September 1, 2006

EA-06-210

Dr. William G. Vernetson
Director of Nuclear Facilities
University of Florida
Department of Nuclear and
Radiological Engineering
202 Nuclear Sciences Center
P.O. Box 118300
Gainesville, Florida 32611-8300

SUBJECT: ISSUANCE OF ORDER MODIFYING LICENSE NO. R-56 TO CONVERT FROM HIGH- TO LOW-ENRICHED URANIUM FUEL (AMENDMENT NO. 26) - UNIVERSITY OF FLORIDA TRAINING REACTOR (TAC NO. MC9037)

Dear Dr. Vernetson:

The U.S. Nuclear Regulatory Commission (NRC) is issuing the enclosed Order, as Amendment No. 26 to Amended Facility Operating License No. R-56, which authorizes the conversion of the University of Florida Training 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. This regulation requires that non-power reactor licensees, such as the University of Florida, convert to LEU fuel under certain conditions which the University of Florida now meets. The Order is being issued in accordance with 10 CFR 50.64(c)(3) and in response to your submittal of December 2, 2005, as supplemented on June 19 and 29, July 20 and 21, and August 4 and 22, 2006. The Order also contains an outline of a reactor startup report that you are required to provide to the NRC within six months following completion of LEU fuel loading.

The portion of the Order that changes License Condition 2.B.(2), 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 Licence Condition 2.C.(2) 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 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/**

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

Docket No. 50-83

Enclosures: 1. Order
2. Safety Evaluation

cc w/enclosures: See next page

cc:

Mayor
City of West Lafayette
609 W. Navajo
West Lafayette, IN 47906

Administrator
Department of Environmental Regulation
Power Plant of Siting Section
State of Florida
2600 Blair Stone Road
Tallahassee, FL 32301

State Planning and Development Clearinghouse
Office of Planning and Budgeting
Executive Office of the Governor
The Capitol Building
Tallahassee, FL 32301

William Passetti, Chief
Bureau of Radiation Control
Department of Health
4052 Bald Cypress Way
Tallahassee, FL 32399-1741

William Vernetson, Ph.D.
Director of Nuclear Facilities
University of Florida
202 Nuclear Science Building
Gainesville, FL 32611-8300

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Sincerely,

**/RA/**

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

Docket No. 50-83

Enclosures: 1. Order
2. Safety Evaluation

cc w/enclosures: See next page

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Tech Specs No.: ML062440059
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7590-01-P

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

| | | |
|---|---|---|
| In the Matter of | ) | |
| | ) | |
| UNIVERSITY OF FLORIDA | ) | Docket No. 50-83 |
| | ) | EA-06-210 |
| (University of Florida Training Reactor) | ) | |

ORDER MODIFYING AMENDED FACILITY OPERATING LICENSE NO. R-56

I.

The University of Florida (the licensee) is the holder of Amended Facility Operating License No. R-56 (the license) issued on May 21, 1959, by the U.S. Atomic Energy Commission, and subsequently renewed on August 30, 1982, by the U.S. Nuclear Regulatory Commission (the NRC or the Commission). The license authorizes operation of the University of Florida Training Reactor (the facility) at a power level up to 100 kilowatts thermal. The facility is a research reactor located on the campus of the University of Florida, in the city of Gainesville, Alachua County, Florida. The mailing address is Department of Nuclear and Radiological Engineering, 202 Nuclear Sciences Center, P.O. Box 118300, Gainesville, Florida 32611-8300.

II.

On February 25, 1986, the Commission promulgated a final rule, Title 10 of the Code of Federal Regulations (10 CFR) Section 50.64, limiting the use of high-enriched uranium (HEU) fuel in domestic non-power reactors (research and test reactors) (see 51 FR 6514). The regulation, which became effective on March 27, 1986, requires that if Federal Government funding for conversion-related costs is available, each licensee of a non-power reactor authorized to use HEU fuel shall replace it with low-enriched uranium (LEU) fuel acceptable to the Commission unless the Commission has determined that the reactor has a unique purpose.

The Commission's stated purpose for these requirements was to reduce, to the maximum extent possible, the use of HEU fuel in order to reduce the risk of theft and diversion of HEU fuel used in non-power reactors.

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

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

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

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

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

Section 2.309 states the requirements for a person whose interest may be affected by any proceeding to initiate a hearing or to participate as a party.

III.

On December 2, 2005, as supplemented on June 19 and 29, July 20 and 21, and August 4 and 22, 2006, the NRC staff received the licensee's conversion proposal, including its proposed modifications and supporting safety analyses. HEU fuel elements are to be replaced with LEU fuel elements. The fuel elements contain fuel plates, typical of the materials test reactor design, with the fuel consisting of uranium silicide dispersed in an aluminum matrix. These plates contain the uranium-235 isotope at an enrichment of less than 20 percent. The NRC staff reviewed the licensee's proposal and the requirements of 10 CFR 50.64 and has determined that public health and safety and common defense and security require the licensee to convert the facility from the use of HEU to LEU fuel in accordance with the attachments to this Order and the schedule included herein. The attachments to this Order specify the changes to the License Conditions, Technical Specifications and Emergency Plan that are needed to amend the facility license and contains an outline of a reactor startup report to be submitted to NRC within six months following completion of LEU fuel loading.

IV.

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

Amended Facility Operating License No. R-56 is modified by amending the License Conditions, Technical Specifications and Emergency Plan as stated in the attachments to this Order. License Condition 2.B.(2), allowing possession of LEU fuel, becomes effective 20 days after the date of publication of this Order in the *Federal Register*. All other changes become effective on the later date of either (1) the day the licensee receives an adequate number and type of LEU fuel elements to operate the facility as specified in the licensee proposal, or (2) 20 days after the date of publication of this Order in the *Federal Register*.

V.

Pursuant to the Atomic Energy Act of 1954, as amended, any person adversely affected by this Order may submit an answer to this Order, and may request a hearing on this Order, within 20 days of the date of this Order. Any answer or request for a hearing shall set forth the matters of fact and law on which the person adversely affected, relies and the reasons why the Order should not have been issued. Any answer or request for a hearing shall be filed (1) by first class mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001, Attention: Rulemaking and Adjudications Staff; or (2) by courier, express mail, and expedited delivery services to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff. Because of continuing disruptions in delivery of mail to the United States Government Offices, it is requested that answers and/or requests for hearing be transmitted to the Secretary of the Commission either by e-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, HEARINGDOCKET@NRC.GOV; or by facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, D.C., Attention: Rulemakings and Adjudications Staff at 301-415-1101 (the verification number is 301-415-1966). Copies of the request for hearing must also be sent to the Director, Office of Nuclear Reactor Regulation and to the Assistant General Counsel for Materials Litigation and Enforcement, Office of the General Counsel, with both copies addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001, and the NRC requests that a copy also be transmitted either by facsimile transmission to 301-415-3725 or by e-mail to OGCMailCenter@nrc.gov.

If a person requests a hearing, he or she shall set forth in the request for a hearing with particularity the manner in which his or her interest is adversely affected by this Order and shall address the criteria set forth in 10 CFR 2.309.

If a hearing is requested by a person whose interest is adversely affected, the Commission shall issue an Order designating the time and place of any hearing. If a hearing is held, the issue to be considered at such hearing shall be whether this Order should be sustained.

In accordance with 10 CFR 51.10(d) this Order is not subject to Section 102(2) of the National Environmental Policy Act, as amended. The NRC staff notes, however, that with respect to environmental impacts associated with the changes imposed by this Order as described in the safety evaluation, the changes would, if imposed by other than an Order, meet the definition of a categorical exclusion in accordance with 10 CFR 51.22(c)(9) and (10). Thus, pursuant to either 10 CFR 51.10(d) or 51.22(c)(9) and (10), no environmental assessment nor environmental impact statement is required.

For further information see the application from the licensee dated December 2, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML062220375), as supplemented on June 19 (ADAMS Accession Nos. ML061720498 and ML062220178) and 29 (ADAMS Accession No. ML061840285), July 20 (ADAMS Accession No. ML062050252) and 21 (ADAMS Accession No. ML062060139), and August 4 (ADAMS Accession No. ML062350107) and 22 (ADAMS Accession No. ML062400265), 2006, the staff's requests for additional information dated May 2 (ADAMS Accession No. ML061220262 with clarification dated May 18, 2006, ADAMS Accession No. ML061420119) and 22, 2006 (ADAMS Accession No. ML061380167), and the cover letter to the licensee, attachments to this Order and staff's safety evaluation dated September 1, 2006 (ADAMS Accession No. ML062440086) available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the ADAMS Public Electronic Reading Room on the Internet at the NRC Web site, http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who have problems in accessing the documents in ADAMS should contact the NRC PDR reference staff by telephone at 1-800-397-4209 or 301-415-4737 or by e-mail to pdr@nrc.gov.

FOR THE NUCLEAR REGULATORY COMMISSION

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Dated this 1st day of September 2006

Attachments: 1. Modifications to Amended Facility Operating License No. R-56
2. Modifications to Emergency Plan
3. Outline of Reactor Startup Report

ATTACHMENT TO ORDER

MODIFICATIONS TO AMENDED FACILITY OPERATING LICENSE NO. R-56

A. License Conditions Revised by This Order

2.B.(2) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess, and use (1) up to 5.0 kilograms of contained uranium-235 of enrichment of less than 20 percent in the form of material test reactor (MTR)-type reactor fuel; (2) a 1-curie sealed plutonium-beryllium neutron source; (3) a 25-curie sealed antimony-beryllium neutron source; and (4) up to 0.2 kilograms of contained uranium-235 of any enrichment in the form of fission chambers, flux foils and other forms, all used in connection with operation of the reactor.

2.C.(2) Technical Specifications

The technical specifications contained in Appendix A, as revised through Amendment No. 26, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the technical specifications.

B. The technical specifications will be revised by this Order in accordance with the "Enclosure to License Amendment No. 26, Amended Facility Operating License No. R-56, Docket No. 50-83, Replacement Pages for Technical Specifications," and as discussed in the safety evaluation for this Order.

ENCLOSURE TO LICENSE AMENDMENT NO. 26

AMENDED FACILITY OPERATING LICENSE NO. R-56

DOCKET NO. 50-83

REPLACEMENT PAGES FOR TECHNICAL SPECIFICATIONS

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

| Remove | Insert |
|---|---|
| 4 | 4 |
| 5 | 5 |
| 6 | 6 |
| 7 | 7 |
| 8 | 8 |
| 9 | 9 |
| 13 | 13 |
| 15 | 15 |
| 16 | 16 |
| 21 | 21 |
| 23 | 23 |
| 24 | 24 |
| 26 | 26 |
| 38 | 38 |

ATTACHMENT TO ORDER

MODIFICATIONS TO EMERGENCY PLAN

Replace the following pages of the Emergency plan for the University of Florida Training Reactor with the enclosed pages.

| Remove | Insert |
|---|---|
| Cover page | Cover page |
| v | v |
| 1-1 | 1-1 |
| 1-6 | 1-6 |
| 1-12 | 1-12 |
| 1-13 | 1-13 |
| 1-14 | 1-14 |
| 5-1 | 5-1 |
| 6-1 | 6-1 |
| 10-4 | 10-4 |

ATTACHMENT TO ORDER

OUTLINE OF REACTOR STARTUP REPORT

Within six months following completion of initial LEU core loading, submit the following information to the NRC. Information on the HEU core should be presented to the extent it exists.

1. Critical Mass

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

2. Excess (operational) reactivity

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

3. Regulating and Safety control rod calibrations

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

4. Reactor power calibration

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

5. Shutdown margin

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

6. Partial fuel element worths for LEU

   Measurements of the worth of the partial loaded fuel elements

7. Thermal neutron flux distributions

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

8. Reactor physics measurements

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

9. Initial LEU core loading

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

10. Primary coolant measurements

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

11. Discussion of results

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING CONVERSION ORDER TO CONVERT FROM

HIGH-ENRICHED TO LOW-ENRICHED URANIUM FUEL

AMENDED FACILITY OPERATING LICENSE NO. R-56

UNIVERSITY OF FLORIDA TRAINING REACTOR

DOCKET NO. 50-83

## 1.0 INTRODUCTION

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.64 requires licensees of research and test reactors to convert from the use of high-enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel, unless specifically exempted. The University of Florida (the licensee) has proposed to convert the fuel in the University of Florida Training Reactor (UFTR) from HEU to LEU. In a letter dated December 2, 2005, as supplemented on June 19 and 29, July 20 and 21, and August 4 and 22, 2006, the licensee submitted its proposal for conversion requesting approval of the fuel conversion and of changes to its Technical Specifications. To support this action the licensee submitted a conversion Safety Analysis Report (SAR) on which the HEU to LEU conversion and the Technical Specification changes are based. The licensee also submitted proposed changes to the Emergency Plan for the University of Florida Training Reactor. This Safety Evaluation Report provides the results of the NRC staff's evaluation of the licensee's conversion proposal. The evaluation was carried out according to the guidance found in NUREG-1537.[1]

## 2.0 EVALUATION

### 2.1 Summary of Reactor Facility Changes

The UFTR is a 100 kW(t) nuclear research reactor that uses Materials Testing Reactor (MTR)-type plate fuel. The HEU to LEU conversion requires the use of a different fuel type and core configuration. The LEU fuel element (19.75% enriched) has the same basic design (MTR-type fuel) as the present HEU fuel element (93% enriched). The LEU fuel element contains 14 fuel plates with $U_3Si_2$-Al fuel meat while the HEU fuel element contains 11 fuel plates of U-Al alloy fuel meat. The cladding of the HEU fuel plates is composed of 1100 Al alloy while the LEU fuel cladding is composed of 6061 Al alloy.

### 2.2 Fuel and Core Design

The major changes in the fuel composition and the fuel element dimensions (as well as other parameters) are given in Table 1. The fuel meat will change from U-Al alloy to a dispersion fuel consisting of $U_3Si_2$ in Al. Generic aspects of the LEU silicide fuel have been reviewed by NRC and the fuel is approved by the NRC for use in research and test reactors with slab fuel plates.[2] The licensee submitted their application for conversion to justify the specific use of the fuel in the UFTR. The NRC generic approval is for fuel with uranium densities up to 4.8 g/cm$^3$. The UFTR LEU fuel will have a uranium density of 3.5 g/cm$^3$.

## Table 1 Summary of Key Nominal Design Parameters of HEU (current) and LEU (proposed) Cores

**DESIGN DATA**

| | HEU | LEU |
|---|---|---|
| Fuel Type | U-Al alloy | $U_3Si_2$-Al |
| Fuel Meat Size | | |
| Width (cm) | 5.96 | 5.96 |
| Thickness (cm) | 0.102 | 0.051 |
| Height (cm) | 60.0 | 60.0 |
| Fuel Plate Size | | |
| Width (cm) | 7.23 | 7.23 |
| Thickness (cm) | 0.178 | 0.127 |
| Height (cm) | 65.1 | 65.1 |
| Cladding | 1100 Al | 6061 Al |
| Cladding Thickness (cm) | 0.038 | 0.038 |
| Fuel Enrichment (nominal) | 93.0 % | 19.75% |
| "Meat" Composition (wt% U) | | 62.98 |
| Mass of $^{235}U$ per Plate (g, nominal) | | 12.5 |
| Number of Plates per Fuel Element | 11 | 14 |
| Number of Full Fuel Elements (current/expected) | 21 | 22 |
| Number of Partial Fuel Elements | 1<br>(5 fuel plates + 5 dummy plates) | 1<br>(13 plates in fueled and dummy pair) |
| Number of Dummy Elements | 2 | 1 |

**REACTOR PARAMETERS**

| | HEU | LEU |
|---|---|---|
| Fresh Core Excess Reactivity (% Δk/k) | 1.09 | 0.925 |
| Shutdown Margin (%Δk/k) | 3.11 | 3.17 |
| Control blade worth, (for LEU from fresh to depleted) | | |
| Regulating (% Δk/k) | 0.87 | 0.63-0.66 |
| Safety 1 (% Δk/k) | 1.35 | 1.62-1.65 |
| Safety 2 (% Δk/k) | 1.63 | 1.81-1.76 |
| Safety 3 (% Δk/k) | 2.06 | 1.42-1.46 |
| Maximum Reactivity Insertion Rate (% Δk/k/s) | 0.042 | 0.045 |
| Average Coolant Void Coefficient (% Δk/k/%void) | -0.148 | |
| Fresh Core | | -0.153 |
| Depleted Core | | -0.146 |
| Coolant Temp. Coefficient (% Δk/k/°C) | | |
| Fresh Core | -5.91E-03 | -5.68E-03 |
| Depleted Core | | -5.26E-03 |
| Fuel Temp. Coefficient (% Δk/k/°C) | | |
| Fresh Core | -2.91E-04 | -1.65E-03 |
| Depleted Core | | -1.49E-03 |
| Effective Delayed Neutron Fraction | | |
| Fresh Core | 0.0079 | 0.0077 |
| Depleted Core | | 0.00756 |
| Neutron Lifetime (ms) | | |
| Fresh Core | 187.4 | 177.5 |
| Depleted Core | | 195.1 |

**THERMAL-HYDRAULIC PARAMETERS** (100 kW(t), 43 gpm, $T_{in}$=30 °C)

| | HEU | LEU |
|---|---|---|
| Max. Fuel Temperature[a] (°C) | 66.5 | 64.5 |
| Max. Clad Temperature (°C) | 66.5 | 64.4 |
| Mixed Mean Coolant Outlet Temperature (°C) | 40.8 | 40.5 |
| Max. Coolant Channel Outlet Temp. (°C) | 58.3 | 59.1 |

| | | |
|---|---|---|
| Minimum onset of nucleate boiling ratio | 1.98 | 2.09 |
| Minimum deviation from nucleate boiling ratio | 354 | 376 |

[a] At nominal operating conditions

In 1991, the Iowa State University successfully converted their Argonaut reactor (UTR-10) facility using the same type of fuel plate. Following closure of the Iowa State reactor, fuel inspection revealed the presence of unexpected corrosion. This issue has been analyzed in an INL/ANL report.[3] To minimize the possibility of corrosion, the manufacturer (BWXT) of the LEU fuel for the UFTR will apply a surface treatment resulting in a protective boehmite layer on the surface of the cladding. In addition to the surface treatment, the licensee has proposed a new limit on primary coolant pH of less than 7.0. The pH of the primary coolant will be measured weekly. The licensee currently controls the resistivity of the primary coolant. The licensee included the effect of the boehmite layer in its analysis and concluded that the presence of this coating will not impact the thermal-hydraulic analysis. The staff has reviewed the use of the surface treatment and proposed pH limits and concludes that corrosion will be acceptably controlled in the LEU core.

There will be more fuel plates per element with the LEU design than with the HEU design (14 verses 11) and the plates will be thinner. The UFTR has proposed a change to its Technical Specification 3.7, “Fuel and Fuel Handling” in order to reflect the change in the number of plates per fuel element. While the height and width dimensions of the fuel elements will remain the same, the thickness of the fuel meat will be reduced from 0.102 cm to 0.051 cm. Each LEU fuel plate will be 0.127 cm thick verses 0.178 cm for HEU fuel. The licensee has proposed changes to Technical Specification 5.4, “Reactor Core,” to reflect the conversion to LEU fuel. Even though the volume of each fuel plate is reduced by a factor of two and the enrichment is reduced from 93.0% to 19.75%, the higher uranium density in the LEU fuel means each fuel plate will contain 12.5 g of U-235 verses 14.5 g of U-235 in the HEU plates. With the additional fuel plates in each fuel element, the total amount of U-235 in each fuel element will be 175 g verses 160 g for HEU fuel and for the entire core the U-235 content increases from 3422 g to 3975 g. The increase in U-235 is to compensate for the increased amount of U-238 in the LEU core relative to the HEU core.

The reactor core has spaces for 24 fuel elements in 6 fuel boxes. The fuel elements are arranged in four rows of six elements with the rows running east and west. The fuel boxes are inserted in reactor grade graphite with 30.48 cm of graphite separating the rows on the north side from the rows on the south side of the core. The reactor is cooled with light water forced upward through the core between and around the fuel plates.

The critical LEU core is calculated to contain 22 full fuel elements and one element containing 10 fuel plates and 3 dummy plates (because of the way partial LEU elements are assembled, they can only contain a maximum of 13 plates as opposed to a full fuel element than can contain 14 plates). The HEU core has 21 fully fueled fuel elements and one fuel element containing 5 fuel plates and 5 dummy plates (because of the way partial HEU elements are assembled, they can only contain a maximum of 10 plates as opposed to a full fuel element than can contain 11 plates).

The licensee calculated neutron flux, power distributions and critical mass for the anticipated startup core. The effects of reconfiguring the core at a later date (e.g., to replace damaged fuel or extend core life) was not considered in the conversion SAR. The licensee expects the present core to have a life expectancy of approximately 20 years. If it becomes necessary to reconfigure the core, the licensee will need to perform a review under 10 CFR 50.59 to determine if changes to the core can be made without prior NRC review and approval.

The control of the reactor is from four control blades which pivot from the side of the core and swivel down into the core. Each blade has a cadmium tip on it. Three of the blades are referred to as safety blades and one is referred to as the regulating blade. Each safety blade contains 40.3 $cm^3$ (348 g) of cadmium and the regulating blade contains 13.6 $cm^3$ (118 g) of cadmium. There are two blades on the north side of the reactor and two on the south side of the reactor. The blades are swung between the fuel elements, so each row of six fuel elements has two fuel elements, then one blade, two elements, one blade and then two elements. No changes to the control blades are proposed as part of the conversion.

The staff has reviewed the proposed fuel and core design of the LEU reactor. The staff concludes that the conversion from HEU to LEU fuel will not impact the overall basic design of the core and its control. The major change will be a physically larger fueled section of the core and a larger number of thinner plates per fuel element. The power of the core will remain at 100 kW(t) so the average power per plate and per fuel element will both be reduced which is conservative. Therefore, the staff finds the fuel and core design acceptable.

## 2.3 Nuclear Design

### 2.3.1 Calculational Methodology

In order to carry out the UTFR neutronic analysis, an MCNP5 model was developed for a fresh core and for a depleted core. The inventories for the depleted core were developed using the SAS2 sequence in the SCALE5 package. Both of these computer codes are state-of-the-art and used for many nuclear reactor analyses. The model was validated by comparing calculated to measured reaction rates for six bare and cadmium covered gold activation foils that were irradiated in-core and by comparing calculated and measured control blade worth. The comparisons show reasonable agreement. Therefore, the staff concludes that the calculational methodology used by the licensee is acceptable.

### 2.3.2 Neutron Flux and Power Distributions

Extensive neutron flux calculations were performed and are presented in Appendix A of the conversion SAR. The results of these calculations show no substantial changes to the neutron flux distribution upon conversion. The power distribution calculations likewise showed only small changes. The reference critical LEU fueled core has a slightly lower power density than the HEU core. The critical mass of the LEU core was slightly larger than the HEU core as discussed above. The largest increase in power in a full LEU fuel element was 5% in element 2-2, but this was not the hottest element. In the HEU core, the hottest fuel element power was calculated to be 5.36 kW(t) and in the LEU core the hottest fuel element is calculated to be 5.09 kW(t), while the total core power will remain at 100 kW(t). The changes to neutron flux and power distributions due to conversion are not significant and are as expected. Therefore, the staff concludes that neutron flux, power distributions and critical mass predictions in the LEU core are acceptable.

### 2.3.3 Excess Reactivity, Shutdown Margin and Control Blade Worth

The excess reactivity and control blade worths were calculated using the MCNP5 model. The excess reactivity calculated for a set of fresh fuel elements is reduced by going from the HEU core (1.09% $\Delta k/k$) to the LEU core (0.93% $\Delta k/k$). A reduction in the allowable maximum excess reactivity in the reactor core has been proposed by the licensee in Technical Specification 3.1. The maximum excess reactivity, without xenon poisoning, was 2.3% Δk/k for the HEU core and is proposed as 1.4% Δk/k for the LEU core. This change is conservative albeit more realistic than the previous value given the licensee's experimental program and operations. It still provides margin for burnup, fission product transients, and operational flexibility. Therefore, the staff concludes that the proposed limit on excess reactivity of the LEU core is acceptable.

Because the power distribution will be shifted from the north to the south, the two blades in the south part of the core have higher worths than those on the north side. There is about an 8% relative decrease in the overall control blade worth with the conversion to LEU, which leaves adequate ability to shut down the reactor. Therefore, the staff concludes that the control blade worths of the LEU core are acceptable.

Technical Specification 3.1.1 states that the minimum shutdown margin, with the most reactive control blade withdrawn be no less than 2% Δk/k. The HEU core was calculated to have a shutdown margin of 3.11% Δk/k and the measured value is 3.01% Δk/k. The calculated value for the LEU core is 3.17% Δk/k, well within the Technical Specification. Therefore, the staff concludes that acceptable ability to shut down the reactor after the conversion to LEU fuel will be maintained.

Technical Specification 3.1 (4), "Maximum Single Blade Reactivity Insertion Rate," sets a limit on the reactivity insertion when removing the most reactive control blade. This limit is 0.06% Δk/k/s. The licensee has not proposed to change this limit because of the conversion from HEU to LEU fuel. With the HEU core, the most reactive blade is #3 and with the LEU core the most reactive blade is #2. The drive mechanisms will not be changed for the conversion so the maximum speed that a blade can be withdrawn is the full range of motion divided by 100 seconds. For the HEU core the maximum rate of reactivity insertion is calculated to be 0.042% Δk/k/s and for the LEU core it is calculated to be 0.045% Δk/k/s. Therefore, the staff concludes that Technical Specification 3.1 (4) will continue to be met after conversion with no changes to the control blades which is acceptable.

2.3.4 Dynamic Parameters

The prompt neutron lifetime, λ, and the effective neutron fraction, $\beta_{eff}$, change slightly as a result of the conversion from HEU to LEU fuel. $\beta_{eff}$ was calculated by determining $k_{eff}$ with and without delayed neutrons and goes from 0.00793 (HEU) to 0.00771 (LEU), a 2.8% decrease which is not significant. The neutron lifetime was calculated using the 1/v method and goes from 187.4 μs (HEU) to 177.5 μs (LEU), a 5.3% reduction which is not significant. For the depleted LEU core the values of $\beta_{eff}$ and neutron lifetime become 0.0076 and 195.1 μs, respectively. The staff concludes that these changes are acceptable and do not significantly change the dynamic behavior of the core.

All reactivity feedback coefficients remain negative (see Table 1). The change from HEU to LEU changes the void coefficient from -0.00148 to -0.00153 Δρ/%void, respectively. The change from HEU to LEU changes the water temperature coefficient from -5.91E-5 to -5.68E-5 Δρ/°C, respectively. The changes with depletion in the LEU core of these coefficients are small. The staff concludes that these changes are acceptable and do not significantly change the dynamic behavior of the core.

The fuel temperature coefficient has the most significant change because of the enhancement of the Doppler effect when additional U-238 is introduced into the reactor core. It goes from -0.291E-5 to -1.65E-5 Δρ/°C based on a fuel temperature change from 21 °C to 227 °C. The fuel temperature coefficient is the most important component of the power coefficient and the conversion to LEU significantly increases the value of the coefficient resulting in a stronger negative reactivity feedback with increasing fuel temperature. For the depleted core, the coefficient decreases to -1.49E-5 Δρ/°C but is still significantly greater than the HEU coefficent. Therefore, the staff concludes that the increase in fuel temperature coefficient in the LEU core is acceptable.

2.3.5 Conclusions

For most of the key neutronic characteristics of the UFTR core, the conversion from HEU to LEU fuel will not cause any significant changes. The most significant change will be in the fuel temperature coefficient which, because of the increase of U-238 in the core, changes from -0.291E-5 to -1.65E-5 Δρ/°C. This would reduce the peak temperature reached during a reactivity addition excursion. The maximum power in the hottest fuel element will also be lowered by the conversion. Therefore, the staff concludes that the changes in neutronic characteristics of the core due to conversion are acceptable.

2.4 Thermal-Hydraulic Design

As noted in Section 2.2, the design of the fuel elements and their placement in the core differs in the LEU core. This design results in a reduction in average power generation in a fuel plate and a narrower water gap between plates than the corresponding HEU fuel elements. However, the fuel meat of an LEU plate only has half the thickness of the fuel meat of a HEU plate and both types of fuel plates have the same clad thickness. In total, the water flow area between plates in each fuel element is slightly larger in the LEU fuel elements than in the HEU fuel elements. The LEU core, with one more fuel element than the HEU core, will have an average power per fuel element that is lower than that for the HEU core.

Section 4.7 of the conversion SAR discusses the thermal design of the LEU core under all anticipated reactor operating conditions. The analysis compares the thermal-hydraulic conditions of the LEU and HEU cores at nominal conditions at the maximum steady state power of 100 kW(t). The calculations include hot channel factors that account for uncertainties in heat flux, enthalpy rise in the coolant channel, film temperature rise, and heat transfer coefficient. The analysis also determines the LEU core limiting safety system settings (LSSSs) on power, coolant flow rate, and core coolant inlet and outlet temperatures taking into account systematic uncertainties in measurements of reactor power, coolant flow rate, and coolant temperatures. Accident analysis presented in Section 13 of the conversion SAR confirms that the LSSSs are selected conservatively to ensure that the maximum fuel and clad temperatures are well below the safety limit of 530 C (986 F).

The thermal-hydraulic design of the LEU core was analyzed by considering four identical LEU fuel elements in one fuel box. The power generated by each element was assumed to correspond to the maximum element power for the LEU core. All fuel plates were assumed to have the same axial power distribution as the hottest plate in the LEU core. A uniform power distribution was assumed across the width of the plates. The analysis assumed the smallest fuel element cross sections and the largest fuel box allowed by the manufacturing tolerances.

This, together with a conservatively small hydraulic resistance for the grid plate, results in more flow bypassing the coolant channels between the fuel plates. Since the flow velocity is lowest in the channels between the fuel plates, the axial pressure is dictated by flows outside the fuel elements. This condition tends to keep the pressure uniform among all flow channels at each axial location and minimizes diversion of flow from the interior channels (between fuel plates).

The computer code PLTEMP/ANL V 3.0 was used for the thermal-hydraulic analysis. The code modeled all fuel plates and coolant channels defined by the four fuel elements inside one fuel box. Heat transfer and pressure drop were modeled for both laminar and turbulent flow with separate laminar forced convection heat transfer coefficients for one-sided and two-sided heating conditions. PLTEMP has been verified by using alternate methods and hand calculations.

Four hot channel factors were used in the analysis to account for random and systematic uncertainties. Random hot channel factors include uncertainties in local heat flux ($F_q$), enthalpy rise in the coolant channel ($F_{bulk}$), and film temperature rise ($F_{film}$). A systematic uncertainty of 20% was assigned to the heat transfer coefficient and the hot channel factor ($F_h$) applied to all fuel plates in the analysis. The hot channel factors were quantified by using either fuel plate manufacturing specifications or engineering judgment.

In quantifying the hot channel factors special considerations were given to the fact that the UFTR core operates under laminar flow conditions and that each fuel plate has two associated coolant channels, one on each side.

It is noted in the UFTR conversion SAR that the peak-to-average width-wise variation in plate power is about 1.3. The thermal-hydraulic analysis did not explicitly account for the width-wise power peaking. A detailed computational fluid dynamics (CFD) solution using the STAR-CD code was used in the conversion SAR to demonstration significant mitigation of local width-wise peaking due to heat conduction across the width of the fuel plate. The CFD peak plate temperature agreed quite well with the one-dimensional PLTEMP solution for a sample problem when a systematic hot channel factor of 1.2 was used to reduce the heat transfer coefficient calculated by PLTEMP. It is noted that the validity of ignoring the width-wise power peaking in the UFTR analyses is partly supported by the hot channel factor for the heat transfer coefficient.

Analysis was performed to determine the thermal-hydraulic parameters of the UFTR under normal operating conditions for both the HEU and LEU cores. In addition, analyses were done to select the LSSSs.

Two thermal-hydraulic phenomena were used to gauge the thermal margin, onset of nucleate boiling (ONB) and deviation from nucleate boiling (DNB). The ratio of the heat flux at the onset of nucleate boiling to the local heat flux (ONBR) was calculated by using the Bergles-Rohsenow correlation. The margin to DNB was evaluated by calculating the DNB ratio (DNBR), the ratio of the DNB heat flux to the local heat flux, using the Groeneveld Lookup Table.

The thermal-hydraulic analyses applied the same code, assumptions and methods to the LEU core and the HEU core. Results (see Table 4-23 of the conversion SAR) are consistent with the nominal conditions of core flow, coolant inlet temperature, and reactor power of 100 kW(t) assumed for the calculations. The predicted maximum fuel and clad temperatures are slightly lower for the LEU core than the HEU core. Although the LEU core has a higher peak-to-average ratio for power generation than the HEU core (1.47 versus 1.46 at beginning-of-life), the additional fuel plates in the LEU core bring the actual heat flux lower than that for the HEU core.

The weight percent of uranium loading in an LEU fuel plate is more than four times higher than that of an HEU fuel plate. The effective thermal conductivity of the LEU fuel meat is lower than that of the HEU fuel meat because of the lower aluminum content. However, the thickness of the fuel meat of an LEU plate is only half that of an HEU plate. These two factors in conjunction with a lower maximum heat flux result in a lower maximum fuel temperature for the LEU core as compared to the HEU core. With a slightly larger total channel (gap) flow area than the HEU core, the LEU core is predicted to have a slightly lower mixed mean coolant outlet temperature than that for the HEU core. The predicted maximum coolant outlet temperature however is about one degree C higher in the LEU core than in the HEU core. Both the minimum ONB ratio and the minimum DNB ratio are higher in the LEU core than in the HEU core. The predicted DNBR value is greater than 350 for both the HEU and LEU cores. Thus, DNB is not a concern for the nominal operation of the UFTR.

ONB is used as a precursor to flow excursion. When the channel pressure drop begins to increase with decreasing flow because of boiling, a necessary condition is created for flow excursion. The UFTR employs a conservative criterion for establishing LSSSs that protects against exceeding the safety limit. It is based on the premise that by preventing the ONBR from getting below 1.0 there would be no boiling induced flow instability that could lead to burnout of the fuel cladding. For a nominal coolant flow rate of 43 gpm, the thermal-hydraulic analysis indicates that initiation of ONB would occur at power levels about twice the nominal operating power of 100 kW(t) for both the HEU and LEU cores. This demonstrates that there is adequate thermal-hydraulic safety margin under normal operating conditions for the HEU and the LEU cores.

In the conversion SAR the licensee has proposed a different safety limit for the LEU core. The safety limit is based on the fuel and clad temperature being below the blister temperature of the 6061 aluminum cladding. The proposed UFTR safety limit is the fuel and clad temperature not exceeding 530 °C (986 F) and it is based on measurements (NUREG-1313[2]) of first fission product release from fuel near the blister temperature (~550 °C).

The thermal analysis of the conversion SAR also contains the derivation of the LSSSs that prevents the UFTR from exceeding the safety limit. The LSSSs for the LEU core based on the thermal analysis are: reactor power, coolant flow rate, and coolant inlet and outlet temperatures. A new LSSS on coolant inlet temperature was introduced because the existing LSSS on coolant outlet temperature cannot be independent of the inlet temperature. Using an ONBR value of 1.0 as the threshold for establishing the thermally based LSSSs, a set of calculations were done for the LEU core to define the operating parameter map for the UFTR core. The map has the reactor power as the ordinate, the coolant flow rate as the abscissa, and the coolant inlet temperature as a parameter. The data for the selection of the LSSSs was generated by calculating the reactor power at which ONBR has a value of 1.0 for several primary coolant flow rates (18, 30, 43, and 50 gpm) and three different coolant inlet temperatures (30, 37.8, and 43.3 °C). The thermally based LSSSs were selected by taking the boundary of the operating map where the ONBR had a value of 1.0 and reducing the boundary value by an amount equal to the estimated systematic uncertainty associated with the boundary parameter.

On the manufacturing drawings for the fuel elements nominal spacing for the water channel (the gap between the fuel plates) at the bolted ends of a fuel element is 110-112 mils with a tolerance of 1 mil. The operation map used in the determination of the LSSSs in the conversion SAR was initially based on a tolerance of 1 mil for the water channel spacing. The conversion SAR was submitted to the NRC before any fuel elements were manufactured. Based on the first fuel elements manufactured, the as-built tolerance was found to vary up to a maximum of 20 mils for a minimum water channel spacing of 90 mils. In order to limit the variability of the water channel gap, additional spacers in the form of aluminum combs were installed at the nominal quarter-points along the fuel plate length (the original design already had aluminum spacers welded onto the edges of the plates at about the half height point). In case any one of the channel spacings is less than 90 mils, another comb will be added to maintain a minimum channel spacing of 90 mils. The “teeth” of the combs have negligible effect on the pressure drop in the water channels between the fuel plates. The addition of the combs repositions the fuel elements in the fuel boxes due to the thickness of the combs extending beyond the edges of the fuel elements (the original spacer at the half height point is flush with the element). Additional analysis was performed by the licensee to generate operating maps taking into consideration the larger tolerance on the water channel spacing and the presence of combs. The larger water channel spacing tolerance has the effect of increasing the hot channel factors for bulk enthalpy rise and film temperature rise. The body of the combs created a bypass flow channel pathway that is located between the fuel plate ends and the box wall. The new analysis assumed that this pathway is a clear channel and conservatively ignored the blockage created by the presence of the combs. The analysis did recognize the reduction in the width of the central water channel slot between fuel elements due to the reposition of the fuel from the box wall. The new analysis was done for water channel spacing tolerance of 10, 15 and 20 mils.

Based on the additional analysis, the following water channel spacing tolerance dependent LSSSs were proposed by the licensee:

- Power level not to exceed 119 kW(t) (includes 5% uncertainty in reactor power) at any flow rate

- Primary coolant flow rate greater than 36 gpm (includes 5% uncertainty in coolant flow) at all power levels greater than 1 watt if the fuel channel spacing tolerance is #15 mils

- Primary coolant flow rate greater than 41 gpm (includes 5% uncertainty in coolant flow) at all power levels greater than 1 watt if the fuel channel spacing tolerance is #20 mils

- Average primary coolant inlet temperature not to exceed 109 F (includes 1 F uncertainty in temperature) when the fuel coolant channel spacing tolerance is #10 mils

- Average primary coolant inlet temperature not to exceed 99 F (includes 1 F uncertainty in temperature) when the fuel coolant channel spacing tolerance is #20 mils

- Average primary coolant outlet temperature not to exceed 155 F when measured at any fuel box outlet (includes 1 F uncertainty in temperature)

One of the design features not explicitly included in the PLTEMP calculations is the presence of a handle that closes off the top of the central coolant channel. This means for the central coolant channel all coolant flow is out the sides of the fuel element. The handle design is similar for both the HEU and LEU fuel elements used in UFTR. The impact of the handle on this coolant flow path was considered by the licensee. The PLTEMP code was modified to include the effect of additional pressure loss due to the handle in the central channel. The new model was used to determine the power level at which ONB occurs for added K-loss factors (due to the handle) of 0 - 100 in the central channel. The analysis shows that the effect of the K-loss factor is not monotonic. Up to a K-loss factor of about 20 the ONB power is increasing with the loss factor. There is a turn around in the trend at a K-loss factor of about 23 when the ONB power begins to decrease with increasing K-loss factor. The reference point of K-loss factor of 0 corresponds to the case of a tolerance of 20 mils on channel spacing as shown in Figure 5 of the licensee's submittal of August 4, 2006. The conditions of the reference point are: a power level of 133 kW(t), a coolant flow rate of 39 gpm, and an inlet temperature of 100 F. The licensee notes that changes relative to this reference point will be very similar for other values of coolant flow rate and inlet temperature. For a K-loss factor of 0 (no central handle) the limiting channel on ONB power is the first interior channel and not the cental channel. The ONB power increases with K-loss factor initially because reducing flow in the central channel increases flow slightly in the other channels. When the K-loss factor exceeds a value of about 20 the limiting channel switches to the cental channel with the handle and from that point on the ONB power varies inversely with the K-loss factor. An estimate on the value of the K-loss factor for the cental handle was obtained by considering a rectangular channel (coolant channel between fuel plates) with a sharp corner in the turn (coolant exiting the cental channel through the side of the flow channel with the side branches created by the presence of the combs). Based on data from an accepted engineering handbook[5] on this subject and considering two 90-degree elbows, the K-loss factor is estimated conservatively to be less than 20. Therefore, the staff concludes that the ONB power will remain the same or somewhat increases slightly with the addition of the handle and combs in the analysis.

The thermal analysis considered the single failure of a LSSS channel. The thermal analysis assumed for the accident analysis that the first LSSS set point reached during the accident progression did not scram the reactor and that the accident was terminated by the second LSSS set point reached. Under these conditions, all analyzed accidents for the UFTR resulted in peak fuel and clad temperatures well below the safety limit of 530 C.

The staff concludes that the thermal-hydraulic analysis reported in the UFTR conversion SAR adequately demonstrates that the conversion from an HEU to an LEU core results in no significant decrease in safety margins in regard to thermal-hydraulic conditions. For most of the thermal-hydraulic parameters, such as the maximum fuel temperature and the core outlet temperature, the predicted values for the LEU core are lower than that for the HEU core. The analyses were done with qualified calculational methods and conservative or justifiable assumptions. The proposed safety limit protects the integrity of the primary barrier (fuel cladding) that protects against the uncontrolled release of radioactivity in accordance with 10 CFR 50.36(c)(1). A conservative criterion of no ONB was used in the development of LSSSs and these limits are incorporated in the revised Technical Specifications. The LSSSs protect the safety limit from being exceeded.

## 2.5 <u>Accident Analysis</u>

### 2.5.1 Reactivity Insertion Transients

The first of two hypothetical reactivity insertion scenarios considered in the conversion SAR is a step insertion of 0.6% Δk/k, which is the Technical Specification limit for the worth of all moveable or non-secured experiments. The assumption was that this reactivity was inserted in a time period of 100 ms. The analysis was done using the RELAP5-3D code with point kinetics. Cases were run both with and without reactor trip (SCRAM) and using conservative assumptions for reactivity feedback coefficients and control blade worth. The calculations were performed with different combinations of coolant flow rate and inlet temperature, including the conservative combination of flow rate at 34 gpm and inlet temperature at 109 F. The assumed flow rate of 34 gpm is lower than the LSSS value of 36 gpm for all power levels greater than 1 watt if the fuel coolant channel spacing tolerance is #15 mils.

The results show very small temperature increases in the fuel and water for the SCRAM cases with a control blade drop time of 1.0 second. The results are similar for the HEU and LEU cores even with different dynamic nuclear properties and the different thermal properties of the fuel plates. For the situation without SCRAM, the core reaches an equilibrium power level of less than 1.5 MW(t) and the coolant temperature increases to the saturation level and boiling begins. However, under boiling conditions, the peak fuel temperature of about 108 C is well below the safety limit of 530 C.

The second hypothetical reactivity insertion accident is a ramp insertion of 0.06% Δk/k/s. This scenario represents the insertion of reactivity due to control blade withdrawal at the maximum rate allowed by the Technical Specifications. Reactor trip is assumed to occur at a reactor power of 125 kW(t). The peak power reaches 127 KW(t) and the increase in coolant and fuel temperatures is modest and similar in both the HEU and LEU cores.

The staff concludes that the licensee has analyzed acceptable reactivity insertion transients. The reactor safety limit is not exceeded during the transients, therefore, the LEU reactor behavior under reactivity insertion transients is acceptable.

### 2.5.2 Loss-of-Coolant Accident

The loss-of-coolant accident (LOCA) was not re-analyzed for the conversion from HEU to LEU fuel because the LEU fuel element has a larger coolant volume fraction and a lower power per fuel plate. These two factors have been identified and discussed in the review of the thermal-hydraulic analysis. Consistent with the results of the thermal-hydraulic analysis for the HEU core and the LEU core, it is reasonable to project that the expected increase in fuel temperature in the LEU core in a LOCA should be approximately equal to the increase for the HEU core. The predicted temperature rise for the HEU core was less than 17 C and this increase results in a cladding temperature that is far below the safety limit of 530 C. The staff concludes that since the cladding integrity is maintained in a LOCA for the HEU core and the LEU core is predicted to behave similarly to the HEU core, the consequences of a LOCA for an LEU core will not be more severe than that for a HEU core and is therefore, acceptable.

### 2.5.3 Fuel Handling Accident

The fuel handling accident consists of damage to fuel equal to stripping the cladding off one fuel plate. The licensee's analytically generated radionuclide inventory for the new LEU fuel was reviewed to determine if the assumptions and boundary conditions were consistent with those previously used for the HEU fuel. Fission product inventories were calculated for both an HEU and LEU fuel element using the ORIGEN-S computer code, which is used within the nuclear industry for such purposes. It was assumed that the reactor had not operated at power (greater than 1 kW(t)) for three days before the accident. To limit the fission product inventory of the core consistent with this assumption, the licensee proposed a new technical specification requirement that at least three days pass since the last reactor operation at power before the last two layers of concrete block shielding is moved from the reactor. Both cases assumed 100% of the gaseous activity produced within the recoil range of the fuel particles (2.7% of the total volatile activity) instantaneously escapes from the fuel plate into the reactor cell, with no water in the reactor tank. Other boundary conditions used for both fuel types were identical. The licensee performed additional calculations to verify that all gaseous fission products except Kr-85 had reached their equilibrium concentrations. Comparison of the results for the HEU and LEU calculated inventories showed that the LEU inventory was lower for all the radionuclides analyzed by approximately 26%, which is to be expected due to the lower power density of the LEU fuel.

Parameters and the methodology used to calculate exposure doses for the UFTR were reviewed against the guidance provided in NUREG-1537. Radiological exposure to the public was calculated at a distance of 16.5 m (UFTR fence which is the closest point of public access) and 190 m (nearest permanent residence) from the UFTR. Other assumptions used in the site boundary dose calculation are found in NUREG/CR-2079[4] and the relevant request for additional information responses. Review of the references, relevant sections of the conversion SAR, and responses to a request for additional information demonstrated that the methodology used to calculate the occupational and public radiation doses was adequate.

Calculations were performed by the licensee to determine the thyroid and whole body doses for a fuel handling accident using both LEU and HEU fuel and using the assumptions described in the previous subsections. Results presented in the conversion SAR for the occupational dose and the public radiation doses at both locations of interest demonstrate that the thyroid and whole body doses are well below the established 30 rem thyroid and 5 rem whole body dose criteria for the occupational exposure limit and the 3 rem to the thyroid and 0.5 rem whole body dose for the public exposure limit established in NUREG-1537 for this reactor. These doses are based on the 10 CFR Part 20 limits in effect prior to January 1, 1994. These doses are used because the reactor was initially licensed prior to that date. Doses calculated for the LEU fuel were approximately 26% lower than those calculated for the HEU fuel. The calculated occupational dose for the 5-minute period of time it is assumed to take the facility staff to leave the reactor cell is 2.4 mrem to the thyroid and much less than 1 mrem whole body. The maximum dose to the most exposed member of the public at the west fenced area, 16.5 m from the reactor building for the 2-hour time period after initiation of the release is less than 7 mrem thyroid and much less than 1 mrem whole body. The maximum dose at the nearest residence located 190 m from the reactor facility for a 24-hour time period after initiation of the release is less than 1 mrem thyroid and much less than 1 mrem whole body. These doses assume a building leak rate of 100 percent volume per hour, a very conservative bounding assumption. The measured reactor cell leak rate with the fan system operating is 11.5 percent volume per hour. The calculated occupational and public radiation exposures for the fuel handling accident were well within the acceptable doses.

2.5.4 Maximum Hypothetical Accident (MHA)

The MHA for the UFTR assumed the reactor core is crushed by a shielding slab to maximize the potential release of fission products. The damage to the fuel is equal to stripping the cladding off the fuel plates of one fuel element. Boundary conditions, assumptions and acceptance criteria were the same as in the fuel handling accident. The highest power fuel element was selected in each case to maximize the radionuclide inventories. The calculated occupational dose for the 5-minute period of time it is assumed to take the facility staff to leave the reactor cell is 194 mrem to the thyroid and less than 0.4 mrem whole body. The maximum dose to the most exposed member of the public at the west fenced area, 16.5 m from the reactor building for the 2-hour time period after initiation of the release, is about 520 mrem thyroid and much less than 1 mrem whole body. The maximum dose at the nearest residence located 190 m from the reactor facility for the 24-hour time period after initiation of the release, is about 21 mrem thyroid and much less than 1 mrem whole body. These doses assume a building leak rate of 100 percent volume per hour, a very conservative assumption. The measured reactor cell leak rate with the fan system operating is 11.5 percent volume per hour. The calculated occupational and public radiation exposures for the MHA fuel handling accident were well within the acceptable doses discussed above.

2.5.5 Conclusions

The licensee has demonstrated that the conversion from HEU to LEU fuel does not introduce the potential of a new reactivity addition accident not previously analyzed for the HEU-fueled reactor or significantly increase the consequences beyond those for an existing HEU-fueled reactor accident. The licensee demonstrated this by presenting the basic neutronic, thermal-hydraulic, and physical similarity between the HEU and LEU cores, and an analysis showing that the conclusions in the HEU analysis regarding the consequences of the maximum credible reactivity addition accident are still applicable to the proposed LEU-fueled reactor. The analyses showed that the reactor safety limit on fuel and clad temperature would not be exceeded. Therefore, the staff concludes that the risk to the health and safety of reactor staff and the public from postulated reactivity addition accidents continues to be acceptable.

Review of the radiation source term, fuel handling accident and MHA calculations performed by the licensee, including the assumptions used, demonstrated that the calculated radionuclide inventory and other boundary conditions used in the analysis were acceptable. The radiological consequences to the public and occupational workers at the UFTR from a postulated fuel handling accident and MHA for the proposed LEU-fueled reactor are acceptable and are less than the radiological consequences calculated for the HEU-fueled reactor. As a result of this review, it is concluded that continued operation of the reactor with LEU fuel poses no undue risk from a radiological standpoint to the public or the staff of the UFTR from the fuel handling accident or MHA.

2.6 <u>Fuel Storage</u>

The licensee has analyzed fresh and spent fuel storage and has determined that the existing storage facilities can hold the LEU fuel and meet the requirements of Technical Specification 3.7(6) that fuel out of the core shall be stored and handled such that $k_{eff}$ is less than 0.8 under optimum conditions of moderation and reflection. The staff has reviewed the licensee's analysis and agrees that Technical Specification 3.7(6) will continue to be met. Therefore, the staff concludes that LEU fuel can be acceptable stored at the UFTR.

2.7 Reactor Start-Up Testing

The licensee plans to make sub-critical measurements for the LEU fuel loading. The start-up testing program also includes control rod and power calibrations, measurements of temperature and void coefficients, excess reactivity, reactivity insertion rates, shutdown margin and experimental facility neutron flux levels, and radiation surveys and effluent measurements. The licensee will also complete a number of normal surveillances to ensure operability of components and systems. The licensee is to submit a start-up report to the NRC on the results of the start-up testing. The staff concludes that the licensee's testing program will provide verification of key LEU reactor functions, and therefore, is acceptable.

2.8 Proposed Changes to the Emergency Plan

The licensee proposed changes to the "Emergency Plan for the University of Florida Training Reactor." The changes proposed by the licensee related to the conversion to LEU fuel such as updating the description of the reactor and updating the descriptions of accidents to agree with the conversion SAR. The licensee concluded that the changes to the Emergency Plan did not decrease the effectiveness of the plan. The NRC staff reviewed the changes using NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors" and concluded that the plan continues to meet the requirements of Appendix E to 10 CFR Part 50. Therefore, the changes to the licensee's emergency plan resulting from the conversion of the reactor to LEU are acceptable.

2.9 Proposed Changes to License Conditions and Technical Specifications

For the UFTR HEU to LEU conversion, the changes proposed to the license conditions change special nuclear material possession limits and change the technical specifications.

2.9.1 Proposed Changes to License Conditions

Existing license condition 2.B.(2) is changed to reflect possession of special nuclear material after conversion. Up to 5.0 kilograms of contained uranium-235 of enrichment of less than 20 percent in the form of reactor fuel replaces the existing possession limit of 4.82 kilograms of uranium of any enrichment or form. After the reactor is converted, the licensee has a continuing need to receive, possess and use small amounts of high enriched uranium to allowed continued operation of the reactor (e.g., fission chambers) and conduct of the experimental program (e.g., flux foils). A new possession limit of up to 0.2 kilograms of contained uranium-235 of any enrichment in the form of fission chambers, flux foils and other forms is added to the license condition.
The staff has reviewed the possession limits associated with conversion of the reactor and concludes that the limits are appropriate for the converted reactor.
License condition 2.C.(2), which incorporates the technical specifications into the license, is changed to incorporate the technical specifications changes needed for conversion into the license.

2.9.2 Proposed Changes to the Technical Specifications

For the UFTR HEU to LEU conversion, many of the changes proposed to the technical specifications involve the fuel type and related specifications. The following paragraphs discuss the proposed changes to the technical specifications.

Section 2.1, Safety Limits: A new safety limit is introduced in the conversion SAR. The safety limit specifies that the fuel and cladding temperatures shall not exceed 986 F (530 C). This safety limit is directly related to protection of the fuel cladding which is the primary barrier to the release of fission products from the fuel. This safety limit replaces safety limits on reactor power, primary coolant flow rate and primary coolant outlet temperature which were not directly related to protection of the primary fission product barrier. These parameters continue as LSSSs. The safety limits are also renumbered to reflect the changes. These changes in the Safety Limits were discussed in Section 2.4 above and found to be acceptable.

Section 2.2, Limiting Safety System Settings: A new LSSS on the coolant inlet temperature is introduced in the conversion SAR. The LSSSs for both the primary coolant flow rate and the coolant inlet temperature are now dependent on the fuel coolant channel spacing tolerance. The proposed LSSS are:

1. Power level at any flow rate shall not exceed 119 kW(t).

2. The primary coolant flow rate shall be

   (a) greater than 36 gpm at all power levels greater than 1 watt if the fuel coolant channel spacing tolerance is 15 mils.

   (b) greater than 41 gpm at all power levels greater than 1 watt if the fuel coolant channel spacing tolerance is 20 mils.

3. The average primary coolant

   (a) inlet temperature shall not exceed 109 °F when the fuel coolant channel spacing tolerance is 10 mils.

   (b) inlet temperature shall not exceed 99 °F when the fuel coolant channel spacing tolerance is 20 mils.

   (c) outlet temperature shall not exceed 155 °F when measured at any fuel box outlet.

These changes in the LSSSs were discussed in Section 2.4 above and found to be acceptable.

Section 3.1(2), Reactivity Limitations: The core excess reactivity at cold critical, without xenon poisoning, is reduced from not exceeding 2.3% Δk/k to not exceeding 1.4% Δk/k. This is a more realistic value given operating conditions and uses of the reactor and increases the safety margin as discussed in Section 2.3.3 above.

Section 3.2.3, Reactor Control and Safety Systems Measuring Channels: The number of operable primary coolant temperature indicators is increased from 6 to 7 to account for the addition of the new LSSS on coolant inlet temperature.

Table 3.1, Specifications for Reactor Safety System Trips: The specification on reactor power and inlet water flow are changed to account for the LSSSs values. A specification on high primary coolant average inlet temperature is added to account for the addition of the new LSSS on coolant inlet temperature.

Table 3.2, Safety System Operability Tests: An operability test with a frequency of "with daily checkout" is added to the table to account for the addition of the new LSSS on coolant inlet temperature.

Section 3.5, Limitations on Experiments: The limit on the total absolute reactivity worth of all experiments is reduced from not exceeding 2.3% Δk/k to not exceeding 1.4% Δk/k. This is consistent with the change in Section 3.1 of the Technical Specifications. The reactivity limitation on moveable or nonsecured experiments of not exceeding 0.6% Δk/k is changed from applicable to any single experiment to all moveable or nonsecured experiments. This change reflects a more realistic reactivity worth of all moveable or nonsecured experiments given operating conditions and uses of the reactor at the UFTR and increases the safety margin.

Section 3.7, Fuel and Fuel Handling: The description of fuel elements is changed to reflect the change in the number of fuel plates in an LEU element.

Section 3.8, Primary Water Quality: A specification is added to maintain primary water pH at less than 7.0. This is to minimize corrosion as discussed in Section 2.2 above. The corresponding weekly surveillance of water quality was added in Technical Specification 4.2.8.

Section 4.2.7, Surveillance Pertaining to Fuel: For both the fuel handling accident and the MHA in the conversion analysis, the radioisotope inventories were calculated with an assumption of three days of decay after shutdown from power operation. This surveillance requirement was augmented to require at least three days since the last operation at power ( 1 kW(t)) before the last two layers of concrete block shielding can be moved to reach the core area and before commencement of fuel handling. This new surveillance requirement is consistent with the assumption in the accident analyses (fuel handling accident and MHA) and would limit the potential consequences of the analyzed accidents.

Section 5.3, Reactor Fuel: The enrichment was changed to specify "no more than about 19.75% U-235" and the loading of U-235 per plate was changed to "nominally 12.5 g of U-235 per fuel plate" based on the LEU fuel selection. The allowable fabrication methodology was changed to allow the use of high purity uranium silicide-aluminum dispersion fuel. The section on the possession limits was changed to reflect the new possession limits.

Section 5.4, Reactor Core: Changes were made to the specifications for the number of plates in an assembly and other design specifications to reflect the conversion to LEU fuel. The

licensee also made some minor changes to the format of the technical specification to improve its readability.

Section 5.5.1, Reactor Control System: Changes in the integral worth values for control blades were made to reflect the small changes expected when the conversion is done.

Section 5.6.1, Primary Cooling System: Changes were made to reflect that the primary coolant inlet temperature will now be monitored.

Section 6.6.3(2), Other Special Reports: Changes are made to require that significant changes in the transient or accident analyses as described in the conversion SAR are reported in writing to the Commission within 30 days similar to changes in the facility SAR.

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

Some of the changes involve 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. Some of the changes involve recordkeeping, reporting, or administrative procedures or requirements.

## 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 UFTR. 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).

## 5.0 REFERENCES

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Nuclear Regulatory Commission, February 1996.

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Principal Contributors: David Diamond, Brookhaven National Laboratory (BNL)
Lap-Yang Cheng, BNL
Albert Hanson, BNL
Richard Deem, BNL
Alexander Adams, Jr., NRC

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