallest subchannel flow area in the core. Since the unit subchannel (see Figure 4-23 in the SAR and updated Figure 4-23 in answer to question 33 of RAI response of June 20, 2008) only encompasses half a fuel rod, the calculation was done by doubling the flow area and increasing the power input to the flow channel to that from one fuel rod. The active fuel region is divided into 20 axial nodes, and the corresponding axial nodalization of the coolant channel is shown in Table 4-19 of the conversion SAR. Radially, the annular fuel region is divided into twenty equally spaced layers, as shown in Figure 4-26 and Table 4-20 of the conversion SAR. The conversion SAR notes that the gap between the fuel and the stainless steel clad can vary from 1.27 E-4 to 1.016 E-3 m (0.05 to 0.4 mils).

The maximum powered element is determined using MCNP5 with all control rods, including the transient rod, conservatively removed from the core. For both HEU and LEU cores the location of the hot rod (maximum powered element) varies according to core configuration and core life but it is not in the normal location of the IFE (B1). The hot rod peak factor is defined as the power generation in the hottest rod (element) relative to the core-average rod power generation. Two other power peak factors are defined for the OSTR. They are the hot rod fuel axial peak factor and the hot rod fuel radial peak factor. The axial peak factor represents the axial peak-to-average power ratio within the hot fuel rod, and the radial peak factor represents the peak-to-average power density in a radial plane at the hottest axial location of the hot fuel rod. Using the values listed in table 4-21 of the SAR, the heat flux peaking factors are shown in Table 2 below. The power peaking factors are given for the HEU and LEU cores in Table 4-21 of the conversion SAR (table revised in answer to question 20 of response to request for additional information (RAI) dated June 20, 2008).

For the type of TRIGA fuel used in the OSTR, the gas gap between the fuel and the cladding is known to vary between 0.05 and 0.4 mils (1.27 E-6 to 1.016 E-5 m). The effect of gap size on fuel temperature was evaluated by performing a series of RELAP5-3D calculations assuming a gap varying from 0.05 mils to 0.4 mils. The calculations were also compared to temperatures measured by the IFE. A gap of 0.1 mil was found to give conservative predictions of fuel temperature as compared to measurement (prediction is 17 to 34 degrees C higher than measurement). The same gap size was used in the pulse calculations. The calculations also assumed the RELAP5-3D default composition for the mixture of gases in the gap. This gap composition was chosen because it incorporates fission product built up in fuel elements and will produce a larger thermal resistance compared with air that is present in fresh fuel.

The thermal hydraulic analysis assumed that all fuel rods in the core have approximately the same axial power shape, and thus the maximum powered rod would produce the maximum local heat flux. Table 4-21 also lists an effective peak factor which is a product of the hot rod factor, the axial hot factor and the radial hot factor but this factor has no physical impact on the determination of DNBR and is included only as a qualitative descriptor to identify the core configuration and lifetime which produces the most limiting maximum fuel temperature during steady state and pulse operation. The only peaking factors that have an impact on the heat flux at the clad surface are the hot rod peaking factor and the axial peak factor. The heat flux peaking factor, a product of the hot rod peak factor and the axial peak factor, is defined as the ratio of the heat flux for the hot node to the core average heat flux. Using the values listed in Table 4-21, the heat flux peaking factor is shown in Table 2 below.

It is observed from the heat flux peaking factors in Table 2 that the maximum heat flux will occur in the LEU-MOL ICIT core and the RELAP5 analysis did confirm that that case resulted in the highest calculated fuel temperature. For the HEU steady-state core the heat flux peaking factor is the maximum at BOL. For the LEU core (steady-state) at all stages of core life (beginning, middle, and end) the ICIT core configuration is the most limiting in terms of the heat flux peaking factor.

The core conversion study for the OSTR considered five different correlations for calculating the critical heat flux (CHF). They were the Groeneveld 1986 (available in RELAP5-3D), 1995, and 2006 Look-up Tables, the McAdams correlation, and the Bernath correlation. It was observed that among the three Groeneveld Look-up Tables, the 1986 version produced the least conservative CHFR values while the 2006 version produced the most conservative values.

Table 2. Summary of Hot Rod Peaking Factors

Core Type	Hot Rod Location	Hot Rod Thermal Power *(kW(t))	Hot Rod Peak Factor Pmax/Pavg	Hot Rod Fuel Axial Peak Factor Pmax/Pavg	Heat Flux Peaking Factor
HEU-BOL Normal	B3	18.02	1.393	1.236	1.722
HEU-MOL Normal	B6	18.37	1.420	1.209	1.717
HEU-EOL Normal	B6	16.48	1.273	1.234	1.571
LEU-BOL ICIT	B6	18.47	1.477	1.221	1.803
LEU-BOL CLICIT	B3	17.03	1.362	1.221	1.663
LEU-BOL Normal	B3	17.77	1.422	1.219	1.733
LEU-MOL ICIT	B6	18.52	1.482	1.225	1.815
LEU-MOL CLICIT	B3	17.03	1.363	1.225	1.670
LEU-MOL Normal	B3	17.80	1.424	1.222	1.740
LEU-EOL ICIT	B6	17.61	1.409	1.181	1.664
LEU-EOL CLICIT	C7	16.35	1.308	1.212	1.585
LEU-EOL Normal	B3	17.02	1.362	1.178	1.604

* Hot rod thermal power corresponds to core power of 1.1 MW(t).

Between McAdams and Bernath, it was the Bernath correlation that gave the more conservative CHFR values. Among all correlations, the Bernath correlation produced the most conservative CHFR values.

The licensee chose to use the 2006 AECL Groeneveld Look-up Tables (being the most current method for calculating CHFR) and the Bernath correlation (being the traditional CHF correlation used by many research reactors and the correlation that produced the most limiting CHF values in the OSTR study). In applying the Groeneveld Look-up Tables to the OSTR a set of correction factors is required to adapt the geometry used in the Look-up Tables to the OSTR geometry. The conversion SAR identified three correction factors that are applicable, K1 (hydraulic diameter), K2 (rod bundle effects), and K4 (axial heated position). The other three correction factors K3 (grid spacer effect), K5 (saturated boiling effect) and K6 (down flow effect) were evaluated and found not applicable to the OSTR and were assumed to be equal to 1.0. In evaluating the CHF using both the Groeneveld Look-up Tables and the Bernath correlation, the outer diameter of the fuel rod was used to represent both the heated and wetted diameter.

For the pulse analysis the transient power was calculated by the point kinetics model in RELAP5-3D. The reactor was modeled with two separate hydraulic channels, the hot rod channel (same as for the steady-state analysis) and an average channel representing the rest of the fuel elements. The only reactivity feedback incorporated in the pulse analysis was the prompt negative fuel temperature coefficient. The fuel temperature used in evaluating the feedback effect is a volume-weighted average temperature of the fuel portion of the heat structure in the RELAP5-3D model of the OSTR core. Analysis has shown that the time lag in heat transfer from the fuel to the moderator makes the moderator feedback inconsequential. The RELAP5-3D analysis of the pulse transient also conservatively neglected direct gamma heating of the moderator. The pulsing analysis was performed with the assumption that the reactor core maintains the same power shape as the steady state. This is considered to be conservative because the hotter rods are expected to experience stronger negative fuel temperature feedback during a pulse, and the core-wide power profile should be flatter than the steady state power profile.

The HEU core at BOL is the most limiting in terms of heat flux and the steady-state results from the RELAP5-3D calculation are presented in Table 4-22 of the conversion SAR. The predicted MDNBR for a reactor power of 1.1 MW(t) is 2.104 and 4.844 by using the Bernath correlation and the Groeneveld look-up table, respectively.

The limiting LEU core is the ICIT core configuration at MOL. The steady-state results for a 1.1 MW(t) operating power are summarized in Tables 4-24, 4-26 and 4-28 of the conversion SAR for BOL, MOL, and EOL respectively. The predicted MDNBR for a reactor power of 1.1 MW(t) is 2.06 and 4.754 by using the Bernath correlation and the Groeneveld look-up table, respectively. The hot rod power is 18.52 kW(t) (core power of 1.1 MW(t)). The MDNBR according to the Bernath correlation will not reach a value of 2.0 (a standard design margin) until the hot rod produces a power of approximately 19.85 kW(t). That is a 7.2 percent margin above the maximum licensed power of OSTR. In all three cases the LEU core has a slightly lower MDNBR (but still above 2.0) than the corresponding values for the HEU core.

The staff concludes that the thermal-hydraulic analysis for the OSTR conversion adequately demonstrates that the conversion from an HEU to LEU core will result in no significant decrease in safety margins in regard to thermal-hydraulic conditions. The analyses were done with qualified calculational methods and conservative or justifiable assumptions. The applicability of the analytical methodology is demonstrated by comparing analytical results with measurements obtained from the HEU core. In comparing the HEU and LEU cores the thermal-hydraulic analyses have accounted for differences in core power distribution and physics parameters due to changes in the design of the fuel.

2.6 Accident Analysis

The conversion SAR analyzed three hypothetical accidents: the maximum hypothetical accident (MHA), insertion of excess reactivity, and loss of coolant accident (LOCA). The conversion SAR also considered other accidents that are identified in NUREG-1537.

2.6.1 Maximum Hypothetical Accident

The OSTR maximum hypothetical accident (MHA) is defined as a cladding rupture of one highly irradiated fuel element in air with no radioactive decay, followed by the instantaneous release of the noble gases and halogen fission products directly into the air. Boundary conditions and assumptions included an infinite irradiation time, no credit for iodine absorption in the reactor pool water, released fission products are uniformly distributed in the reactor room air, and 100 percent of the noble gases and 25 percent of the iodine inventory become instantaneously released and airborne. The occupational dose was calculated for an individual in the reactor room and the doses at various distances from the OSTR (the closest distance being 33 ft) were selected for the radiological exposure to the public by the licensee. Review of the methodology showed it was consistent with NUREG-1537 guidance and adequate to calculate occupational and public radiation doses. Assumptions and boundary conditions used were consistent with accepted nuclear industry practices and representative of the proposed new LEU fuel.

Radioactivity releases from OSTR operations can only occur if the fuel cladding is breached. The licensee analytically generated a radionuclide inventory for the MHA for both the HEU and LEU fuel. Using these inventories, three different scenarios were evaluated for each fuel type. Variation of the leakage rate was the only difference between the three scenarios evaluated.

Review of the calculated results for the MHA using both the HEU and LEU fuel source terms showed the occupational and public exposures were similar. For HEU fuel, the reactor room occupational total effective dose equivalent (TEDE), which is based on a 5-minute building evacuation time for the OSTR staff, was calculated to be 22 mrem. This is within the regulatory limit given in 10 CFR 20.1201(a)(1)(i) of 5,000 mrem. The thyroid dose (sum of the deep-dose equivalent and the committed dose equivalent) was calculated to be 575 mrem. This is within the regulatory limit given in 10 CFR 20.1201(a)(1)(ii) of 50 rems (50,000 mrem). The licensee stated that 5 minutes is a reasonable amount of time to perform an evacuation including the time it takes to check out systems to eliminate a potential false alarm. For the LEU fueled reactor the calculated values are 22 mrem TEDE and 580 mrem thyroid dose. These doses are also well within the regulatory limits.

For the public exposure doses, the calculated TEDE for the scenario with the highest doses are 17 mrem for both the HEU and LEU fuel. The dose at the nearest office building is 3 mrem and at the nearest residence is less than 1 mrem for both the HEU and LEU fuel. All the analyzed cases demonstrated that doses are within the regulatory limit given in 10 CFR 20.1301 for doses for individual members of the public of 100 mrem.

The staff finds that the results of the MHA with the reactor fueled with LEU fuel is similar to that for the HEU fueled reactor. Because the doses are within the regulations, the results of the MHA continue to be acceptable to the staff.

2.6.2 Insertion of Excess Reactivity

Pulsing of the OSTR is discussed above in Section 2.4.3. For the accident analysis, beam port flooding and uncontrolled withdrawal of control rods are considered. Conversion from HEU to LEU fuel does not change the analysis of beam port flooding which was previously shown to lead to acceptable results.

Uncontrolled withdrawal of a control rod was reanalyzed for the LEU core taking into account the changes in rod worth and scram timing. The resulting reactivity insertion was, in all cases, less than the reactivity normally inserted under pulsing operation (and less than $1.00). Even in the unlikely event that all control rods simultaneously withdraw, the inserted reactivity is only $1.13; again, less than what is inserted in pulsed operation.

The staff concludes that the results of accidents involving the insertion of excess reactivity into the core continues to be acceptable with the conversion to LEU fuel.

2.6.3 Loss-of-Coolant Accident (LOCA) Analysis

The scenarios leading to a LOCA are not affected by the type of fuel present in the reactor and, therefore, the analysis previously done for the HEU core is applicable to the LEU core. The analysis assumes all water is lost from the pool. The power density of the maximum power fuel element in the LEU core is 18.52 kW(t) for a reactor power of 1.1 MW(t) which is less than the limiting power density of 23.5 kW(t). Natural convective air cooling of the fuel is found to be sufficient to keep the fuel temperature below the temperature for cladding failure even with conservative assumptions about core nominal power and operating time.

The main consequence from a catastrophic loss of coolant is the gamma ray dose from the exposed core. The licensee calculated doses at the top of the reactor pool and on the floor of the reactor room. The calculated dose rates for the LEU fueled reactor are less than that calculated for the HEU fueled reactor. While the licensee did not calculate doses in the unrestricted environment for the LEU fueled reactor, because the LEU fuel doses are lower than the HEU fuel doses inside the reactor building, the HEU fuel doses in the environment will bound the LEU fuel doses. The doses to the public for the HEU fueled reactor were within the regulations and acceptable. The LEU fuel doses will likewise be within the regulations and acceptable.

The staff concludes that the results of the LOCA for the LEU fueled reactor are similar to that for the HEU fueled reactor and continue to be acceptable.

2.6.4 Other Accidents

Other accidents have been considered for the HEU core and do not change for the LEU core. They have all been shown to either lead to acceptable results, to be bounded by the MHA, or not to be possible. The events discussed in the conversion SAR that fall into this category are:

- Loss of Coolant Flow
- Mishandling or Malfunction of Fuel
- Loss of Normal Electrical Power
- External Events
- Mishandling or Malfunction of Equipment

2.6.5 Conclusions

The staff has reviewed the radiation source term and MHA calculation for the HEU to LEU fuel conversion at the OSTR. Review of the calculations, including assumptions, demonstrated that the inventory of radioactivity assumed and other boundary conditions used in the analysis were acceptable. The radiological consequences to the public and occupational workers at the OSTR from a postulated MHA for the proposed LEU fueled reactor are similar to the radiological consequences calculated for the HEU-fueled reactor, which were in conformance with 10 CFR Part 20 requirements. As a result of this review, the staff has concluded that continued operation of the reactor poses no undue risk from a radiological standpoint to the public or the staff of the OSTR from the maximum hypothetical accident.

The conversion does not change the assumptions, analyses or consequences of a LOCA and hence, since the consequences were acceptable for HEU fuel, they are acceptable for LEU fuel.

The licensee did not identify any new reactivity addition accidents not previously analyzed for the HEU-fueled reactor. The design features and administrative restrictions that prevent accidental pulsing from occurring at full power and preclude damage if such pulsing occurs continue to exist for the LEU core. The licensee also found that other accidents considered for HEU fuel either lead to acceptable results, are bounded by the MHA, or are not possible for the core fueled with LEU fuel. Therefore, risk to the health and safety of reactor staff and the public does not increase above that previously found acceptable for the HEU core from reactivity addition accidents or from the list of accidents discussed in Section 2.6.4 above.

2.7 Fuel Storage

The licensee will fabricate and install a sufficient number of fuel storage racks to store the existing HEU fuel and the new LEU fuel. The licensee was originally planning to store both HEU and LEU fuel in similar storage racks, as discussed in its conversion safety analysis report. As described in its February 11, 2008, response to the NRC's RAI, the licensee subsequently decided to fabricate different storage racks for the HEU and LEU fuel. The conversion safety analysis report discusses the storage racks for the HEU fuel, while the licensee's answer to RAI No. 2 describes the LEU storage racks. The staff reviewed information about both storage racks.

In accordance with the existing TS 5.5(a), all reactor fuel assemblies shall be stored in a geometric array where the multiplication factor, k-eff, is less than 0.9 for all conditions of moderation.

For the HEU storage racks, the licensee provided information on the storage of the HEU and LEU fuel in the racks. The racks are on a square pitch of 8.0 centimeters. The licensee modeled five racks, each with a capacity of two rows of 20 fuel elements. The licensee analyzed three cases with the storage racks in water—storage of all HEU fuel in the racks, storage of all LEU fuel in the racks, or storage of an equal mixture of HEU fuel and LEU fuel in the racks. The licensee's analysis showed that the TS limit is satisfied with a considerable margin under all three cases. The analysis showed a k-eff of less than 0.548 for the case with HEU fuel only, a k-eff of 0.612 for the case with LEU fuel only, and a k-eff of 0.577 for the case with an equal mixture of both fuel types. A critical array would have a k-eff of 1.000. The licensee also performed calculations for the same cases with the storage racks in air. The multiplication factors were reduced as compared with water for the three fuel cases. These values are below the multiplication factor limit of 0.9, so the fresh and spent fuel elements can be stored safely in these storage racks.

For the LEU storage racks, the licensee modeled an infinite two-dimensional array of fresh LEU fuel elements with a 10 centimeter fuel storage lattice pitch. The use of an infinite array provided conservative results as compared to a finite array. If air surrounds the fuel, k-infinity is 0.668 (k-infinity in this case can be compared against the TS k-eff); if water surrounds the fuel, k-infinity is 0.646. These values are below the multiplication factor limit of 0.9, so fresh fuel elements can be stored safely in these storage racks.

The licensee used the Monte Carlo N-Particle Transport code MCNP. This is a state-of-the-art code frequently used for this type of analysis. The staff has reviewed the licensee's use of the code for all other aspects of its conversion analysis and found that it is knowledgeable in the application of the code. Hence, the staff has a high level of confidence in the licensee's application of the code and in the results.

The staff has reviewed the licensee's analysis and concludes that TS 5.4.a will continue to be met. Therefore, the staff finds the fuel storage to be acceptable.

2.8 LEU Startup Plan

With the LEU 30/20 fuel elements, the core will contain a total of 85 fuel elements. The licensee plans to use a standard 1/M plot for loading of a critical assembly. The licensee's calculations show that the LEU core should go critical when 60-70 fuel elements have been loaded. This is a standard technique for approaching initial criticality.

The startup plan was used previously for loading HEU fuel at the OSTR and is similar to those used in other reactors. The licensee's Reload and Startup Guidelines are found in Appendix A of the conversion SAR. The staff concludes that the licensee's procedures are sufficiently detailed to result in the safe loading of the reactor with the LEU fuel.

The licensee is to submit a start-up report to the NRC on the results of the start-up testing. The licensee's Startup Acceptance Criteria is found in Appendix B of the conversion SAR. The staff concludes that the licensee's testing program will provide verification of key LEU reactor functions and, therefore, is acceptable.

2.9 Proposed Changes to License Conditions and Technical Specifications

For the OSTR HEU to LEU conversion, the licensee has proposed changes to the license conditions for special nuclear material possession limits and TSs.

2.9.1 Proposed Changes to License Conditions

License condition 2.B.(2) is changed to reflect receipt, possession, and use of special nuclear material after conversion. The license conditions currently reads as follows:

2.B.(2) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material,"

a. to receive, possess, but not use, in connection with operation of the facility, up to 16.30 kilograms of contained uranium-235 enriched to less than 20 percent in the form of TRIGA reactor fuel;

b. to receive, possess, but not use, up to 656 grams of uranium-235 enriched to less than 20 percent in the form of the core from the AGN-201 reactor;

c. to receive, possess, use, but not separate, in connection with operation of the facility, such special nuclear material as may be produced by operation of the facility; and

d. to receive, possess and use in connection with operation of the facility up to 12.83 kilograms of contained uranium-235.

Based on the licensee proposed possession limits, the license condition reads as follows:

2.B.(2) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material,"

a. to receive, possess and use, in connection with operation of the facility, up to 16.30 kilograms of contained uranium-235 enriched to less than 20 percent in the form of TRIGA reactor fuel;

b. to receive, possess and use, in connection with operation of the facility, up to 100 grams of contained uranium-235 of any enrichment in the form of fission chambers and flux foils;

c. to receive, possess, but not use, up to 656 grams of uranium-235 enriched to less than 20 percent in the form of the core from the AGN-201 reactor;

d. to receive, possess, use, but not separate, in connection with operation of the facility, such special nuclear material as may be produced by operation of the facility; and

e. to possess, but not use, up to 12.83 kilograms of contained uranium-235 at equal to or greater than 20 percent enrichment in the form of TRIGA fuel until the existing inventory of this fuel is removed from the facility.

License condition 2.B.(2)a. is amended to allow use of the LEU TRIGA fuel needed for the conversion. This license condition was added to the license by order dated April 4, 2008, as part of the conversion process to allow the licensee to possess the LEU fuel needed for conversion prior to this order. After the reactor is converted, the licensee has a continuing need to receive, possess, and use small amounts of HEU to allow continued operation of the reactor (e.g. fission chambers) and conduct of the experimental program (e.g. flux foils). A new license condition 2.B.(2)b. is added with a possession limit of up to 100 grams of contained uranium-235 of any enrichment in the form of fission chambers or flux foils. (The licensee had also requested uranium in the form of fueled experiments for this license condition. As part of license renewal being evaluated by the staff during the same timeframe as this conversion, fuel experiments were removed from the TS. Removing fueled experiments from this license condition was discussed with and agreed to by the OSTR Facility Director during a telephone conversation on August 22, 2008.) Because of the addition of new license condition 2.B.(2)b., the existing license conditions are re-lettered. License condition 2.B.(2)e. is revised to allow possession, but not use, of the existing HEU fuel until it is removed from the licensee's site.

License condition 2.C.(2), which incorporates the TSs into the license, is changed to incorporate the TS changes needed for conversion as discussed below into the license.

The staff has reviewed the proposed possession limits associated with the conversion of the reactor and concludes that the limits are appropriate for the converted reactor.

2.9.2 Proposed Changes to Technical Specifications

TS 2.2, "Limiting Safety System Setting" (LSSS): TS 2.2 states that the LSSS shall be equal to or less than 510 °C (950 °F) as measured in an instrumented fuel element (IFE) and that the IFE shall be located in the B-ring. The licensee has not proposed any changes to the LSSS. The licensee has proposed changes to the basis for TS 2.2 to reflect the LEU core operating characteristics. Because the proposed changes reflect the characteristics of the proposed LEU fuel, the staff finds these proposed changes to the basis for TS 2.2 to be acceptable.

TS 3.1.4, "Pulse Mode Operation:" The licensee has proposed changes to TS 3.1.4, and its basis to change the limiting condition during pulsing from reactivity insertion to maximum fuel element temperature. The specification currently reads:

3.1.4 The reactivity to be inserted for pulse operation shall be determined and limited by a mechanical block and electrical interlock on the transient rod, such that the reactivity insertion shall not exceed $2.55.

The licensee has proposed changing the specification to read:

3.1.4 The reactivity to be inserted for pulse operation shall be determined and limited by a mechanical block and electrical interlock on the transient rod, such that the maximum fuel element temperature shall not exceed 830 °C.

The TS limits the reactivity to be inserted for pulse operation. The amount of inserted reactivity shall not produce a peak fuel temperature of 830 °C or above. It is also proposed to revise the basis of the technical specification to reflect limiting the pulse based on maximum fuel element temperature. A basic limit for TRIGA fuel is dictated by the dissociation of hydrogen from the uranium-zirconium hydride fuel. The maximum fuel temperature is based on limiting the pressure of the dissociated hydrogen for the core to restrict the reactivity limit for the pulse mode of operation. It has been observed that after extensive steady-state operation at higher power levels (1 MW(t)), hydrogen in the TRIGA hydride fuel will migrate from the central high temperature regions to the cooler outer regions of the fuel element. If the fuel temperature exceeds 874 °C during pulsing the hydrogen pressure will be sufficient to cause expansion of the microscopic holes in the fuel that grows with each pulse. This could lead to fuel damage such as fuel bowing. A decrease in fuel temperature from 874 °C to 830 °C reduces the hydrogen pressure by a factor of two and provides an acceptable safety factor. Because the proposed maximum fuel temperature will help protect the fuel cladding from damage it is acceptable to the staff.

TS 3.2.3, "Reactor Safety System:" The licensee has proposed changes to TS 3.2.3, Table 3 – Minimum Interlocks, and affected basis (e.g. 1 kW Pulse Interlock; Shim, Safety, and Regulating Rod Drive Circuit; and Transient Rod Cylinder Position Interlock) to address limitations on reactivity insertion based on maximum fuel temperature. The affected portion of the specification currently reads:

Table 3 – Minimum Interlocks

Interlock	Function	Effective Mode		
		S.-S.	Pulse	S.-W.
Transient Rod Cylinder Position	Prevents pulse insertion of reactivity greater than $2.55	-	1	1

The licensee has proposed changing the affected portion of the specification to read:

Table 3 – Minimum Interlocks

Interlock	Function	Effective Mode		
		S.-S.	Pulse	S.-W.
Transient Rod Cylinder Position	Prevents pulse insertion of reactivity greater than that which would produce a maximum fuel element temperature of 830 °C	-	1	1

The changes for the specification and basis from a reactivity insertion limit to a maximum fuel temperature is consistent with the proposed changes as discussed above to TS 3.1.4 and, therefore, are acceptable to the staff.

TS 3.8, “Limitations on Experiments – Reactivity Limits:” The licensee has proposed changes to TS 3.8.1 and its basis to address limitations on reactivity insertion based on maximum fuel temperature. The affected portion of the specification currently reads:

> 3.8.1.b The sum of the absolute values of the reactivity worths of all experiments shall be less than $2.55.

The licensee has proposed changing the affected portion of the specification to read:

> 3.8.1.b The sum of the absolute values of the reactivity worths of all experiments shall be less than $2.30.

The changes to the specification and basis reduce the reactivity limit to be consistent with the maximum fuel temperature as discussed above in TS 3.1.4 and, therefore, are acceptable to the staff. The licensee had originally proposed changing this TS to a temperature limit similar to the other changes above. During a telephone discussion on August, 25, 2008, between the Facility Director and the NRC Senior Project Manager it was agreed to base the experiment limit on reactivity.

TS 5.3.3, “Reactor Fuel:” The licensee has proposed changes to TS 5.3.3 (1) and (3) and its basis to reflect the changes in the fuel alloy. The affected portions of the specifications currently read:

1. Uranium content: maximum of 9 wt% enriched to a nominal 70% ^{235}U;

3. Natural erbium content (homogeneously distributed): between 1.1 and 1.6 wt%;

The licensee has proposed changing the affected portions of the specifications to read:

1. Uranium content: nominal 30 wt% enriched to less than 20% ^{235}U;

3. Natural erbium content (homogeneously distributed): nominal 1.1 wt%;

The licensee had proposed in TS 5.3.3 (3) limiting uranium-235 enrichment to a nominal 20 percent. During a discussion between the NRC Project Manager and the OSTR Director on August 22, 2008, it was agreed to change the specification to less than 20 percent to clearly reflect the definition of LEU. Because the proposed changes reflect the characteristics of the proposed LEU fuel, the staff finds these proposed changes to TS 5.3.3 to be acceptable.

2.9.3 Conclusions

The staff has reviewed all of the proposed changes to the TSs. The staff concludes that these changes to the TSs are needed for the conversion of the reactor to LEU fuel. The licensee has justified the technical bases for these changes as discussed above. The staff concludes that the changes to the TSs continue to meet the regulations in 10 CFR 50.36 and that the changes to the TSs are, therefore, acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

In accordance with 10 CFR 51.10(d), an Order is not subject to Section 102 of the National Environmental Policy Act. The NRC staff notes, however, that even if these changes were not being imposed by an order, pursuant to 10 CFR 51.22(b), the changes would not require an environmental impact statement or environmental assessment.

The changes involve the use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes in inspection and surveillance requirements. The NRC staff has determined that the changes involve no significant hazards consideration, no significant increase in the amounts, and no significant change in the types, of any effluents that may be released off site, and no significant increase in individual or cumulative occupational radiation exposure. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Therefore, pursuant to 10 CFR 51.10 (d) or 51.22 (b), no environmental impact statement or environmental assessment is required.

4.0 CONCLUSIONS

The NRC staff has reviewed and evaluated the operational and safety factors affected by the use of LEU fuel in place of HEU fuel in the OSTR. The staff has concluded, on the basis of the considerations discussed above that (1) the proposal by the licensee for conversion of the reactor to LEU fuel is consistent with and in furtherance of the requirements of 10 CFR 50.64; (2) the conversion, as proposed, does not involve a significant hazards consideration because the amendment does not involve a significant increase in the probability or consequences of accidents previously evaluated, create the possibility of a new kind of accident or a different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety; (3) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed activities; and (4) such activities will be conducted in compliance with the Commission's regulations, and the issuance of this Order will not be inimical to the common defense and security or the health and safety of the public. Accordingly, it is concluded that an enforcement order as described above should be issued pursuant to 10 CFR 50.64(c)(3) for the conversion of the OSTR.

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