September 1, 2006

EA-06-211

Dr. W. D. Reece, Director
Nuclear Science Center
Texas Engineering Experimental Station
Texas A&M University
3575 TAMU
College Station, Texas 77843-3575

SUBJECT: ISSUANCE OF ORDER MODIFYING LICENSE NO. R-83 TO CONVERT FROM HIGH- TO LOW-ENRICHED URANIUM FUEL (AMENDMENT NO. 17) - TEXAS A&M UNIVERSITY NUCLEAR SCIENCE CENTER TRIGA RESEARCH REACTOR (TAC NO. MC9449)

Dear Dr. Reece:

The U.S. Nuclear Regulatory Commission (NRC) is issuing the enclosed Order, as Amendment No. 17 to Amended Facility Operating License No. R-83, which authorizes the conversion of the Texas A&M University Nuclear Science Center TRIGA Research Reactor from high-enriched uranium fuel to low-enriched uranium (LEU) fuel. This Order modifies the license, including the technical specifications, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.64. This regulation requires that non-power reactor licensees, such as the Texas A&M University, convert their reactors to LEU fuel under certain conditions which Texas A&M University now meets. The Order is being issued in accordance with 10 CFR 50.64(c)(3) and in response to your submittal of December 29, 2005, as supplemented on July 17 and August 4 and 21, 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 Order becomes effective on the later date of either the day of receipt of an adequate number and type of LEU fuel elements that are necessary to operate the facility as specified in your submittal and supplements, or 20 days after the date of its publication in the *Federal Register*, provided there are no requests for a hearing.

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

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

Sincerely,

/RA/

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

Docket No. 50-128

Enclosures: 1. Order
2. Safety Evaluation

cc w/enclosures: See next page

cc:

Mayor, City of College Station
P.O. Box Drawer 9960
College Station, TX 77840-3575

Governor's Budget and
Planning Office
P.O. Box 13561
Austin, TX 78711

Texas A&M University System
ATTN: Dr. Warren D. Reece, Director
Nuclear Science Center
Texas Engineering Experiment Station
F. E. Box 89, M/S 3575
College Station, Texas 77843

Texas State Department of Health
Radiation Control Program Director
Bureau of Radiation Control
Dept. of Health
1100 West 49th Street
Austin, Texas 78756-3189

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

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
TEXAS A&M UNIVERSITY	)	Docket No. 50-128
(Nuclear Science Center	)	EA-06-211
TRIGA Research Reactor)	)	
	)	

ORDER MODIFYING AMENDED FACILITY OPERATING LICENSE NO. R-83

I.

The Texas A&M University (the licensee) is the holder of Amended Facility Operating License No. R-83 (the license). The license was issued on December 7, 1961, by the U.S. Atomic Energy Commission and subsequently renewed on March 30, 1983, by the U.S. Nuclear Regulatory Commission (the NRC or the Commission). The license includes authorization to operate the Nuclear Science Center TRIGA Research Reactor (the facility) at a power level up to 1,000 kilowatts thermal (1,300 kilowatts thermal for purposes of testing and calibration) and to receive, possess, and use special nuclear material associated with the operation. The facility is on the campus of the Texas A&M University, in the city of College Station, Brazos County, Texas. The mailing address is Nuclear Science Center, Texas Engineering Experimental Station, Texas A&M University, 3575 TAMU, College Station, Texas 77843-3575.

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 29, 2005, as supplemented on July 17, and August 4 and 21, 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 reactor core contains fuel bundles, each fuel bundle contains up to four fuel elements of the TRIGA design, with the fuel consisting of uranium-zirconium hydride with 30 weight percent uranium. These fuel elements contain the uranium-235 isotope at an enrichment of less than 20 percent. The NRC staff reviewed the licensee's proposal and the requirements of 10 CFR 50.64 and has determined that public health and safety and common defense and security require the licensee to convert the facility from the use of HEU to LEU fuel in accordance with the attachments to this Order and the schedule included herein. The attachments to this Order specify the changes to the License Conditions and Technical Specifications that are needed to amend the facility license and 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-83 is modified by amending the License Conditions and Technical Specifications as stated in the attachments to this Order. The Order 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's 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 licensee, or other 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). Thus, pursuant to either 10 CFR 51.10(d) or 51.22(c)(9), no environmental assessment nor environmental impact statement is required.

For further information see the application from the licensee dated December 29, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML062200390), as supplemented on July 17, and August 4 and 21, 2006 (ADAMS Accession

Nos. ML062220189, ML062220278 and ML062410495), the staff's request for additional information dated June 1, 2006 (ADAMS Accession No. ML061500125), and the cover letter to the licensee, attachments to the Order, and the staff's safety evaluation dated September 1, 2006 (ADAMS Accession No. ML062410474), 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

/RA/

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. Outline of Reactor Startup Report

ATTACHMENT TO ORDER

MODIFICATIONS TO AMENDED FACILITY OPERATING LICENSE NO. R-83

A. License Conditions Revised by This Order

II.B.(2) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to possess, but not use, up to 12.0 kilograms of contained uranium-235 at equal to or greater than 20 percent enrichment in the form of TRIGA FLIP-type reactor fuel until the existing inventory of this fuel is removed from the facility.

II.B.(8) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess, and use up to 15.0 kilograms of contained uranium-235 of enrichment of less than 20 percent in the form of TRIGA-type reactor fuel in connection with operation of the reactor.

II.C.(2) 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.

B. The technical specifications will be revised by this Order in accordance with the "Enclosure to License Amendment No. 17, Amended Facility Operating License No. R-83, Docket No. 50-128, Replacement Pages for Technical Specifications," as discussed in the safety evaluation for this Order.

ENCLOSURE TO LICENSE AMENDMENT NO. 17

AMENDED FACILITY OPERATING LICENSE NO. R-83

DOCKET NO. 50-128

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
Cover page	Cover page
Table of Contents 1	Table of Contents 1
Table of Contents 2	Table of Contents 2
2	2
3	3
4	4
5	5
6	6
7	7
8	8
9	9
11	11
31	31
32	32
33	33
35	35
36	36

TECHNICAL SPECIFICATIONS

FOR THE TEXAS ENGINEERING EXPERIMENT STATION

TEXAS A&M UNIVERSITY SYSTEM

NUCLEAR SCIENCE CENTER

REACTOR FACILITY

DOCKET NO. 50-128

LICENSE NO. R-83

MARCH 1983

REVISED THROUGH AMENDMENT NO. 17

TECHNICAL SPECIFICATIONS FOR THE

NUCLEAR SCIENCE CENTER REACTOR

FACILITY LICENSE NO. R-83

March 1983

Revised through Amendment No. 17

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Page

1.5 Core Lattice Position

The core lattice position is that region in the core (approximately 3" x 3") over a grid plug hole. It may be occupied by a fuel bundle, an experiment, or a reflector element.

1.6 Experiment

An operation, hardware, or target (excluding devices such as detectors, foils etc.) which is designed to investigate non-routine reactor characteristics or which its intended for irradiation within the pool, on or in a beam port or irradiation facility and which is not rigidly secured to a core or shield structure so as to be a part of their design.

1.7 Experimental Facilities

Experimental facilities shall mean beam ports, including extension tubes with shields, thermal columns with shields, vertical tubes, through tubes, in-core irradiation baskets, irradiation cell, pneumatic transfer systems and in-pool irradiation facilities.

1.8 Experiment Safety Systems

Experiment safety systems are those systems, including their associated input circuits, which are designed to initiate a scram for the primary purpose of protecting an experiment or to provide information which requires manual protective action to be initiated.

1.9 Fuel Bundle *

A fuel bundle is a cluster of two, three or four elements and/or non-fueled elements secured in a square array by a top handle and a bottom grid plate adapter. Non-fueled elements shall be fabricated from stainless steel, aluminum or graphite materials.

1.10 Fuel Element *

A fuel element is a single TRIGA fuel rod of LEU type. *

1.11 Instrumented Element *

An instrumented element is a special fuel element in which a sheathed chromal-alumel or equivalent thermocouple is embedded in the fuel near the horizontal center plane of the fuel element at a point approximately 0.3 inch from the center of the fuel body.

1.12 LEU Core *

A LEU core is an arrangement of TRIGA-LEU fuel in a reactor grid plate. *

1.13 Limiting Safety System Setting

The limiting safety system setting is the setting for automatic protective devices related to those variables having significant safety functions.

1.14 Measuring Channel

A measuring channel is the combination of sensor, interconnecting cables or lines, amplifiers, and output device which are connected for the purpose of measuring the value of a variable.

1.15 Measured Value

The measured value is the value of a parameter as it appears on the output of a channel.

1.16 Movable Experiment *

A movable experiment is one for which it is intended that the entire experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

1.17 Operable *

Operable means a component or system is capable of performing its intended function.

1.18 Operating *

Operating means a component or system is performing its intended function.

1.19 Steady State Operational Core *

A steady state operational core shall be an LEU core for which the core parameters of shutdown margin, fuel temperature and power calibration have been determined. *

1.20 Pulse Operational Core *

A pulse operational core is a steady state operational core for which the maximum allowable pulse reactivity insertion has been determined.

1.21 Pulse Mode *

Pulse mode operation shall mean any operation of the reactor with the mode selector switch in the pulse position.

1.22 Reactivity Worth of an Experiment *

The reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter the experiment position or configuration.

1.23 Reactor Console Secured *

The reactor console is secured whenever all scrammable rods have been fully inserted and verified down and the console key has been removed from the console.

1.24 Reactor Operating *

The reactor is operating whenever it is not secured or shutdown.

1.25 Reactor Safety Systems *

Reactor safety systems are those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action. Manual protective action is considered part of the reactor safety system.

1.26 Reactor Secured *

A reactor is secured when:

a) It contains insufficient fissile material or moderator present in the reactor and adjacent experiments to attain criticality under optimum available conditions of moderation and reflection, or

b) The reactor console is secured, and

1) No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods, and

2) No experiments in or near the reactor are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value of one dollar.

1.27 Reactor Shutdown *

The reactor is shut down when the reactor, at ambient temperature and xenon-free condition and including the reactivity worth of all experiments, is subcritical by at least one dollar.

1.28 Reportable Occurrence *

A reportable occurrence is any of the following which occurs during reactor operation:

a) Operation with actual safety system settings for required systems less conservative than the limiting safety-system settings specified in the Technical Specifications 2.2.

b) Operation in violation of limiting conditions for operation established in the technical specifications.

c) A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns. (Note: Where components or systems are provided in addition to those required by the technical specifications, the failure of the extra components or systems is not considered reportable provided that the minimum number of components or systems specified or required perform their intended reactor safety function.)

d) An unanticipated or uncontrolled change in reactivity greater than one dollar.

e) Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or containment boundary (excluding minor leaks) where applicable which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both.

f) An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

1.29 Rod-Control *

A control rod is a device fabricated from neutron absorbing material or fuel which is used to establish neutron flux changes and to compensate for routine reactivity losses. A control rod may be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged.

1.30 Rod-Regulating *

The regulating rod is a low worth control rod used primarily to maintain an intended power level that need not have scram capability and may have a fueled follower. Its position may be varied manually or by the servo-controller.

1.31 Rod-Shim Safety *

A shim-safety rod is a control rod having an electric motor drive and scram capabilities. It may have a fueled follower section.

1.32 Rod-Transient *

The transient rod is a control rod with scram capabilities that is capable of providing rapid reactivity insertion to produce a pulse.

1.33 Safety Channel *

A safety channel is a measuring channel in the reactor safety system.

1.34 Safety Limit *

Safety limits are limits on important process variables which are found to be necessary to reasonably protect the integrity of certain physical barriers which guard against the uncontrolled release of radioactivity.

1.35 Scram Time *

Scram time is the time measured from the instant a simulated signal reaches the value of the LSSS to the instant that the slowest scrammable control rods reaches its fully inserted position.

1.36 Secured Experiment *

A secured experiment is any experiment, experiment facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible malfunctions.

1.37 Shall, Should and May *

The word “shall” is used to denote a requirement; the word “should” to denote a recommendation; and the word “may” to denote permission, neither a requirement nor a recommendation. In order

to conform to this standard, the user shall conform to its requirements but not necessarily to its recommendations.

1.38 Shutdown Margin *

Shutdown margin shall mean the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition, if the most reactive rod is stuck in its most reactive position, and that the reactor will remain subcritical without further operator action.

1.39 Steady State Mode *

Steady state mode operation shall mean operation of the reactor with the mode selector switch in the steady state position.

1.40 True Value *

The true value is the actual value of a parameter.

1.41 Unscheduled Shutdown *

An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions which could adversely affect safe operation, not to include shutdowns which occur during testing or check out operations.

2.0 Safety Limit and Limiting Safety System Setting

2.1 Safety Limit Fuel Element Temperature

Applicability

This specification applies to the temperature of the reactor fuel.

Objective

The objective is to define the maximum fuel element temperature that can be permitted with confidence that no damage to the fuel element cladding will result.

Specifications

The temperature in a stainless steel-clad TRIGA LEU fuel element shall not exceed *
2100 °F (1150°C) under any conditions of operation. *

Bases

The important parameter for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification especially since it can be measured. A loss in the integrity of the fuel element cladding could arise from a buildup of excessive pressure between the fuel-moderator and the cladding if the fuel temperature exceeds the safety limit. The magnitude of this pressure is determined by the fuel-moderator temperature and the ratio of the hydrogen to zirconium in the alloy.

The temperature safety limit for the LEU fuel element is based on data which indicates that the *
internal stresses within the fuel element due to hydrogen pressure from the dissociation of the *
zirconium hydride will not result in compromise of the stainless steel cladding if the fuel *
temperature is not allowed to exceed 2100°F (1150°C) and the fuel element cladding is water *
cooled. *

2.2 Limiting Safety System Setting

Applicability

This specification applies to the scram setting which prevents the safety limit from being reached.

Objective

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

Specification

The limiting safety system setting shall be 975°F (525°C) as measured in an instrumented fuel element. The instrumented element shall be located adjacent to the central bundle with the exception of the corner positions.

Basis

The limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated preventing the safety limit from being exceeded. The temperature safety limit for LEU fuel is 2100°F (1150°C). Due to various errors in measuring temperature in the core, it is necessary to arrive at a Limiting Safety System Setting (LSSS) for the fuel element safety limit that takes into account these measurement errors. One category of error between the true temperature value and the measured temperature value is due to the accuracy of the fuel element channel and any overshoot in reactor power resulting from a reactor transient during steady state mode of operation. Although a lesser contributor to error, a minimum safety margin of 10% was applied on an absolute temperature basis. Adjusting the fuel temperature safety limit to degrees Kelvin, °K, and applying a 10% safety margin results in a safety limit reduction of 150°C. Applying this first margin of safety, the safety setting would be 1000°C for LEU. However, to arrive at the final LSSS it is necessary to allow for the difference between the measured temperature value and the peak core temperature, which is a function of the location of the thermocouple within the core. For example, if the thermocouple element were located in the hottest position in the core, the difference between the true and measured temperatures would be only a few degrees since the thermocouple junction is at the mid-plane of the element and close to the anticipated hot spot. However, at the TAMU this core position is not available due to the location of the transient rod. For the TAMU the location of the instrumented elements is therefore restricted to the positions closest to the central element. Calculations indicate that, for this case, the true temperature at the hottest location in the core will differ from the measured temperature by no more than 40%. When applying this 40% worst case measurement scenario and considering the previously mentioned sources of error between the true and measured values, a final LSSS temperature of 975°F (525°C) is imposed on operation. Viewed on an absolute temperature scale, °K, this represents a 37% safety margin in the LEU safety limit.

In the pulse mode of operation, the above temperature limiting safety system setting will apply. However, the temperature channel will have no effect on limiting peak powers generated because of its relatively long time constant (seconds) as compared with the width of the pulse (milliseconds). In this mode, however, a temperature trip will act to reduce the amount of energy generated in the entire pulse transient by cutting the "tail" off the energy transient in the event the pulse rod remains stuck in the fully withdrawn position.

3.0 Limiting Conditions for Operation

3.1 Reactor Core Parameters

3.1.1 Steady State Operation

Applicability

This specification applies to the energy generated in the reactor during steady state operation.

Objective

The objective is to assure that the fuel temperature safety limit will not be exceeded during steady state operation.

Specifications

The reactor power level shall not exceed 1.3 megawatts under any condition of operation. The normal steady state operating power level of the reactor shall be 1.0 megawatts. However, for purposes of testing and calibration, the reactor may be operated at higher power levels not to exceed 1.3 megawatts during the testing period.

Basis

Thermal and hydraulic calculations indicate the TRIGA fuel may be safely operated up to power levels of at least 2.0 MW with natural convection cooling.

3.1.2 Pulse Mode Operation

Applicability

This specification applies to the peak temperature generated in the fuel as the result of a pulse insertion of reactivity.

Objective

The objective is to assure that respective pulsing will not induce damage tot he reactor fuel.

Specification

a) The reactivity to be inserted for pulse operation shall not exceed that amount which will produce a peak fuel temperature of 1526°F (830°C). In the pulse mode the pulse rod shall be limited by mechanical means or the rod extension physically shortened so that the reactivity insertion will not inadvertently exceed the maximum value.

b) Until the LEU fuel core has been calibrated, maximum pulse shall be limited to $2.00. *

Basis

TRIGA fuel is fabricated with a nominal hydrogen to zirconium ratio of 1.6 for LEU fuel. This yields delta phase zirconium hydride which has a high creep *

3.1.4 Core Configuration Limitation

Applicability

This specification applies to a full LEU core. *

Objective

The objective is to assure that the fuel temperature safety limit will not be exceeded due to power peaking effects in full LEU cores. *

Specifications

a) The TRIGA core assembly shall be LEU. *

b) The reactor shall not be taken critical with a core lattice position vacant except for positions on the periphery of the core assembly. Water holes in the inner fuel region shall be limited to single rod positions. Vacant core positions shall contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core.

c) The instrumented element shall be located adjacent to the central bundle with the exception of the corner positions (Reference: 2.2 Limiting Safety System Setting).

Bases

a) Safety and accident analysis were only performed for a TRIGA core using LEU fuel. *
*

b) Vacant core positions containing experiments or an experimental facility will prevent accidental fuel additions to the reactor core. They will be permitted only on the periphery of the core or a single rod position to prevent power peaking in regions of high power density.

c) Reference: 2.2 Limiting Safety System Setting.

3.1.5 Maximum Excess Reactivity

Applicability

This specification applies to the maximum excess reactivity, above cold critical, which may be loaded into the reactor core at any time.

Objective

The objective is to ensure that the core analyzed in the safety analysis report approximates the operational core within reasonable limits.

5.0 Design Features

5.1 Reactor Fuel

Applicability

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

Objective

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

Specifications

TRIGA-LEU Fuel *
The individual unirradiated LEU fuel elements shall have the following characteristics: *

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

2) Hydrogen-to-zirconium ratio (in the ZrH_x): 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 0.90 wt%. (See bases below for contract specifications.) * *

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

Bases

The fuel specification permits a maximum uranium enrichment of 19.95%. This is about 1% * greater than the design value for 19.75% enrichment. Such an increase in loading would result in * an increase in power density of less than 1%. An increase in local power density of 1% reduces * the safety margin by less than 2%. *

The fuel specification for a single fuel element permits a minimum erbium content of about 5.6% * less than the design value of 0.90 wt%. (However, the quantity of erbium in the full core must * not deviate from the design value by more than –3.3%). This variation for a single fuel element * would result in an increase in fuel element power density of about 1-2%. Such a small increase * in local power density would reduce the safety margin by less than two percent. *

The maximum hydrogen-to-zirconium ratio of 1.65 could result in a maximum stress under * accident conditions in the fuel element clad about a factor of two greater than for a hydrogen-to- * zirconium ratio of 1.60. This increase in the clad stress during an accident would not exceed the * rupture strength of the clad. *

5.2 Reactor Core

Applicability

This specification applies to the configuration of fuel and in core experiments.

Objective

The objective is to assure that provisions are made to restrict the arrangement of fuel elements and experiments so as to provide assurance that excessive power densities will not be produced.

Specifications

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

b) The reflector, excluding experiments and experimental facilities, shall be water or a combination of graphite and water or D_2O.

Bases

a) Standard TRIGA cores have been in use for years and their characteristics are well documented. LEU cores including 30/20 fuel have also been operated at General *
Atomics and their successful operational characteristics are available. General Atomics *
and Texas A&M have done a series of studies documenting the viability of using LEU *
fuel in TRIGA reactors. *

b) The core will be assembled in the reactor grid plate which is located in a pool of light water. Water in combination with graphite reflectors can be used for neutron economy and the enhancement of experimental facility radiation requirements.

5.3 Control Rods

Applicability

This specification applies to the control rods used in the reactor core.

Objective

The objective is to assure that the control rods are of such a design as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications

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

b) The regulating control rod need not have scram capability and shall be a stainless rod or contain the materials as specified for shim-safety control rods. This rod may incorporate a fueled follower.

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

Bases

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

5.4 Radiation Monitoring System

Applicability

This specification describes the functions and essential components of the area radiation monitoring equipment and the system for continuously monitoring airborne radioactivity.

Objective

The objective is to describe the radiation monitoring equipment that is available to the operator to assure safe operation of the reactor.

Specification

The radiation monitoring equipment listed in the following table will have these characteristics.

Radiation Monitoring Channel and Function

Area Radiation Monitor (gamma sensitive instruments)
Function: Monitor radiation fields in key locations, alarm and readout at control console and readout in reception room.

Specifications

a) The reactor shall be housed in a facility designed to restrict leakage. The minimum free volume in the facility shall be 180,000 cubic feet.

b) The reactor building shall be equipped with a ventilation system designed to filter and exhaust air or other gases from the reactor building and release them from a stack at a minimum of 85 feet from ground level.

c) Emergency shutdown controls for the ventilation system shall be located in the reception room and the system shall be designed to shut down in the event of a substantial release of fission products.

Bases

The facility is designed such that the ventilation system will normally maintain a negative pressure with respect to the atmosphere so that there will be no uncontrolled leakage to the environment. The free air volume within the reactor building is confined when there is an emergency shutdown of the ventilation system. Controls for startup, emergency filtering, and normal operation of the ventilation system are located in the reception room. Proper handling of airborne radioactive materials (in emergency situations) can be conducted from the reception room with a minimum of exposure to operating personnel.

5.7 Reactor Pool Water Systems

Applicability

This specification applies to the pool containing the reactor and to the cooling of the core by the pool water.

Objective

The objective is to assure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding.

Specifications

a) The reactor core shall be cooled by natural convective water flow.

b) The pool water inlet and outlet pipe to the demineralizer shall not extend more than 15 feet below the top of the reactor pool when fuel is in the core.

c) Diffuser and skimmer pumps shall be located no more than 15 feet below the top of the reactor pool.

d) Pool water inlet and outlet pipes to the heat exchanger shall have emergency covers within the reactor pool for manual shut off in case of pool water loss due to external pipe system failure.

e) A pool level alarm shall indicate loss of coolant if the pool level drops approximately 10% below operating level.

Bases

a) This specification is based on thermal and hydraulic calculations which show that the TAMU TRIGA-LEU core can operate continuously in a safe manner at power levels up *
to 2,420 kW with natural convection flow and sufficient bulk pool cooling. *

*

b) In the event of accidental siphoning of pool water through inlet and outlet pipes of the demineralizer system, the pool water level will drop no more than 15 feet from the top of the pool.

c) In the event of pipe failure and siphoning of pool water through the skimmer and diffuser water systems, the pool water level will drop no more than 15 feet from the top of the pool.

d) Inlet and outlet coolant lines to the pool heat exchanger terminate at the bottom of the pool. In the event of pipe failure, these lines must be manually sealed from within the reactor pool. Covers for these lines will be stored in the reactor pool. The time required to uncover the reactor core due to failure of a single pool coolant pipe system is 17 minutes.

e) Loss of coolant alarm after 10% loss requires corrective action. This alarm is observed in the reactor control room and the reception room.

5.8 Physical Security

The licensee shall maintain in effect and fully implement all provisions of the NRC staff approved physical security plan, including amendments and changes made pursuant to the authority of 10 CFR 50.54 (p). The approved security plan consists of documents withheld from public disclosure pursuant to 10 CFR 2.70, collectively titled “Texas A&M University System, Nuclear Science Center Reactor Security Plan.”

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. Pulse Measurements

   Measurements of any test pulses performed

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 HEU if available.

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-83

TEXAS A&M UNIVERSITY NUCLEAR SCIENCE CENTER REACTOR

DOCKET NO. 50-128

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. Texas A&M University (TAMU or the licensee) has proposed to convert the fuel in the Nuclear Science Center Reactor (NSCR) from HEU to LEU. In a letter dated December 29, 2005, as supplemented on July 17, and August 4 and 21, 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. 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 NSCR is a TRIGA-conversion reactor similar in design to many others operating in the U.S. 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 replaced the MTR plate-type fuel assemblies.

The reactor normally operates at a maximum thermal power level 1 MW(t) although the Technical Specifications allow operation up to 1.3 MW(t) for calibration and testing purposes. The reactor uses natural convection for cooling. It is presently fueled with either TRIGA fuel lifetime improvement program (FLIP) fuel or a mixture of FLIP and Standard TRIGA fuel. The HEU FLIP fuel has an enrichment of 70% uranium-235. The HEU to LEU conversion only requires changes in the fuel type and core configuration, it does not require any changes to the remainder of the facility. The LEU fuel will have an enrichment of about 19.75% in a fuel element with the same geometry and with the same cladding as the FLIP fuel.

2.2 Fuel and Core Design

The physical size of the fuel assemblies or elements will not change in the conversion from HEU to LEU. The licensee plans to replace the fuel elements and reuse the existing fuel assemblies. The outer diameter of the fuel meat will remain at 34.82 mm with a fuel meat length of 381 mm. Only the fuel meat composition changes.

The FLIP fuel elements contain erbium poison at 1.48 weight percent (w/o). With the LEU fuel, the erbium poison will be lowered to 0.90 w/o. The TRIGA FLIP and TRIGA Standard fuel elements have uranium contents of 9 w/o with the balance of the fuel being ZrH_x, where x~1.6 for the FLIP and ~1.7 for the Standard fuels. The fresh LEU fuel has a nominal uranium content of up to 30 w/o uranium with the balance being ZrH_x, where x~1.6 (the fuel being called 30/20 fuel). This means there is less hydrogen in the fuel, the hydrogen being one of the main reasons for the very large negative temperature coefficient of reactivity. The reduced hydrogen content changes the value of the temperature coefficient of reactivity, α, which is discussed in Section 2.3.4, "Dynamic Parameters." The uranium density for the LEU fuel is 2.14 g/cc. The amount of uranium-235 per LEU fuel element is 149 g. Calculations indicate that there will be 90 fuel elements in the LEU core compared to the present 94. Generic aspects of the LEU TRIGA fuel with this high uranium content has been previously approved for use in research and test reactors by the Nuclear Regulatory Commission (NUREG-1282).[2] The licensee submitted this application to justify the specific use of the LEU fuel in the NSCR.

Table 1 shows the major changes in the fuel composition (as well as other parameters) between the Puerto Rico Nuclear Center (PRNC) reactor using HEU FLIP fuel and the proposed NSCR LEU reactor. The TAMU HEU FLIP fuel was initially used in the PRNC core and when it was shutdown, the fuel was transferred to the NSCR. The NSCR never had fresh HEU FLIP fuel loaded into it and the burnup of the fuel when it came from the PRNC was not known precisely. Therefore, the neutronic code validation, which was for a fresh core, is accomplished by comparison of calculations with measurements from the PRNC. The geometry of the two cores is similar but not identical; nor are the control rods. Additional issues resulting from the use of the PRNC reactor as a benchmark for thermal-hydraulic analysis are discussed in Section 2.4 below.

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

Conversion from TRIGA FLIP fuel to the LEU fuel will result in a decrease in the life expectancy of the core, from 2350 MWd to 2000 MWd. This calculation was based on no fuel shuffling during the life of the core, which could increase the life expectancy of the core. Based on historical operations, the LEU core is expected, with no shuffling, to last 18 to 35 years and is considered to be a "lifetime core." A concern addressed in Section 12.6 of the conversion SAR, "Reactor Reload Consideration," is if an instrumented fuel element (IFE) needs to be replaced due to thermocouple failure, what would be the effect on the core peaking factors. A set of calculations were performed where a fresh fuel element is replaced in each of the three locations where the IFE is located. For steady state operation, the calculated peaking in any one of those three positions was less than the maximum peaking for a beginning-of-life core,

Table 1. HEU and LEU Design Data, Reactor Parameters, and Safety Parameters for Conversion of the TAMU Resarch Reactor.

(Note: The HEU data are taken from the Puerto Rico Nuclear Center FLIP and TAMU Core.)

DESIGN DATA	**HEU (FLIP) PRNC Core**	**LEU (30/20) NSCR CORE**
Number of Fuel Rods	95	90
Fuel Type	UZrH-Er	UZrH-Er
Uranium Enrichment, %	70	19.75
Erbium, w/o	1.48	0.90
Zirconium Rod Outer Diameter, mm	6.35	6.35
Fuel Meat Outer Diameter, mm	34.823	34.823
Fuel Meat Length, mm	381	381
Clad Thickness, mm	0.508	0.508
Clad Material	304 SS	304 SS
REACTOR PARAMETERS		
Reactor Steady State Operation		
Routine Power, MW(t)	1.4	1.0
Testing Maintenance, MW(t)	-	1.3
Maximum Fuel Temperature at 1 MW(t), ºC	362	373
Maximum Pulsing Operation with ∅ Limited to 830ºC, MW(t)	2400	1325
Cold Clean Excess Reactivity, Δk/k/β ($)	6.26	7.73
Prompt Negative Temperature Coefficient of Reactivity, Δk/k/ºC 23-700 ºC	-0.47 to -1.79 x 10^{-4}	-0.53 to -1.31 x 10^{-4}
Coolant Void Coefficient, Δk/k/% void		-1.30 x 10^{-3}
Maximum Rod Power at 1 MW(t), kW(t)/element	-1.07 x 10^{-3}	17.4
Average Rod Power at 1 MW(t), kW(t)/element	10.5	11.1
Maximum Rod Power at 1.4 MW(t), kW(t)/element	16.3	-
Average Rod Power at 1.4 MW(t), kW(t)/element	22.8	-
Maximum Rod Power at deviation from nucleate	14.7	42
boiling (DNB) = 1.0, kW(t)/element	44	
DNB Ratio at Operating Power		2.42
Prompt Neutron Lifetime, μs	1.93	26.3
Effective Delayed Neutron Fraction	22.5	0.0070
Shutdown Margin, Δk/k/β ($)	0.0071	
(with most reactive rod stuck out)		
(with most reactive rod and regulating rod stuck out)	-2.12	-1.15

Table 1 (Cont'd.)

OPERATING LIMITS	**HEU (FLIP) PRNC CORE**	**LEU (30/20) NSCR Core**
Reactor Power, MW(t)	2.2	1.25
Measured Fuel Temperature, ºC at reactor power	600	525
Minimum DNB ratio at 1.0 MW(t)	-	2.42
Minimum DNB ratio at 1.4 MW(t)	1.93	-
Maximum Positive Pulsed Reactivity Insertion to reach ∅= 830 ºC, Δk/k/β ($)	2.05	2.10
Peak Pulsed Power, MW(t)	2400	1325
Peak Pulsed Fuel Temperature, ºC	830	830

which is acceptable. Under pulsing conditions, one of the core positions, 4D4, had what the licensee believes is an unacceptable peaking factor, so they will not place a fresh fuel element into that position in a high burnup core. The exclusion of placing a fresh IFE in location 4D4 is controlled by Technical Specification 3.1.4 that requires the IFE be placed adjacent to the central bundle with the exception of the corner positions. Power peaking in pulse mode operation is controlled by Technical Specification 3.1.2 that imposes a limiting condition for operation (LCO) on the reactivity addition from a pulse requiring that the resultant peak fuel temperature does not exceed 830 EC.

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 to the reactor because of conversion will be the use of high-density, low-enriched fuel meat which calculations show will result in a slightly smaller core (90 verses 94 fuel elements). Therefore, the staff finds the fuel and core design acceptable.

2.3 Nuclear Design

2.3.1 Calculational Methodology

The reactor was analyzed with two computer codes that have been used extensively for nuclear reactor applications: the finite difference three-dimensional (3-D) diffusion theory code, DIF3D, and the Monte Carlo transport code, MCNP5. The computations were benchmarked by comparing results to the PRNC TRIGA HEU core for parameters such as fuel elements needed for criticality, excess reactivity, and power defect. The BURP depletion code, a GA computer code developed for TRIGA reactors was used for burnup calculations. Because the licensee used well developed codes, the staff concludes that the calculational methodology used by the licensee is acceptable.

2.3.2 Power Distributions

The SAR presents detailed analysis of the power distributions, but does not present any flux maps that would result from the conversion from HEU to LEU fuel. However, the differences in power distributions between the HEU and LEU cores are not significant enough to result in any

major changes in the neutron flux distributions. The SAR presents axial power profiles for the PRNC HEU and the proposed TAMU LEU cores along with four power peaking values: the relative average power in the hottest rod; the axial peak-to-average power ratio within a fuel rod; and the average and peak power distributions within a rod. For the beginning-of-life core, compared to the PRNC FLIP core, the relative average power in the hottest rod increased approximately 1%. This is expected since the proposed TAMU LEU core has 90 fuel elements and the PRNC FLIP core contained 95 fuel elements. The hottest spot within a fuel rod was calculated to increase 17%. This increase is taken into account in the safety analysis for the core. As expected, the end-of-life core had less power peaking. The staff finds that the changes to power distributions due to conversion are not significant and are as expected. Therefore, the staff concludes that power distribution predictions for the LEU core are acceptable.

2.3.3 Excess Reactivity, Shutdown Margin and Control Blade Worth

For the TAMU LEU core with all rods inserted, the reactivity was calculated to be -$8.61 and the control rods have a collective worth of $16.34. With all control rods withdrawn, the core reactivity was calculated to be $7.73, which is within the limits of Technical Specification 3.1.5, “Maximum Excess Reactivity,” which states that the maximum core reactivity must be less than $7.85 (5.5 %Δk/k). This value is unchanged from the Technical Specification for the HEU core. Therefore, the staff concludes that the proposed limit on excess reactivity of the LEU core is acceptable.

The shutdown margin was calculated for two conditions, both had the highest worth control rod fully withdrawn along with the reactor being in the cold condition without xenon and with all other control rods inserted. One case included the highest worth non-secured experiment in its most reactive state with the regulating rod inserted and the other case considered the regulating rod also being withdrawn. The latter case had the least negative reactivity which calculated to be -$1.15. This exceeds the minimum allowed (absolute) value of $0.25 specified in Technical Specification 3.1.3, “Shutdown Margin.” As part of startup testing with the new LEU core, the actual value of the shutdown margin of the core will be measured and the licensee will verify that Technical Specification 3.1.3 is met. Therefore, the staff concludes that acceptable ability to shut down the LEU fueled reactor will be maintained.

In the neutronic analyses, differential control rod values were not presented, only integral values were presented. Differential rod worths are typically used to define conditions for rod withdrawal reactivity accidents. However, in the case of the TAMU, the reactivity insertion events leading to normal pulsed behavior are bounding.

Based on the acceptable calculated values for excess reactivity and shutdown margin, the staff concludes that the control blade worths of the LEU core are acceptable.

2.3.4 Dynamic Parameters

The delayed neutron fraction, β_{eff}, was calculated for both the PRNC FLIP and the TAMU LEU cores using the 3-D diffusion theory code with first only the prompt fission spectrum and then with the prompt and delayed fission spectrum. For the PRNC FLIP and the TAMU LEU cores, the values for β_{eff} were calculated to be 0.0071 and 0.0070, respectively. The calculation of β_{eff} was done using a formula developed at GA. A more rigorous treatment was also provided and

the calculations produced results for β_{eff} whose values were insignificantly different from those calculated with the GA methodology. The licensee has shown that the result is still accurate to first order and therefore, is useable in the neutron kinetics analysis. Therefore, the staff concludes that conversion to LEU fuel will not have a significant impact on β_{eff}.

The prompt neutron lifetime (*l*) was calculated using the 1/v absorber method with the 3-D diffusion theory code. This is a standard method and is performed by adding small amounts of boron uniformly throughout the system. The prompt neutron lifetime can be determined from the change in the reactivity due to the boron. For the PRNC FLIP core *l* was calculated to be 22.5 μs and for the TAMU LEU core it is calculated to be slightly higher at 26.3 μs. For an end-of-life LEU core the prompt neutron lifetime is calculated to be 27.3 μs. The staff concludes that conversion to LEU fuel will not have a significant impact on the prompt neutron lifetime.

The void coefficient of reactivity is slightly negative (-0.130 (%Δk/k)/% void) in the LEU core as with the HEU core (-1.07 (%Δk/k)/% void). The LEU value of the coefficient is somewhat larger than the HEU value. A negative void coefficient means that boiling in the core will add negative reactivity to the core causing reactor power to decrease. A negative void coefficient also means that reactivity is added if there is flooding of a normally dry in-core experiment. Flooding of a 281 cm^3 (volume of an experimental facility the size of a fuel element) dry experiment results in an increase in reactivity of $0.26, smaller than the $1.00 insertion needed for prompt critical and within the technical specification limits of $1.00 for a movable experiment and $2.00 for any single experiment. TRIGA reactors are designed for significantly larger additions of reactivity with no fuel damage. The impact on safety of the experiment flooding is insignificant and is similar to the impact of the same type of failure in the HEU core. The staff concludes that this change is not significant and will not significantly change the dynamic behavior of the core.

One of the inherent safety features of the TRIGA fuel design is the strong negative temperature coefficient of reactivity, α. Because TRIGA reactors were designed for pulsing, the original TRIGA fuels could be subject up to $5.00 of prompt positive reactivity insertion without damaging the fuel. One of the reasons is the fuel contains considerable zirconium hydride. With conversion to LEU fuel, the amount of uranium is increased from 8.42 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 91.58 w/o to 70 w/o. It is expected that this, along with the increase in the amount of uranium-238 in the fuel will impact the temperature coefficient.

At 23 °C the value of α has been calculated to be $-4.8x10^{-5}$ Δk/k/EC for the HEU fuel and $-5.3x10^{-5}$ Δk/k/EC for the LEU fuel. This increase in magnitude is expected due to the added uranium-238 in the LEU fuel. However, when pulsing, it is the zirconium hydride that dominates the value of α, so at 700 °C the numbers are $-17.9x10^{-5}$ Δk/k/EC for the HEU fuel and $-13.1x10^{-5}$ Δk/k/EC for the LEU fuel. Although the value of the fuel temperature coefficient in LEU fuel decreases during pulsing, Technical Specification limitations on pulsing will continue to protect the integrity of the fuel. Technical Specification 2.1, “Safety Limit – Fuel Element Temperature,” for the existing TRIGA-FLIP HEU fuel and the proposed Technical Specification for the TRIGA-LEU fuel is that the fuel shall not exceed a temperature of 1150 °C under any condition of operation. The licensee has not proposed any change to the limiting safety system setting (LSSS) of 525 EC. The basis for Technical Specification 3.1.2, “Reactor Core Parameters, Pulse Mode Operation,” for the TRIGA FLIP HEU fuel and the proposed basis of

the Technical Specification for the proposed LEU fuel is that the temperature of the fuel should not exceed 874 °C during pulse mode operation because hydrogen gas can build up pressure. In order to assure this temperature is never exceeded, this Technical Specification limits the maximum fuel temperature, during pulse mode operation, to a peak of 830 °C, a value that is unchanged between the current HEU and proposed LEU Technical Specifications. Technical Specification 3.1.2 limits the initial pulse insertions for a new or reconfigured core to $2.00 until the peak fuel temperature can be established. The pulse insertion to limit the temperature to 830 °C will then be determined by the licensee.

The performance of the pulse mode operation is analyzed with the GA computer program TRIGA-BLOOST. Calculated and measured data are presented in Table 4-12 in the conversion SAR for the PRNC FLIP fuel. This table indicates that the BLOOST calculation, at least for the PRNC, predicts higher power, energy release and fuel temperatures than are measured, which is conservative. Calculations show that a temperature of 830 °C will be reached for the fresh LEU fuel with a reactivity insertion of $2.10 (compared to the PRNC FLIP fuel of $2.05). For highly irradiated LEU fuel (2000 MWd burnup) it is calculated that a reactivity insertion of $2.10 will result in a maximum fuel temperature of 810 °C. There is a mechanical interlock that is set to limit the maximum pulse to $2.95, the purpose of which is to keep the maximum fuel temperature below the safety limit of 1150 °C. Based on the discussion above, the staff concludes that the LEU fuel temperature coefficient and the proposed Technical Specifications including the approach to limiting pulse reactivity for a new core configuration to $2.00 until the core is calibrated are acceptable. (See also the discussion of pulsing with respect to the thermal-hydraulics (Section 2.4) and Technical Specifications (Section 2.8)).

2.3.5 Conclusions

The calculations of neutronic and dynamic parameters indicate that neutronic conditions between the HEU and LEU cores are not significantly different and that conversion of the TAMU NSCR from the present HEU FLIP fuel to LEU (30/20) fuel should pose no significant problems or hazards. The safety limit, LSSS, and limiting conditions of operation for pulsing are not changed 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

The TAMU NSCR conversion SAR presents the thermal-hydraulic analysis of steady-state operation in two parts. The first part presents the methodology and results together with a discussion of the suite of computer codes used in the evaluation of the PRNC HEU (FLIP) core. The same computational technique was then used in the second part to analyze the TAMU LEU (30/20) core. Results from the analysis of the PRNC TRIGA serve as a benchmark for the computational technique utilized in the thermal-hydraulic analysis of the NSCR.

The conversion SAR did not provide a side-by-side performance comparison of the TAMU TRIGA core loaded with HEU (FLIP) fuel and LEU (30/20) fuel because the TAMU reactor never had a fresh FLIP core (their FLIP was used). Instead, the comparison is between the fresh PRNC FLIP core and the TAMU LEU (30/20) core. However the two reactors have differences in their designs that need to be considered. In the thermal design three features that distinguished the two reactors are:

(1) The PRNC core had water-followed control rods and transient rod while the TAMU core has fuel-followed control rods and a void-followed transient rod. This difference implies the two cores will have different flow areas for a three-element bundle (bundle with three fuel rods and one control rod). Also, the pattern of power peaking in a reactivity pulse will be different for the two reactors.

(2) Unlike the PRNC core, the peak power density (steady-state) for the TAMU core does not occur in the fuel element (rod) with the maximum rod power.

(3) Unlike the PRNC core, the hottest fuel element for the TAMU core does not coincide with the location for one of the IFEs.

The TAMU conversion SAR has addressed the differences in design features, as noted above, in conducting the thermal-hydraulic analysis of both steady-state and pulse operation.

For both the PRNC and the TAMU TRIGA, the fuel elements are cooled by natural convection. The flow rate and wall heat transfer coefficient are calculated by the computer code STAT. The STAT code and its associated methodology have been applied to the licensing of TRIGA reactors for the past 40 years. For example, the methodology was applied to the original Torrey Pines TRIGA reactor to calculate the flow rate in the coolant channel between the inner-most (B and C) rings of fuel elements. The calculated flow rate was conservative as it was 35% lower than that deduced from the measured channel exit temperatures.

Steady-state calculations were done for a single flow channel based on a four-rod cluster of fuel elements. The STAT model examines a single fuel element and a representative flow area associated with that single fuel element. The model ignores cross flow from one channel to another. Given the inlet water temperature, system pressure, local pressure loss coefficients and the axial power distribution for the bounding fuel elements, STAT calculates the natural circulation flow rate, and along the axial length of the flow channel, the coolant temperature, wall heat flux, and the clad temperature. STAT analysis was done for an average flow channel and a maximum powered channel using hydraulic characteristics of a typical flow channel. The maximum powered element is determined by the DIF3D diffusion code.

The channel total pressure drop listed in Tables 4-17 and 4-21 of the conversion SAR for the PRNC and TAMU core, respectively, does not include the hydrostatic pressure change in the channel. However, the approach used by the licensee is acceptable. A common core pressure drop is actually assumed for the average and maximum powered channel in calculating flow rates. This can be demonstrated by adding the hydrostatic head to the channel pressure drop as listed in the Tables. The uniform pressure condition in part supports the assumption of no cross flow between channels.

The STAT code uses the larger of the convection and the boiling heat transfer coefficient to calculate the clad-to-fluid temperature difference. The code uses a wall friction factor that is proportional to the Reynolds number to the quarter power. Flow through the TAMU core is by natural circulation and the flow is driven by the buoyancy difference between the coolant heated in the core and the cold pool water. This natural circulation flow is similar to a low velocity flow driven by a low-head pump and thus appropriate forced flow correlations can be used to model the thermal-hydraulic conditions of the core. The use of turbulent heat transfer and friction correlations in the STAT code is supported by a study[3] which showed that the transition to the

turbulent regime for a closely packed core, such as the TAMU reactor, is at Reynolds numbers less than 2000.

The STAT code also calculates the minimum deviation from nucleate boiling ratio (MDNBR). The ratio is determined by increasing the reactor power until the maximum core heat flux reaches the predicted critical heat flux (CHF). The lower of the CHF calculated from the McAdams correlation and the Bernath correlation was used to determine the MDNBR. Both of these correlations have been used in TRIGA licensing for decades and their predictions of CHF agree within 10% for 1-12 ft/s flow in narrow annuli.

The presence of guide tubes reduces slightly the flow area (about 3.5%) associated with the maximum powered fuel rod that is next to the transient rod. By using the actual reduced flow area instead of the representative flow area used in all calculations, the flow rate for the maximum powered rod is reduced from 0.089 kg/s to 0.087 kg/s. The corresponding MDNBR is reduced from 2.42 to 2.34. This demonstrates that the use of a representative flow area instead of a specific flow area has little effect on the thermal limit of the TAMU core.

In the conversion SAR the steady-state fuel temperature was calculated by a finite-difference code TAC2D. The code calculates temperatures in two-dimensional problems. The fuel rod model consists of the central zirconium rod, the fuel annulus, the fuel-to-clad gap, and the stainless steel clad. TAC2D requires as an input, a clad surface temperature or a coolant temperature with the corresponding clad surface heat transfer coefficient. The fuel temperature durning pulse operation is determined in the BLOOST calculation.

The calculated steady-state fuel temperatures are shown in Tables 4-8 and 4-11 of the conversion SAR for the PRNC core and the TAMU core respectively. The measured temperature was 45 EC lower than the calculated temperature at 0.5 MW(t) for the PRNC core which is conservative. At 1.0 MW(t) and 1.4 MW(t) the measured and calculated fuel temperatures for the PRNC core agreed well. The good agreement serves to assure that peak fuel temperature would have similar good agreement since the IFEs are located in the best possible positions to represent the peak fuel temperature.

Table 4-11 of the conversion SAR provides the predicted temperatures for the TAMU LEU (30/20) fuel elements. The predicted steady-state temperatures are for the hottest fuel element (in core position 5D3), the IFE (in the IF3 position), and the average core. The other thermal and hydraulic parameters for the TAMU LEU (30/20) core are shown in Table 4-22 of the conversion SAR. It is noted that for the TAMU LEU core under steady-state operation, the peak fuel temperature occurs in the maximum power rod and the corresponding MDNBR is calculated for this rod.

Water in position 3D of the TAMU core leads to power peaking especially in core locations 4D3 and 4D4. The fuel element in core position 4D3 has the maximum rod peaking factor (2.297 at beginning of life), defined as the peak to average power in a radial plane within a fuel rod. This peaking factor becomes important only in pulse operation. The IFE closest to core position 4D3 is located in 5D4, a location where uncertainty in neutron flux distribution is expected to be similar to that at 4D3. The conservative calculation of peak fuel temperature in pulse operation is supported by measured and calculated temperatures of pulses in the PRNC core as shown in Table 4-12 of the conversion SAR. The calculation assumes that the fuel is essentially

adiabatic without heat transfer to the clad and coolant and the calculated temperature is conservatively higher than the measured temperature.

The maximum steady-state power to initiate a pulse was set at 1 kW(t) by GA forty years ago. At this power level the fuel will be essentially at the ambient (coolant) temperature. Analysis has shown that below 1 kW(t) there is no change in peak fuel temperatures during a reactivity pulse.

The staff concludes that the thermal-hydraulic analysis in the TAMU conversion SAR adequately demonstrates that the proposed conversion from an HEU to an LEU core will result in no significant decrease in safety margins in regard to thermal-hydraulic conditions. The analyses were done with qualified calculational methods and conservative or justifiable assumptions. The applicability of the analytical methodology is demonstrated by comparing analytical results with measurements obtained from the PRNC reactor (loaded with FLIP fuel). In comparing the HEU (PRNC) and LEU (TAMU) cores the thermal-hydraulic analyses have accounted for design differences between the two reactors. The safety limit continues to protect 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). Conservative temperature limits as measured in an IFE were used in the development of the LSSS and pulsing LCOs and these limits continue unchanged in the Technical Specifications.

2.5 Accident Analysis

2.5.1 Loss-of-Coolant Accident (LOCA) Analysis

Based on previous analyses, two rod power limits were derived to assure that air cooling is sufficient to prevent excessive fuel temperatures (950 EC when the cladding temperature is greater than 500 EC) in a LOCA event (complete draining of the reactor pool). This temperature will not be exceeded if the power generated in a fuel element is less than 21 kW(t)/element for long term operation of a core at 1 MW(t) for the situation where there is no delay between reactor scram and loss of coolant (this is a very conservative assumption). A second limit of
28 kW(t)/element is derived for 1.0 MW(t) operation limited to less than or equal to
70 MWhr/week assuming a 15 minute delay between reactor scram and complete loss of coolant. For the LEU (30/20) core operating at 1.0 MW(t), the peak rod power is predicted to be 17.6 kW(t)/element. This is well below the 21 kW(t)/element limit. Therefore, the staff concludes that fuel integrity will continued to be maintained during a LOCA with the proposed LEU core.

2.5.2 Accidental Pulsing from Full Power

A pulsing accident from full power can be initiated from the ejection of an inserted transient rod or the unplanned removal of an experiment. Based on the accident analysis limiting the withdrawal of the transient rod at full power (1 MW(t)) to less than $2.95 in reactivity will produce fuel temperatures less than the safety limit of 1150 EC. Accidental pulsing from full power is an unlikely event because there are existing design and procedural steps that prevent it from happening. Some of these are:

- An interlock is in place at TAMU to prevent the initiation of pulsing from an initial reactor power greater than 1 kW(t).

- Procedures require that during a startup to steady-state power, the transient rod is raised to fully withdrawn before other rod movement.

- Pulsing procedures require the insertion of a mechanical pulse-stop prior to pulsing thus limiting the reactivity insertion of a pulse.

C Non-secured experiments shall have reactivity worth less than $1 (according to Technical Specification 3.6.1).

The staff concludes that there are sufficient design features and administrative restrictions in place to make accidental pulsing of the reactor unlikely.

2.5.3 Radiological Consequences of Design Basis and Maximum Hypothetical Accident

The licensee's proposed 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. Both the HEU and LEU analysis assumed a simultaneous loss of fuel cladding integrity with a simultaneous loss of reactor pool water, and a 2.6 x10^{-5} fission product release fraction. The release fraction used was based on experiments performed by GA on TRIGA fuel. Review of the report that documented the experimental results[4] showed that for steady-state operation at low temperatures (less than about 400 EC), the release of fission products from the fuel into the gap is by recoil and results in the release fraction of 2.6 x10^{-5}. At high temperature the release of fission products is by a temperature dependent diffusion process with the release fraction increasing with increasing fuel temperature. Since the axial peak fuel temperature (about 375 EC) establishes the release mechanism, the diffusion based release mechanism would not contribute significantly to the fission product release from the fuel. Therefore, use of a 2.6 x10^{-5} release fraction was appropriate. Other boundary conditions used for both fuel types were identical and were consistent with the previously accepted SAR. Comparison of the results for the HEU and LEU calculated fission product inventories showed that the LEU inventory was not significantly different for the radiological inventory analyzed, except for the xenon-135, which was marginally larger for the LEU core by approximately 6.0%. Review of the references and relevant sections of the NSCR conversion SAR 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 LEU fuel and the assumptions described in the previous subsection. Radiological exposure to the licensee staff was for a period of 5 minutes. The five minute stay time is based on times recorded for practice emergency evacuation drills performed for the reactor building. Radiological exposure to the public was calculated at just outside the site boundary and at 800 m which is the closest permanently occupied area. 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.

Review of the licensee's calculated results in the conversion SAR showed that both the thyroid and whole body doses for the LEU core at the licensee's fence line for a ground level release resulted in a maximum exposure of 71 mrem thyroid and 6.2 mrem whole body, and for the nearest permanently occupied area, a maximum exposure of 1.8 mrem thyroid and 0.14 mrem whole body, respectively. For an individual who remained inside the reactor building for five minutes subsequent to the radiological release, the dose was estimated to be 3.7 rem to the thyroid and 0.25 mrem whole body for the LEU core.

The calculations performed by the licensee and the assumptions made demonstrated that the inventory of radioactivity assumed and other boundary conditions used in the analysis were acceptable. The radiological consequences to the public and occupational workers at the NSCR from a postulated MHA for the proposed LEU-fueled reactor were found to be within acceptable doses. As a result of this review, the conclusion is that continued operation of the reactor poses no undue risk, from a radiological standpoint, to the public or the staff of the NSCR from the maximum hypothetical accident.

2.5.4 Conclusions

The licensee identified fuel element power levels that under various operating conditions could result in fuel temperatures that could lead to cladding failure under LOCA conditions. The licensee showed that the peak power generated in a fuel element in the LEU core is less than the power levels that could lead to fuel failure. Therefore, the staff concludes that air cooling the reactor core after a LOCA is sufficient to prevent cladding failure.

The licensee did not identify any new reactivity addition accidents not previously analyzed for the HEU-fueled reactor. The design features and administrative restrictions that prevent accidental pulsing from occurring at full power continue unchanged with 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 the reactivity addition accidents.

Review of the radiation source term and MHA calculations performed by the licensee, including the assumptions used, demonstrated that the inventory of radioactivity assumed and other boundary conditions used in the analysis were acceptable. The radiological consequences to the public and occupational workers at the NSCR from a postulated MHA for the proposed LEU-fueled reactor were found to be acceptable. As a result of this review, it is concluded that continued operation of the reactor poses no undue risk from a radiological standpoint to the public or the staff of the NSCR from the MHA.

2.6 Fuel Storage

The licensee has analyzed the fresh and spent fuel storage and has determined that the existing storage facilities can hold the LEU fuel and meet the requirement of Technical Specification 5.5 a. that all fuel elements shall be stored in a geometrical array for which the k-effective is less than 0.8 for all conditions of moderation. The staff has reviewed the licensee's analysis and agrees that Technical Specification 5.5 a. will continue to be met. The

licensee does not propose any modifications to the existing facilities. Therefore, the staff concludes that LEU fuel can be acceptable stored at the NSCR.

2.7 LEU Startup Plan

With the LEU 30/20 fuel elements, the core will contain a total of 90 fuel elements. The licensee plans to perform a criticality experiment by adding fuel elements and measuring the multiplication factor, k, after each insertion. For this type of reactor the licensee expects that criticality should occur when 58-68 fuel elements have been inserted. This is a standard technique for approaching initial criticality.

After criticality has been achieved (less than a full core), other standard measurements will be performed on the new core, such as core excess reactivity using the period method, and reactivity worth of each control rod using the rod drop technique. For the control rod aggregate, the acceptance criteria using the rod drop technique will be a combined value of $1.50 to $3.00. Each individual control rod will be calibrated with acceptance criteria of $1.00 to $3.75 reactivity worth (for this measurement, more fuel will be added to the core).

Once criticality has been reached, loading will continue with all but two control rods fully inserted. After the insertion of four fuel elements the control rods will be recalibrated. The final excess reactivity will be measured. The expected excess reactivity with all 90 fuel elements installed and the core next to the graphite Coupler Box is $7.79.

This approach to loading is acceptable since Technical Specification 3.1.3, “Shutdown Margin,” sets a value for the minimum shutdown margin to be no less than $0.25 when the highest worth control rod and the regulating rod withdrawn. As long as the shutdown margin never drops below $0.25 the reactor loading should proceed safely.

After reloading the core, the licensee will calibrate the power channels. The NSCR is cooled by natural convection, so there is a constant temperature rise of the water over time. The temperature rise of the water over time is one of the techniques for determining the power level. The licensee will first operate the reactor at ~250 kW(t) and measure the temperature rise of the water. The power will then be raised to 750 kW(t) followed by another measurement of the water temperature rise. This temperature rise is correlated with the actual power and this is compared to the other power channel monitors. Acceptance is that at 1 MW(t) of power, all channels agree within 2%. These steps complete the low and intermediate power testing. This is the standard approach to power calibration at a TRIGA natural convection cooled pool type reactor.

The licensee is aware that in fresh TRIGA fuel the cladding will permanently expand during the first few power increases to full power. This increases the gap between the fuel meat and the cladding and increases the fuel temperature at power. Therefore the licensee will increase the power in small increments, measuring the temperature rise in the fuel. After the initial operation at 1MW(t), the power will be decreased and again raised incrementally with fuel temperatures being measured. Because of fuel element expansion, there is an expected reactivity loss from the core of $1.50 to $2.00.

One of the power channel indicators is a fission chamber. On a log-log plot the output should be linear with power between 0.1 and 1.0 MW(t). This will be checked. The final check will be to raise the reactor power to 1.25 MW(t) to test the scram which is set to trip at 125% of

nominal power. The 1.25 MW(t) level is less than the maximum level set in Technical Specification 3.1.1 which limits the maximum steady state power to 1.3 MW(t), which can only be approached during maintenance and testing.

The licensee proposes to calibrate the pulsing mode operation by starting with a low reactivity insertion of $1.25 followed by pulses in $0.25 increments of reactivity insertion. The maximum fuel temperature is recorded until the 830 EC temperature limit in Technical Specification 3.1.2 is approached. This will establish the largest reactivity insertion that will be allowed with the LEU core. This limit will be checked over time since it is assumed that the negative prompt coefficient of reactivity will decrease in magnitude with burnup.

Once the low and intermediate power level tests are successfully completed, the licensee has a plan to bring the reactor to full power operations. This plan is adequate to assure there are no inherent problems with the reactor and that all systems are verified to be functional.

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 License Conditions and Technical Specifications

The licensee has proposed changes to the license conditions and technical specifications to reflect the conversion to LEU fuel. These changes are discussed in the following sections.

2.8.1 License Conditions

On July 21, 2006, the NRC issued an Order to the licensee which changed special nuclear material possession license conditions to allow the licensee to start preparations for the conversion to LEU.

The July 21, 2006, Order changed License Condition II.B.(2) to allow possession and use of the current HEU fuel in connection with operation of the reactor while removing the authorization from the license to receive additional HEU fuel. This conversion Order further changes this license condition removing the authority to use the HEU fuel. The licensee is authorized to possess the material until it is removed from the facility.

The July 21, 2006, Order added a new License Condition II.B.(8) to receive and possess the new LEU reactor core. This conversion Order changes this license condition to allow the use of the LEU core in connection with operation of the reactor.

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 II.C.(2), which incorporates the technical specifications into the license, is updated to incorporate the technical specifications changes needed for conversion into the license.

2.8.2 Technical Specifications

For the TAMU HEU to LEU conversion, the changes proposed to the technical specifications primarily involve the fuel type and certain related specifications. However, others technical specifications have been changed such as the safety limit. The following paragraphs discuss the proposed changes in the technical specifications.

Changes are proposed to the cover page and table of contents to reflect the updated technical specifications. These changes are administrative in nature and reflect the other proposed changes to the technical specifications.

Changes are proposed to the definition section of the technical specifications to reflect the conversion to LEU fuel. The definition section of the technical specifications would be renumbered to reflect the changes. The proposed changes are:

1.9 FLIP Core (previous number) - Definition removed to reflect the conversion to LEU fuel.

1.11 Fuel Element - All references to fuels other than the proposed LEU fuel were removed from the definition.

1.17 Mixed Core (previous number) - The definition of a Mixed Core was removed since it will not apply to an all LEU core.

1.20 Steady State Operational Core - The references to fuels other than the new LEU fuel were removed.

1.40 Standard Core (previous number) - The definition was removed because it is no longer needed with the conversion to LEU fuel.

A change is proposed to Technical Specification 2.1, “Safety Limit Fuel Element Temperature” which contains two safety limits on fuel temperature which are dependent on fuel type. The safety limit for standard TRIGA fuel elements would be removed since with the conversion standard fuel will not be used. The properties and performance of the TRIGA LEU higher uranium weight percent fuels, including the LEU (30/20) fuel proposed for the TAMU NSCR, have been evaluated by the NRC in NUREG-1282 and generically approved for use in TRIGA reactors. This approval came with the provision that each licensee using the fuel perform a case-by-case analyses to discuss individual reactor operating conditions. The TAMU LEU (30/20) core adopts the fuel safety limit of 1150 EC as established in NUREG-1282 for water-cooled fuel. The safety limit is unchanged from that for the FLIP HEU fuel. Reference to FLIP fuel in the technical specification would be replaced by LEU fuel.

Changes are proposed to the basis for Technical Specification 2.2, “Limiting Safety System Setting.” A fuel temperature of 525 EC as measured in an IFE in the TAMU LEU (30/20) core is the proposed LSSS (same LSSS temperature as for the existing HEU core). The basis for selecting this limit is to have sufficient safety margin to the safety limit. In particular, the licensee’s technical basis addressed the impact of the IFE not being the hottest fuel element in the core. The value of the LSSS allows for this. It is noted in the conversion SAR that due to the thermal capacity of the fuel and the short time duration of a pulse, the LSSS on fuel temperature will have no effect on limiting the peak power generated during a pulse. Peak power is controlled by technical specification limits on pulse reactivity addition. Also in the case of pulse mode operation, a reactor scram is initiated by the present timer which causes the

transient rod to scram. This cuts off the tail of the pulse and limits total energy release. The basis of the technical specification would be revised to reflect the use of LEU fuel.

Changes are proposed for Technical Specification 3.1.2, “Pulse Mode Operation.” The Technical Specification states in part that until a core has been calibrated, the maximum pulse reactivity insertion will be limited to $2.00. The technical specification also limits the reactivity to be inserted for pulse operation. The amount of inserted reactivity shall not produce a peak fuel temperature of 830 EC or above. These values are not changed with the conversion to LEU fuel. Reference to a full FLIP fuel core in the technical specification would be changed to LEU fuel core to reflect the conversion to LEU fuel. It is also proposed to revise the basis of the technical specification to reflect the use of LEU fuel. A basic limit for TRIGA fuel is dictated by the dissociation of hydrogen from the uranium-zirconium hydride fuel. The temperature limit is based on limiting the pressure of the dissociated hydrogen for the core to restrict the reactivity limit for the pulse mode of operation. It has been observed that after extensive steady state operation at higher power levels (1 MW(t)), hydrogen in the TRIGA hydride fuel will migrate from the central high temperature regions to the cooler outer regions of the fuel annulus. If the fuel temperature exceeds 874 EC during pulsing the hydrogen pressure will be sufficient to cause expansion of the microscopic holes in the fuel that grows with each pulse. This could lead to fuel damage. A decrease in temperature from 874 EC to 830 EC reduces the hydrogen pressure by a factor of two and provides an acceptable safety factor.

Changes are proposed for Technical Specification 3.1.4, “Core Configuration Limitation.” All references to fuels other than the LEU fuel would be removed.

Changes are proposed for Technical Specification 5.1, “Reactor Fuel.” The proposed changes to this Technical Specification removes references to fuels other than the proposed LEU fuel and includes the proposed design specifications for the LEU fuel.

Changes are proposed for Technical Specification 5.2, “Reactor Core,” Technical Specification 5.3, “Control Rods,” and Technical Specification 5.7, “Reactor Pool Water Systems.” The basis for these technical specifications would be changed to reflect the conversion to LEU by referring to analysis performed for the LEU conversion core.

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) of the regulations, 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 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.

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 NSCR. 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

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.

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3. S-H. Kim et al., "Heat Transfer Experiments and Correlations for Natural and Forced Circulations of Water in Rod Bundles at Low Reynolds Numbers," Eleventh Biennial U.S. TRIGA Users Conference, Washington, D.C., April 10-13, 1988.

4. F.C. Foushee and R.H. Peters, "Summary of TRIGA Fuel Fission Product Release Experiments," GULF-EES-A10801, TRIGA Reactors, Gulf Energy and Environmental Systems, Inc., September 1971.

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