EA-09-141

June 11, 2009

Robert J. Agasie, Director
Nuclear Reactor Laboratory
University of Wisconsin – Madison
1513 University Avenue, Room 1215 ME
Madison, WI 53706-1687

SUBJECT: ISSUANCE OF ORDER MODIFYING LICENSE NO. R-74 TO CONVERT FROM HIGH- TO LOW-ENRICHED URANIUM FUEL (AMENDMENT NO. 17) – UNIVERSITY OF WISCONSIN NUCLEAR REACTOR (TAC NO. MD9592)

Dear Mr. Agasie:

The U.S. Nuclear Regulatory Commission (NRC) is issuing the enclosed Order, as Amendment No. 17 to Facility Operating License No. R-74, which authorizes the conversion of the University of Wisconsin Nuclear Reactor from high-enriched uranium fuel to low-enriched uranium (LEU) fuel. This Order modifies the license, including the technical specifications and emergency plan, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.64, "Limitations on the Use of Highly Enriched Uranium (HEU) in Domestic Nonpower Reactors." This regulation requires that non-power reactor licensees, such as the University of Wisconsin, convert to LEU fuel under certain conditions which the University of Wisconsin now meets. The Order is being issued in accordance with 10 CFR 50.64(c)(3) and in response to your submittal of August 25, 2008, as supplemented on April 10, 2009, May 1, 2009, and June 4, 2009. 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 portion of the Order that changes License Condition 2.B, to allow possession of the LEU fuel, will become effective 20 days after the date of its publication in the *Federal Register*, provided there are no requests for a hearing. The portions of the Order that change License Condition 3.B modifying the technical specifications to be applicable to LEU fuel, and change portions of the facility emergency plan to be applicable to LEU fuel become 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, technical specifications, and emergency plan 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/

William C. Schuster IV, Project Manager
Research and Test Reactors Branch A
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Docket No. 50-156

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

cc w/o enclosure 4: See next page

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NAME	WSchuster	GLappert	KBrock	BMizuno	NHilton (DFurst for)
DATE	5/19/09	5/27/09	5/27/09	6/4/09	6/8/09
OFFICE	DPR/D	QTE*	NRR/D	PRTA/PM	
NAME	TMcGinty	KAzariah-Kribbs	ELeeds	WSchuster	
DATE	6/9/09	6/1/09	6/11/09	6/11/09	

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cc:

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FACILITY OPERATING LICENSE NO. R-74

DOCKET NO. 50-156

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
License Page 2	License Page 2
License Page 3	License Page 3

Accordingly, License No. R-74, as amended, is hereby amended in its entirety, effective as of the date of issuance, to read as follows:

1. The license applies to the University of Wisconsin's nuclear reactor with the TRIGA nuclear core and control system (herein "the reactor") owned by the University of Wisconsin (herein "the licensee"), and located on the University's campus in Madison, Wisconsin, and described in the licensee's application for license dated July 13, 1966, and amendments thereto including the amendment dated June 6, 1973, and supplements dated August 1, and August 21, 1973, (herein "the application").

2. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the University of Wisconsin:

   A. Pursuant to Section 104c of the Act and Title 10, CFR, Chapter I, Part 50, "Licensing of Production and Utilization Facilities," to possess, use and operate the reactor in accordance with the procedures and limitations described in the application and in this amendment;

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

      (1) to receive, possess and use, in connection with operation of the facility, up to 15.0 kilograms of contained uranium-235 enriched to less than 20 percent in the form of TRIGA reactor fuel;

      (2) to receive, possess and use, in connection with operation of the facility, up to 150 grams of contained uranium-235 of any enrichment in the form of neutron detectors;

      (3) to receive, possess and use, in connection with operation of the facility, up to 16 grams of contained plutonium in the form of plutonium-beryllium neutron source;

      (4) 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

      (5) to possess, but not use, up to 18.0 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.

C. Pursuant to the Act and Title 10, CFR, Chapter I, Part 30, "Rules of General Applicability to Licensing of Byproduct Material," to possess, but not separate, such byproduct material as may be produced by operation of the reactor.

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: (Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70;) 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:

   A. Maximum Power Level

   The licensee may operate the reactor at steady state power levels up to a maximum of 1000 kilowatts (thermal).

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 17, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   C. Physical Security Plan

   The licensee shall fully implement and maintain in effect all provisions of the physical security plan approved by the Commission and all amendments and changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p), respectively. The approved plan, which is exempt from public disclosure pursuant to the provisions of 10 CFR 2.790(d) and 10 CFR 73.21, is entitled, "University of Wisconsin Nuclear Reactor Security Plan," Revision 4, submitted by letter dated June 17, 1991.

4. This amendment is effective as of the date of issuance and shall expire at midnight June 7, 2000.

FOR THE ATOMIC ENERGY COMMISSION

/RA/

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Attachment: Appendix "A" (Change No. 4
to the Technical Specifications)

Amendment No. 17
June , 2009

Date of Issuance: February 4, 1974

FACILITY OPERATING LICENSE NO. R-74

DOCKET NO. 50-156

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

Remove	Insert
TS Page Table of Contents	TS Page Table of Contents
TS Page TS 3	TS Page TS 3
TS Page TS 4	TS Page TS 4
TS Page TS 7	TS Page TS 7
TS Page TS 8	TS Page TS 8
TS Page TS 9	TS Page TS 9
TS Page TS 11	TS Page TS 11
TS Page TS 13	TS Page TS 13
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TS Page TS 28	TS Page TS 28
TS Page TS 29	TS Page TS 29
TS Page TS 31	TS Page TS 31
TS Page TS 32	TS Page TS 32

FACILITY OPERATING LICENSE NO. R-74

DOCKET NO. 50-156

Replace the following pages of the University of Wisconsin Nuclear Reactor Emergency Plan (EP) with the enclosed pages.

Remove	Insert
EP Page 1	EP Page 1, Rev 6
EP Page 5	EP Page 5, Rev 7
EP Page 6	EP Page 6, Rev 6
EP Page 7	EP Page 7, Rev 6
EP Page 13	EP Page 13, Rev 7
EP Page 14	EP Page 14, Rev 7
EP Page 15	EP Page 15, Rev 7

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING THE ORDER TO CONVERT FROM

HIGH-ENRICHED TO LOW-ENRICHED URANIUM FUEL

FACILITY OPERATING LICENSE NO. R-74

UNIVERSITY OF WISCONSIN NUCLEAR REACTOR

DOCKET NO. 50-156

1.0 INTRODUCTION

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.64, "Limitations on the Use of Highly Enriched Uranium (HEU) in Domestic Nonpower Reactors," requires licensees of research and test reactors to convert from the use of HEU fuel to low-enriched uranium (LEU) fuel, unless specifically exempted. University of Wisconsin (the licensee) has proposed to convert the fuel in the University of Wisconsin Nuclear Reactor (UWNR) from HEU to LEU. In a letter dated August 25, 2008, as supplemented by letters dated April 10, 2009, May 1, 2009, and June 4, 2009, the licensee submitted its proposal for conversion requesting approval of the fuel conversion and of changes to its technical specifications (TS). To support this action the licensee submitted a conversion safety analysis report (SAR) on which the HEU to LEU conversion and the TS changes are based. The licensee also submitted proposed changes to the UWNR Emergency Plan. This safety evaluation report provides the results of the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the licensee's conversion proposal. The evaluation was carried out according to the guidance found in NUREG-1537[1].

2.0 EVALUATION

2.1 Summary of Reactor Facility Changes

The UWNR is a TRIGA-conversion reactor, similar in design to many others operating in the United States and abroad. The reactor was originally designed for Material Testing Reactor (MTR)-type fuel assemblies. General Atomics (GA) developed a fuel system with fuel assemblies that can hold up to four TRIGA fuel elements each. These fuel assemblies were designed to replace the MTR plate-type fuel assemblies.

The reactor normally operates at a maximum thermal power level of 1 megawatt (thermal) (MW(t)). The reactor uses natural convection for cooling. It is presently fueled with highly

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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.

enriched (70 percent uranium-235) TRIGA fuel lifetime improvement program (FLIP) fuel. The new conversion LEU fuel will have a nominal enrichment of 19.75 percent uranium-235 in a fuel element with the same geometry and with the same cladding as the FLIP fuel. The HEU-to-LEU conversion 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 calculations performed by the licensee indicate that four additional graphite reflectors will be inserted in the core as part of the conversion. This is being done to (1) reduce the number of fuel bundles to meet TS 3.1 “Reactivity Limitations,” and (2) increase the core lifetime from 150 to 1,800 megawatt-days (MWd). These new reflectors will be constructed to conform to the specifications of the present reflectors. The impact of this change on the operation or safety of the LEU core was reviewed and found acceptable by the staff in Section 2.4.1, “Calculational Methodology” and Section 2.4.2, “Excess Reactivity, Control Element Worth, and Shutdown Margin,” of this report.

The fuel assembly hardware (e.g. bottom adapter, top handle, locking plate, bolts) and transient rod guide tube will be replaced as part of the conversion. This is being done to minimize the radiation dose to the staff in accordance with the as-low-as-reasonably-achievable, or ALARA, principle. The new hardware and guide tube will be constructed to conform to the specifications of the present hardware and guide tube. The impact of this change on the operation or safety of the LEU core was reviewed by the staff and determined to be inconsequential.

2.2 Comparison with Similar Facilities Already Converted

Similar TRIGA conversion reactors at Texas A&M University and Washington State University converted using the same design of LEU fuel elements as proposed for the UWNR conversion. In addition, the TRIGA Mark F reactor at GA in San Diego, CA, 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 of the fuel assemblies or 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 1.48 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 0.9 w/o, and 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 element has a greater uranium mass than the HEU fuel element. The additional mass is uranium-238. The main fuel characteristics are shown in Table 1.

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

Design Data	HEU FLIP	LEU 30/20
Uranium content [w/o]	8.5	30
U-235 enrichment [w/o]	70	19.75
Erbium content [w/o]	1.48	0.9
Fuel alloy inner diameter [mm (in)]	6.35 (0.25)	6.35 (0.25)
Fuel alloy outer diameter [mm (in)]	34.798 (1.37)	34.798 (1.37)
Fuel alloy length [mm (in)]	381.0 (15)	381.0 (15)
Cladding material	Type 304 SS	Type 304 SS
Cladding thickness [mm (in)]	0.508 (0.02)	0.508 (0.02)
Cladding outer diameter, mm (in)	35.8349 (1.411)	35.8349 (1.411)

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 NRC (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 UWNR.

The erbium poison content 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 0.9 w/o is within the erbium poison content range of 0.0 to 1.8 w/o approved for use.

Due to the increase in uranium content from the HEU core to the LEU core, the w/o or amount of ZrH_x is reduced, meaning there is less hydrogen in the fuel. The hydrogen is one of the main reasons for the very large negative fuel temperature coefficient of reactivity. The reduced hydrogen content changes the value of the fuel temperature coefficient of reactivity, which is discussed in Section 2.4.3, “Dynamic Parameters,” of this report.

Calculations indicate that there will be 83 fuel elements in the LEU core compared to the present 91 in the HEU core. This is due to the removal of two four-element fuel assemblies. As discussed in Section 2.1, “Summary of Reactor Facility Changes,” graphite reflectors will be inserted in place of those assemblies. The LEU core will also have two instrumented fuel elements (IFE), which is a fuel element containing thermocouples to measure fuel temperature.

The initial LEU core is expected to go critical when the core contains either: (1) 16 fuel assemblies loaded with 14 graphite reflectors installed, or (2) 18 fuel assemblies loaded with no graphite reflectors installed. Note that this analysis assumes that the fuel assemblies will be loaded in a symmetric pattern from the center of the reactor core.

The UWNR control system consists of five control elements: three Boral shim-safety blades, one stainless steel regulating blade, and a water-followed boron carbide transient rod. The HEU to LEU conversion will not require replacement or modification of these control elements; however, the guide tube for the transient rod will be replaced at the time of conversion.

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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.

Using REBUS-MCNP[3], the licensee calculated middle-of-life (MOL) and end-of-life (EOL) for both the HEU and LEU cores, assuming operation at a constant power of 1 MW(t) and no fuel shuffling to extend the life of the core. 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 1,400 MWd for MOL and 2,800 MWd for EOL. An initial analysis of the LEU core was conducted using one of the current approved HEU operational core geometries with 12 graphite reflectors. Results of that analysis showed EOL or an excess reactivity of 0.5 %Δk/k at approximately 150 MWd. Additional licensee calculations performed showed that core life could be extended to 1,800 MWd by using 14 graphite reflectors. All of the analysis for the LEU core was conducted using 14 graphite reflectors. The values for the LEU core are 800 MWd for MOL and 1,800 MWd for EOL. The values for MOL and EOL for both the HEU and LEU cores were then used in the analysis of all significant parameters.

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

2.4 Nuclear Design

This section of the safety evaluation discusses the nuclear design of the reactor and the impact that conversion will have.

2.4.1 Calculational Methodology

In order to carry out the UWNR neutronic analysis, a Monte Carlo N-Particle (MCNP) model was developed by the licensee using MCNP5[4]. REBUS-MCNP, an Argonne National Laboratory (ANL) computer code developed to analyze reactor fuel cycles, was used for the burnup and inventory analyses. Two different cores were modeled for the analysis. The first is the present core, consisting of 91 HEU FLIP fuel elements, and the second contains 83 new 30/20 LEU fuel elements. The first model was used to benchmark the analytical methods and the second was used to predict what will happen after conversion.

The licensee benchmarked the MCNP core model by comparing the calculated and measured values of the three control blades, regulating blade, and transient rod. The fully inserted position of the transient rod is used to determine the worths of the control and regulating blades. The fully inserted position of the transient rod for the HEU core was calculated to be 22.83 centimeters (cm) (8.99 inches), and agrees well with the measured value of 21.84 cm (8.60 inches). Two methods were used to calculate the transient rod worth in this position. Using MCNP to model the transient rod worth using the rod drop methodology of measuring reactivity, the calculated worth of the transient rod for the HEU core is 1.334 %Δk/k ± 0.0453 %Δk/k. Using MCNP to model the transient rod worth using the rising period rod bump methodology of measuring reactivity, the calculated integral worth of the transient rod for the HEU core is 1.467 %Δk/k ± 0.105 %Δk/k. Both values agree well with the measured value for the HEU core of 1.374 %Δk/k. The fully-inserted position of the transient rod for the LEU core was calculated to be 24.94 cm (9.82 inches). The calculated integral worth of the transient rod for the LEU core is 1.369 %Δk/k.

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3 "The REBUS-MCNP Linkage", J.G. Stevens, ANL (Draft), REBUS-MCNP is a computer code under development at ANL.

4 "MCNP – A General Monte Carlo N-Particle Transport Code, Version 5", X-5 Monte Carlo Team, LA-UR-03-1987, dated April 24, 2004

The staff has reviewed the licensee's use of codes for the conversion analysis and found the licensee to be knowledgeable in the application of the codes. Hence, the staff has a high level of confidence in the licensee's application of the codes and in the results. Because the licensee used documented codes that are well-accepted and have been validated against data for other TRIGA reactors, including the one at UWNR, the staff concludes that the calculational methodology used by the licensee is acceptable.

2.4.2 Excess Reactivity, Control Element Worth, and Shutdown Margin

Excess Reactivity

The licensee provided calculated and measured values for excess reactivity. The calculated excess reactivity at beginning-of-life (BOL) is 2.745 %Δk/k ± 7.550E-4 %Δk/k for the HEU core and 2.752 %Δk/k ± 0.011 %Δk/k for the LEU core. The measured excess reactivity for the HEU core at BOL is 3.16 %Δk/k. The calculated excess reactivity at MOL is 2.9 %Δk/k for the HEU core and 1.45 %Δk/k for the LEU core. Curves of excess reactivity as a function of burnup are given in the conversion SAR in Figure 4.5.3 for the HEU core and in Figure 4.5.10 for the LEU core. Referring to Figures 4.5.3 and 4.5.10, both the HEU and LEU fuels have erbium as a burnable poison so they both have an initial decrease in the excess reactivity approaching 50-200 MWd followed by an increase in the excess reactivity to MOL. If there were to be a direct replacement of the current HEU-fueled core with new LEU fuel, the excess reactivity of the LEU core would be high enough that the core would be supercritical under the shutdown margin conditions defined in TS 3.1. Existing TS 3.1 requires the shutdown margin provided by the control elements to be at least 0.2 %Δk/k with: (1) the highest worth non-secured experiment in its most reactive state, (2) the highest worth control rod and the regulating rod (if not scrammable) fully withdrawn, and (3) the reactor in the cold condition without xenon. The solution to the increase in excess reactivity was to replace eight fuel elements in the LEU core with graphite reflectors. The LEU core excess reactivity values are considerably lower than the HEU core but are expected to be sufficient to permit all modes of operation.

The staff has reviewed the values for excess reactivity. Because the conversion from HEU to LEU fuel will not impact the ability of the licensee to meet TS 3.1, the staff finds the values for excess reactivity acceptable.

Control Element Worth

Using MCNP to model the control element worth, the licensee calculated the control element worths by either using the rod drop methodology or rising period rod bump methodology of measuring reactivity. For the HEU core, calculations of integral control elements compared favorably with measured values for the control elements. Calculated integral worth of the control elements for the HEU and LEU cores and the measured integral worth of the control elements for the HEU core are shown in Table 2.

Table 2 Calculated and Measured Values for the Control Elements

Control Element	HEU FLIP		LEU 30/20
	Calculated Integral Worth (%Δk/k)	Measured Integral Worth (%Δk/k)	Calculated Integral Worth (%Δk/k)
Control Blade 1	2.35	1.95	2.13
Control Blade 2	2.26	1.85	2.09
Control Blade 3	2.79	2.53	2.60
Regulating Blade	0.49	0.41	0.38
Transient Rod	1.47	1.37	1.37

The staff has reviewed the values for control element worth. Given the reduction in excess reactivity of the core, the staff concludes that the worths of the control elements in the LEU core are acceptable.

Shutdown Margin

The shutdown margin was calculated by the licensee for both the HEU and LEU cores using the MCNP5 model with: (1) the reactor core in a cold condition and no xenon present, (2) the regulating blade (which is non-scrammable) and control blade 3 fully withdrawn, and (3) the maximum allowable experiment installed. The calculated shutdown margin at BOL with no experiment installed is 0.903 %Δk/k ± 2.577 x 10^{-4} %Δk/k for the HEU core and 0.994 %Δk/k ± 0.011 %Δk/k for the LEU core. If the maximum allowable experiment of 0.7 %Δk/k were installed, both would continue to meet TS 3.1, which states that this margin should be at least 0.2 %Δk/k. Therefore, the staff concludes that the ability to shut down the reactor will be maintained after conversion.

The staff has reviewed the values for shutdown margin. Because the conversion from HEU to LEU fuel will not impact the ability of the licensee to meet TS 3.1, the staff concludes that the ability to shut down the reactor will be maintained after conversion and the values for shutdown margin are, therefore, acceptable.

2.4.3 Dynamic Parameters

Effective Delayed Neutron Fraction (β_{eff})

The effective delayed neutron fraction, β_{eff}, was calculated by the licensee using MCNP for the HEU and LEU cores. The value of β_{eff} for the HEU core was calculated to be 0.0075 at BOL, 0.0076 at MOL, and 0.0073 at EOL. The value of β_{eff} for the LEU core was calculated to be 0.0078 at BOL, 0.0077 at MOL, and 0.0073 at EOL. The staff notes that the calculated values of β_{eff} are consistent with values determined 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 staff has reviewed the values for β_{eff} for various times during core life. Because the changes in the β_{eff} due to the conversion from HEU to LEU fuel are not expected to significantly change the basic behavior of the core, the staff concludes that the changes in β_{eff} are acceptable.

Prompt neutron lifetime (ℓ_p)

The prompt neutron lifetime, ℓ_p, changes slightly as a result of the conversion from HEU to LEU fuel. The value of ℓ_p was calculated by the licensee 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 then correlated to the associated changes in core reactivity as the boron concentration becomes infinitely small. The value of ℓ_p for the HEU core was calculated to be 27.0 μs ± 0.8 μs at BOL, 20.3 μs ± 1.8 μs at MOL, and 23.0 μs ± 1.7 μs at EOL. The value of ℓ_p for the LEU core was calculated to be 27.1 μs ± 0.8 μs at BOL, 27.9 μs ± 0.8 μs at MOL, and 29.4 μs ± 0.8 μs at EOL. The staff notes that the calculated values of neutron lifetime are consistent with values determined for other research reactors.

The staff has reviewed the values for ℓ_p for various times during core life. Because the changes in the ℓ_p due to the conversion from HEU to LEU fuel are not expected to significantly change the basic behavior of the core, the staff concludes that the changes in ℓ_p are acceptable.

Fuel Temperature Coefficient of Reactivity (α_{Fuel})

One of the inherent safety features of the TRIGA fuel design is the large negative fuel temperature coefficient of reactivity, α_{Fuel}. Because TRIGA reactors were designed for pulsing, TRIGA fuels can be subject to prompt insertions of reactivity 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 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 licensee's calculated values of α_{Fuel} for the HEU and LEU cores at BOL, MOL, and EOL as a function of temperature in kevlins (K) are shown in Table 3.

Table 3 Calculated Fuel Temperature Coefficients of Reactivity for HEU and LEU Cores

Temperature, K (°C)	Negative Prompt Fuel Temperature Coefficient ([%Δk/k / K]) x 10^3					
	BOL		MOL		EOL	
	HEU	LEU	HEU	LEU	HEU	LEU
350 (77)	4.57 ± 0.06	4.08 ± 0.04	3.75 ± 0.16	3.26 ± 0.08	3.64 ± 0.17	3.05 ± 0.09
500 (227)	7.17 ± 0.03	6.57 ± 0.02	6.00 ± 0.09	5.65 ± 0.04	5.64 ± 0.08	5.33 ± 0.04
700 (427)	10.38 ± 0.03	8.20 ± 0.02	7.63 ± 0.09	6.75 ± 0.04	6.41 ± 0.07	5.86 ± 0.04
1000 (727)	19.18 ± 0.02	13.34 ± 0.01	15.44 ± 0.04	11.00 ± 0.02	13.29 ± 0.04	9.54 ± 0.02

The decrease in magnitude from the HEU core to the LEU core 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_{Fuel}. Although the value of the fuel temperature coefficient decreases in magnitude from the HEU core to the LEU core during pulsing, the limitations on pulsing will continue to protect the integrity of the fuel.

Existing TS 3.2 limits prompt reactivity insertions during pulsing to 1.4 %Δk/k. There are electrical and mechanical interlocks on the transient rod that are set to limit the maximum pulse reactivity insertion to 1.4 %Δk/k.

GA has recommended that the temperature of the fuel should not exceed 830 degrees Celsius (°C) (1,526 degrees Farenheit (°F)) 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). Assuming the most limiting core configuration (LEU core at MOL) and a reactor scram at 5 seconds following the pulse, the licensee calculated that a prompt reactivity insertion of 1.4 %Δk/k will result in a prompt peak fuel temperature of 790 °C (1,454 °F). Even if the pulse continued for the maximum pulse timer setting of 15 seconds, fuel temperature will not exceed 830 °C (1,526 °F). Calculations provided by the licensee show the maximum fuel temperature following a prompt reactivity insertion of 1.4 %Δk/k is 826 °C (1,519 °F).

During startup 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 (1.00 %Δk/k) followed by gradual increases (+0.05 %Δk/k) in reactivity. After each pulse, the peak power, peak fuel temperature, and integrated pulse power will be recorded. These measured values will be checked against the calculated values for that given reactivity insertion. This will be repeated as prompt reactivity insertions approach the maximum reactivity limit of 1.4 %Δk/k as specified in TS 3.2.

The staff has reviewed the values for α_{Fuel} for various times during core life. Because the licensee will continue to meet the α_{Fuel} continues to provide for the inherent safety of the reactor with increasing fuel temperature, the staff concludes that the changes in α_{Fuel} are acceptable.

Void Coefficient of Reactivity

The void coefficient of reactivity for both the HEU and the LEU cores was calculated by the licensee using the MCNP model by uniformly reducing coolant density 2.5 percent to introduce a void in the system. The calculated void coefficients, as shown in Table 4, are more negative for the LEU core than for the HEU core.

Table 4 Calculated Void Coefficients of Reactivity for HEU and LEU Cores

Fuel Condition		HEU ([Δk/k / %void]) x 10^3	LEU ([Δk/k / %void]) x 10^3
BOL	Cold (27°C)	-1.13 ± 0.03	-1.49 ± 0.05
	Hot (327°C)	-1.06 ± 0.03	-1.35 ± 0.05
MOL	Cold (27°C)	-0.971 ± 0.03	-1.35 ± 0.05
	Hot (327°C)	-0.962 ± 0.03	-1.28 ± 0.05
EOL	Cold (27°C)	-0.673 ± 0.03	-1.36 ± 0.05
	Hot (327°C)	-0.633 ± 0.06	-1.25 ± 0.06

The staff has reviewed the values for the void coefficient of reactivity for various times during core life. Because the void coefficient of reactivity continues to provide for the inherent safety of the reactor should voids form in the system, the staff concludes that the changes in the void coefficient of reactivity are acceptable.

Coolant Temperature Coefficient of Reactivity

The temperature of the coolant affects the overall reactivity of the core. The coolant temperature coefficient of reactivity was calculated by the licensee, as shown in Table 5, using the MCNP model to uniformly increase coolant temperature by 100 K. Heating only the coolant has a small positive temperature coefficient; however, raising the coolant temperature would also raise fuel temperature. The magnitude of the coolant temperature coefficient is small when compared to the magnitude of the negative fuel temperature coefficient, as discussed above. The overall reactivity effect of heating the reactor is negative when considering both the coolant and fuel temperature coefficients. Hence, the staff does not consider a small positive coolant temperature coefficient of reactivity to have a significant safety impact.

Table 5 Calculated Coolant Temperature Coefficients of Reactivity for HEU and LEU Cores

Fuel Condition		HEU ([$\Delta k/k$ / K]) x 10^4	LEU ([$\Delta k/k$ / K])x 10^4
BOL	Cold (27°C)	0.931 +/- 0.002	0.816 +/- 0.013
	Hot (327°C)	0.350 +/- 0.002	0.905 +/- 0.011
MOL	Cold (27°C)	1.01 +/- 0.02	0.848 +/- 0.013
	Hot (327°C)	1.09 +/- 0.02	0.923 +/- 0.013
EOL	Cold (27°C)	1.21 +/- 0.02	0.910 +/- 0.013
	Hot (327°C)	1.39 +/- 0.02	1.01 +/- 0.015

The staff has reviewed the coolant temperature coefficient of reactivity for various times during core life. Because the overall reactivity effect of heating the reactor is negative, the staff concludes that the changes in the coolant temperature coefficient of reactivity are acceptable.

2.4.4 Power Peaking

The average power per rod is higher for the LEU fuel, compared to the HEU fuel, since there are eight fewer fuel elements in the LEU core, compared to the HEU core. This results in the power peaking factors being slightly higher for the LEU core than for the HEU core. The licensee calculated values for steady-state power peaking factors, as shown in Table 6, using MCNP for both the HEU and LEU cores at BOL, MOL, and EOL conditions.

Table 6 Calculated Steady-State Power Peaking Factors for HEU and LEU Cores

Fuel Condition	Minimum		Maximum	
	HEU	LEU	HEU	LEU
BOL	0.44	0.46	1.60	1.61
MOL	0.44	0.47	1.58	1.60
EOL	0.49	0.49	1.45	1.57

During the life of the LEU core, it is conceivable that some new fuel may be introduced. If a fresh fuel element were to be placed in the hottest fuel element location, the peaking factor would increase from the values in Table 6. The peaking factor would increase to 1.68, if the element were inserted at MOL, and to 1.74, if at EOL. None of the safety limits would be exceeded during steady-state operation or accident conditions. According to the conversion SAR, if an interior fuel element requires replacement, a partially burned fuel element from an outer location will be moved to the inner-core location, and the fresh fuel element will be placed in the periphery of the core. One of the IFEs is presently located near the center of the core, and the other is in the periphery of the core. If the IFE located near the center of the core needs to be replaced, the IFE in the periphery can be moved to the central portion of the core. In response to question 30 of response to request for additional information (RAI), the licensee stated that review based on the criteria in 10 CFR 50.59, “Changes, Tests, and Experiments,” would be performed as part of the fuel rod replacement and core rearrangement.

The staff has reviewed power peaking factors for various times during core life. Because the licensee has stated that a 10 CFR 50.59 review would be conducted as part of fuel rod replacement and core rearrangement, the staff finds the licensee’s approach to control power peaking acceptable. Because power peaking will not change significantly following the conversion from HEU to LEU fuel, the staff concludes that the changes in the power peaking are acceptable.

2.4.5 Conclusions

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

2.5 Thermal-Hydraulic Design

The UWNR 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 and the core pulse analysis, calculations were performed using RELAP5/MOD3.3. RELAP5 was benchmarked for the UWNR by comparing calculated results and current operational data for the HEU core.

RELAP5 was used to determine the natural convection flow rate, fuel temperature profile, cladding temperature profiles, and reactor power in a pulse transient. RELAP5 also calculated the 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 include: inlet coolant temperature, system pressure at the top of the core, radial and axial heat source distribution, spacing of heat source nodes, inlet and exit pressure loss coefficients, and geometric parameters for the coolant channels and the fuel rods. The parameters that were produced as outputs from RELAP5 include: channel flow rate, axial fuel centerline temperature distribution, fuel radial temperature distribution, axial clad temperature distribution, axial bulk coolant temperature distribution, and axial departure-from-nucleate-boiling ratio (DNBR).

The version of RELAP5 code used does not incorporate recent fixes that corrected an error in the logic that determined when to apply a quasi-steady form of the point kinetics equation. Calculations were performed using the uncorrected and corrected version of the code for a $2.00 step reactivity insertion in the UWNR. The calculated peak powers using both versions of the code were found to be identical to six significant digits. Therefore, the version of RELAP5 used will not impact the results of pulse calculations presented by the UWNR.

The staff has reviewed the licensee's use of codes for the thermal-hydraulic analysis and found the licensee to be knowledgeable in the application of the codes. Hence, the staff has a high level of confidence in the licensee's application of the codes and in the results. Because the licensee used documented codes that are well-accepted and have been validated against data for other TRIGA reactors, including the one at UWNR, 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 three different power levels: 1.5 MW(t), 1.3 MW(t) (1.25 MW(t) trip set point + uncertainty), and 1.0 MW(t) (the licensed operating power) at a water inlet temperature of 54.44 °C (130 °F) (normal operating temperature is between 25-30 °C (77-86 °F) at 1.0 MW(t)). A pool water temperature limit of 54.44°C (130°F) is a proposed addition to TS 3.3.3, "Reactor Safety System," under TS 3.3.3(j) (discussed below in Section 2.10.2). A pool water level of 5.79 meters (m) (19 feet (ft)) above the top of core was also assumed in the steady-state analysis. This is consistent with TS 3.3.3(d), "Reactor Safety System – Reactor Pool-water Level." Results of the steady-state analysis include fuel and coolant temperatures and the minimum DNBR (MDNBR). For the pulse analysis, calculated peak fuel temperature and pulse peak power are presented.

The steady-state analysis considered the maximum powered 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 loss coefficients carries uncertainties. The effect of uncertainties in the loss coefficients on the thermal analysis was evaluated by UWNR in a sensitivity study by altering both the inlet and outlet loss coefficients by ± 20 percent. The results of the calculations demonstrate that the changes in the loss coefficients have insignificant impacts on temperatures: fuel, clad, coolant (less than 2 percent), mass flow rate (less than 4 percent), and MDNBR (less than 2 percent). The loss coefficients are given in Table 4.7.1 of the conversion SAR.

An examination of the pressure differential between the hot channel and an adjacent channel indicated that cross flow would occur from the cold to the hot channel. The net increase in flow in the hot channel tends to increase the margin to critical heat flux (CHF). The thermal-hydraulic calculations conservatively ignored cross flow between adjacent channels in the conversion analysis.

Given the inlet water temperature (54.44 °C (130°F)), 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 4.7.2 of the conversion SAR, and the parameters are identical for both the

HEU and LEU cores. Fuel rods in the UWNR are arranged in square pitch. There are three types of unit flow channels (subchannels), in positions adjacent to the transient rod, in positions adjacent to the control blades, and an "average" core position in which there were no transient rod or control blades neighboring the channel. The hydraulic parameters of the three types of unit flow channel are shown in Table 4.7.3 of the conversion SAR.

The active fuel region is divided into 15 axial nodes, and the corresponding axial nodalization of the coolant channel is shown in Table 4.7.4 of the conversion SAR. Radially, the annular fuel region is divided into 21 equally spaced layers, as shown in Figure 4.7.4 and Table 4.7.5 of the conversion SAR.

The maximum powered element is determined using MCNP5, with hot conditions and all the control blades at the critical bank position. For both HEU and LEU cores, the location of the hot rod (maximum powered element) is in the same core position at all times in core life. The hot rod core position is adjacent to the transient rod but not in one of the two IFE positions. Relative power distribution in the fuel elements is characterized by three power factors, and there are corresponding peak factors. The hot channel 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 UWNR. They are the hot channel fuel axial peak factor and the hot channel 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 radial power in the hot fuel rod. The axial power profiles were derived from MCNP, with the critical bank height adjusted for core burnup. The power peaking factors are given in the conversion SAR as follows:

- Figures 4.7.5, 4.7.6 and 4.7.7 – pin power peaking factors for the HEU BOL, MOL, and EOL cores, respectively
- Figures 4.7.30, 4.7.31 and 4.7.32 – pin power peaking factors for the LEU BOL, MOL, and EOL cores, respectively
- Figure 4.7.8 – axial power peaking factors for the hot rod, HEU BOL core
- Figure 4.7.34 - axial power peaking factors for the hot rod, LEU BOL, MOL, and EOL cores (revised in answer to question 28 of response to RAI dated April 10, 2009)
- Figure 4.7.9 – radial power density factors for the hot rod, HEU BOL core
- Figure 4.7.33 – radial power density factor for the hot rod, LEU BOL, MOL, and EOL cores

For the type of TRIGA fuel used in the UWNR, the gas gap between the fuel and the cladding is known to vary. The effect of gap size on fuel temperature was evaluated by performing a series of RELAP5 calculations, assuming a gap varying from 0.05 mils to 0.15 mils. The calculations also assumed the RELAP5 default composition for the mixture of gases (helium, krypton, xenon) in the gap. This gap composition was chosen to incorporate fission product buildup and will produce a larger thermal resistance, compared with air that is present in fresh fuel. The licensee confirmed the gap model by comparing calculated fuel temperatures against temperatures measured by the IFE in a number of core locations. These temperatures were obtained at the axial location of the highest axial power peaking factor. The UWNR conversion

SAR assumed a 0.1 mil (2.54×10^{-6} m) gap. A gap of 0.1 mil was found to give conservative predictions of fuel temperature as compared to measurement (prediction is 26 to 51 °C (79 to 124 °F) higher than measurement). Results of these calculations are shown in Figure 4.7.11 of the conversion SAR. The same 0.1 mil gap was used in the pulse calculations.

The steady-state thermal-hydraulic analysis was performed for the maximum powered channel. For both the HEU and the LEU core, the maximum powered fuel element is next to the transient rod. In the single-channel, steady-state model, the coolant subchannel was conservatively assumed to be bordered on all sides by a fuel rod having the same characteristics of the hot rod. Because of channel geometry, Table 4.7.3 of the conversion SAR demonstrates that the channel with the highest thermal power (e.g. hot rod) is not the channel with the minimum flow rate (e.g. rod next to control blade shroud). Thermal analysis was performed for fuel elements in three types of coolant channels. It was established that the channel associated with the hot rod had the lowest MDNBR and is the limiting channel.

The conversion SAR discussed replacement of the hot rod with a fresh fuel pin at various times during core life. Currently, the highest pin power peaking factor is 1.61 at BOL; if a fresh fuel rod replaced the hot rod, this would increase to 1.68 at MOL and 1.74 at EOL. This would represent a decrease in the margin of safety; however in response to RAI number 30, UWNR stated that it will ensure the pin power peaking factor does not exceed 1.61 when replacing the hot rod. UWNR proposed to address this issue by substituting a burned fuel pin in the hot rod location and putting the fresh fuel in the periphery of the core to decrease the pin power peaking effect. The analysis discussed above also shows that inserting fresh fuel next to a control blade could be more limiting due to less flow in that channel. As stated in the response to RAI number 30, UWNR stated that it will also ensure the pin power peaking factor does not exceed 1.47 when placing fuel next to a control blade shroud. A review based on the criteria in 10 CFR 50.59, “Changes, Tests, and Experiments,” would be performed by the licensee as part of the fuel rod replacement and core rearrangement.

Results of the steady-state thermal analysis are summarized in Table 7 for both the HEU and LEU core. The number of fuel elements in the core was 91 for the HEU core and 83 for the LEU core. Compared to the HEU core, the LEU core, with fewer fuel elements, is expected to have higher hot rod power and higher maximum fuel temperatures. Results of the analysis do show the expected trend, and the differences in MDNBR and the maximum fuel temperatures between the HEU and LEU cores are consistent with the predicted hot rod powers. Since the HEU analysis was done to illustrate the model and to compare with measurements, only BOL was analyzed. For the LEU core, a separate analysis was done for the three stages of the core lifetime: BOL, MOL, and EOL.

Table 7 Summary of Steady-State Analysis at 1.5 MW(t)

Core Type	Hot Channel Peak Factor	Hot Rod Thermal Power* (kW(t))	Maximum Fuel Temperature (°C)	MDNBR (Bernath)	Maximum Exit Bulk Coolant Temperature (°C)
HEU-BOL	1.60	26.40	642.0	1.29	100.4
LEU-BOL**	1.61	29.04	673.9	1.23	101.0
LEU-MOL	1.60	28.89	665.1	1.22	101.2
LEU-EOL	1.57	28.33	641.9	1.23	100.7

* Hot rod thermal power corresponds to core power of 1.5 MW(t).
** The LEU-BOL results are the revised values from the UWNR supplemental information.

The core conversion study for the UWNR considered two different correlations for calculating the CHF. The licensee chose to use the Groeneveld 2006 Look-up Tables (being the most current method for calculating the critical heat flux ratio (CHFR)) and the Bernath correlation (being the traditional CHF correlation used by many research reactors. Between the two correlations, the Bernath correlation produced the more conservative CHFR values. At the nominal operating core power of 1.0 MW(t), the MDNBR, according to the Bernath correlation, was 1.60 and 1.53 (revised value from the UWNR supplemental information) for the HEU and LEU cores at BOL, respectively. This represents a 5 percent reduction in the MDNBR from the HEU to LEU core for a corresponding 10 percent increase in hot rod power (29.04/26.4 -1).

In applying the Groeneveld Look-up Tables to the UWNR, a set of correction factors is required to adapt the geometry used in the Look-up Tables to the UWNR 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 UWNR 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.

A typical method of determining the DNBR is to run RELAP5 at a specified power level and then calculate the CHF and DNBR based on the thermal-hydraulic conditions obtained for that power level (DNBR = CHF/maximum heat flux corresponding to the specified power).

The licensee used a modified definition to calculate the DNBR:

DNBR = (rod power corresponding to CHF calculated at the predicted flow rate) / (rod power input into RELAP5 to calculate the flow rate in the hot channel)

The numerator of the above relation is determined in two steps:

(1) A hot channel power was selected (this becomes the denominator), and the corresponding flow rate was calculated by RELAP5.

(2) Using the flow rate from (1), the power at which CHF was reached for the hot rod was calculated using one of the CHF correlations. This step was done external to the

RELAP5 calculation, using a spreadsheet program to systematically vary the rod power while holding the flow rate constant until DNBR reached 1.0.

Results of the flow versus power (step (1) above) and the corresponding critical power (step (2) above) are illustrated in Figure 4.7.16 of the conversion SAR for the HEU BOL case and in Figures 4.7.45, 4.7.51, and 4.7.57 for the LEU at BOL, MOL, and EOL, respectively. In all cases, the Bernath correlation predicted more conservative DNBR values than the Groeneveld 2006 Look-up Tables. It was noted in the conversion SAR that RELAP5 started to predict oscillatory flow in the hot channel when the hot rod power was above a threshold value of 28 to 29 kW(t)/rod. In the supplemental information provided by UWNR, the region of stable flow was extended by using a pseudotransient approach to achieve a steady-state solution in which power is raised slowly from zero with flow set initially to zero. Replacement figures for coolant flow versus hot rod power are provided in the supplemental information. The revised figures show that the LEU flow rate using the pseudotransient approach is stable at 1.5 MW(t) and thus, flow oscillations are not predicted in steady-state operation at all times in core life. The calculated critical rod power is summarized in Table 8.

Table 8 Critical Rod Power

CHF Correlation	Power [MW(t)]	HEU-BOL [kW(t)/rod]	LEU-BOL* [kW(t)/rod]	LEU-MOL [kW(t)/rod]	LEU-EOL [kW(t)/rod]
Groneveld 2006	1.5	52.38	53.11	52.83	54.55
	1.3	49.57	51.45	49.63	54.65
	1.0	47.58	49.89	51.58	51.31
Bernath	1.5	33.99	35.63	35.82	35.49
	1.3	31.85	33.40	33.52	33.15
	1.0	28.21	29.60	29.41	29.12

* The LEU-BOL results are the revised values from the UWNR supplemental information.

The licensee calculated thermocouple temperatures for the IFEs in the LEU-BOL core at 1.0 MW(t), 1.3 MW(t), and 1.5 MW(t). As shown in Tables 9a-9c, in order for the HEU core to meet the limiting safety system setting (LSSS) of 400 °C (752 °F), the IFE connected to the fuel temperature safety channel is located in the core periphery. It is evident that a higher temperature limit than the existing 400 °C (752 °F) is required to accommodate an IFE near the center of the core. The conversion SAR proposes a change to TS 2.2, "Limiting Safety System Setting," to provide the licensee greater flexibility in placing an IFE connected to the fuel temperature safety channel in the central region of the core (see Section 2.10.2). The licensee conducted analysis in support of locating an IFE connected to the fuel temperature safety channel in the periphery and central region of the core. The licensee analyzed the current LSSS; if the IFE is located where the IFE power peaking factor is between 0.87 and 1.16, the scram at 400 °C (752 °F) ensures the maximum fuel temperature will not exceed 678 °C (1,252 °F). Additionally, the licensee analyzed the proposed LSSS; if an IFE was located where the IFE power peaking factor is at least 1.16, the scram at 500 °C (932 °F) ensures the maximum fuel temperature will not exceed 678 °C (1,252 °F). The current and proposed LSSS values provide a margin of 472 °C (882 °F) to the safety limit of 1,150 °C (2,100 °F) (TS 2.1, "Safety Limits"). If the IFE is located in either the periphery or central region of the core, both would continue to meet TS 2.1, which states that the temperature in a TRIGA LEU 30/20 fuel element shall not exceed 1,150 °C (2,100 °F) under any conditions of operation. Based on the staff's review of the licensee's analysis, the staff concludes that the LSSS will continue to protect the reactor after conversion and the proposed changes to TS 2.2 are, therefore, acceptable.

Table 9a IFE Temperatures for the LEU-BOL Core at 1.0MW(t)

IFE/Thermocouple Location		0.1 mil gap		0.05 mil gap		0.15 mil gap	
		°C	°F	°C	°F	°C	°F
Near Center of Core	Bottom	444.4	831.9	397.2	747.0	488.2	910.7
	Center	429.5	805.1	384.2	723.6	471.6	881.0
	Top	412.6	774.6	369.5	697.0	452.8	847.0
In Core Periphery	Bottom	299.3	570.7	270.9	519.6	326.3	619.3
	Center	291.4	556.5	264.1	507.4	317.3	603.2
	Top	279.9	535.8	254.3	489.7	304.3	579.7

Table 9b IFE Temperatures for the LEU-BOL Core at 1.3MW(t)

IFE/Thermocouple Location		0.1 mil gap		0.05 mil gap		0.15 mil gap	
		°C	°F	°C	°F	°C	°F
Near Center of Core	Bottom	535.2	995.4	476.7	890.0	589.2	1,092.5
	Center	516.4	961.5	460.1	860.2	568.3	1,054.9
	Top	494.9	922.8	441.2	826.2	544.5	1,012.0
In Core Periphery	Bottom	349.0	660.1	313.5	596.3	382.4	720.3
	Center	339.0	642.2	304.9	580.8	371.2	700.1
	Top	324.5	616.1	292.4	558.4	354.8	670.7

Table 9c IFE Temperatures for the LEU-BOL Core at 1.5MW(t)

IFE/Thermocouple Location		0.1 mil gap		0.05 mil gap		0.15 mil gap	
		°C	°F	°C	°F	°C	°F
Near Center of Core	Bottom	594.8	1,102.6	529.0	984.2	654.9	1,210.9
	Center	573.3	1,064.0	510.1	950.1	631.3	1,168.3
	Top	548.8	1,019.9	488.5	911.4	604.3	1,119.7
In Core Periphery	Bottom	382.4	720.3	342.3	648.1	420.0	788.0
	Center	371.1	699.9	332.5	630.4	407.3	765.1
	Top	354.6	670.2	318.2	604.8	388.7	731.7

For the pulse analysis, the transient power was calculated by the point kinetics model in RELAP5. 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 model of the UWNR core. The RELAP5 analysis of the pulse transient included direct gamma heating of the moderator from standard fission product decay. Analysis has shown that the time lag in heat transfer from the fuel to the moderator makes the moderator feedback inconsequential. The pulsing analysis was performed with the assumption that the reactor core maintains the same power shape as the steady state. To show that the assumptions are valid, the licensee performed a sensitivity analysis with a more severe pin power peaking (12 percent higher, LEU-MOL with transient rod full out and control blades at the cold critical bank height) and the same 1.4 %Δk/k pulse at the most limiting state-point of the LEU core. The resulting peak pulse temperature was increased from 727 °C to 790 °C (1,341 °F to 1,454 °F) but is still below GA's recommended fuel temperature pulsing limit of 830 °C (1,526 °F). All pulse transients were analyzed with an initial reactor power of 1 kW(t). The analysis also assumed that the transient rod remained out of the core for 15 seconds and then all scrammable control elements were fully inserted into the core. This assumption is consistent with the TS 3.3.3(g), "Reactor Safety System – Preset Timer," requirement that a scram must occur in 15 seconds or less after the initiation of a pulse.

A 1.34 %Δk/k pulse experiment for the HEU BOL core was used for comparison with the model predictions of pulse power and total energy of pulse. The two-channel model predictions were conservative compared to the experimental measurements (predicted peak power and integrated power were 1,960 MW(t) and 24.6 MJ while the measured values were 859 MW(t) and 12.7 MJ). An analysis was conducted at various stages in core life on the maximum pulse of 1.4 %Δk/k allowable by TS 3.2. An additional analysis was performed to determine the pulse sizes needed to reach GA's recommended fuel temperature pulsing limit of 830 °C (1,526 °F) and the fuel temperature safety limit of 1,150 °C (2,100 °F). Results of the pulse analyses are summarized in Tables 10 and 11.

Table 10 Summary of Results for a 1.4 %Δk/k Pulse

Core Type	Maximum Fuel Temperature (°C)	Peak Pulse Power (GW)	Total Pulse Energy (MJ)
HEU-BOL	661.1	2.29	28.05
LEU-BOL	663.1	2.06	26.9
LEU-MOL	727.0	2.52	31.9
LEU-EOL	723.5	3.06	35.2

Table 11 Pulse Sizes to Exceed the Fuel Temperature Limits

CBore Type	Pulse to Exceed Fuel Temperature Operational Limit (%Δk/k)	Pulse to Exceed Fuel Temperature Safety Limit (%Δk/k)
HEU-BOL	1.6	2.2
LEU-BOL	1.6	2.1
LEU-MOL	1.5	2.0
LEU-EOL	1.5	2.0

Pulse results presented in Table 10 above indicate that the highest predicted maximum fuel temperature occurs in the LEU core at MOL. The maximum fuel temperatures shown in Table 10 are prompt peaks occurring near the outside edge of the fuel. As heat is redistributed over time, the maximum fuel temperature occurs at the fuel centerline. For the LEU-MOL core, UWNR calculations show a maximum fuel temperature of 826 °C (1,519 °F) at 15 seconds after a 1.4 %Δk/k pulse. Even though the maximum pulse power is higher in the LEU-EOL core than the LEU-MOL core, the maximum prompt fuel temperature is higher in the LEU-MOL core. This is due to a combination of higher rod power, axial peaking, and outside radial power peaking in the LEU-MOL core. It is noted from Table 11 that at no time during the reactor life of the LEU core will a 1.4 %Δk/k pulse ever exceed the 830 °C (1,526 °F) fuel temperature operational limit or the 1,150 °C (2,100 °F) fuel temperature safety limit.

Based on the staff's review of the thermal-hydraulic design of the UWNR core, the conversion from HEU to LEU fuel will not cause any significant changes in the thermal-hydraulic characteristics. The analyses have been calculated using qualified 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. The staff concludes that the thermal-hydraulic analysis for the UWNR conversion adequately demonstrates that the conversion from an HEU to an LEU core will result in no significant decrease in safety margins in regard to thermal-hydraulic conditions and is, therefore, acceptable.

2.6 Accident Analysis

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

2.6.1 Maximum Hypothetical Accident

To support the conversion from HEU to LEU fuel, the licensee performed a complete reanalysis of the MHA. The UWNR MHA is defined as an assumed cladding failure of one highly irradiated fuel element after continuous operation at 1.3 MW(t), followed by the instantaneous release of the noble gases and halogen fission products. Since 1.0 MW(t) is the licensed power and the scram set point of the UWNR is 1.25 MW(t), 1.3 MW(t) is representative of the maximum power accounting for power level uncertainty. The previous HEU SAR assumed infinite 1.25 MW(t) operation of the fuel. Boundary conditions and assumptions for the most conservative cases analyzed for both the HEU and LEU cores included the following: no credit is given for iodine absorption in the reactor pool water, released fission products are uniformly distributed in the reactor room air space, the building ventilation system is inoperable, and 100 percent of the fuel pin air gap noble gases and halogen inventory become instantaneously released and airborne, except for the pool water activity calculations performed, which assumed that all of the gap activity released is dissolved in the pool water. A 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 existing HEU core and proposed new LEU fuel.

The ORIGEN2 computer code version 2.1[5] with the PWRUS cross section library was used to calculate the revised HEU and new LEU core fission product inventories. This code is well known and used throughout the nuclear industry to calculate core fission product inventories.

The review of Table 13.1.1 in the conversion SAR showed the near saturation inventories for

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5 "A Users' Manual for the ORIGEN2 Computer Code", A.G. Croft ORNL/TM-7175 (July 1980), with the v.2.1 update, S. Ludwig, August 1, 1991.

the LEU core are generally higher than the revised HEU core for most isotopes. This is reasonable because of the increased power density due to fewer fuel bundles in the core. The fission product release fraction from the fuel into the fuel pin air gap is totally dependent on the fuel temperature. Since the previous HEU SAR assumed a maximum fuel temperature of 600 °C (1,112 °F), a calculation was performed in Section 4.7.4 of the conversion SAR to determine the maximum fuel temperature for the revised HEU core and LEU core parametric analysis. The review of the analysis showed a maximum fuel temperature of 576 °C (1,069 °F) for the HEU core and 596 °C (1105 °F) for the LEU MOL core. A comparison performed by the licensee of the calculated maximum fuel temperature with thermocouple readings taken inside the HEU operating core show reasonable agreement.

Radioactivity releases from UWNR 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, the licensee calculated both occupational doses and public doses.

The licensee also calculated the dose rate in the vicinity of the reactor demineralizer resulting from the MHA halogen activity being released into the pool water and captured by the demineralizer. Since it would require 24 hours at the existing maximum pumping rate to remove all the activity from the pool water and deposit it in the demineralizer, the licensee assumed 24 hours of pumping. Two calculations were performed; one with all soluble fission products being deposited, and the other applying the release fraction for the LEU core. Both calculations also included a decay time of 24 hours. From a review of the calculations, the staff estimates that the occupational limit set in 10 CFR 20.1201, “Occupational Dose Limits for Adults,” of 5 rem whole-body and 50 rem thyroid would not be exceeded in a short period of time, therefore, the resulting dose rates are low enough for any post-accident operations requiring access to the demineralizer area.

In calculations for occupational and non-occupational doses in the conversion SAR, the licensee neglected isotopes with a half-life of less than 10 minutes. Since the licensee used an exposure time of 5 minutes, the staff asked the licensee to provide revised calculations or provide justification for excluding those isotopes. Revised dose calculations for whole body dose and total effective dose equivalent (TEDE) were submitted, but the short-lived halogens were not included for the thyroid dose. The licensee rationale for excluding the halogens was based on the fact that the transport of halogens from the blood to the thyroid takes approximately 6 hours. Therefore, the excluded short-lived halogens would not make a contribution to the thyroid dose. Only whole body and TEDE doses were revised to include the short-lived contributing isotopes. The previously excluded short-lived isotopes had a contribution to those doses that ranged from 4 percent to 11 percent. Those revised doses are the ones referred to in subsequent paragraphs of this section.

The occupational dose was calculated for an individual in the reactor room. Boundary conditions for these calculations included: assuming failure of the hottest fuel element, incorporating the calculated release fractions, and assuming the reactor room has a volume of 2,000 m^3, an added conservatism, since this is at least 20 percent smaller than the actual volume. With these conservative assumptions, the maximum whole-body (which occurs for the MOL condition) and thyroid doses are below the occupational limit set in 10 CFR 20.1201.

For the non-occupational dose, the licensee analyzed three situations: (1) a person who is co-located inside the Mechanical Engineering Building but exterior to the reactor confinement, (2) a person standing by the wall outside the Mechanical Engineering Building, and (3) a person located at the nearest residence, which is approximately 80 m (263 ft) from the Mechanical Engineering Building.

The UWNR is located in a reactor confinement room inside the University of Wisconsin's Mechanical Engineering Building, which has several floors with non-restricted access that are occupied by people not associated with the UWNR. Therefore, the other building occupants are considered members of the general public. The licensee assumed the following: an evacuation time of 5 minutes from the reactor room and an evacuation time of 10 minutes from the Mechanical Engineering Building; no pool water scrubbing of the halogens; fission products disperse uniformly through the reactor confinement plus the auxiliary support space; and after the confinement space is filled, that fission products will disperse uniformly throughout the first four floors of the Mechanical Engineering Building. The first four floors are connected to an open-air atrium in the central wing of the building, which allows for ready mixing of the building air. The licensee's calculations excluded the volume of the fifth floor of the building in the calculations because there is no ready flow path for mixing of building air. This is conservative and provides a higher concentration of fission products when calculating the dose rates to all of the building occupants. Actual dose rates to building occupants would be lower if the volume of the fifth floor would be included in the calculations.

The dose that an individual inside the Mechanical Engineering Building would receive under these conservative assumptions is below the public dose limit of 100 millirem (mrem) (10 CFR 20.1301, "Dose Limits for Individual Members of the Public"). The dose to an individual standing outside the Mechanical Engineering Building was calculated using the X/Q methodology described in NUREG/CR-2387[6] and is below the public dose limit of 100 mrem (10 CFR 20.1301). The dose at the nearest residence is bounded by the dose to an individual outside the Mechanical Engineering Building.

The licensee provided other calculations, in which it assumed that the reactor pool was not drained, so there was scrubbing of the bromine and iodine isotopes. In these cases, the scrubbing resulted in lowering the occupational whole-body dose by slightly more than a factor of 2 and the thyroid dose by almost a factor of 10. The same reduction to the non-occupational doses was determined for the non-staff person being evacuated from the Mechanical Engineering Building and the case of a person standing just outside the Mechanical Engineering Building. According to the licensee, the ventilation system is intended to be in operation for this event. If the ventilation system is operating, the dose to a person outside the building would be further reduced because it would reduce the concentration of radioactivity released in the air through dilution.

The staff reviewed the MHA for the LEU core. Because the doses are within the requirements in 10 CFR Part 20, "Standards for Protection Against Radiation," the results of the MHA continue to be acceptable to the staff with the conversion to LEU fuel.

2.6.2 Insertion of Excess Reactivity

Pulsing of the UWNR is discussed above in Section 2.4.3. An accidental pulsing accident from full power is hypothesized to be initiated by the ejection of the transient rod or the ejection of an experiment up to the value of the transient rod (1.4 %Δk/k, or $1.79). The analysis assumed the reactor was operating at 1.3 MW(t), the water inlet temperature was 54 °C (130 °F) and the pool

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6 "Credible Accident Analysis for TRIGA and TRIGA-fueled Reactors," NUREG/CR-2387, U.S. Nuclear Regulatory Commission, April 1, 1982.

water level is 5.79 m (19 ft) above the core. The analysis also assumed that the other control blades do not fall for 2 seconds after the ejection. This 2-second time is the limit set in TS 3.3.1, “Scram Time,” and is the time after the initiation of a scram for the blades to become fully inserted. For this analysis, it was assumed that the blades did not move for 2 seconds and then instantaneously are set at the fully inserted position (including the transient rod, which is not in pulse mode). The analysis of the transient utilized RELAP5 and the licensee calculated the temperature of the highest power fuel pin.

For the LEU fuel, the analyses were performed for the BOL, MOL, and EOL conditions. The maximum temperature was calculated to occur under the EOL condition, because the magnitude of the negative fuel temperature coefficient decreases with burnup, and is lowest at EOL. The calculated maximum temperature is calculated to reach 997 °C (1,826 °F) after 2.1 seconds. This compares with the present HEU fuel at BOL of 864 °C (1,587 °F) after 2.0 seconds. The maximum fuel temperature for this accident is 152 °C (273 °F) below the maximum safety limit for this fuel, 1,150 °C (2,100 °F). Therefore, this accident is not expected to result in any fuel failure.

The staff reviewed the insertion of excess reactivity accident for the LEU core. Because the accident does not result in fuel failure, the staff concludes that the results of the insertion of excess reactivity accident continue to be acceptable with the conversion to LEU fuel.

2.6.3 Loss-of-Coolant Accident (LOCA) Analysis

The LOCA analysis calculated what would happen under the limiting assumption that water was removed from the pool. In calculating the fuel temperature in a LOCA, the transient was divided into two portions: a water-cooled portion and an air-cooled portion. The water-cooled portion spans the period from the start of the LOCA to the time when the pool water reached the top of the fuel. As soon as the pool water reached the top of the fuel, the remaining water in the vessel was assumed to disappear immediately. This initiated the start of the air-cooled portion of the LOCA. The licensee’s acceptance criterion for the maximum air-cooled fuel temperature is given as 950 °C (1,740 °F). This value is long-standing and has been previously found acceptable to the staff (NUREG-1282). Assuming a starting fuel temperature of 506 °C (943 °F), the water level drops to the reactor pool-water level scram set point and the reactor scrams. The fuel is cooled to a temperature of 75 °C (167 °F) as the water is removed from the pool. When the water level is low enough and the fuel is cooled by air, the maximum temperature reached in the hot rod is 652 °C (1,206 °F).

The supplemental information also provided an independent analysis of the LOCA performed by ANL. The ANL analysis examined both a complete and partial LOCA. In a complete LOCA, the water in the reactor tank is assumed to drain completely. For this scenario, the maximum fuel temperature was calculated by using RELAP5-3D, assuming air cooling of the fuel rods by natural convection. In a partial LOCA, the water initially drained down to the bottom of the beam port. For this scenario, the maximum fuel temperature was determined by using a computational fluid dynamics code (STAR-CD) that accounted for heat conduction from the fuel rod to the remaining water in the reactor tank and steam cooling from steam generated in the heated water. Assuming the reactor operated continuously at 1.02 MW(t) for 7 days a week, maximum fuel temperatures for the hot rod (19.7 kW(t)) were calculated to be 585 °C (1,085 °F) for the complete LOCA and 578 °C (1,072 °F) for the partial LOCA. In both the UWNR analysis and the ANL independent analysis, the complete LOCA was shown to be more limiting, resulting in a higher maximum fuel temperature but still below the maximum air-cooled temperature limit of 950 °C (1,742 °F).

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 pool, at the console, and above the reactor in an adjacent classroom. In the event of a LOCA, people in the classroom will be alerted by an alarm to evacuate the building. Assuming a hypothetical member of the public remains in the third floor classroom for 5 minutes following the evacuation alarm and evacuates the building between the 5 and 10 minute mark following the alarm, the total dose received is below the public dose limit of 100 mrem (10 CFR 20.1301). Assuming a delay of 24 hours to carry out the emergency procedures to refill the pool with water, the total dose estimated at the site boundary is below the public dose limit of 100 mrem (10 CFR 20.1301).

The staff reviewed the LOCA for the LEU core. Because the maximum fuel temperature is below the air-cooled limit of 950 °C (1,742 °F) and doses to the public at the site boundary are within the requirements in 10 CFR Part 20, the results of the LOCA continue to be acceptable to the staff with the conversion to LEU fuel.

2.6.4 Other Accidents

Other accidents have been considered for the HEU core and do not change for the LEU core. The other accidents have shown to: (1) lead to acceptable results as previously reviewed by the staff; (2) be bounded by: the MHA, the insertion of excess reactivity accident, or the LOCA for the LEU core; or (3) 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
- 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 UWNR. The review of the calculation, including assumptions, demonstrated that the inventory of radioactivity assumed and other boundary conditions used in the analysis are acceptable. While the radiological consequences to the public and occupational workers at the UWNR from a postulated MHA for the proposed LEU-fueled reactor are expected to be slightly higher than the radiological consequences calculated for the HEU-fueled reactor, these doses are in conformance with the requirements in 10 CFR Part 20. As a result of this review, the staff has concluded that, from a radiological standpoint, continued operation of the reactor poses no undue risk to the public or the staff of the UWNR from the MHA.

The licensee did not identify any reactivity addition accidents that are not bounded by those analyzed for the LEU-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. 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.

The review of the calculations, including assumptions, demonstrated that a LOCA would not result in fuel element temperatures that would be unacceptable. The radiological consequences to the public and occupational workers at the UWNR from a postulated LOCA for the proposed LEU-fueled reactor are expected to be 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, from a radiological standpoint, air cooling the reactor after a LOCA is sufficient to prevent cladding failure and that continued operation of the reactor poses no undue risk to the public or the staff of the UWNR after a LOCA.

2.7 Fuel Storage

In support of the conversion of the UWNR from HEU to LEU, new 30/20 LEU fuel must be delivered to and stored at the UWNR. The new 30/20 LEU fuel elements will be placed in a dry storage container acquired from the Oregon State TRIGA Reactor.

In accordance with the existing TS 5.5(a), "Fuel Storage," all fuel elements must be stored in a geometric array where the value of the multiplication factor (k_{eff}) is less than 0.8 for all conditions of moderation. The licensee provided information on the storage of the LEU fuel in the fuel storage racks. The storage racks consist of two aluminum support plates with holes drilled in a square matrix with a 10 cm (3.9 inch) fuel storage pitch. The lower support plate is located 15.24 cm (6 inches) above the bottom of the fuel element and the upper support plate is located 20.32 cm (8 inches) below the top of the fuel element. Using MCNP for this analysis, the licensee determined k_{eff} values for an infinite array. If the box is not flooded (dry storage), k_{eff} is 0.694 ± 0.0003. If the box is flooded with water, k_{eff} is 0.6568 ± 0.0003. These values are below the multiplication factor limit of 0.8, so fresh LEU fuel elements can be stored safely in these storage racks.

The staff has reviewed the licensee's use of MCNP in this instance and for all other aspects of its conversion analysis and found that it is knowledgeable in the application of the code. Additionally, the staff note that MCNP is a state-of-the-art code frequently used for this type of analysis. 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 5.5(a) will continue to be met. Therefore, the staff finds the licensee's analysis of fuel storage for the new LEU fuel to be acceptable.

2.8 LEU Startup Plan

With the LEU 30/20 fuel elements, the licensee calculated that the core will contain a total of 83 fuel elements bundled in groups of four, with the exception of one bundle that will contain three fuel elements and the transient rod. A total of 21 fuel bundles will be used. The licensee plans to use a standard 1/M plot for loading a critical assembly. The licensee's calculations show that the LEU core should go critical when it contains: (1) 18 fuel bundles and no graphite reflectors, or (2) 16 bundles and 14 graphite reflectors. The initial loading will be without the graphite reflectors and only to the point where initial criticality is achieved.

The licensee's startup plan is similar to those used in other reactors. The licensee's Reload and Startup Plan is found in Chapter 12.7 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 required to submit a startup report to the NRC on the results of the startup testing. The licensee's acceptance criteria are found in Chapter 12.7.1 of the conversion SAR.

Based on the staff's review of the startup plan, the staff finds the licensee's testing program will provide verification of key LEU reactor functions and, therefore, is acceptable.

2.9 Proposed Changes to the Emergency Plan

The licensee proposed changes to the UWNR Emergency Plan as provided in supplemental information dated April 10, 2009, and June 4, 2009. The changes proposed by the licensee are related to the conversion to LEU fuel, such as revising the definitions and boundaries of the emergency planning zone and evacuation zone and updating the descriptions and consequences of accidents to agree with the conversion SAR. The NRC staff reviewed the changes using NUREG-0849[7] and concluded that the plan continues to meet the requirements of Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." 10 CFR Part 50 Appendix E references Regulatory Guide 2.6[8] as guidance for the acceptability of research and test reactor emergency response plans. Therefore, the changes to the licensee's emergency plan resulting from the conversion of the reactor to LEU are acceptable.

2.10 Proposed Changes to License Conditions and Technical Specifications

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

2.10.1 Proposed Changes to License Conditions

During a May 19, 2009 discussion between the NRC Project Manager and the UWNR Director agreed that a change needed to be made to License Condition 1 in order to reflect the conversion to LEU fuel. The license condition currently reads as follows:

1. The license applies to the University of Wisconsin's nuclear reactor with the TRIGA-FLIP nuclear core and control system (herein "the reactor") owned by the University of Wisconsin (herein "the licensee"), and located on the University's campus in Madison, Wisconsin, and described in the licensee's application for license dated July 13, 1966, and amendments thereto including the amendment dated June 6, 1973, and supplements dated August 1, and August 21, 1973, (herein "the application").

The proposed license condition reads as follows:

1. The license applies to the University of Wisconsin's nuclear reactor with the TRIGA nuclear core and control system (herein "the reactor") owned by the University of Wisconsin (herein "the licensee"), and located on the University's campus in Madison, Wisconsin, and described in the licensee's application for license dated July 13, 1966, and amendments

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7 "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors," NUREG-0849, U.S. Nuclear Regulatory Commission, October 1983.

8 "Emergency Planning for Research and Test Reactors," Regulatory Guide 2.6, U.S. Nuclear Regulatory Commission, March 1983.

thereto including the amendment dated June 6, 1973, and supplements dated August 1, and August 21, 1973, (herein "the application").

The staff finds the licensee's proposed change to be acceptable because removal of the reference to a FLIP nuclear core is consistent with the conversion of the reactor to LEU.

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

2. B. Pursuant to the Act and Title 10 CFR, Chapter I, Part 70, "Special Nuclear Materials," to receive, possess, and use up to a maximum of 17.9 kilograms of contained uranium 235 at various enrichments and 16 grams plutonium contained in a plutonium-beryllium neutron source in connection with operation of the reactor. Without exceeding the foregoing maximum possession limits, the specific categories of maximum limits are as follows:

	Maximum U-235	Maximum Pu	% Enrichment	Exempt Status*
(1)	4.3 kg		19.30	Exempt 10 CFR 73.6(a)
(2)	13.5 kg		70	Exempt 10 CFR 73.6(b)
(3)	.14 kg		93.22	Not Exempt
(4)	.01 kg		93.00	Not Exempt
(5)		16 grams		Exempt 10 CFR 73.6(c)
(6)	4.5 kg		70	Not Exempt

*Material is exempt provided that it meets the requirements for exemption pursuant to the cited provisions of 10 CFR 73.

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

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

(1) to receive, possess and use, in connection with operation of the facility, up to 15.0 kilograms of contained uranium-235 enriched to less than 20 percent in the form of TRIGA reactor fuel;

(2) to receive, possess and use, in connection with operation of the facility, up to 150 grams of contained uranium-235 of any enrichment in the form of neutron detectors;

(3) to receive, possess and use, in connection with operation of the facility, up to 16 grams of contained plutonium in the form of plutonium-beryllium neutron source;

(4) 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

(5) to possess, but not use, up to 18.0 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.

Up to 15 kilograms of contained uranium-235 of enrichment of less than 20 percent in the form of TRIGA reactor fuel replaces the existing possession limit. The reactor fuel consists of the new 30/20 LEU conversion fuel needed for conversion of the reactor. The licensee has provided justification for the proposed limit to allow possession of fuel on site. After the reactor is converted, the licensee has a continuing need to receive, possess and use small amounts of HEU and plutonium to allow continued operation of the reactor (e.g., nuclear chambers and neutron source) and conduct of the experimental program (e.g., flux foils and fueled experiments). Proposed license condition (5) above contains the authority to possess, but not use, the existing HEU core until it is removed from the facility. The license condition is rewritten to agree with current formatting.

License Condition 3.B, which incorporates the TS 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 changes to the license, including possession limits, associated with the conversion of the reactor and concludes that the changes are appropriate for the converted reactor.

2.10.2 Proposed Changes to Technical Specifications

TS 1, “Definitions”: The licensee has proposed changes to the definitions section of the TS to reflect the conversion to LEU fuel. The specifications currently read:

> 1.14 FUEL ELEMENT
>
> A fuel element is a single TRIGA fuel rod of either standard or FLIP type
>
> 1.18 STANDARD CORE
>
> A standard core is an arrangement of standard TRIGA fuel in the reactor grid plate.
>
> 1.19 MIXED CORE
>
> A mixed core is an arrangement of standard TRIGA fuel elements with at least 35 TRIGA-FLIP fuel elements located in a central region of the core.
>
> 1.20 FLIP CORE
>
> A FLIP core is an arrangement of TRIGA-FLIP fuel in the reactor grid plate.
>
> 1.21 OPERATIONAL CORE
>
> An operational core may be a standard core, mixed core, or FLIP core for which the core parameters of shutdown margin, fuel temperature, power calibration, and maximum allowable reactivity insertion have been determined to satisfy the requirements of the Technical Specifications.

The licensee has proposed deleting TS 1.19 and 1.20 and changing TS 1.14, 1.18, and 1.21 to read:

> 1.14 FUEL ELEMENT
>
> A fuel element is a single TRIGA fuel rod of LEU 30/20 type.
>
> 1.18 LEU 30/20 CORE
>
> A LEU 30/20 core is an arrangement of TRIGA LEU 30/20 fuel in the reactor grid plate.
>
> 1.21 OPERATIONAL CORE
>
> An operational core is an LEU 30/20 core for which the core parameters of shutdown margin, fuel temperature, power calibration, and maximum allowable reactivity insertion have been determined to satisfy the requirements of the Technical Specifications.

Because the proposed changes reflect the conversion to LEU fuel, the staff finds these proposed changes to TS 1 to be acceptable.

TS 2.1, “Safety Limits”: The licensee has proposed changes to TS 2.1 under “Specifications” and “Bases” to reflect the conversion to LEU fuel and clarify that LEU fuel is to be used for pulsing operations. The specifications currently read:

a. The temperature in a TRIGA-FLIP fuel element shall not exceed 1150°C under any conditions of operation.

b. The temperature of a standard TRIGA fuel element shall not exceed 1000°C under any conditions of operation.

c. The reactor power level shall not exceed 1500 kW under any conditions of operation.

The licensee has proposed changing the specifications to read:

a. The temperature in a TRIGA LEU 30/20 fuel element shall not exceed 1150°C under any conditions of operation.

b. The steady-state reactor power level shall not exceed 1500 kW under any conditions of operation.

The licensee has proposed changing the bases for TS 2.1 to reflect the analysis performed in the conversion SAR specific to the design of the UWNR LEU core. Because the proposed changes reflect the conversion to LEU fuel, the staff finds these proposed changes to TS 2.1 to be acceptable.

TS 2.2, “Limiting Safety System Setting”: The licensee has proposed changes to TS 2.2 under “Specifications” and “Bases” to reflect the limitations on the placement of the IFE in the core. The specifications currently read:

1. The limiting safety system setting for fuel temperature shall be 400ºC (750ºF) as measured in an instrumented fuel element. For a mixed core, the instrumented element shall be located in the region of the core containing FLIP type elements.

2. The limiting safety system setting for reactor power level shall be 1.25 MW.

The licensee has proposed changing the specifications to read:

a. The limiting safety system setting for fuel temperature shall be 400ºC as measured in an instrumented fuel element with a pin power peaking factor between 0.87 and 1.16, or 500ºC as measured in an instrumented fuel element with a pin power peaking factor of at least 1.16.

b. The limiting safety system setting for reactor power level shall be 1.25 MW.

The licensee has proposed changing the bases for TS 2.2(a) and TS 2.2(b) to reflect the analysis performed in the conversion SAR specific to the design of the UWNR LEU core. The specification is renumbered to agree with current formatting. Because the proposed changes to the limiting safety system setting (LSSS) continue to protect the safety limit and reflect the conversion to LEU fuel, the staff finds these proposed changes to TS 2.2 to be acceptable.

TS 3.2, “Pulse Mode Operation”: The licensee has proposed changes to TS 3.2 under “Bases” to reflect the analysis performed in the conversion SAR specific to the design of the UWNR LEU core. Because the proposed changes reflect the conversion to LEU fuel, the staff finds these proposed changes to TS 3.2 to be acceptable.

TS 3.3.3, “Reactor Safety System”: The licensee has proposed changes to TS 3.3.3 under “Specifications” in Table 1, and affected bases to include a new specification for pool water temperature and address changes to the LSSS and the associated fuel element scram. The affected portions of the specification read:

TABLE 1

	Safety System or Measuring Channel	Minimum No. Operable	Function & Operating Mode in Which Required
a.	Fuel Element Temperature	1	Scram at 400°C. All modes.

The licensee has proposed changing the specifications to read:

TABLE 1

	Safety System or Measuring Channel	Minimum No. Operable	Function & Operating Mode in Which Required
a.	Fuel Element Temperature	1	Scram at 400°C for IFE peaking factors 0.87-1.16 or 500°C for IFE peaking factors >1.16. All modes.
j.	Reactor Pool-water Temperature	1	Scram if water temperature is greater than 130ºF; All modes.

Calculations performed in the conversion SAR are based on a maximum core inlet temperature of 54 °C (130 °F). Initiating a reactor scram when the pool water temperature exceeds 54 °C (130 °F) will ensure the reactor does not operate in a condition that was not previously analyzed. The licensee has proposed changing the bases for TS 3.3.3(a) and (j) to reflect the analysis performed in the conversion SAR specific to the design of the UWNR LEU core. Because the proposed changes are consistent with changes discussed above to TS 2.2 and the analysis performed for the conversion to LEU fuel, the staff finds these proposed changes to TS 3.3.3 to be acceptable.

TS 5.1, “Reactor Fuel”: The licensee has proposed changes to TS 5.1 under “Specifications” and “Bases” to reflect the conversion to LEU fuel. The specification currently reads:

a. TRIGA-FLIP Fuel

The individual unirradiated FLIP fuel elements shall have the following characteristics:

(1) Uranium content: maximum of 9 Wt-% enriched to nominal 70% Uranium 235.

(2) Hydrogen-to-zirconium atom ratio (in the ZrHx): nominal 1.6 H atoms to 1.0 Zr atoms.

(3) Natural erbium content (homogeneously distributed): nominal 1.5 Wt-%.

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

(5) Identification: Top pieces of FLIP elements will have characteristic markings to allow visual identification of FLIP elements employed in mixed cores.

b. Standard TRIGA fuel

The individual unirradiated standard TRIGA fuel elements shall have the following characteristics:

(1) Uranium content: maximum of 9.0 Wt-% enriched to a nominal 20% Uranium 235.

(2) Hydrogen-to-zirconium atom ratio (in the ZrHx): nominal 1.7 H atoms to 1.0 Zr atoms.

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

The licensee has proposed changing the specifications to read:

a. TRIGA LEU 30/20 Fuel

The individual unirradiated TRIGA LEU 30/20 fuel elements shall have the following characteristics:

(1) Uranium content: maximum of 30 Wt-% enriched to maximum of 19.95 Wt-% with nominal enrichment of 19.75 Wt-% Uranium 235.

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

(3) Natural erbium content (homogeneously distributed): nominal 0.9 Wt-%.

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

The licensee has proposed changes to the bases to reflect the design features of the LEU fuel. Because the proposed changes reflect the conversion to LEU fuel, the staff finds these proposed changes to TS 5.1 to be acceptable.

TS 5.2, “Reactor core”: The licensee has proposed changes to TS 5.2 under “Specifications” and “Bases” to reflect the conversion to LEU fuel. The specification currently reads:

a. The core shall be an arrangement of TRIGA uranium-zirconium hydride fuel-moderator bundles positioned in the reactor grid plate.

b. The Triga core assembly may be standard, FLIP, or a combination, thereof (mixed core) provided that any FLIP fuel be comprised of at least thirty-five (35) fuel elements, located in a contiguous, central region.

c. The reactor shall not be operated with a core lattice position vacant except for positions on the periphery of the core assembly.

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

The licensee has proposed changing the specifications to read:

a. The core shall be an arrangement of TRIGA LEU 30/20 uranium-zirconium hydride fuel-moderator bundles positioned in the reactor grid plate.

b. The reactor shall not be operated with a core lattice position vacant except for positions on the periphery of the core assembly.

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

The licensee has proposed changes to the bases to reflect the design features of the LEU fuel. The specifications are renumbered due to the deletion of a specification no longer needed following the conversion to LEU fuel. Because the proposed changes reflect the conversion to LEU fuel, the staff finds these proposed changes to TS 5.2 to be acceptable.

TS 5.6, “Reactor Building”: The licensee has proposed changes to TS 5.6 under “Specifications” and “Bases” to reflect: (1) the analysis in the conversion SAR specific to the design of the UWNR LEU core, and (2) recent changes to the building surrounding the reactor laboratory. The specification currently reads:

b. All air or other gas exhausted from the reactor room and associated experimental facilities shall be released to the environment a minimum of 17 meters above ground level.

The licensee has proposed changing the specification to read:

b. All air or other gas exhausted from the reactor room and associated experimental facilities shall be released to the environment a minimum of 30.5 meters above ground level.

The licensee has proposed changes to the bases to reflect the current stack design and analysis performed in the conversion SAR specific to the design of the UWNR LEU core. The increase in minimum stack exhaust height is due to recent changes to the building surrounding the reactor laboratory. Because the proposed changes are consistent with the current stack design and analysis performed for the conversion to LEU fuel, the staff finds these proposed changes to TS 5.6 to be acceptable.

TS 5.7, “Reactor Pool Water Systems”: The licensee has proposed additions and changes to TS 5.7 under “Specifications” and “Bases” to reflect the updated thermal-hydraulic analysis in the conversion SAR specific to the design of the UWNR LEU core. The licensee has proposed adding the following specification:

f. A pool water temperature alarm shall indicate if water temperature reaches 130ºF.

The licensee has proposed changes to the bases of TS 5.7(a) to reflect the analysis performed in the conversion SAR specific to the design of the UWNR LEU core. Because the proposed changes are consistent with changes discussed above to TS 3.3.3(j) and the analysis performed for the conversion to LEU fuel, the staff finds these proposed changes to be acceptable.

2.10.3 Conclusions

The staff has reviewed all of the proposed changes to the TSs. The staff concludes that these changes to the TS are necessitated by the conversion of the reactor to LEU fuel. The staff has reviewed the licensee’s technical bases for these changes as discussed above and found that the bases support the proposed changes to the TS. The staff concludes that the changes to the TS continue to meet the regulatory requirements in 10 CFR 50.36, “Technical Specifications,” and that the changes to the TS 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 10 CFR 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 UWNR. 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 UWNR.

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