Jerk

July 13, 1987

SECY-87-171

For: The Commissioners

From: Victor Stello, Jr.
Executive Director for Operations

Subject: ORDER MODIFYING LICENSE TO CONVERT FROM HIGH- TO LOW-ENRICHED URANIUM FUEL--RENSSELAER POLYTECHNIC INSTITUTE

Purpose: To provide to the Commission, for its information, the first order issued under 10 CFR 50.64, which requires that non-power reactors convert from high- to low-enriched uranium fuel.

Background: On March 27, 1986, 10 CFR 50.64, "Limiting the Use of Highly Enriched Uranium in Domestically Licensed Research and Test Reactors," became effective. This regulation is applicable to all NRC-licensed non-power reactors; this includes four owned by commercial entities and one owned by the National Bureau of Standards, as well as those owned by universities. In part, the regulation requires conversion from the use of high- to low-enriched uranium fuel, provided suitable fuel and funding are available through Department of Energy or another appropriate Federal agency.

Discussion: The first facility to convert to low-enriched uranium is the Rensselaer Polytechnic Institute (RPI). The regulation in 10 CFR 50.64(c)(3) requires that the conversion be implemented by an order issued by the Director of the Office of Nuclear Reactor Regulation.

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The Enclosure is the Order that has been issued for the conversion of the RPI facility. Future orders should be substantially similar. Consequently, unless requested by the Commission, the staff does not intend to submit subsequent conversion orders to the Commission for information.

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

Victor Stello, Jr.
Executive Director
for Operations

Enclosure:
Order Modifying License - RPI

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| In the Matter of<br><br>Rensselaer Polytechnic Institute<br>Critical Facility<br>Department of Nuclear Engineering<br>and Science<br>Troy, New York 12181 | Docket No. 50-225<br>Facility Operating License No. CX-22<br><br><br>Amendment No. 7 |

ORDER MODIFYING LICENSE

I

Rensselaer Polytechnic Institute (licensee) is the holder of Facility Operating License No. CX-22 (License) issued on July 23, 1964 (Possession Only) and July 20, 1965 (Operating), and with subsequent renewals on June 19, 1969 and December 2, 1983 issued by the U.S. Nuclear Regulatory Commission (Commission). The license authorizes the operation of the Rensselaer Polytechnic Institute Critical Facility (facility) at a power level not in excess of 100 watts (thermal). The facility is a research reactor located on the south side of the Mohawk River in Schenectady, New York, adjacent to a site belonging to the General Electric Company. The mailing address, however, is Rensselaer Polytechnic Institute Critical Facility, Department of Nuclear Engineering and Science, Troy, New York 12181.

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 became effective on March 27, 1986.

The rule requires that a licensee of an existing non-power reactor (as a separate matter, the rule also covers newly licensed non-power reactors) 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 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) 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) 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).

III

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.

IV

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.

V

On October 3, 1986, the Director received the licensee's proposal, including its proposed modifications, supporting safety analyses and schedule for conversion. The conversion consists of the installation of low-enriched $UO_2$ fuel pins, enriched to about 4.8% in the U-235 isotope, produced in the early-to-mid 1960s for use in the Special Power Excursion Reactor Test (SPERT) Program. The HEU fuel has been removed from the facility. The Licensing Conditions and Technical Specification changes needed to amend the facility license are included in the attachment to this Order. Also, the Technical Specifications have been revised to include minor editorial changes. On the bases of the licensee's submittals and on 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 based upon the schedule set out below.

VI

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:

Within 30 days of the date of publication of this Order in the FEDERAL REGISTER, Facility License No. CX-22 is modified by amending the License Conditions and Technical Specifications as stated in the Attachment to this Order.

VII

The licensee or any other person whose interest is adversely affected by this Order may request a hearing within 30 days of its date of publication in the FEDERAL REGISTER. Any request for a hearing, petition for leave to intervene, or answer to this Order shall be submitted to the Director of the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555. A copy of the request, petition or answer shall also be sent to the Office of the General Counsel at the same address, and to the Regional Administrator, Region I, USNRC, 631 Park Avenue, King of Prussia, PA 19406. If a person other than the licensee requests a hearing, that person shall set forth with particularity the manner in which the petitioner's interest is adversely affected by this Order and should address the criteria set forth in 10 CFR 2.714(d).

If a hearing is to be held about this Order, the Commission will issue an order designating the time and place of the hearing and the issue to be considered in that hearing be whether this Order shall be sustained.

This Order shall become effective within 30 days of its date of publication in the FEDERAL REGISTER or, if a hearing is held, on the date specified in an order following further proceedings on this Order.

An Environmental Assessment with a finding of no significant impact on the environment has been prepared in conjunction with this Order and is available in the Nuclear Regulatory Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555.

FOR THE NUCLEAR REGULATORY COMMISSION,

Thomas E. Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 7th day of July, 1987.

Enclosure:
As stated

ATTACHMENT TO ORDER
MODIFYING LICENSE NO. CX-22 DATED July 7, 1987

A. License Conditions Revised 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 at any one time up to 21.12 kilograms of contained uranium-235 in SPERT (F-1) fuel pins and up to 80 grams of plutonium encapsulated in plutonium-beryllium neutron sources, both in connection with operation of the facility.

No. 2.B.5. Delete. (This condition is deleted because it authorized possession of 21.12 kilograms of SPERT (F-1) fuel pins. This condition has now been included in License Condition No. 2.B.2)

No. 2.C.2. The Technical Specifications contained in Appendix A, dated November 1983, as revised through Amendment 7, 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

TABLE OF CONTENTS

1.0 DEFINITIONS

D. Control Rod Assembly - A control mechanism consisting of a stainless steel basket that houses two absorber sections, one above the other. These absorber sections may contain either enriched boron in iron, $EuO_3$ in a stainless steel cermet, stainless steel, or an alloy of silver-cadmium-indium. All absorber sections except the one containing silver-cadmium-indium are clad in stainless steel. All are of the same dimensions, nominally 2.6 inches square, with their poisons uniformly distributed. The absorbers, when fully inserted, shall extend above the top and to within one inch of the bottom of the active core.

N. Reactor Secured - (1) The full insertion of all control rods has been verified, (2) the console key is removed, and (3) no operation is in progress which involves moving fuel pins in the reactor vessel, the insertion or removal of experiments from the reactor vessel, or control rod maintenance.

O. Reactor Shutdown - The control rods are fully inserted and the reactor is shut down by at least 1.00$. The reactor is considered to be operating whenever this condition is not met and more than 60% of the total number of fuel pins required for criticality in a given configuration (Core A or Core B) have been loaded in the core.

Q. Reportable Occurrence - The occurrence of any facility condition that:

6. Results in uncontrolled or unanticipated change in reactivity of greater than 0.60$;

T. Secured Experiment - Any experiment, experimental facility, or component of an experiment is deemed to be secured, or in a secured position, if it is held in a stationary position relative to the reactor. The restraining forces must be equal to or greater than those that hold the fuel pins themselves in the reactor core.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits - Fuel Pellet Temperature

Applicability

Applies to the maximum temperature reached in any incore fuel pellet as a result of either normal operation or transient effects.

Objective

To identify the maximum temperature beyond which material degradation of the fuel and/or its cladding is expected.

Specification

Fuel pellet temperature at any point in the core, resulting from normal operation or transient effects, shall be limited to no more than 2000°C.

Bases

Specific determination of the melting point of the SPERT fuel has not been reported. A safety limit of 2000°C is below the listed melting point of $UO_2$ under a wide variety of conditions. The chosen value is conservative in view of variations that might result because of the presence of small quantities of impurities and the comparatively high vapor pressure of $UO_2$ at elevated temperatures. The safety limit specified is about 700°C below the measured melting point of $UO_2$ in a helium atmosphere.*

2.2 Limiting Safety System Setting-Reactor Power

Bases

The maximum power level trip setting of 135 watts on Log Power and Period Channel 2(PP2) correlates with a reading of not greater than 90% on the highest scale of either of the two Linear Power Channels (LP1, LP2) as established by activation techniques. These scram setpoints ensure reactor shutdown and prevent significant energy deposition or enthalpy rise in the core in the event of any credible accident scenario.

The minimum flux level has been established at 2 cps to prevent a source-out startup and provide a positive indication of proper instrument function before any reactor startup.

The minimum 5-second period is specified so that the automatic safety system channels have sufficient time to respond in the event of a very rapid positive reactivity insertion. Power increase and energy deposition subsequent to scram initiation are thereby limited to well below the identified safety limit.

## 3.0 LIMITING CONDITIONS FOR OPERATION

### 3.1 Reactor Control and Safety Systems

Specifications

1. The excess reactivity of the reactor core above cold, clean critical shall not be greater than 0.60$. The maximum reactivity worth of any clean fuel pin shall be 0.20$.

2. There shall be a minimum of four operable control rods. The reactor shall be subcritical by more than 0.70$ with the most reactive control rod fully withdrawn.

(Paragraphs 3-9 have not changed and therefore are not included.)

10. The thermal power level shall be controlled so as not to exceed 100 watts, and the integrated thermal power for any consecutive 365 days shall not exceed 200 kilowatt-hours.

*Reference: W. A. Duckworth, ed., "Physical Properties of Uranium Dioxide," Uranium Dioxide: Properties and Nuclear Applications (Washington, D.C.: Naval Reactors, Division of Reactor Development), 1961, pp. 173-228.

Bases

The minimum number of four control rods is specified to ensure that there is adequate shutdown capability even for the stuck control rod condition.

(Paragraphs 2-6 have not changed and therefore are not included.)

Limitations imposed on core reactivity, control rod worth, and reactor power preclude conditions that could allow the development of a potentially damaging accident. The limitations are conservative in view of core energy deposition, yet permit adequate flexibility in the research and instruction for which the facility is intended.

3.2 Reactor Parameters

Specifications

1. Above 100°F the isothermal temperature coefficient of reactivity shall be negative. The net positive reactivity insertion from the minimum operating temperature to the temperature at which the coefficient becomes negative shall be less than 0.15$.

2. The void coefficient of reactivity shall be negative, when the moderator temperature is above 100°F, within all standard fuel assemblies and have a minimum average negative value of 0.00043$/cc within the boundaries of the active fuel region.

Bases

The minimum absolute value of the temperature coefficient of reactivity is specified to ensure that negative reactivity is inserted when reactor temperature increases above 100°F. It is of note that even in the worst postulated accident scenario, such as considered in Section 4 of the SAR (1964), reactivity insertion because of temperature change would be negligible. The minimum average negative value of the void coefficient is specified to ensure that the negative reactivity inserted because of void formation is greater than that which was calculated in the SAR.

3.4 Experiments

Bases

The 7th or last paragraph is amended to correct references to the regulations, i.e., 10 CFR 20.105 and 10 CFR 20.106 vs. 10 CFR 105(1) and 10 CFR 106 as follows:

Specifications 8 and 9 will ensure that the quantities of radioactive materials contained in experiments will be so limited that their failure will not result in exposures to individuals in restricted or unrestricted areas to exceed the maximum allowable exposures stated in 10 CFR 20. The restricted area maximum is defined in 10 CFR 20.101 and 10 CFR 20.103. The unrestricted area maximum is defined in 10 CFR 20.105 and 10 CFR 20.106.

# 5.0 DESIGN FEATURES

## 5.4. Reactor

### 5.4.2 Reactor Core

The reactor core shall consist of uranium fuel provided in the form of 4.8 weight percent enriched $UO_2$ pellets in stainless steel cladding, arranged in roughly a cylindrical fashion with four control rods placed symmetrically about the core periphery. Fuel pins, with an effective length of 91.44 cm are set on a square pitch of 1.43 cm to yield an effective core radius of approximately 35 cm. Two fuel pin arrangements have been evaluated. The first, referred to as "Core A," is the solid array of pins shown in Figure 4.3 of the SAR. The second, referred to as "Core B," is the annular array of pins shown in Figure 4.4 of the SAR. The pins themselves are supported and positioned on a fuel pin support plate, drilled with 1/4-inch-diameter holes to accept tips on the end of each pin. The support plate rests on a carrier plate which forms the base of a three-tiered overall core support structure. An upper fuel lattice plate rests on the top plate, and both are drilled through with 1/2-inch-diameter holes on the prescribed pitch to accommodate the upper ends of the fuel pins. The lower fuel pin support plate, a middle plate, and the upper fuel pin lattice plate are secured with tie rods and bolts. The entire core structure is supported vertically and anchored by four posts set in the floor of the reactor tank. Finally, in the event the fuel pins are bowed but still satisfactory for use in the core, a plastic spacer plate may be installed on the middle plate. Figure 4.6 of the SAR depicts the total core assembly.

### 5.4.3 Fuel Pins

Fuel pins to be utilized are 4.8 weight percent enriched SPERT (F-1) fuel rods. Each fuel rod is made up of sintered $UO_2$ pellets, encased in a stainless steel tube, capped on both ends with a stainless steel cap and held in place with a chromium-nickel spring. An aluminum oxide ($Al_2O_3$) insulator between the fuel pellets and stainless steel caps on each end of the rod is installed. Gas gaps to accommodate fuel expansion are also provided at both the upper end and around the fuel pellets. Figure 4.7 of the SAR depicts a single fuel pin and its pertinent dimensions.

### 5.4.4 Control Rod Assemblies

Four control rod assemblies are installed, spaced 90 degrees apart at the core periphery. Each rod consists of a 6.99-cm square stainless steel tube which passes through the core and rests on a hydraulic buffer on the bottom carrier plate of the support structure. Housed in each of these "baskets" are two neutron-absorber sections, one positioned above the other as depicted in Figure 4.8 of the SAR. The combination of the four rods must meet the values given in Table 5.2 of the SAR, with regard to reactivity with one stuck rod and shutdown margin.

5.6 Fuel Storage and Transfer

Old Section 5.6 is deleted and Section 5.7 is renumbered 5.6 and the last paragraph is modified as follows:

For a known system, up to a quadrant of fuel pins may be removed from the core or a single stationary fuel pin be replaced with another stationary pin only under the following conditions:

1. The net change in reactivity has been previously determined by measurement or calculation to be negative or less than 0.20$.

2. The reactor is subcritical by at least 1.00$ in reactivity.

6.0 ADMINISTRATIVE CONTROLS

6.2 Procedures

2) Installation and removal of fuel pins, control rods, experiments, and experimental facilities.

6.5 Reports

In addition to the requirements of applicable regulations, and in no way substituting therefore, all written reports shall be sent to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, D.C. 20555, with a copy to the Region I Administrator.

6.5.2 Non-Routine Reports

a) Reportable Operational Occurrence Reports. Notification shall be made within 24 hours by telephone and telegraph to the Administrator of Region I followed by a written report within 10 days in the event of a reportable operational occurrence as defined in Section 1.0. The written report on these reportable operational occurrences, and to the extent possible, the preliminary telephone and telegraph notification shall: (1) describe, analyze, and evaluate safety implications; (2) outline the measures taken to ensure that the cause of the condition is determined; (3) indicate the corrective action (including any assurance program) taken to prevent repetition of the occurrence and of similar occurrences involving similar components or systems; and (4) evaluate the safety implications of the incident in light of the cumulative experience obtained from the record of previous failures and malfunctions of similar systems and components.

b) Unusual Events. A written report shall be forwarded within 30 days to the Administrator of Region I in the event of discovery of any substantial errors in the transient or accident analyses or in the methods used for such analyses, as described in the Safety Analysis Report or in the bases for the Technical Specifications.