POLICY ISSUE
(Information)

August 31, 1995 SECY-95-222

FOR: The Commissioners

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: ORDER MODIFYING LICENSE TO CONVERT FROM HIGH-ENRICHED TO LOW-ENRICHED URANIUM FUEL (GEORGIA TECH RESEARCH REACTOR)

PURPOSE:

To inform the Commission about an order issued to a non-power reactor to convert from high-enriched uranium (HEU) to low-enriched uranium (LEU) fuel in accordance with Section 50.64 of Title 10 of the Code of Federal Regulations (10 CFR 50.64).

DISCUSSION:

As cited in 10 CFR 50.64, conversion from the use of HEU to LEU fuel is required if suitable fuel and funding are available through the U.S. Department of Energy or another Federal agency. In accordance with COMLZ-87-43, wherein the Commission asked the Office of Nuclear Reactor Regulation (NRR) to tell it of subsequent conversion orders, the staff informed the Commission when it issued orders to convert from HEU to LEU fuel in SECY-87-171 (Rennselaer Polytechnic Institute), SECY-89-001 (Ohio State University and Worcester Polytechnic Institute), SECY-90-184 (Manhattan College), SECY-90-402 (Iowa State University), SECY-91-122 (University of Missouri, Rolla), SECY-93-154 (Rhode Island Atomic Energy Commission), and SECY-93-208 (University of Virginia). In SECY-90-184, the staff summarized the status of the conversions from HEU to LEU fuel.

NOTE: TO BE MADE PUBLICLY AVAILABLE IN 5 WORKING DAYS FROM THE DATE OF THIS PAPER

Contact:
Marvin M. Mendonca ONDD/NRR
415-1128

Enclosed is the order that NRR issued to the Georgia Institute of Technology on June 16, 1995. The required 20-day period for hearing requests has expired and a request for hearing has been received. The request for hearing is under consideration by an Atomic Safety and Licensing Board.

COORDINATION:

The Office of the General Counsel has reviewed this paper and has no legal objection to it.

James M. Taylor
Executive Director
for Operations

Enclosure:
Order Modifying License
for Georgia Institute
of Technology

cc Service List
See next page

DISTRIBUTION:
Commissioners
OGC
OCAA
OIG
OPA
OCA
REGION II
EDO
SECY

Georgia Institute of Technology

Docket No. 50-160

cc:

Mr. Charles H. Badger
Office of Planning and Budget
Room 608
270 Washington Street, S.W.
Atlanta, Georgia 30334

Mayor of City of Atlanta
55 Trinity Avenue, S.W.
Suite 2400
Atlanta, Georgia 30303

Dr. G. Poehlein
Vice President for Interdisciplinary Programs
Georgia Institute of Technology
225 North Avenue
Atlanta, Georgia 30332

Dr. William Vernetson
Director of Nuclear Facilities
Department of Nuclear Engineering Sciences
University of Florida
202 Nuclear Sciences Center
Gainesville, Florida 32611

Mr. Pedro B. Perez, Associate Director
Nuclear Reactor Program
North Carolina State University
P. O. Box 7909
Raleigh, North Carolina 27695-7909

Dr. R. U. Mulder, Director
Reactor Facility
University of Virginia
Charlottesville, Virginia 22901

Joe D. Tanner, Commissioner
Department of Natural Resources
47 Trinity Avenue
Atlanta, Georgia 30334

Dr. Rodney Ice, MORS
Neely Nuclear Research Center
Georgia Institute of Technology
900 Atlantic Drive
Atlanta, Georgia 30332-0425

Ms. Pamela Blockey-O'Brien
D23 Golden Valley
Douglasville, Georgia 30134

Mr. E. F. Cobb
Southern Nuclear Company
42 Inverness Center
Birmingham, Alabama 35242

Dr. G. Wayne Clough, President
Georgia Institute of Technology
Carnegie Building
Atlanta, Georgia 30332-0325

Ms. Glenn Carroll
139 Kings Highway
Decatur, Georgia 30030

Charles Bechhoefer, Chairman
Atomic Safety and Licensing Board Panel
U.S. NRC
Washington, D.C. 20555

Dr. Jerry R. Kline
Atomic Safety and Licensing Board Panel
U.S. NRC
Washington, D.C. 20555

Dr. Peter S. Lam
Atomic Safety and Licensing Board Panel
U.S. NRC
Washington, D.C. 20555

Mr. James C. Hardeman, Jr.
Manager Environmental Radiation Program
Environmental Protection Division
Dept. of Natural Resources
State of Georgia
4244 International Parkway Suite 114
Atlanta, Georgia 30354

Dr. Ratib A. Karam, Director
Neely Nuclear Research Center
Georgia Institute of Technology
Atlanta, Georgia 30332

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 16, 1995

Dr. Ratib A. Karam, Director
Neely Nuclear Research Center
Georgia Institute of Technology
Atlanta, Georgia 30332

SUBJECT: ISSUANCE OF ORDER MODIFYING LICENSE NO. R-97 TO CONVERT FROM HIGH-TO LOW-ENRICHED URANIUM - GEORGIA INSTITUTE OF TECHNOLOGY (TAC NO. M85896)

Dear Dr. Karam:

The U.S. Nuclear Regulatory Commission (NRC) is issuing an Order (Enclosure 1) modifying Facility Operating License No. R-97, for the Georgia Tech Research Reactor (GTRR), to authorize the conversion from high-enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel. This Order modifies the license in accordance with Section 50.64 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.64), which requires that a non-power reactor, such as the reactor at the Georgia Institute of Technology, convert to LEU fuel under certain conditions. The Order is being issued in accordance with 10 CFR 50.64(c)(3) and in response to your proposal of January 21, 1993, as supplemented on March 2, March 21, and July 15, 1994. Enclosure 2 contains the changes to the GTRR Technical Specifications, and Enclosure 3 is the staff's Safety Evaluation Report.

The portions of the Order that allow possession of the LEU fuel [License Condition 2.B(4)] and require submission of a startup report within six months of achieving initial criticality with LEU fuel [License Condition 2.C(4)], are to be implemented 30 days after the date of publication of this Order in the *Federal Register*. The portions of the Order that change License Condition 2.B(2) to allow the possession but not use of the HEU fuel and that change License Condition 2.C(2) and the Technical Specifications to be applicable to LEU fuel, are to be implemented on 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.

Please inform me when you receive the LEU fuel elements and when the HEU fuel is completely removed from the facility.

A copy of the Order is being sent to the *Federal Register* for publication.

Sincerely,

M. M. Mendonca

Marvin M. Mendonca, Senior Project Manager
Non-Power Reactors and Decommissioning
Project Directorate
Division of Project Support
Office of Nuclear Reactor Regulation

Docket No. 50-160

Enclosures:
1. Order
2. Replacement pages for
   Technical Specifications
3. Safety evaluation

cc w/enclosures:
See next page

Georgia Institute of Technology

Docket No. 50-160

cc:

Mr. Charles H. Badger
Office of Planning and Budget
Room 608
270 Washington Street, S.W.
Atlanta, Georgia 30334

Mayor of City of Atlanta
55 Trinity Avenue, S.W.
Suite 2400
Atlanta, Georgia 30303

Dr. G. Poehlein
Vice President for Interdisciplinary Programs
Georgia Institute of Technology
225 North Avenue
Atlanta, Georgia 30332

Dr. William Vernetson
Director of Nuclear Facilities
Department of Nuclear Engineering Sciences
University of Florida
202 Nuclear Sciences Center
Gainesville, Florida 32611

Mr. Pedro B. Perez, Associate Director
Nuclear Reactor Program
North Carolina State University
P. O. Box 7909
Raleigh, North Carolina 27695-7909

Dr. R. U. Mulder, Director
Reactor Facility
University of Virginia
Charlottesville, Virginia 22901

Joe D. Tanner, Commissioner
Department of Natural Resources
47 Trinity Avenue
Atlanta, Georgia 30334

Dr. Rodney Ice, MORS
Neely Nuclear Research Center
Georgia Institute of Technology
900 Atlantic Drive
Atlanta, Georgia 30332-0425

Mr. E. F. Cobb
Southern Nuclear Company
42 Inverness Center
Birmingham, Alabama 35242

Dr. G. Wayne Clough, President
Georgia Institute of Technology
Carnegie Building
Atlanta, Georgia 30332-0325

Ms. Glenn Carroll
139 Kings Highway
Decatur, Georgia 30030

Dr. Ratib A. Karam, Director
Neely Nuclear Research Center
Georgia Institute of Technology
900 Atlantic Drive
Atlanta, Georgia 30332-0425

Charles Bechhoefer, Chairman
Atomic Safety and Licensing Board Panel
U.S. NRC
Washington, D.C. 20555

Dr. Jerry R. Kline
Atomic Safety and Licensing Board Panel
U.S. NRC
Washington, D.C. 20555

Dr. Peter S. Lam
Atomic Safety and Licensing Board Panel
U.S. NRC
Washington, D.C. 20555

Mr. James C. Hardeman, Jr.
Manager Environmental Radiation Program
Environmental Protection Division
Dept. of Natural Resources
State of Georgia
4244 International Parkway
Suite 114
Atlanta, Georgia 30354

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
GEORGIA INSTITUTE OF TECHNOLOGY	)	Docket No. 50-160
(Georgia Tech Research Reactor)	)	

ORDER MODIFYING FACILITY OPERATING LICENSE NO. R-97

I.

The Georgia Institute of Technology (Georgia Tech or the licensee) is the holder of Facility Operating License No. R-97 (the license) issued on December 29, 1964, by the U.S. Atomic Energy Commission. The license, as amended on June 6, 1974 (Amendment No. 1) and by subsequent amendments, authorizes operation of the Georgia Tech Research Reactor (GTRR or the facility) at steady-state power levels up to 5 megawatts thermal (MWt). The research reactor is located in the Neely Nuclear Research Center, in the north central portion of the Georgia Tech campus in Atlanta, Georgia.

II.

On February 25, 1986, the U.S. Nuclear Regulatory Commission (NRC or the Commission) promulgated a final rule in Section 50.64 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.64) limiting the use of high-enriched uranium (HEU) fuel in domestic research and test reactors (non-power reactors) (see 51 *FR* 6514). The rule, which became effective on March 27, 1986, requires that each licensee of a non-power reactor (NPR) replace its HEU fuel with low-enriched uranium (LEU) fuel acceptable to the Commission. This replacement is contingent upon Federal Government funding for conversion-related costs, and is required unless the Commission has determined that the reactor has a unique purpose as defined in 10 CFR 50.2. The rule is intended to promote the common defense and security by reducing the risk of theft or diversion of HEU fuel used in non-power reactors and the consequences to public health, safety and the environment from such potential theft or diversion.

Sections 50.64(b)(2)(i) and (ii) require that a licensee of an NPR (1) not initiate acquisition of additional HEU fuel, if LEU fuel that is acceptable to the Commission for that reactor is available when the licensee proposes that acquisition, 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).

Section 50.64(c)(2)(i) requires, among other things, that each licensee of an NPR authorized to possess and to use HEU fuel, develop and submit to the Director of the Office of Nuclear Reactor Regulation (Director, NRR) by March 27, 1987, and at 12-month intervals thereafter, a written proposal for conforming to the requirements of the rule.

Section 50.64(c)(2)(i) also requires the licensee to have the following in its proposal: (1) a certification that Federal Government funding for conversion is available through the U.S. Department of Energy (DOE) or another appropriate Federal agency and (2) a schedule for conversion, based upon the availability of replacement fuel acceptable to the Commission for that reactor, and upon consideration of other factors such as the availability of shipping casks, implementation of arrangements for available financial support, and reactor usage.

Section 50.64(c)(2)(iii) requires the licensee to include in its proposal, to the extent required to effect conversion, all necessary changes to the license, facility, or procedures. This paragraph also requires the licensee to submit supporting safety analyses so as to comply with the schedule established for conversion.

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

Section 50.64(c)(3) requires the Director, NRR, 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 the public health and safety, any necessary changes to the license, facility, or procedures. In the *Federal Register* notice of the final rule, the Commission indicated that in most cases, if not all, an Enforcement Order would be issued to modify the license.

Section 2.202, the current authority for issuing Orders of all types, including Orders to modify licenses, provides, among other things, that the Commission may modify a license by serving an Order on the licensee. The licensee or other person adversely affected by the Order may demand a hearing with respect to any part or all of the Order within 20 days from the date of the notice or such other period as the notice may provide.

III.

On January 21, 1993, as supplemented on March 2, March 21, and July 15, 1994, the licensee submitted a proposal to convert from the use of HEU to the use of LEU. This proposal contained descriptions of the modifications, supporting safety analyses, and plans for conversion. The conversion consists of replacing HEU with LEU fuel elements. The LEU fuel elements contain material test reactor (MTR)-type fuel plates, with the fuel consisting of uranium silicide dispersed in an aluminum matrix and completely clad in aluminum alloy. These plates contain an enrichment of less than 20 percent uranium-235.

The NRC staff has reviewed the licensee's proposal for conversion to LEU fuel and the requirements of 10 CFR 50.64 and has determined that the public health and safety and the common defense and security support a conversion of the facility from the use of HEU to LEU fuel in accordance with the attachment to this Order and the schedule requirements that follow. The attachment to this Order specifies the changes to the license and Technical Specifications that are needed to implement the requirements of this Order.

IV.

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

Facility Operating License No. R-97 be modified as stated in the "ATTACHMENT TO ORDER MODIFYING FACILITY OPERATING LICENSE NO. R-97" by adding License Conditions 2.B(4) and 2.C(4) on the thirtieth day after the date of publication of this Order in the *Federal Register* and by revising the License Conditions 2.B(2) and 2.C(2) and Technical Specifications on the day the licensee receives an adequate number and type of LEU fuel elements that are necessary to operate the facility as specified in the licensee's proposal as supplemented.

V.

In accordance with 10 CFR 2.202, the licensee or any other 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. The answer may consent to this Order. Unless the answer consents to this Order, the answer shall, in writing and under oath or affirmation, set forth the matters of fact and law on which the licensee or other person adversely affected relies and the reasons as to why the Order should not have been issued. Any request for a hearing shall be submitted to the Secretary, U.S. Nuclear Regulatory Commission, ATTN: Chief, Docketing and Service Section, Washington, DC 20555. Copies also shall be sent to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, to the Assistant General Counsel for Hearings and Enforcement at the same address, and to the licensee if the hearing request is by a person other than the licensee. If a person other than the licensee requests a hearing, that person shall set forth with particularity the manner in which the person's interest is adversely affected by this Order and shall address the criteria set forth in 10 CFR 2.714(d).

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

In the absence of any request for a hearing, the provisions specified in this Order shall be effective and final 20 days from the date of this Order without further order or proceedings.

FOR THE NUCLEAR REGULATORY COMMISSION

Frank J. Miraglia

Frank J. Miraglia, Acting Director
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland
this 16th day of June 1995

Attachments:
As stated

ATTACHMENT TO ORDER

MODIFYING FACILITY OPERATING LICENSE NO. R-97

A. License Conditions Revised and Added by This Order

2.B(2) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to possess, but not use, up to 4.9 kilograms of contained uranium-235 at enrichments greater than 20 percent in the form of MTR-type reactor fuel until the existing inventory of this fuel is removed from the facility.

2.B(4) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess, and use at any one time in connection with the operation of the reactor up to 8.85 kilograms of contained uranium-235 at enrichments less than 20 percent in the form of MTR-type reactor fuel.

2.C(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through the Order Modifying Facility Operating License No. R-97, dated June 16, 1995, and Amendment No. 10 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

2.C(4) Startup Test Report

The licensee shall submit a startup test report within six months after achieving initial criticality with low-enriched uranium reactor fuel in accordance with the Order Modifying Facility Operating License No. R-97, dated June 16, 1995. This report shall be sent as specified in 10 CFR 50.4, "Written Communications."

B. The Technical Specifications will be revised by this Order in accordance with the Enclosure to the Order Modifying Facility Operating License No. R-97, dated June 16, 1995, Docket No. 50-160, and as discussed in the safety evaluation for this Order.

ENCLOSURE TO THE ORDER MODIFYING FACILITY OPERATING LICENSE NO. R-97

DATED June 16, 1995

DOCKET NO. 50-160

Remove and insert the pages of the Technical Specifications (Appendix A) as indicated below. The revised pages are identified by amendment or revision number and the area of change is marked by a vertical line in the margin.

Remove Page	Insert Page
6	6
Figure II-1	Figure II-1
8	8
37	37

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 SAFETY LIMITS IN THE FORCED CONVECTION MODE

APPLICABILITY

This specification applies to the interrelated variables associated with core thermal and hydraulic performance in the steady state with forced convection flow. The variables are reactor thermal power, reactor coolant flow, reactor coolant inlet temperature, and the moderator level in the reactor vessel.

OBJECTIVE

To maintain the integrity of the fuel element cladding and prevent the release of significant amounts of fission products.

SPECIFICATION

a. The reactor power shall not exceed the limit specified in Figure II-1 corresponding to values of reactor coolant flow.
b. The reactor coolant inlet temperature shall not exceed 123°F.
c. The moderator level shall be within 12 inches of overflow.

BASIS

Gross fuel element failure and concomitant fission product release will not occur until after there is onset of flow instability. The integrity of the fuel element cladding can be assured by control of the reactor power, the reactor coolant flow rate, and reactor coolant outlet (or inlet) temperature.

The basis for establishing the safety limits on reactor power, coolant flow, and outlet temperature is thermal hydraulic analysis calculating the values of these parameters at which flow instability and departure from nucleate boiling occur.

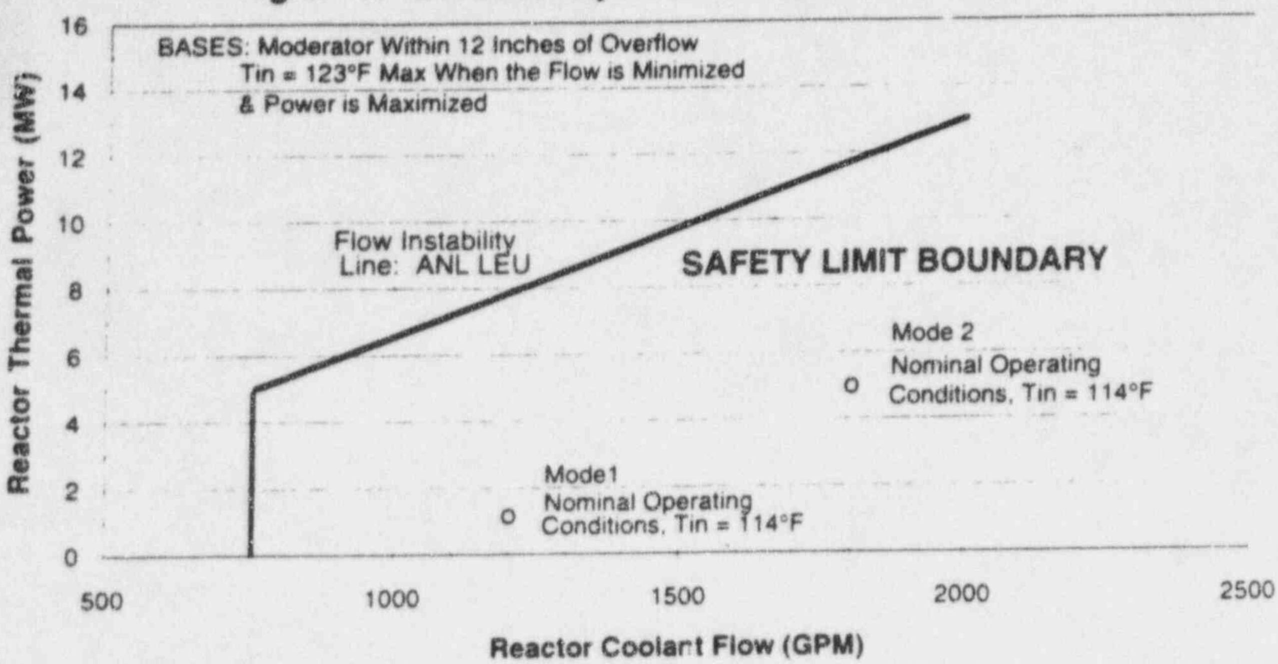
This analysis[1,1a] establishes that flow instability will not occur at power levels up to 10.6 MW and departure from nucleate boiling will not occur at power levels up to 10.8 MW. These results were obtained with the coolant outlet temperature and coolant flow at their respective limiting safety system settings. The analysis is not extended below 760 GPM because the orifices are not designed for extremely low flow.

---

(1) Letter, R. S. Kirkland to USAEC, June 23, 1972, Enclosure 5.

(1a) Letter, R. A. Karam to Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, January 21, 1993, Attachment 1 "Analyses for Conversion of the Georgia Tech Research Reactor from HEU to LEU Fuel," J. M. Matcs, S. C. Mo, and W. L. Woodruff, Argonne National Laboratory, September 1992

Fig. II - 1. GTRR Safety Limit for Forced Convection



BASIS

The trip settings are chosen so that the reactor is operated with no onset of nucleate boiling. Analyses incorporating all the engineering uncertainty factors were made at 1800 gallons per minute total coolant flow, five MW thermal power and an inlet reactor coolant temperature of 114°F. The results showed that a maximum fuel surface temperature 11°F less than the local D_2O saturation temperature would be obtained.[1,1a]

Operation during the period 1964 to 1973 has demonstrated that a 1000 GPM flow trip setting provides for safe operation of the reactor at power levels less than or equal to one MW. The 1.25 MW power trip setting has been chosen to ensure that no onset of nucleate boiling occurs with the reduced coolant flow.

REFERENCES

(1) Letter, R. S. Kirkland to USAEC, June 23, 1972, Enclosure 5.

(1a) Letter, R. A. Karam to Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, January 21, 1993, Attachment 1 "Analyses for Conversion of the Georgia Tech Research Reactor from HEU to LEU Fuel," J. M. Matos, S. C. Mo, and W. L. Woodruff, Argonne National Laboratory, September 1992

2.2.2 LIMITING SAFETY SYSTEM SETTINGS IN NATURAL CONVECTION MODE

APPLICABILITY

Applies to the values of safety system settings when operating in the natural convection mode.

OBJECTIVE

To assure the reactor is not operated at a power level sufficient to cause fuel damage.

SPECIFICATION

The reactor thermal power safety system setting shall not exceed 1.1 kW when operating in the natural convection mode.

BASIS

In the natural convection mode of reactor operation the main coolant pumps are not operating. The reactor isolation valves may be closed so that only internal, natural convection is available for cooling. Experience with the GTRR has shown that the reactor can be operated at one kW indefinitely without exceeding a bulk reactor temperature of 123 F.

5.0 SITE DESCRIPTION

5.1 SPECIFICATION

a. The reactor facility is located on the Georgia Institute of Technology campus in the city of Atlanta, Georgia.
b. The restricted area is formed by the six-foot security fence on the east, south and west of the containment building and the laboratory building on the north. The closest unrestricted area is 40 meters from the reactor stack exhaust.
c. The exclusion area is the area inside the circle formed by a 100 meter (328 foot) radius centered at the reactor.
d. The low population zone outer boundary is formed by a 400 meter (1312 foot) radius from the containment building.
e. The population center distance for the GTRR is established as a radius of 523 meters (1750 feet) from the containment building.

5.2 FUEL ELEMENTS

SPECIFICATIONS

The LEU fuel elements shall be of the MTR type consisting of 18 fuel plates of uranium silicide with an enrichment less than 20%. Each LEU fuel plate will have a nominal loading of 12.5 grams of U-235. The HEU fuel elements shall also be of the MTR type consisting of 16 fuel plates of uranium aluminide with an enrichment of 93%. Each HEU fuel plate will have nominal loading of 11.75 grams of U-235.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING THE ORDER MODIFYING FACILITY OPERATING

LICENSE NO. R-97 TO CONVERT FROM

HIGH-ENRICHED TO LOW-ENRICHED URANIUM FUEL

GEORGIA INSTITUTE OF TECHNOLOGY

DOCKET NO. 50-160

1 INTRODUCTION

Section 50.64 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.64) requires licensed non-power reactors (NPRs) to convert from the use of high-enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel (<20 percent U-235), unless specifically exempted from the conversion.

Starting in 1978, the U.S. Department of Energy (DOE) took a leading role in an international program to provide the capability for NPRs to be converted from HEU to LEU. Among the activities in this program were reactor analyses, implementation of a demonstration conversion, and an extensive fuel development and qualification program. In an effort to standardize and control costs of the conversion of the material test reactor (MTR) fuel designs in the United States, it was agreed among the DOE staff, NRC licensees, and the NRC staff, that only a single plate-type fuel design would be made available (Reference 1).

The NRC furnished guidance to all non-power reactor (NPR) licensees in preparing their revised safety analyses, and advised licensees to concentrate on and consider those conditions and characteristics of the reactor, the facility operating license, and Technical Specifications that are dependent on fuel enrichment and, therefore, that might be changed by the fuel conversion.

The Georgia Institute of Technology (Georgia Tech or the licensee) has proposed to convert the fuel in the Georgia Tech Research Reactor (GTRR) from HEU to LEU. By letter dated January 21, 1993, the licensee submitted a supplement to the safety analysis report (SAR) describing the changes needed in connection with conversion to LEU fuel. In addition, the licensee submitted proposed necessary changes to the facility Technical Specifications in order to operate the reactor with LEU fuel. The staff transmitted requests for additional information by letters dated January 13 and May 17, 1994. The licensee responded by letters dated March 2, March 21, and July 15, 1994. The analyses by the licensee show that no changes in facility design or operating procedures except those specifically related to the LEU fuel are required.

This safety evaluation was prepared by M. M. Mendonca (Senior Project Manager, Non-Power Reactors and Decommissioning Project Directorate, Division of Project Support, Office of Nuclear Reactor Regulation) and R. E. Carter and J. R. Miller of Lockheed Idaho Technologies Company, Idaho National Engineering Laboratory.

2 EVALUATION

2.1 General Facility Description

The GTRR is a heavy-water-moderated and cooled tank-type reactor, currently fueled with highly enriched material test reactor (MTR)-type fuel. The reactor core is approximately a right circular cylinder about 0.71 meter (m) [2 ft, 4 in] in diameter and 0.61 m [2 ft] in height with provision for up to 19 fuel assemblies spaced about 0.15 m [6 in] apart in a triangular array. The reactor core is centrally located in a closed tank. The tank is about 1.83 m [6 ft] in diameter and about 3.14 m [10 ft] high. This tank is suspended within a graphite cup that provides about 0.71 m [2 ft] of additional neutron reflection.

The GTRR is licensed to operate in two forced-flow modes: at steady thermal power levels not to exceed 5 megawatts thermal (MWt) when cooled by forced convection at a nominal flow rate of 6813 liters per minute (l/m) [1800 gallons per minute (gpm)], and at steady thermal power levels not to exceed 1 MWt when cooled by forced convection at a nominal flow rate of 3785 l/m [1000 gpm]. The reactor is also authorized to operate at power levels up to 1.1 kilowatts thermal (kWt) when cooled by natural convection.

2.2 Fuel Construction and Geometry

The HEU fuel assemblies at the GTRR have 16 equally spaced, curved fuel plates with 2 aluminum side plates and 2 curved-end plates (MTR design). The uranium in the HEU fuel meat is enriched to about 93 percent uranium-235, and each plate contains approximately 11.75 grams (g) of uranium-235 before irradiation.

The LEU fuel elements are designed with the same outer dimensions as the HEU fuel elements with 18 fuel plates, 2 aluminum side plates, and 2 curved end-plates. Each of these fuel plates consists of uranium silicide dispersed in an aluminum matrix (U_3Si_2-Al) and completely clad in aluminum alloy. In the LEU fuel plates design, the nominal dimensions of the fuel meat is 0.510 millimeters (mm) thick, and the cladding is 0.380 mm thick. The fresh LEU fuel plates design contains 12.5 g of uranium-235 enriched to less than 20 percent. Use of this design increases the uranium-235 content per element from 188 g for HEU to 225 g for LEU. This design is similar to that used in previously converted research reactors. Table 1 compares these and other key characteristics for the HEU and LEU fuel assemblies.

Table 1. HEU and LEU Fuel Assembly Design Characteristics

Characteristics	HEU	LEU
Fuel Plates/Assembly, no.	16	18
Non-Fuel Plates/Assembly, no.	2	2
Fissile Loading/Plate, g U-235	11.75	12.5
Fissile Loading/Assembly, g U-235	188	225
Fuel Meat Composition	U-Al Alloy	U_3Si_2-Al
Cladding Material	1100 Al	6061 Al
Fuel Meat Dimensions, mm		
Thickness,	0.51	0.51
Width,	63.5	58.9-62.8
Length,	584-610	572-610
Cladding Thickness, mm	0.38	0.38

The fuel plate design for the GTRR low-enriched core was pre-specified (Reference 1) so that there would be uniformity in reactors undergoing the conversion from HEU to LEU. The Argonne National Laboratory (ANL) developed these fuel element plates for conversion to LEU fuel at the U.S. NPRs. Prototypes of these fuel plates were tested extensively in the Oak Ridge Research Reactor under operational and hostile environmental conditions that would be extreme for most NPRs, including the GTRR, and demonstrated acceptable fuel performance. The NRC staff approved the use of this type of fuel in NUREG-1313 (Reference 2). The characteristics of the fuel proposed for the LEU conversion at the GTRR are within the boundaries of the tested fuel evaluated in NUREG-1313. This fuel design is also similar to designs previously found acceptable and operating successfully at several converted reactors. Therefore, the proposed fuel construction and geometry are acceptable.

2.3 Core Configuration

The GTRR core grid plate has provisions for up to 19 fuel assemblies, with the number in use determined by the fissile burnup and experimental conditions and limited by Technical Specifications requirements (e.g., shutdown margin and excess reactivity as subsequently discussed). Openings in the grid plate are plugged, when not containing a fuel assembly, to ensure that coolant flow is forced through the assemblies. The HEU core currently in use and the proposed LEU core consist of 17 fuel assemblies, and allow for between 14 and 19 fuel assemblies. No changes are proposed to the core except for the direct replacement of fuel assemblies. Therefore, the core configuration proposed by the licensee is acceptable.

2.4 Fuel Storage

The licensee anticipates the possible need to safely possess but not use the HEU fuel after conversion, and has requested the necessary license conditions to permit possession of the irradiated HEU and LEU fuel. Underwater storage conditions and limitations applicable to HEU or LEU fuel are in the GTRR Technical Specifications, which require that the reactivities not exceed 0.85 for these storage facilities with optimum neutron moderation by water. Calculations that were performed against measurements at the Oak Ridge Research Reactor have shown that the reactivity of an array that is similar to the storage for the GTRR type LEU fuel would not exceed 0.72 (Reference 3). New unirradiated fuel assemblies will be stored in a secure storage vault, either in shipping containers or similar geometries. Analyses of this vault have shown that the storage continues to meet the Technical Specifications (TSs) requirements (*Keff* <0.85, TS 3.8.a). This facility has been used with HEU fuel for about 20 years, consistent with NRC requirements. There is no significant change in the fuel configuration, and the storage and control capacities have been shown to be acceptable. Further, these capabilities and designs are consistent with other non-power reactors that have successfully converted. Therefore, fuel storage and possession are acceptable for both the LEU and HEU fuels.

2.5 Basic Reactor Physics Parameters

Reactor physics parameters, such as prompt neutron lifetime and effective delayed neutron fraction (*Beff*) necessary for determining reactor kinetics and reactivity conditions of the reactor were calculated. The calculated reference core prompt neutron lifetime decreased from 780 microseconds for HEU to 745 microseconds for LEU, primarily because of the increased neutron absorption in the LEU fuel. The *Beff* for the LEU core is calculated to be 7.5×10^{-3} to 7.6×10^{-3}, compared with a measured 7.55×10^{-3} for the HEU core. The results are similar to those calculated and confirmed by measurements of reactivity conditions for other NPR conversions to LEU fuel. Therefore, these calculations are acceptable for the GTRR.

2.6 Critical and Operating Masses of Uranium-235

The licensee submitted both Monte Carlo and diffusion-type calculations that were benchmarked against initial HEU fuel loading measurements. The excess reactivities of both HEU and LEU cores with larger numbers of assemblies were also calculated using these methods (as discussed below). The GTRR startup program is to measure subcritical multiplication for the LEU core.

Calculations were performed for 14 assemblies, the minimum number allowed to be used in an operating core, and 17 assemblies, the number in the current HEU core and the number planned for the initial operating LEU core. With 17 fuel assemblies, the operating HEU core of fresh fuel contained about 3.2 kg of uranium-235, and the initial LEU core is projected to contain 3.8 kg. The calculated increase in fissile material is consistent with previous experience with reactors converted from HEU to similar LEU fuel.

Because the calculated changes are as expected, and the startup program will verify critical and operating configurations, the proposed LEU fuel loading is acceptable.

2.7 Excess Reactivity

The excess reactivity, 9.4 percent ± 0.4 percent $\Delta k/k$, has been calculated with a 17-element core of fresh LEU fuel. This value is within the TSs required limit of 11.9 percent $\Delta k/k$. The LEU core excess reactivity is slightly smaller than that for the HEU core, because of the increased non-fissile neutron absorption in the uranium-238. This is consistent with earlier calculations and measurements at converted reactors. The GTRR LEU startup program is to include experimental verification of the excess reactivity of the operating core. On this basis, the excess reactivity is acceptable for the proposed LEU conversion at the GTRR.

2.8 Control Rod Worths and Shutdown Margin

The proposed LEU core uses the same shim-safety control blades and regulating rod that have been used in the existing HEU core. Four shim-safety control blades and one regulating rod are made of aluminum-clad cadmium. A detailed Monte Carlo program was used to compare the reactivity worths in the HEU and LEU reactors. The results indicate that the reactivity worths will be slightly larger in the proposed LEU reactor, and will control reactivity at least as well as in the HEU reactor. Also, the reactivity worths are within the bounds assumed in the accident and transient analysis and, thus, will not introduce any new reactivity-related events. Detailed measurements of both differential and integral reactivity worths for the shim-safety control blades and regulating rod will be in the GTRR LEU startup program.

The shutdown margin ensures that the reactor can be shut down from any operating condition, even if the shim-safety blade of maximum worth and non-scrammable regulating rod are in their fully withdrawn position. The shutdown margin is a requirement in the TSs (GTRR TS 3.1.a). Monte Carlo calculations of the reactivities of the HEU and LEU reactors, loaded with 17 fresh fuel assemblies and with control rods as stipulated above, showed that reactivity of the HEU core would be about 4 percent $\Delta k/k$ negative, and the reactivity of the LEU core would be about 5 percent $\Delta k/k$ negative. Thus, the proposed LEU reactor can readily meet the TS shutdown margin of 1 percent $\Delta k/k$. The shutdown reactivity is to be determined during the LEU reactor startup program to confirm compliance with the shutdown margin TS.

Therefore, the proposed LEU core design is acceptable from the standpoint of reactivity control.

2.9 D_2O Reflector Worth

In addition to the neutron-absorbing control-safety rods, the GTRR has provisions for rapidly removing the top D_2O neutron reflector to a storage tank. Calculations have confirmed that the negative reactivity introduced by removing this specific section of reflector is essentially the same for the HEU and LEU reactors. This demonstrates an acceptable alternative shutdown mechanism.

2.10 Core Power Characteristics

The GTRR proposal contained analyses of the neutron flux densities and thermal power distributions in the HEU and proposed LEU reactor cores. The calculations used standard neutron diffusion and Monte Carlo programs that were previously validated on several converted NPRs. The power distributions and the peak-to-average power ratios for both 14 and 17 fresh fuel assemblies per core were calculated. The calculations showed that the HEU and LEU power densities are very similar for the same number of fuel assemblies, and the maximum peak-to-average power density ratio of the LEU core is slightly higher than the HEU core. The calculations also confirm that power densities in the 14-assembly cores are higher than in the 17-assembly cores, but that the maximum peak-to-average ratios are not significantly different. These core sizes were selected for analysis because the minimum core size is 14 assemblies, and the probable number of assemblies for routine operations will be 17. The peak power densities in fuel assemblies will tend to decrease as burnup continues with use. Neutron flux measurements are to be included in the LEU startup program to validate the core characteristics. Therefore, the core power characteristics for the proposed LEU reactor are acceptable.

2.11 Thermal-Hydraulics

The proposal for converting the GTRR to the use of LEU fuel contained the results of thermal-hydraulic calculations. Both the HEU and LEU cores were conservatively analyzed. The analyses acceptably considered the power peaking, thermal conductivities of the fuel plates, coolant hydraulic conditions, and fuel assembly and core configurations (including differences in the gap between fuel plates and in the number of assemblies in the core). Nominal and limiting steady-power operating conditions were analyzed for 1 MWt and 5 MWt operation with forced coolant flow, and for natural thermal convection coolant flow.

For the existing HEU fuel (14 assemblies) with coolant flow at the minimum TS-allowed value for operation up to 5 MWt, the calculations predict a departure from nucleate boiling (DNB) power of 11.5 MWt, and flow instability power of 10.6 MWt; for 14 assemblies of LEU fuel, the calculations predict a DNB power at 10.8 MWt and flow instability power at 10.6 MWt. The limiting LEU fuel power levels are lower than those for HEU because of larger assumed tolerances for some of the procurement specifications. The licensee presented plots of calculated thermal power level against coolant flow rate for both DNB and onset of flow instability with the coolant inlet temperature at the TS limiting safety system setting (LSSS) value of 50.6 °C (123 °F). The plots show the envelope of the safety limits for the HEU and the proposed LEU core containing 14 fuel assemblies. Additional calculations have shown that the safety limits for 17 fuel assemblies are not significantly different and are conservatively bound by the proposed plots. On this basis, the licensee has included the plot for LEU fuel in proposed revised Technical Specifications using the safety criterion of onset of flow instability (as discussed in Section 3 below).

For forced convection at power levels not greater than 1 MWt, the limiting power level calculations for both the HEU and proposed LEU cores assumed operation at the LSSS values. For the HEU core, the peak fuel-cladding temperature was calculated to be 72.2 °C (162 °F), and for the proposed LEU core, a peak cladding temperature of 73.3 °C (164 °F), both well below the D_2O saturation temperature. For both HEU and LEU cores, the power levels for initiation of flow instability were predicted to be between 6.5 and 7.0 MWt.

In the natural thermal convection cooling mode, the licensee has operated for extended periods of time with HEU at 1 kWt without the bulk coolant temperature exceeding 50.6 °C (123 °F). Because the comparisons for power densities and distributions show no significant differences between the LEU and HEU fuels, the same operating conditions for natural convection flow are shown to be applicable and acceptable for the proposed LEU fuel as for the HEU fuel. This conclusion is also supported by operation of other NPRs with similar LEU fuel in the natural convection mode of operation.

The preceding discussions and results of the thermal-hydraulic analyses show large margins between allowed routine operating conditions and conditions that could lead to fuel failure. On this basis, the analytical methods and assumptions are acceptable for the operation of the GTRR with the proposed LEU fuel in all operating modes.

2.12 Reactivity Feedback Coefficients

The proposal for conversion presented calculations of reactor core temperature and void coefficients of reactivity for the HEU and proposed LEU reactors. Calculated values were compared with previous measurements for the HEU reactor. Reactivity changes were calculated separately by means of a widely used three-dimensional diffusion theory program for changes in coolant temperature, coolant density, and fuel temperature, while conservatively assuming all heavy water outside of the fuel region to be at constant temperature. In addition, a void coefficient was calculated, assuming a 1 percent void uniformly distributed in the coolant/moderator of all fuel assemblies.

The calculations showed only small differences between 14 fuel assemblies and 17 assemblies. The sum of the individual temperature coefficients for HEU was calculated to be about -0.73×10^{-4} $\Delta k/k/{}^{\circ}C$, and for LEU it was calculated as approximately -0.85×10^{-4} $\Delta k/k/{}^{\circ}C$. Included in these values was the Doppler broadening of neutron absorption resonances in U-238 of essentially zero for HEU, and about -0.19×10^{-4} for LEU.

The average void coefficient was calculated to be approximately -3.9×10^{-4} $\Delta k/k$ percent void for HEU and about -3.4×10^{-4} $\Delta k/k$ percent for LEU.

All of these reactivity coefficients are sufficiently negative for the proposed LEU fuel to help ensure reactor stability. The LEU startup program will include experimental measurement of the sign and magnitude of these coefficients. Therefore, the reactivity feedback coefficients for the proposed LEU core are acceptable.

2.13 Fission-Product Inventory and Containment

Because the basic components and characteristics of the LEU and HEU fuel are similar, the formation and buildup of the inventory of fission products per megawatt-hour of operation of the proposed LEU core at GTRR will not differ significantly from those of the HEU core. Furthermore, because the licensee proposes no significant differences in operating schedules or utilization programs, results of previously postulated releases of fission products from a fuel assembly remain valid for the LEU reactor. However, because the proposed LEU fuel assemblies will contain 18 fuel plates instead of the 16 in the current HEU fuel, the fission-product inventory per plate will be decreased accordingly.

As indicated in Section 2.2, ANL has developed the LEU fuels and has performed extensive tests under more extreme operational conditions than the fuel could experience during normal operation of the GTRR. These tests demonstrated excellent performance for the proposed LEU fuel, comparable to that for HEU fuel. Further, use of similar fuel assemblies and plates in other NPRs has continued to demonstrate the excellent fission-product retention capability of the LEU fuel.

The fission-product inventory or containment for the LEU fuel is not significantly different from that of the HEU fuel and is, therefore, acceptable.

2.14 Potential Accident Scenarios

2.14.1 Design-Basis Accident

For NPRs, the release of radioactive material from fuel that has been postulated to have lost cladding integrity is often analyzed as a design-basis accident (DBA). By supplement to the SAR for the 5 MWt HEU GTRR, dated June 23, 1972, the self-heating, melting, and release of all contained iodine, xenon, and krypton radioisotopes from a fuel assembly to the air in the reactor containment room was postulated and analyzed as the DBA.

This analysis assumes moving a fuel element from the core to the reactor room shortly after reactor shutdown contrary to the licensee's procedures and Technical Specifications. Furthermore, the GTRR is a closed-tank reactor, so the physical effort to remove a fuel assembly from the core is considerable such that the assumed movement of a fuel element a short-time after shutdown is a conservative assumed accident scenario.

The licensee compared the effects of decay heat in HEU and LEU fuel assemblies, discussing the differences in thermodynamic characteristics. If the hottest HEU fuel assembly were removed from its coolant 8 hours after reactor shutdown, its highest temperature after about 45 minutes is predicted to be 425 °C [797 °F]. For an LEU assembly, with similar assumptions, its maximum temperature is predicted to be 400 °C [752 °F] after 50 minutes. The principal change in the assemblies causing these differences is the increase from 16 to 18 plates in the LEU fuel. These results indicate that the LEU fuel would have more margin to melting failure under the assumed scenario than the currently used HEU, which has been found acceptable. The licensee has proposed no changes in the current procedures that limit the delay times between reactor shutdown and removal of fuel from the reactor coolant.

These facts notwithstanding, the licensee compared the potential consequences of the DBA scenario for the LEU fuel with those for the HEU. As noted in Section 2.13 above, the inventory of fission products in an LEU assembly is predicted not to be significantly different from an HEU assembly with the same fissile burnup. For the proposed LEU fuel, no significant difference is expected in predicted doses from that for HEU and, therefore, the analysis is acceptable.

2.14.2 Reactor Startup Accident

The licensee assumed the potential reactor startup accident with the most limiting consequences (i.e., the simultaneous withdrawal of a shim-safety blade and the regulating rod). This could occur because the regulating rod is not interlocked to prevent simultaneous withdrawal. However, the scenario assumes an operator error to withdraw both rods contrary to established procedures. The analysis considered this event for both the HEU and proposed LEU reactors, assuming that all other systems functioned normally. If reactivity were inserted continuously at the maximum rate for the two-rod motion starting at a power level of 5 kWt, both the HEU and LEU reactors are conservatively predicted to be scrammed by one of the two reactor period trips by about 6.5 kWt. Therefore, there would be no significant difference between the proposed LEU reactor and the existing acceptable HEU reactor in the event of a startup malfunction.

The licensee's safety analysis also considered a similar operator error while the reactor was operating at its rated 5 MWt. In these cases for both HEU and LEU, the reactor would be scrammed by either the period trip or the 5.5 MWt power level trip. For both HEU and LEU in the GTRR, the peak power predicted was 5.9 MWt, with peak surface cladding temperatures of 81 °C [178 °F] and 78 °C [174 °F], respectively. For this postulated event, the safety margins for the LEU are shown to be at least as large as for the existing acceptable HEU, and are, therefore, acceptable.

2.14.3 Reactivity Change From Fuel Plate Melting

The licensee analyzed an event for reactivity change of the 5 MWt HEU reactor in which a single fuel plate was assumed to melt. The licensee found that removal of a single plate caused a negative reactivity change in HEU fuel. The licensee's calculations involving diffusion theory confirmed the previous results for HEU fuel, and showed that the removal of a single plate from an LEU assembly would also produce a negative reactivity, though not quite as large. Therefore, the assumed melting of a fuel plate in the proposed LEU reactor would lead to a decrease in reactor power similar to that in the existing acceptable HEU reactor and the result is, therefore, acceptable.

2.14.4 Rapid Positive Reactivity Insertion

The SAR for the 5 MWt GTRR had analyzed the change in reactivity if an experiment of the maximum worth allowed by the Technical Specifications were to fail. The amount allowed and analyzed is 1.5 percent $\Delta k/k$.

The postulated scenario assumes that the GTRR is operating normally at the licensed 5 MWt and that an experiment fails or is ejected, thereby inserting a positive reactivity. Reactor power immediately starts to increase, but an operable power level trip scrams the reactor at 5.5 MWt. Analyses were performed for two amounts of reactivity: 0.4 percent $\Delta k/k$ and 1.5 percent $\Delta k/k$. For the 0.4 percent $\Delta k/k$ insertion, the peak surface cladding temperature was calculated as of about 85 °C [185 °F] for both HEU and LEU fuel. For 1.5 percent $\Delta k/k$, the peak surface cladding temperature was calculated as about 131 °C [268 °F] for LEU and 136 °C [287 °F] for the existing acceptable HEU core.

In order to gain additional confidence that the accident analysis methodology used by the licensee remains acceptable, the Idaho National Engineering Laboratory (INEL) performed an independent calculation of the most limiting reactivity transient. The scenario assumed the addition of 1.5 percent $\Delta k/k$, the reactivity of the most limiting experiment allowed by the Technical Specifications. INEL analyzed this transient using the RELAP 5/MOD 3 code. The results of this separate and independent analysis were within approximately 2.7 percent of the results obtained by the licensee on peak cladding temperature. This small difference in fuel cladding temperature is not significant and confirms the licensee's analysis methods. Further, since the maximum cladding temperature calculated by both the licensee and INEL is well below the solidus temperature of the cladding, the results are acceptable.

On this basis, this accident analysis and results are acceptable.

2.14.5 Fuel Loading Accident

In the original SAR (December 1967), as supplemented, the licensee discussed an accident scenario in which it was postulated that a fresh HEU fuel assembly was dropped into the center core position of the GTRR while it was just critical. This would result in a rapid insertion of 2.5 percent $\Delta k/k$ positive reactivity, and a reactor excursion with a short period. Comparing SPERT tests with a similar D_2O-moderated HEU reactor indicates that fuel temperature in GTRR would not reach the melting point. The proposed LEU fuel has more fuel plates per assembly and a larger prompt negative reactivity feedback coefficient (from the Doppler effect) in uranium-238 neutron absorption resonances. Therefore, the maximum temperature in LEU fuel would be expected to be lower than in the existing acceptable HEU fuel for the same accident scenario, and, therefore, is acceptable.

2.14.6 Summary of Accident Analyses Evaluation

For all of the relevant postulated accidents, safety margins for the proposed LEU fuel are at least as large as for the existing HEU fuel. Converting the GTRR to the proposed LEU fuel would not introduce any new safety considerations and would not increase any radiological risks to the public.

2.15 Reactor Startup Testing

The licensee plans to make sub-critical measurements for the LEU fuel loading. The startup testing program also includes control rod and power calibrations, and temperature coefficient, flux distribution, shutdown margin, excess reactivity, and void coefficient measurements. The licensee is to submit a startup report to the NRC on the results of this startup testing. This startup testing will provide verification of key LEU reactor functions, and, therefore, is acceptable.

3 CHANGES TO TECHNICAL SPECIFICATIONS

In its proposal for converting the GTRR from the use of HEU fuel to LEU fuel, the licensee requested changes to the facility Technical Specifications; these are discussed below. Changes to the TSs in connection with conversion were limited to those necessary to accommodate the differences in fuel, reactor operating characteristics, and established procedures. These changes are based on analyses and representations given in the January 21, 1993, GTRR proposal and supplements, are consistent with the proposed conversion, and as discussed below are acceptable.

3.1 Technical Specification 2.1.1: Safety Limits in the Forced Convection Mode

The licensee proposed replacing Figure II-1 (old) with Figure II-1 (new) for Technical Specification 2.1.1a, because of different operating parameters of the LEU fuel. As noted in Section 2.11, these figures present calculated values of reactor thermal power level as a function of total coolant flow rate, with a fixed coolant inlet temperature of 50.6 °C [123 °F]. Combinations of parameters to the right and below the plotted lines correspond to fuel temperatures low enough that integrity is ensured. That is, the plotted lines correspond to the envelope of reactor safety limits. Figures II-1 (old) and II-1 (new) have the same shapes and are based on setting conservative limits to avoid fuel damage. Also, the bases for this Technical

Specification were changed to reflect the proposed LEU core. Therefore, the proposed plot for LEU on Figure II-1 as the envelope of safety limits and associated bases for the converted GTRR reactor are acceptable.

3.2 Technical Specification 5.2: Fuel Elements

This Technical Specification is changed as follows to describe the LEU and HEU fuels:

The LEU fuel elements shall be of the MTR type consisting of 18 fuel plates of uranium silicide with an enrichment less than 20 percent. Each LEU fuel plate will have a nominal loading of 12.5 g of U-235. The HEU fuel elements shall also be of the MTR type consisting of 16 fuel plates of uranium aluminide with an enrichment of 93 percent. Each HEU fuel plate will have nominal loading of 11.75 g of U-235.

4 CONCLUSION

The operational and safety factors that could be impacted by the use of the NRC-accepted LEU fuel in place of HEU fuel in the GTRR have been reviewed and evaluated. The conversion, as proposed, would not reduce any safety margins, would not introduce any new safety issues, and would not lead to increased radiological risk to the public or the environment. Therefore, the conversion of the GTRR to the use of LEU fuel is acceptable.

REFERENCES

1. H. H. Young et al., "Conversion and Standardization of University Reactor Fuels Using Low Enrichment Uranium-Plans and Schedules," 1986 International Meeting on Reduced Enrichment of Research and Test Reactors, ANL,RERTR/TM-9, Gatlinburg, TN, November 3, 1986.

2. U.S. Nuclear Regulatory Commission, "SER Related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Nonpower Reactors," NUREG-1313, July 1988.

3. R. B. Pond and J. E. Matos, "Nuclear Criticality Assessment of LEU and HEU Fuel Element Storage," Proceedings of the 1983 International Meeting on Reduced Enrichment for Research and Test Reactors," Japan Atomic Energy Research Institute, Tokai, Japan, October 24-27, 1983, JAERI-M 84-073, pp. 416-425 (May 1984).

Date: June 16, 1995