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

September 12, 1988

Docket No. 50-134

Mr. Thomas H. Newton, Jr., Director
Nuclear Reactor Facility
Worcester Polytechnic Institute
Worcester, Massachusetts 01609

Dear Mr. Newton:

SUBJECT: ISSUANCE OF ORDER MODIFYING LICENSE NO. R-61 TO CONVERT FROM HIGH TO LOW-ENRICHED URANIUM--AMENDMENT NO. 10 - WORCESTER POLYTECHNIC INSTITUTE

The Commission has approved an order modifying Facility Operating License No. R-61--Amendment No. 10--for the Worcester Polytechnic Institute Training and Reactor facility. The Order modifies the license in accordance with the Commission's regulations, 10 CFR 50.64, which require that a non-power reactor such as yours convert to low-enriched uranium (LEU) fuel under certain conditions. The Order is being issued in accordance with 10 CFR 50.64(c)(3) and in response to your submittals dated September 17, 1987 and February 10, 1988. While these submittals contained the necessary information for us to approve the conversion to LEU, we found that some information should be corrected and clarified for the record. Please respond to the statements in Enclosure 4 in this regard within 30 days after receipt of this letter or as indicated by the statement.

The Order will become effective on the later date of either receipt of LEU fuel elements or 30 days following the date of publication of this Order in the Federal Register, provided there are no requests for a hearing. If a hearing is requested the Order will become effective on the date specified in an order following further proceedings on this Order. Please inform me when you receive the LEU fuel elements and when the HEU will be completely removed from the facility.

DF01
1/1

Enclosed with the Order, which is being sent to the Federal Register for publication, is a copy of the Safety Evaluation and replacement pages for the Technical Specifications.

Sincerely,

Theodore S. Michaels

Theodore S. Michaels, Project Manager
Standardization and Non-Power
Reactor Project Directorate
Division of Reactor Projects - III, IV,
V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Order on Amendment No. 10
2. Replacement pages for Technical Specifications
3. Safety Evaluation
4. Information to be supplied by WPI

cc: See next page

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Theodore S. Michaels, Project Manager
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Enclosed with the Order, which is being sent to the Federal Register for publication, is a copy of the Safety Evaluation and replacement pages for the Technical Specifications.

Sincerely,

original signed by
Theodore S. Michaels, Project Manager
Standardization and Non-Power
Reactor Project Directorate
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*See Previous Concurrence

PDSNP*	PDSNP*	AD:PDSNP*	OGC-Rockville*
EHylton	TMichaels:cw	CMiller	EChan
07/20/88	07/20/88	07/26/88	08/15/88

DF01
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cc: Francis J. McGrath
City Manager
Worcester, Massachusetts 01608

Office of the Attorney General
Environmental Protection Division
19th Floor
One Ashburton Place
Boston, Massachusetts 02180

Department of Environmental
Quality Engineering
100 Cambridge Street
Boston, Massachusetts 02180

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of WORCESTER POLYTECHNIC INSTITUTE Worcester, Massachusetts 01609	Docket No. 50-134 Facility Operating License No. R-61 Amendment No. 10

ORDER MODIFYING LICENSE

I

Worcester Polytechnic Institute (licensee or WPI) is the holder of Facility Operating License No. R-61 (License) issued on December 16, 1959 and subsequently renewed on December 30, 1982 by the U.S. Nuclear Regulatory Commission (Commission). The license originally authorized operation of the Worcester Polytechnic Institute Training and Research Reactor (facility) at a power level of up to 1 kilowatt (thermal). In 1967, the license was amended to allow operation up to 10 kilowatts (thermal) at which level it is now limited. The facility is a training and research reactor located in Worcester, Massachusetts, approximately 45 miles west-southwest of Boston, Massachusetts, and is located in the Washburg Laboratory on the east side of the WPI campus. The mailing address is Worcester Polytechnic Institute, Nuclear Reactor Facility, Worcester, Massachusetts 01609.

II

On February 25, 1986, the Commission promulgated a final rule in 10 CFR 50.64 of its regulations 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 a licensee of an existing non-power reactor replace HEU fuel at its facility with low-enriched uranium (LEU) fuel acceptable to the Commission: (1) unless the Commission has determined that the reactor has a unique purpose and (2) contingent upon Federal Government funding for conversion-related costs. The rule is intended to promote the common defense and security by reducing the risk of theft and diversion of HEU fuel used in non-power reactors and the adverse consequences to public health and safety and the environment from such theft or diversion.

10 CFR 50.64(b)(2)(i) and (ii) require that a licensee of a non-power reactor: (1) not initiate acquisition of additional HEU fuel, if LEU fuel acceptable to the Commission for that reactor is available when it 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).

10 CFR 50.64(c)(2)(i) of the rule, among other things, requires each licensee of a non-power reactor, authorized to possess and to use HEU fuel, to develop and to 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 (proposal) for meeting the rule's requirements.

10 CFR 50.64(c)(2)(i) also requires the licensee to include in its proposal: (1) a certification that Federal Government funding for conversion is available through the Department of Energy (DOE) or other appropriate Federal agency, and (2) a schedule for conversion, based upon availability of 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 the available financial support, and reactor usage.

10 CFR 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, to the facility, and to the licensee's procedures (all three types of changes hereafter called modifications). This paragraph also requires the licensee to provide supporting safety analyses so as to meet the schedule established for conversion.

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

10 CFR 50.64(c)(3) requires the Director to review the licensee's supporting safety analyses and to issue an appropriate enforcement order directing both the conversion and, to the extent consistent with protecting the public health and safety, any necessary modifications. The Commission explained in the statement of considerations of the final rule that in most cases, if not all, the enforcement order would be in the form of an order to modify the license under 10 CFR 2.204 (see 51 FR 6514).

10 CFR 2.204 provides, among other things, that the Commission may modify a license by issuing an amendment on notice to the licensee that it may demand a hearing with respect to any part or all of the amendment within 20 days from the date of the notice or such longer period as the notice may provide. The amendment will become effective on the expiration of this 20-day-or-longer period. If the licensee requests a hearing during this period, the amendment will become effective on the date specified in an order made after the hearing.

10 CFR 2.714 sets out 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 September 17, 1987, the Director received the licensee's proposal, including its proposed modifications, supporting safety analyses and schedule for conversion. The conversion consists of replacement of high-enriched with low-enriched uranium fuel elements. The fuel elements contain MTR-type fuel plates with the fuel meat in the form of uranium aluminides dispersed in an aluminum matrix. The enrichment is less than 20% in the U-235 isotope. The Licensing Conditions and Technical Specification changes needed to amend the facility license are included in the attachment to this Order. On the bases of the licensee's submittals and the requirements of 10 CFR 50.64, I have made a determination that the public health and safety and the common defense

and security require the licensee to convert from the use of HEU to LEU fuel pursuant to the modifications set forth in the attachment in accordance with the schedule set out below.

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 the Commission's regulations in 10 CFR 2.204 and 50.64, IT IS HEREBY ORDERED THAT:

On the later date of either receipt of low-enriched uranium fuel elements by the licensee or 30 days following the date of publication of this Order in the Federal Register, Facility Operating License No. R-61 is modified by amending the License Conditions and Technical Specifications as stated in the Attachment to this Order.

V

Pursuant to the Atomic Energy Act of 1954, as amended, the licensee or any other person adversely affected by this Order may request a hearing within 30 days of the date of this Order. Any request for a hearing shall be submitted to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, with a copy to the Assistant General Counsel for Enforcement at the same address. If a person other than the licensee requests a hearing, that person shall set forth with particularity in accordance with 10 CFR 2.714 the manner in which the person's interest is adversely affected by this Order.

If a hearing is requested by the licensee or 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 hearings is whether this Order should be sustained.

This Order shall become effective on the later date of either the receipt of low-enriched uranium fuel elements by the licensee or 30 days following the date of publication of this Order in the Federal Register or, if a hearing is requested, on the date specified in an order following further proceedings on this Order.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas E. Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland
this 12th day of September 1988.

Enclosure:
As stated

ATTACHMENT TO ORDER

MODIFYING FACILITY OPERATING LICENSE NO. R-61

A. License Conditions Revised And Added By This Order

No. 2.B.(2) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess and use up to a maximum of 5.2 kilograms of contained uranium 235 at enrichments equal to or less than 20% and 16 grams of plutonium as Pu-Be source for use in connection with operation of the reactor.

No. 2.B.(4) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to possess, but not to use, a maximum of 4004 grams of contained uranium at greater than 20% enrichment until the existing inventory of high enriched uranium is removed from the facility.

No. 2.C.(2) The Technical Specifications contained in Appendix A, as revised through Amendment 10, are hereby incorporated in the license. The licensee shall operate the reactor in accordance with these Technical Specifications.

B. Technical Specifications Revised By This Order

2.0 SAFETY LIMITS AND OPERATING RESTRICTIONS

2.1 Safety Limits

Shutdown Margin: The minimum shutdown margin under any conditions with the highest worth control blade fully withdrawn shall be no less than 1% Δ k/k.

Control Blade Withdrawal: The maximum withdrawal rate for a control blade shall be 7.5 in./min. The maximum reactivity addition rate through movement of the regulating blade shall be 0.006% Δ k/k sec. Interlocks shall prevent simultaneous withdrawal of more than one control blade and shall prevent withdrawal of any control blade unless the regulating blade is fully inserted.

Temperature and Void Coefficients: The temperature and void coefficients of reactivity shall be more negative than $-2 \times 10^{-5} \Delta$ k/k °F and -2×10^{-3} Δ k/k % void, respectively, at 80°F and shall not be positive at any average core temperature above 80°F.

3.0 SURVEILLANCE REQUIREMENTS

3.1 Frequency Of Surveillance

Semiannually:

(4) The minimum shutdown margin with the highest worth control blade fully withdrawn shall be verified to be no less than $1\%\Delta k/k$.

(5) The reactivity worth of the regulating blade shall be measured.

Annually: At least once each year, all fuel elements shall be removed from the core to the storage racks. While the fuel elements are thus stored, the control blades shall be brought to the surface and visually inspected and the blade drives lubricated. Blade drop times and magnet release times shall be measured for each control blade, and a plot of blade drop times versus distance shall be obtained for each safety blade and compared with data of previous years. Abnormal deviation from previous data will be investigated and reviewed by the Radiation, Health, and Safeguards Committee.

4.0 SITE AND DESIGN FEATURES

4.4 Reactor Core

Fuel Elements: Standard fuel elements shall be flat plate type consisting of uranium-aluminum alloy clad with aluminum. The width and depth of each fuel element shall be 3 in. x 3 in. Each element shall have an active length of 24 in. There shall be a maximum of 10 g of U-235 in each fuel plate and not more than 170 g of U-235 in any fuel element. The fuel shall be enriched to less than 20% U-235. Standard fuel elements have 18 fuel plates, each plate 1.52 mm thick with a clad thickness of 0.381 mm on each side. A maximum of 28 standard fuel elements may be installed in the core.

Not more than two experimental fuel elements with sixteen removable fuel plates similar to standard fuel plates and fitted with removable top end boxes may be installed in the core. These elements may be used as part of the core assembly either as complete elements or as partial assemblies, loaded with from 2 to a total of 18 fuel plates each.

4.5.2 Control Blades

There shall be three control blades intersecting the core, each consisting of vertical blades 10.5 in. wide x 40.5 in. long with a poison section composed of boron carbide and aluminum 0.375 in. thick sandwiched between aluminum side plates.

4.5.3 Regulating Blade

There shall be one regulating blade consisting of a vertical stainless-steel blade 10.65 in. wide x 40.5 in. long x 0.125 in. thick. It shall have a reactivity worth of less than $0.7\%\Delta k/k$.

5.7 Reports

In addition to reports otherwise required under this license and applicable regulations

(1) The licensee shall inform the Commission of any incident or condition relating to the operation of the reactor that prevented or could have prevented a nuclear system from performing its safety function as described in the Technical Specifications. For each such occurrence, WPI shall promptly notify, by telephone or telegraph, the Administrator of the appropriate NRC Regional Office listed in Appendix D of 10 CFR 20 and shall submit within 10 days a report in writing to the Director, Division of Reactor Projects-III/IV/V & Special Projects (DRSP), with a copy to the Regional Office.

(2) The licensee shall report to the Director, DRSP, in writing within 30 days, any observed occurrence of substantial variance disclosed by operation of the reactor from performance specifications contained in the Safety Analysis Report or the Technical Specifications.

(3) The licensee report to the Director, DRSP, in writing within 30 days, any occurrence of significant changes in transient or accident analysis as described in the SAR.

5.8 Annual Operating Reports

A report covering the previous year shall be submitted to the Administrator of the appropriate Regional Office not later than March 31 of each year. It shall include

(1) Operations Summary: a summary of operating experience having safety significance occurring during the report period, including

(e) a brief summary of those changes, tests, and experiments that did not require authorization from the Commission pursuant to 10 CFR 50.59(a).

ENCLOSURE TO LICENSE AMENDMENT NO. 10

FACILITY OPERATING LICENSE NO. R-61

DOCKET NO. 50-134

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change.

Remove Page	Insert Page
3	3
6	6
8	8
9	9
12	12
13	13

2.0 SAFETY LIMITS AND OPERATING RESTRICTIONS

2.1 Safety Limits

Criticality: The reactor shall be subcritical when the three control blades are at their fully withdrawn positions and the regulating blade is in its fully inserted position.

Shutdown Margin: The minimum shutdown margin under any condition with the highest worth control blade fully withdrawn shall be no less than 1%Δk/k.

Magnet Release and Blade Drop Times: The interval between the occurrence of cutoff voltage (scram) and the separation of each control blade from its magnet shall not exceed 100 msec. Total time of insertion of the first 24 in. of the control blades following initiation of a scram signal shall be less than 600 msec, including the magnet release time.

Maximum Excess Reactivty: The maximum excess reactivity above cold, clean, critical shall be 0.5%Δk/k.

Radiation Alarms: Upon indication of radiation levels in excess of 50 mrems/hr (20 mrems/hr for fuel storage) area monitors shall actuate audible evacuation alarms in the reactor room and in the second and third floor areas above the reactor pool.

Radiation Levels: The maximum radiation levels 1 m above the pool surface and at the surface of the concrete shield, when the beam port and thermal column are closed, shall be less than 50 mrems/hr.

Control Blade Withdrawal: The maximum withdrawal rate for a control blade shall be 7.5 in./min. The maximum reactivity addition rate through movement of the regulating blade shall be 0.006%Δk/k·sec. Interlocks shall prevent simultaneous withdrawal of more than one control blade and shall prevent withdrawal of any control blade unless the regulating blade is fully inserted.

Startup Source Requirement: During reactor startup, a neutron source producing at least 10^6 neutrons/sec shall be located adjacent to the fuel region. When readings on the log count rate meter are below 50 counts/sec, an interlock shall prevent withdrawal of any control blade.

Temperature and Void Coefficients: The temperature and void coefficients of reactivity shall be more negative than -2×10^{-5} Δ k/k·°F and -2×10^{-3} Δ k/k·% void, respectively, at 80°F, and shall not be positive at any average core water temperature above 80°F.

Water Level: The minimum depth of water above the top of the end box of the core fuel elements in the reactor pool shall be 10 ft.

Water Purity: Corrective action shall be taken promptly if the following limits for the pool water are not met:

(1) pH less than 8.0 and greater than 6.0
(2) resistivity greater than 5×10^{-5} ohm-cm
(3) pool water activity less than 10^{-5} uCi/ml

3.0 SURVEILLANCE REQUIREMENTS

3.1 Frequency of Surveillance

Daily: Before each day's critical operation (with the exception of those experiments that require the reactor to be operated continuously for more than one full day, the two safety channels, the log-N period channel, and the console annunciator system shall be checked and ensured to be operational.

Quarterly: The area radiation monitoring systems and the pool water level switch shall be checked and ensured to be operational quarterly.

Semiannually: At least semiannually, a reactor inspection shall be performed consisting of

(1) The excess reactivity of the core above cold, clean, critical shall be measured.
(2) The console instrumentation shall be calibrated by a foil activation measurement of reactor power where applicable, or calibrated by other means, and checked for proper conditions.
(3) Pool water pH shall be measured and conductivity and pH devices shall be calibrated.
(4) The minimum shutdown margin with the highest worth control blade fully withdrawn shall be verified to be no less than 1% $\Delta k/k$.
(5) The reactivity worth of the regulating blade shall be measured.

Annually: At least once each year, all fuel elements shall be removed from the core to the storage racks. While the fuel elements are thus stored, the control blades shall be brought to the surface and visually inspected and the blade drives lubricated. Blade drop times and magnet release times shall be measured for each control blade, and a plot of blade drop times versus distance shall be obtained for each safety blade and compared with data of previous years. Abnormal deviation from previous data will be investigated and reviewed by the Radiation, Health, and Safeguards Committee.

3.2 Action to be Taken

If maintenance or recalibration is required for any of the items, it shall be performed and the instrument shall be rechecked before reactor startup proceeds.

3.3 Radiation Detection

Area Monitors: Area radiation sensors capable of detecting gamma radiation in the range of 0.1 to 100 mrems/hr shall be installed near the beam port, demineralizer, thermal column door, fuel storage area, and less than 1 m above the core pool surface. Upon indication of radiation levels in excess of 50 mrems/hr (20 mrems/hr for fuel storage) these monitors shall actuate audible alarms in the reactor room and in the second and third floor areas above the reactor pool.

4.0 SITE AND DESIGN FEATURES

4.1 Site

The reactor and associated equipment is housed in the Washburn Laboratories located between West Street and Boynton Street on the campus of Worcester Polytechnic Institute in Worcester, Massachusetts.

4.2 Restricted Area and Exclusion Area

The reactor room shall constitute a restricted area as defined in 10 CFR 20 and shall be controlled by partitions and normally locked doors. In addition, two small areas, one each on the second and third floors of Washburn Laboratories, directly above the reactor control drives, shall become restricted areas whenever the radiation levels in any of the rooms exceed those specified in 10 CFR 20.105. The exclusion areas, as defined in 10 CFR 100, shall consist of the reactor room and the areas above the reactors.

4.3 Reactor Building and Ventilation System

The reactor shall be housed in a closed room that is designed to restrict leakage. The ventilation system shall provide at least two changes of air per hour in the reactor room whenever the reactor is operating.

4.4 Reactor Core

Fuel Elements: Standard fuel elements shall be flat plate type consisting of uranium-aluminum alloy clad with aluminum. The width and depth of each fuel element shall be 3 in. x 3 in. Each element shall have an active length of 24 in. There shall be a maximum of 10 g of U-235 in each fuel plate and not more than 170 g of U-235 in any fuel element. The fuel shall be enriched to less than 20% U-235. Standard fuel elements have 18 fuel plates, each plate 1.52 mm thick with a clad thickness of 0.381 mm on each side. A maximum of 28 standard fuel elements may be installed in the core.

Not more than two experimental fuel elements with sixteen removable fuel plates similar to standard fuel plates and fitted with removable top end boxes may be installed in the core. These elements may be used as part of the core assembly either as complete elements or as partial assemblies, loaded with from 2 to a total of 18 fuel plates each.

4.5 Reactor Safety and Control Systems

The safety system shall be designed so that no single electrical fault that partially or completely disables the automatic scram function can, in any manner, impair or disable the manual scram function, and vice versa. The safety system shall be fail safe with respect to loss of voltage.

4.5.1 Nuclear Instrumentation

The channels of nuclear instrumentation (listed below with their minimum operating ranges) shall during all reactor critical operations be operational and shall be connected to the safety system, except as noted in Table 4.1.

(1) startup channel; background to 10^{-2}% full power, i.e., background to 1 W

(2) log-N period channel; 2×10^{-3}% to 150% full power; i.e., 0.2 W to 15 kW

(3) linear safety channels 1 and 2, 2×10^{-3}% to 150 full power; i.e., 0.2 W to 15 kW

4.5.2 Control Blades

There shall be three control blades, intersecting the core, each consisting of vertical blades 10.5 in. wide x 40.5 in. long with a poison section composed of boron carbide and aluminum 0.375 in. thick sandwiched between aluminum side plates.

4.5.3 Regulating Blade

There shall be one regulating blade consisting of a vertical stainless-steel blade 10.65 in. wide x 40.5 in. long x 0.125 in. thick. It shall have a reactivity worth of less than 0.7% $\Delta k/k$.

4.5.4 Blade Position Indicators

The blade position indicator on the console shall provide an indication of the blade position to within ±.02 in. Signal lights shall be provided for each control blade drive and for the regulating blade to indicate the upper and lower limits of travel and, in the case of control blades, an armature engaged by a magnet.

Temporary procedures that do not change the intent of the initial approval procedures may be authorized by two members of the facility staff at least one of whom shall be a licensed senior operator. Such procedures shall be subsequently reviewed by the Radiation, Health, and Safeguards Committee.

5.6 Operating Records

In addition to records required elsewhere in the license application, the following records shall be kept of

(1) reactor operation, including power levels and periods of operation at each power level

(2) maximum radioactivity released or discharged into the air or water beyond the effective control of the licensee as measured at or before the point of such release or discharge

(3) emergency shutdowns and inadvertent scrams, including reasons for emergency shutdowns

(4) maintenance operations involving substitution or replacement of reactor equipment or components

(5) experiments installed including description, reactivity worths, locations, exposure time, total irradiation and any unusual events involved in their performance and in their handling

(6) tests and measurements performed pursuant to the Technical Specifications

(7) incore irradiations

5.7 Reports

In addition to reports otherwise required under this license and applicable regulations

(1) The licensee shall inform the Commission of any incident or condition relating to the operation of the reactor that prevented or could have prevented a nuclear system from performing its safety function as described in the Technical Specifications. For each such occurrence, WPI shall promptly notify, by telephone or telegraph, the Administrator or the appropriate NRC Regional Office listed in Appendix D of 10 CFR 20 and shall submit within 10 days a report in writing to the Director, Division of Reactor Projects-III/IV/V & Special Projects (DRSP), with a copy to the Regional Office.

(2) The licensee shall report to the Director, DRSP, in writing within 30 days, any observed occurrence of substantial variance disclosed by operation of the reactor from performance specifications contained in the Safety Analysis Report or the Technical Specifications.

(3) The licensee report to the Director, DRSP, in writing within 30 days, any occurrence of significant changes in transient or accident analysis as described in the SAR.

5.8 Annual Operating Reports

A report covering the previous year shall be submitted to the Administrator of the appropriate Regional Office not later than March 31 of each year. It shall include

(1) Operations Summary: a summary of operating experience having safety significance occurring during the reporting period, including

- (a) changes in facility design
- (b) performance characteristics (e.g., equipment and fuel performance)
- (c) changes in operating procedures that relate to the safety of facility operations
- (d) any abnormal results of surveillance tests and inspections required by these Technical Specifications
- (e) a brief summary of those changes, tests, and experiments that did not require authorization from the Commission pursuant to 10 CFR 50.59(a)
- (f) changes in the plant operating staff serving in the positions of Reactor Facility Director, Health Physicist, or Radiation, Health, and Safety Committee members

(2) Power Generation: the most current summary of thermal output of the facility available together with a summary of the total thermal power generated over the life of the reactor

(3) Shutdowns: a listing of unscheduled shutdowns which have occurred during the reporting period, tabulated according to cause, and a brief discussion of the actions taken to prevent recurrence

(4) Maintenance: a discussion of corrective maintenance (excluding preventative maintenance) performed during the reporting period on safety related systems and components

(5) Changes, Tests, and Experiments: a brief description and a summary of the safety evaluation for those changes, tests, and experiments that were carried out without prior Commission approval, pursuant to the requirements of 10 CFR 50.59(a)

(6) Radioactive Effluents Releases: a statement of the quantities of radioactive effluents released from the plant

5.9 Fuel Storage

Two fuel storage racks are located on opposite sides of the reactor pool. Each rack shall be designed to contain not more than 18 fuel elements. When the reactor contains a critical mass, all additional fuel elements not in the core shall be locked in place except as authorized by the licensed senior operator in charge.

A fuel element shall not be stored outside of the reactor pool unless it produces radiation dose levels of less than 100 mrems/hr at the storage container surface. Storage containers of fuel elements shall be locked closed when unattended.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING CONVERSION ORDER TO CONVERT FROM

HIGH ENRICHED TO LOW ENRICHED URANIUM FUEL

FACILITY OPERATING LICENSE NO. R-61

WORCESTER POLYTECHNIC INSTITUTE

DOCKET NO. 50-134

1.0 INTRODUCTION

In accordance with 10 CFR 50.64, which requires that non-power reactors convert to a low-enriched uranium (LEU) fuel, except under certain conditions, the Worcester Polytechnic Institute (WPI) has proposed to convert the fuel in its pool-type training and research reactor (the reactor) from high-enriched uranium (HEU) to LEU. WPI submitted a Safety Analysis Report (SAR) and revised Technical Specifications (TS) dealing with the changes needed to convert to LEU fuel on September 17, 1987. The staff's safety review with respect to issuing an order for conversion from HEU to LEU fuel has been based on an analysis of WPI's SAR and the proposed TS as well as information provided by WPI on February 10, 1988 in response to staff questions. This material is available for review at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. This Safety Evaluation (SE) was prepared by T. S. Michaels, Project Manager, Division of Reactor Projects III, IV, V, and Special Projects, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. Major contributors to the technical review include R. E. Carter, W. R. Carpenter, and C. H. Cooper of EG&G, Idaho National Engineering Laboratory (INEL).

In addition to the changes associated with the HEU to LEU conversion some minor TS changes, which do not pertain to the conversion, were made to update and clarify the intent of the TS.

2.0 EVALUATION

General

The WPI reactor currently is licensed for operation at power levels not to exceed 10 kW thermal, using MTR-type flat-plate fuel, and cooled by natural thermal convection of the pool water. The Licensee has proposed no changes to any reactor systems or operating characteristics except for the replacement of the HEU fuel elements by new LEU fuel elements. The following evaluations and conclusions are based on that assumption.

Fuel Construction and Geometry

The HEU fuel elements currently installed at WPI contain 10 plates each, in which the fuel meat is a 93.4% enriched uranium aluminum alloy. Each fueled plate contains approximately 13.6 grams of U-235 for a total U-235 loading of about 136 grams per fuel element. The new LEU fuel elements will have the same outer dimensions as the HEU fuel elements, but will contain 18 plates each, with the fuel meat in the form of uranium aluminide (enriched to less than 20%) dispersed in an aluminum matrix. The LEU fueled plates will each contain approximately 9.3 grams of U-235 for a total U-235 loading of about 167 grams of U-235 per fuel element. The standardized LEU plates are thinner than the HEU plates, with thinner aluminum cladding, so, even though there are more LEU plates per fuel element, the metal-to-water ratio for the LEU and HEU fuel elements is very nearly equal, with the LEU fuel element being slightly higher, i.e., slightly less water volume in the new LEU fuel elements. Fuel elements with plates and uranium composition essentially identical with the proposed WPI plates were reviewed by NRC and licensed in 1981 for use in the 2MW University of Michigan reactor[1]. These fuel elements have operated successfully at this higher power level with no unpredicted events having a safety significance.

The licensee also proposes to acquire and use a fuel element with removable fueled plates. The use of this element in positions at corners of the core, B-3 or F-3, have been considered and such use is deemed acceptable.

Critical Operating Masses of U-235

The critical mass of the WPI HEU reactor is approximately 3.3 kg U-235. The critical mass of the LEU reactor is predicted to be approximately 4.0 kg U-235. This increase is due to the large increase in concentration of U-238, which absorbs low energy (thermal) and epithermal (resonance) energy neutrons, and causes a hardening of the thermal neutron spectrum. The increase of uranium is achieved by increasing its concentration in the fuel matrix and increasing the number of fuel plates in the fuel elements. The proposed concentration is similar to that successfully achieved and tested in the licensed University of Michigan reactor.

In an Argonne National Laboratory (ANL) report[2] the sensitivity calculations of reactivity for different core configurations is discussed. In this study, fuel elements were shifted and/or removed to form various core configurations in the WPI HEU and LEU cores. These fuel element perturbations were identical between the two cores and the resultant reactivity changes were compared. In all cases the reactivity effects of core rearrangements between the HEU and LEU cores is virtually identical except for the removal of the center fuel element where the reactivity effect in the HEU core is slightly higher. The results of this study, again, demonstrate the neutronic similarity between the HEU and LEU cores at WPI.

Hydraulics and Thermal-hydraulics

At WPI, there are 18 plates in each new LEU fuel element as compared to 10 plates in each HEU fuel element. The LEU plates, however, are thinner and contain less U-235 (per plate). This results in less heat generation per plate, due to the thinner clad. Also, the increased number of LEU fuel plates per element is offset by the plates being thinner, which results in a very similar metal to water ratio for the two fuels with the cross section for coolant flow being slightly smaller in each LEU fuel element. Accordingly, the ANL calculations indicate that maximum fuel plate temperatures will not be significantly different between the HEU and LEU cores (the LEU plates being cooler), and at the 10 kW power level these temperatures are well below the temperature for onset of nucleate boiling or fuel degradation.

Power Density and Power Peaking

Power densities and power density peaking, including both nuclear and engineering factors, were computed by ANL. The power distribution among the fuel elements is essentially the same for the HEU and LEU cores. The power densities in individual fuel plates and maximum peaking (hot channel) factors were also very similar, but slightly higher in the LEU fuel. However, the over-power at which onset of nucleate boiling would occur in the hottest channel was almost twice as high in the LEU core as in the existing HEU core, primarily because of the lower initial heat generation per plate. Thus the safety margin between the licensed power and the power at which temperatures might lead to fuel damage is much higher in the LEU core than the currently licensed HEU core.

Control Rod Worths

The reactivity worths of the control and regulating blades were computed by acceptable methods for both the LEU and HEU cores. The values for the LEU core were somewhat smaller, as expected, because of the increased neutron absorption in the U-238, but the LEU rod-worths are fully acceptable for safe reactor operation and control.

Shutdown Margin

The criterion used for shutdown margin is that reasonable assurance be provided that a nonpower reactor can be shut down from any operating condition even if the control/safety rod of maximum worth is in its most reactive position (fully withdrawn). On the basis of the computed control rod worths and the authorized excess reactivity, the WPI reactor would be subcritical by approximately 6% $\Delta k/k$ with the rod of maximum worth fully withdrawn. This is substantially larger than the Technical Specification margin of at least 1% $\Delta k/k$, and is acceptable.

Excess Reactivity

Additional reactivity above cold, clean critical is required to allow a reactor to perform programmatic and academic functions. The Licensee's submittal discussed and presented calculated changes in reactivity caused by various LEU core configurations. There is reasonable assurance that the excess reactivity permitted by the Technical Specifications, which is the same for the LEU and HEU cores, can be achieved. Because the authorized maximum excess is 0.5% Δ k/k, inadvertent insertion of all of this excess will not allow the reactor to become prompt critical so any possible transient power increase would be quickly terminated by a power level scram or operator intervention and would increase fuel temperatures only a few degrees C, which is acceptable for both the HEU and the proposed LEU cores.

Reactivity Feed-back Coefficients

The temperature coefficient of reactivity and the void coefficient of reactivity were computed for both the HEU and LEU cores. Both coefficients are more negative than required by the Technical Specifications. The void coefficient of the LEU core is significantly more negative than for the HEU core because of the more under-moderated LEU core condition. The temperature coefficient of the LEU core is also more negative than in the HEU core because of the epithermal Doppler effect in the neutron capture resonances of the relatively much more abundant U-238 present in the LEU fuel. The Doppler feedback is prompt, and therefore more effective in countering a reactor transient in the LEU core than is the heat transfer dependent moderator temperature coefficient in the HEU core. Thus, these reactivity coefficients for the LEU core are larger and more effective in leading to reactor stability than the reactivity coefficients for the HEU core, and therefore are acceptable.

Fission Product Inventory and Containment

The total inventory of fission products will not be significantly different between the HEU and LEU cores. Furthermore, because there are 18 plates in LEU versus 10 in HEU fuel elements, the inventory per plate is less in the LEU core. The aluminum cladding however is thinner on the LEU plates which may tend to reduce the integrity of the fission product barrier. This cladding thickness, however, is currently in use on the University of Michigan LEU fuel, and has been used on HEU fuel for many years in most of the licensed research reactors. Because there have been no failures or significant releases of fission products, attributable to this cladding thickness, there is reasonable assurance that the new LEU fuel will perform satisfactorily in containing fission products in the relatively benign environment in the 10 kW WPI reactor.

Potential Accident Scenarios

Among the various potential accidents considered by the Licensee or the staff at the time of the 1982 license renewal for WPI[3], only two could be affected by the conversion from HEU to LEU fuel. These two scenarios are addressed below.

(1) Inadvertent Insertion of Excess Reactivity.

The ANL[2] report presents the results of computations using the modified PARET code[4] for stepwise insertion of reactivity in both the current HEU core and the proposed LEU core. It was assumed that only inherent reactivity feedback mechanisms limit the transient, and the assumed values are given. The computation determined the minimum step insertion that would just raise the temperature to the melting point of the cladding for each core. For the HEU core the computed insertion was 1.5% Δ k/k. For the LEU core the insertion was 2.0% Δ k/k. The difference is due principally to the prompt Doppler coefficient and the larger average void coefficient of reactivity in the LEU core as discussed previously.

Because the WPI Technical Specifications limit the excess reactivity to 0.5% Δ k/k, a step insertion of this amount was also analyzed. The computed maximum fuel temperatures were slightly above 100°C for both cores, well below temperatures at which fuel or cladding damage would occur. Also, the calculated maximum fuel temperature in the LEU core was slightly lower than in the HEU core, which is acceptable.

(2) Fuel Handling Accident.

An accident leading to structural damage in air of a fully irradiated fuel element was considered to be the maximum hypothetical accident by both the licensee and the staff in the evaluation of the WPI reactor facility for renewal of its operating license in 1982. This same accident was also considered by the ANL group in its assessment of the LEU fuel at WPI. The only significant difference between the inventory of radioactivity in the LEU and in the HEU element is the plutonium-239 formed by neutron capture in the uranium-238, which is much more abundant in the LEU fuel. However, on the basis of the licensed power level and consequent burn-up of fuel at WPI, the additional build-up of plutonium in the LEU fuel is radiologically insignificant, and release of the fission products including this plutonium from damaged LEU fuel would result in maximum potential radiation exposures in the unrestricted areas of only a small fraction of those allowed by 10 CFR 20 guidelines. Therefore, damage to LEU fuel in place of HEU fuel would cause no significant change in the risk to the health and safety of the public, which was already acceptably low for the current HEU core.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes in inspection and surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously evaluated, or create the possibility of a new or different kind of accident from any accident previously evaluated, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed activities, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or the health and safety of the public.

5.0 REFERENCES

1. Amendment No. 27 to the University of Michigan Ford Nuclear Reactor License, dated February 10, 1981.

2. Analysis for Conversion of the Worcester Polytechnic Institute Reactor from HEU to LEU Fuel, Argonne National Laboratory, RERTR Program, J. E. Matos and K. E. Freese, August 1987.

3. Amendment No. 7 to the Worcester Polytechnic Institute Open Pool Training Reactor and NUREG-0912, Safety Evaluation Report, December 1982.

4. William L. Woodruff, "A Kinetics and Thermal-Hydraulics Capability for the Analysis of Research Reactor," Nuclear Technology 64 pp. 196-206, (February 1984).

Dated: September 12, 1988

Enclosure 4

INFORMATION TO BE SUPPLIED BY WPI

1. In order to provide a comparison between measured values with computed predictions please provide a report six months following fuel loading that addresses the items in the outline of the attachment to this enclosure.

2. Please submit a Fuel Load and Reactor Start-Up Planning document, with specific procedures and instrumentation requirements. This document should also identify the personnel who will be involved with the startup and who have previous experience with initial fuel-loading, power calibration and start-up of a non-power reactor. Please inform me of the expected date when fuel loading will begin and any changes that may occur to this date.

3. In response to question 3 of our January 21, 1988 letter to you requesting additional information, you submitted two figures, labeled Figure 9 and Figure 10 stating that they are from the current SAR. We note that (1) these figures do not, in themselves, respond to the question, and (2) these figures are different from Figures 8, 9 and 10 included in your transmittal of September 17, 1987. Please identify which figures are to be considered as official parts of the SAR and explain their relevance to the Technical Specifications requirements on void coefficient and temperature coefficient of reactivity.

4. Please submit the following corrections to the table of Section 2.3, of your February 10, 1988 response, to clarify the record as follows:

   (a) Nuclear Characteristics--(Calc): (a) for the flux entries under LEU, the symbol "nv" appears to have no use and should be deleted; (b) for the flux entries under HEU, it is recommended that the units of n/cm^2 sec be substituted for the symbol "nv".

   (b) Control, Safety Elements, Reactivity Control: it is recommended that the unit of control be made more complete by making it read "3.5% Δ k/k each, minimum".

Attachment to Enclosure 4

OUTLINE OF REACTOR START-UP

REPORT AND COMPARISONS WITH CALCULATIONS

1. Critical Mass

   Measurement with HEU
   Measurement with LEU
   Comparisons with calculations for both LEU and HEU.

2. Excess (operational) reactivity

   Measurement with HEU
   Measurement with LEU
   Comparison with calculations for both LEU and HEU.

3. Control and regulating rod calibrations

   Measurements of differential and total rod worths, and comparisons with calculations for both HEU and LEU.

4. Reactor power calibration

   Methods and measurements that assure operation within the license limit. Comparison between HEU and LEU nuclear instrumentation setpoints, detector positions, and detector output.

5. Shutdown margin

   Measurement with HEU
   Measurement with LEU
   Comparisons between these, and with computations for both.

6. Partial fuel element worths for LEU

   Measured for different numbers of plates for which the fuel is capable; comparison with calculations.

7. Thermal neutron flux distributions.

   Measurements with HEU and LEU, and comparisons with each other and calculations.

8. Discussion of how compliance with void and temperature coefficient values in Technical Specifications is to be assured. Comparisons with any calculations for both HEU and LEU fuel.

9. Comparison of the various results, and discussion of the comparison, including an explanation of any significant differences which have an impact on both normal operation and potential accidents with the reactor.

10. 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.