Docket Files

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

March 17, 1993

Docket No. 50-193

Mr. Terry Tehan, Director
Nuclear Science Center
Rhode Island Atomic Energy Commission
South Ferry Road
Narrangansett, Rhode Island 02882-1197

Dear Mr. Tehan:

SUBJECT: ISSUANCE OF ORDER MODIFYING LICENSE NO. R-95 TO CONVERT FROM HIGH- TO LOW-ENRICHED URANIUM (AMENDMENT NO. 17) - RHODE ISLAND ATOMIC ENERGY COMMISSION (TAC NO. M84084)

The U.S. Nuclear Regulatory Commission (NRC) is issuing this Order, Amendment No. 17 to Facility Operating License No. R-95, 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), which requires that a non-power reactor, such as the reactor at the Rhode Island Atomic Energy Commission Nuclear Science Center, 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 submittal of November 11, 1991, as supplemented on July 23, December 22, 1992, and January 13, 1993.

The Order will become effective on the later date of either (1) the day of receipt of an adequate number and type of LEU fuel elements that are necessary to operate the facility as specified in your submittal and supplements or (2) 30 days after the date of publication of this Order in the Federal Register if no hearing is requested. 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 fuel is completely removed from the facility. Additionally, you are required by the Order to provide, within six months after the initial criticality with LEU fuel, your startup report.

DF03

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

Sincerely,

ORIGINAL SIGNED BY

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

Enclosures:
1. Order and Amendment No. 17
2. Replacement pages for Technical Specifications
3. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION: [RIHEUORD.MMM] [M.Mendonca diskette]

Docket file 50-193	MMendonca
PDRs	OGC
ONDD R/F	EJordan
TMurley/FMiraglia	GHill (2)
JPartlow	ACRS (10)F
SECY (Orders Only)	OPA
EHylton	BGrimes
SWeiss	OC/LFOCB
ERossi	Region I
JLieberman	RDudley
VMcCree, EDO	

*See previous concurrence

ONDD:LA	ONDD:PM*	TECH EDITOR*	OGC*
EHylton	MMendonca:lb	JMain	CBarth
3/11/93	2/12/93	1/22/93	2/24/93
ONDD:D*	DORS:DD	ADP	DONRR
SWeiss	BGrimes	JPartlow	TMurley
3/9/93	3/10/93	3/11/93	3/15/93

OFFICIAL RECORD COPY

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

Sincerely,

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

Enclosures:
1. Order and Amendment No. 17
2. Replacement pages for Technical Specifications
3. Safety Evaluation

cc w/enclosures:
See next page

Rhode Island Atomic Energy Commission Docket No. 50-193

cc:

President, Town Council
Town of Narragansett
Town Hall
Narragansett, Rhode Island 02882

Governor of Rhode Island
Providence, Rhode Island 02903

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
RHODE ISLAND ATOMIC ENERGY COMMISSION	)	Docket No. 50-193
	)	
	)	
(Rhode Island Nuclear Science Center Research Reactor)	)	
	)	

ORDER MODIFYING LICENSE

I.

The Rhode Island Atomic Energy Commission (the licensee) is the holder of Facility Operating License No. R-95 (the license) issued on July 21, 1964, by the U.S. Atomic Energy Commission. The license, as amended by Amendment No. 1 on September 10, 1968, authorizes operation of the Rhode Island Nuclear Science Center Research Reactor (the facility) at a power level up to 2 megawatts (Mw) thermal (t). The facility is a research reactor located in the Narragansett Bay Campus of the University of Rhode Island (formerly called Fort Kearney) in Narragansett, Rhode Island. The research reactor is contained in the Rhode Island Nuclear Science Center, which is located on the south central portion of the Narragansett Bay Campus. The mailing address is Nuclear Science Center, Rhode Island Atomic Energy Commission, South Ferry Road, Narragansett, Rhode Island 02882-1197.

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 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 Commission issued the rule to promote the common defense and security by reducing the risk of theft and diversion of HEU fuel used in non-power reactors.

Sections 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 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 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 requirements of the rule.

Section 50.64(c)(2)(i) also requires the licensee to include 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 other appropriate Federal agency and (2) a schedule for conversion,

based upon availability of replacement fuel acceptable to the Commission for that reactor and upon consideration of other factors such as the availability of shipping casks, implementation of arrangements for available financial support, and reactor usage.

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

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

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

Section 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.

Section 2.714 states the requirements for a person whose interest may be affected by any proceeding to initiate a hearing or to participate as a party.

III.

On November 18, 1991, as supplemented on July 23, 1992, December 22, 1992, and January 13, 1993, the NRC staff received the licensee proposal, including its proposed modifications, supporting safety analyses, and plans for conversion. The conversion consists of replacing high-enriched with low-enriched uranium fuel elements. The fuel elements contain materials test reactor (MTR)-type fuel plates, with the fuel meat consisting of uranium silicide dispersed in an aluminum matrix. These plates contain an enrichment of less than 20 percent with the uranium-235 isotope. The NRC staff reviewed the licensee proposal and the requirements of 10 CFR 50.64 and has determined that the public health and safety and the common defense and security require the licensee to convert the facility from the use of HEU to LEU fuel in accordance with the Attachment to this Order and the schedular requirements included herein following. The Attachment to this Order specifies the changes to the license conditions and discusses the changes to Technical Specifications that are needed to amend the facility license.

IV.

Accordingly, pursuant to Sections 51, 53, 57, 101, 104, 161b., 161i., and 161o. of the Atomic Energy Act of 1954, as amended, and to Commission regulations in 10 CFR 2.204 and Section 50.64, IT IS HEREBY ORDERED THAT:

Facility Operating License No. R-95 is modified by amending the license conditions and Technical Specifications as stated in the Attachment to this Order on the later date of either (1) 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 proposal or (2) 30 days after the date of publication of this Order in the Federal Register.

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, D.C. 20555, with a copy to the Assistant General Counsel for Hearings and 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 man in which their 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 hearing is whether this Order should be sustained.

This Order shall become effective on the later date of either 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 proposal or 30 days after the date of publication of this Order in the Federal Register or, if a hearing is requested, on the date specified in an order after 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 17th day of March 1993

Attachments:
As stated

ATTACHMENT TO ORDER

MODIFYING FACILITY OPERATING LICENSE NO. R-95

A. License Conditions Revised and Added by this Order

2.b. Pursuant to the Act and 10 CFR Part 70, "Special Nuclear Material," to receive, possess, and use at any one time up to 10.4 kilograms of contained uranium-235 at enrichments equal to or less than 20 percent in the form of MTR-type reactor fuel in connection with operation of the reactor and up to 32 grams of plutonium encapsulated in two plutonium-beryllium neutron sources for reactor startup.

2.d. Pursuant to the Act and 10 CFR Part 70, "Special Nuclear Material," to possess, but not use, up to 8.0 kilograms of contained uranium-235 at greater than 20 percent enrichment in the form of MTR-type reactor fuel until the existing inventory of this fuel is removed from the facility.

3.b. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 17, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3.d.(4) The licensee shall provide a startup test report within six months after initial criticality with low enriched uranium reactor fuel in accordance with Amendment No. 17. 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 License Amendment No. 17, Facility Operating License No. R-95, Docket No. 50-193, Replacement Pages for Technical Specifications," and as discussed in the Safety Evaluation for this Order.

ENCLOSURE TO LICENSE AMENDMENT NO. 17

FACILITY OPERATING LICENSE NO. R-95

DOCKET NO. 50-193

REPLACEMENT PAGES FOR TECHNICAL SPECIFICATIONS

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

RHODE ISLAND NUCLEAR SCIENCE CENTER REACTOR

TECHNICAL SPECIFICATIONS

APPENDIX A

TO

FACILITY LICENSE R-95

TABLE OF CONTENTS

TABLE OF CONTENTS (CONTINUED)

A. Site

1. Location

The reactor shall be located at the Rhode Island Nuclear Science Center on three acres of a 27-acre former military reservation, originally called Fort Kearney and now called the Narragansett Bay Campus of the University of Rhode Island. The University of Rhode Island is a state agency. The 27-acre reservation is controlled by the State of Rhode Island through the University of Rhode Island. The reservation is in the Town of Narragansett, Rhode Island on the west shore of Narragansett Bay approximately 22 miles south of Providence, Rhode Island and approximately six miles north of the entrance of the Bay from the Atlantic Ocean. The Rhode Island Nuclear Science Center and various buildings used for research, education and training purposes are located on this 27-acre campus.

2. Exclusion Area

Figure A.1 is a drawing of the Narragansett Bay Campus showing the three acre Nuclear Science Center site. The boundary of this area shall be posted with conspicuous signs to delineate the area. This three acre area shall be the exclusion area as defined in 10 CFR 100.

3. Restricted Area

Figure A.1 also shows the location of the reactor building on the three acre area. The reactor building and attached office laboratory wing shall be considered the restricted area as defined in 10 CFR 20.

4. Principal Activities

The principal activities carried on within the restricted and exclusion area shall be those associated with operation and utilization of the reactor. It shall be permissible to locate additional Nuclear Science Center or University of Rhode Island buildings within the exclusion area provided that these additional buildings are capable of timely evacuation and do not interfere with the operation of the reactor.

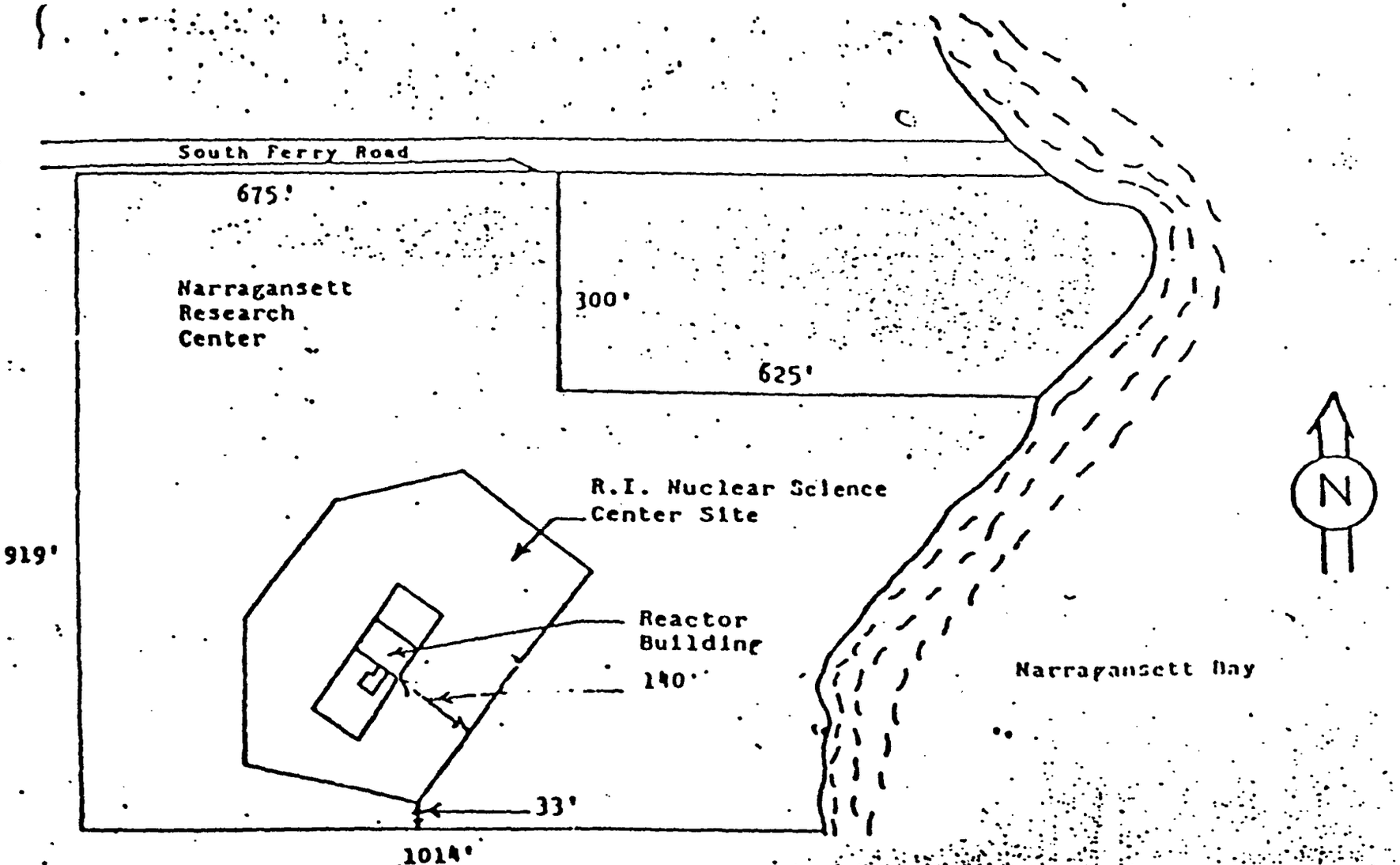

FIGURE A.1 PORTION OF NARRAGANSETT RESEARCH CENTER SHOWING LOCATION OF REACTOR BUILDING

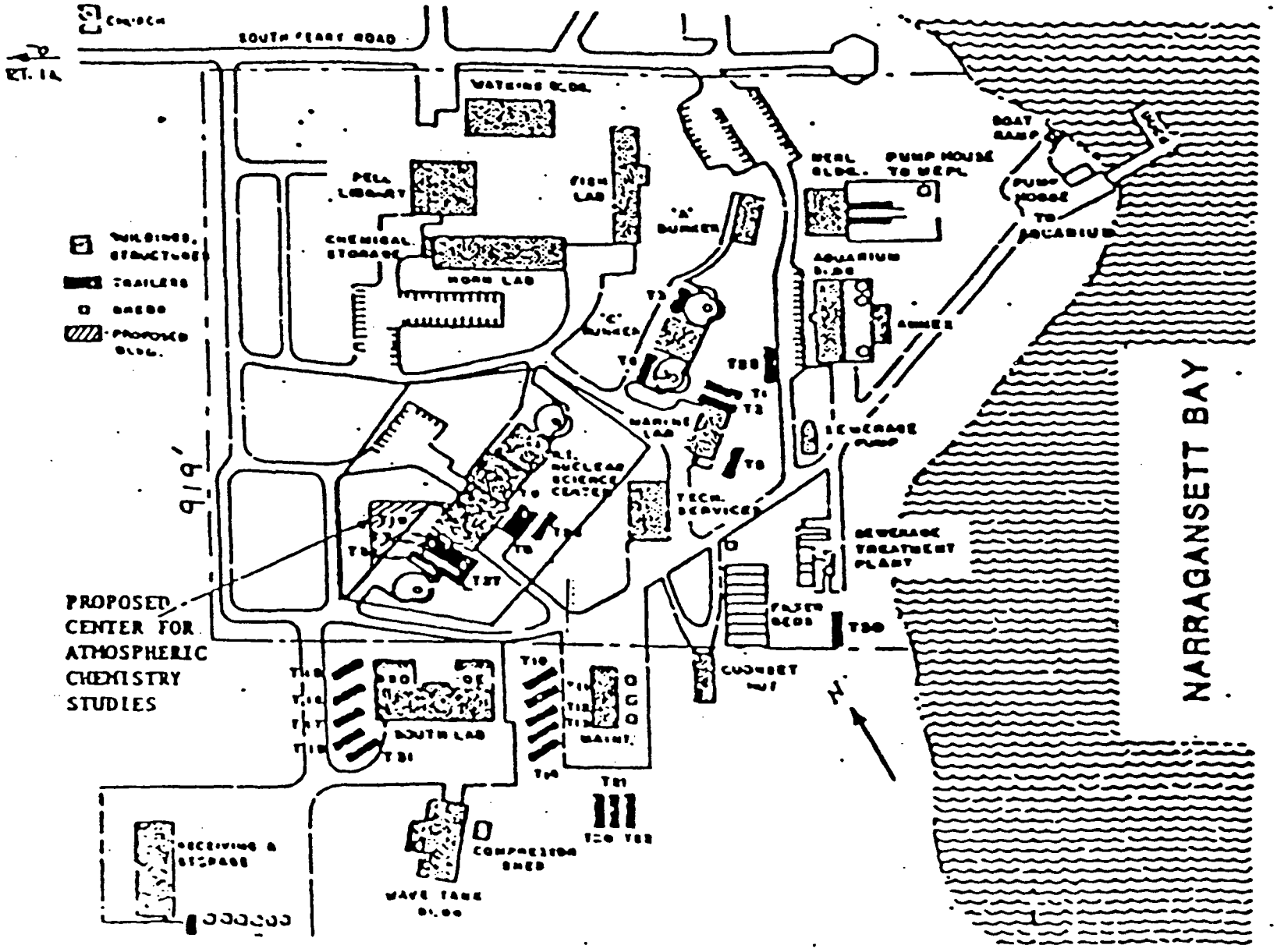

FIGURE A.1 PORTION OF NARRAGANSETT BAY CAMPUS OF URI SHOWING LOCATION OF REACTOR BUILDING

Amend #5 Oct. 1987

Amendment No. 17

B. CONTAINMENT

1. Reactor Building

The reactor shall be housed in a building capable of meeting the following functional requirements:

> In the event of an accident which could involve the release of radioactive material, the confinement building air shall be exhausted through a clean-up system and stack creating a flow of air into the building with a negative differential pressure between the building and the outside atmosphere. The building shall be gas tight in the sense that a negative differential pressure can be maintained dynamically with all gas leaks occurring inward. The confinement and clean-up systems shall become operative when a building evacuation button is pressed. This action shall: (1) turn off all ventilation fans and the air conditioner system and (2) close the dampers on the ventilation and air conditioning system intakes and exhaust, other than those which are a part of the clean-up system. No further action shall be required to establish confinement and place the clean-up system in operation. An auxiliary electrical power system shall be provided at the site to insure the availability of power to operate the clean-up system.

The reactor building exhaust blower, which is designed to exhaust at least 4000 cfm, operates in conjunction with additional exhaust blower(s) which provide an additional exhaust of at least 10000 cfm from non-reactor building sources and in conjunction with the air handling unit which takes air into the reactor building at less than 4000 cfm. The total exhaust rate through the stack is at least 14000 cfm. During normal operation, the building is at a pressure somewhat belong atmospheric. The control room air conditioner shall be a self-contained unit, thermostatically controlled, providing constant air temperature for the control room. If it is installed with a penetration through the wall of the reactor building, it shall have a damper at this penetration which closes when an evacuation button is pressed.

Upon activation, the clean-up system shall exhaust air from the reactor building through a filter and a 115 foot high stack, creating a pressure less than atmospheric pressure. The clean-up filter shall contain a roughing filter, an absolute particulate filter, a charcoal filter for removing radioiodine, and an absolute filter for removing charcoal dust which may be contaminated with radioiodine. Each absolute filter cartridge shall be individually tested and certified by the manufacturer to have an efficiency of not less than 99.97% when tested with 0.3 micron diameter dioctylphthalate smoke. The minimum removal efficiency of the charcoal filters shall be 99%, based on ORNL data and measurements performed locally.

Gases from the beam ports, thermal column, pneumatic system, and all other radioactive gas exhaust points shall be exhausted to the stack through a roughing and absolute filter system.

C. REACTOR POOL AND PRIMARY COOLANT SYSTEM

1. General

The primary coolant system shall consist of the reactor pool, delay tank, heat exchanger, coolant pump, and the associated valves, piping, flow channels and sensors. During forced convection cooling, coolant water shall be supplied to the core by an aluminum line connected to the inlet flow channel which is on one side of the suspension frame. The coolant water shall flow from the inlet flow channel downward through the core to a plenum below the grid box. The coolant water shall then flow into the outlet flow channel on the opposite side of the suspension frame and then through a discharge line to the delay tank, coolant pump, heat exchanger and then return to the coolant inlet line.

2. Reactor Pool

The reactor pool shall be constructed of ordinary concrete with 1/4" thick 6061-T6 aluminum liner and shall have a volume of approximately 36,300 gal.

3. Shielding

The reactor pool and primary system shielding shall be adequate to meet the applicable personnel radiation protection requirements of 10 CFR 20.

4. Primary Coolant System

The primary coolant system shall conform to the following:

a. Heat Exchanger

The heat exchanger shall be designed to remove heat at the rate generated by the reactor at maximum licensed steady state power from the primary water and shall be designed to perform under the maximum primary system operating temperature and pressure. Replacement heat exchanger shell and tube bundles shall be constructed from stainless steel according to the requirements of Section III, Class C of the ASME Boiler and Pressure Vessel Code.

b. Primary Pump

Number of pumps	1
Type	Horizontal mounted, Centrifugal, Single Suction

Materials of construction	Worthite
Rating	1500 gpm
Head	59 feet
Design Pressure	75 psig minimum
Design Temperature	150°F minimum
Motor Type	Drip proof, induction, 440 v, 3-phase, 60 cycle

c. Delay Tank

Number of tanks	1
Material of construction	Aluminum Association Alloy 5083 and 5086
Material Thickness	
Walls	0.25 inch
Dished Heads	0.375 inch
Capacity	3000 gal., minimum

d. Primary Recirculation Piping

Material and thickness	Sch. 40 Al. type 3003 aluminum
Size	8 and 10 inch
Design temperature	150°F, minimum
Design pressure	100 psig, minimum

e. Make-up System

A check valve shall be installed in the line between the potable water supply and the make-up and cleanup demineralizer to prevent entry of potentially contaminated water into the potable water supply.

Water source	Potable water from city main
Make-up demineralizer type	Mixed-bed single shell, regenerative
Make-up demineralizer capacity	
Normal	25 gpm
Emergency	50 gpm
Water softener capacity	
Normal	50 gpm

f. Cleanup System for Primary Coolant Water

Cleanup pump	
Capacity	40 gpm
Head	100 ft
Cleanup demineralizer	
Type	Mixed-bed, single shell, regenerative
Cleanup demineralizer capacity	
Normal	40 gpm
Emergency	50 gpm

D. SECONDARY COOLANT SYSTEM

The secondary coolant system shall carry the heat rejected from the primary coolant at the heat exchanger to the atmosphere at cooling towers. It shall be composed of the heat exchanger, cooling towers, pumps and associated valves, piping and sensors. In this system, water flows from the heat exchanger through a control valve to the cooling towers. From the cooling tower basins, the water is then pumped back to the heat exchanger.

E. REACTOR CORE AND CONTROL ELEMENTS

The reactor core and control elements shall have the following characteristics and nominal dimensions:

1. Principal Core Materials

Fuel matrix	U_3Si_2-Al dispersion
U-235 enrichment	Approximately 20%
Fuel clad	6061 aluminum
Fuel element side plates	6061 aluminum
End fittings	3'6-T6 or 6061 aluminum
Moderator	Water
Reflector-Graphite	AGOT grade (or equivalent graphite and/or water
Reflector-Beryllium	Beryllium-aluminum clad
Control elements	Mixture of B_4C and aluminum, clad with aluminum
Servo Element	Stainless steel 304

2. Fuel Elements

Plate width overall	2.81 inches
Active plate width	2.40 inches maximum
Plate length overall	25.00 inches
Active plate length	23.50 inches
Plate thickness	0.06 inch
Clad thickness	0.02 inch
Fuel matrix thickness	0.02 inch
Water gap between plates	0.08 inch

Number of plates per fuel element	22
U-235 per fuel element	275 grams, nominal
Overall fuel element dimensions	3 in x 3 in. x 40 in.

3. Reflector Elements - Graphite and Beryllium

Overall reflector element dimensions, nominal	3 in x 3 in. x 40 in.
Nominal clad thickness	.1 in.
Nominal graphite dimensions	2.8 in. x 2.8 in. x 28.7 in.
Standard Beryllium element dimensions	2.94 in. x 2.94 in. x 29 in.

3a. Beryllium Flux Trap — 2.94 in. x 2.94 in. x 29 in. with a 1.5 inch diameter thru hole

4. Control Elements

Width	10.6 in.
Thickness	0.38 in.
Overall length	54.1 in.
Active length	52.1 in.

5. Servo Regulating Element

Shape	Square stainless steel
Width	2.1 in.
Overall length	28.8 in.
Active	24.9 in.

6. Control Element Drive

Type	Electromechanical screw
Drive to safety element connection	Electromagnet
Stroke	32 in. maximum

7. Servo Regulating Element Drive

Type	Electromechanical screw
Drive to element connection	Lock screw (no scram)
Stroke	26 in. maximum
Position indication accuracy	± 0.02 in.

8. Neutron Sources

Start-up Source

Number	2
Type	Plutonium-beryllium
Unit Source Strength	1×10^6 neutrons/sec. minimum
Maximum Power Level with Plutonium-beryllium sources installed	10 Kw

Operational Source

Number	1
Type	Antimony-beryllium
Source Strength	2×10^6 neutrons/sec. minimum

F. REACTOR SAFETY SYSTEMS

1. Modes of Power Operation

There shall be two modes of power operation:

a. Power Operation - Natural Circulation (NC)

Power operation - NC shall be any reactor operation performed with the reactor cooling provided by natural circulation. The reactor power shall not exceed 0.1 MW during NC operation.

b. Power Operation - Forced Circulation (FC)

Power operation - FC shall by any reactor operation performed with reactor cooling provided by forced circulation. The reactor power shall not exceed 2 MW during FC operation.

2. Design Features

a. The Reactor Control System

The reactor safety system shall consist of sensing devices and associated circuits which automatically sound an alarm and/or produce a reactor scram. The systems shall be designed on the fail-safe principle (de-energizing shall cause a scram). Table F.1 and F.2 describe the arrangement and requirements of the safety system.

b. Process Instrumentation

Process instrumentation with readout in the control room shall be provided to permit measurement of the flow rate, temperature, and conductivity of the primary coolant and the flow rate of the secondary coolant. In addition, a second primary flow indicating device with readout in the control room shall be located between the reactor outlet plenum and the reactor outlet header.

After normal working hours, an independent protection system, separate from the system described in Section K.3.a, shall be used to monitor certain items in the reactor building and alarm in the event of an abnormal condition. The alarm channels provided are:

(1) A fire in the reactor room,

(2) A fire in a location other than the reactor room,

(3) A decrease of 2 inches in reactor pool water level,

(4) A power failure in the reactor building,

(5) An alarm condition from the radiation monitors reading out in the control room,

(6) An alarm condition from any other selected feature.

c. Master Switch

A key lock master switch shall be provided with three positions; "off", "test", and "on". These positions shall have the following functions:

(1) The "off" position shall de-energize the reactor control circuit.

(2) The "test" position shall energize the reactor control circuit exclusive of the control blade magnets.

(3) The "on" position shall energize the reactor control circuit including the control blade magnets.

d. Power Level Selector Switch

A power level selector switch shall be provided with four positions; "0.1 MW", "1 MW", "2 MW", and "5 MW". These positions shall have the following functions:

(1) The "0.1 MW" position shall activate all safety system sensors except those indicated in Table F.1.

(2) The "1 MW" and "2 MW positions shall activate all safety system sensors.

(3) The "5 MW" position shall scram the reactor.

e. Control Element Withdrawal Interlocks

Interlocks shall prevent control rod withdrawal unless all of the following conditions exist:

(1) The master switch is in the "on" position,

(2) The safety system has been reset,

(3) The Log N amplifier switch is in the "operate" position,

(4) The startup channel neutron count rate is three counts per second or greater, and

(5) The start-up counter is not being withdrawn.

It shall not be possible to withdraw more than one control element at a time.

f. Servo System Control Interlock

Interlocks shall prevent switching to servo control unless the period as indicated by the Log N channel is thirty seconds or greater. The Servo control system shall be designed so that immediately following a scram the Servo control shall automatically return to the manual mode of operation.

TABLE F.1 - REACTOR SAFETY SYSTEM

Sensor or Trip Device	No. of Switches or Sensors	Trip Set Point	Alarm Set Point
Short Period	1	3 sec. min.	7 sec. min.
High Neutron Flux	2	Max. of 115% of full scale with a 2.3 MW max.	110% max.
High Temperature of Primary Coolant Entering Core During Forced Convection Cooling*			113°F max.
High Temperature of Primary Coolant Leaving Core During Forced Convection Cooling*		125°F max.	123°F max.
Low Flow Rate of Primary Coolant*	1	1600 gpm, min.	1650 gpm, min.
Low Pool Water Level	1	2" max.decrease	2" max.decrease
Seismic Disturbance	1	IV on Modified Mercalli Scale max.	
Bridge Misalignment*	1	X	X
Coolant Gates Open*	1 per gate	X	X
Neutron Detector High Voltage Failure in Linear Level Safety Channels	1 per power supply	Decrease of 50 volts max.	
Manual Scram (Switch at bridge and on console)	2	X	X
High Conductivity of Primary Coolant	1		Equivalent to 2μmho/cm at 25°C, max.
Safety Blade Disengaged	1		X
Log N - Period Amplifier Failure	1	X	X
Regulating Rod at Either Limit of Travel	1		X
Low Flow Rate of Secondary Coolant*	1		800 gpm, min.
Bridge Movement	1	X	X
No Flow Thermal Column*	1	X	X

*These functions are bypassed when the Power Level Selector Switch is in the "0.1 MW" position.

TABLE F.2

REACTOR NUCLEAR INSTRUMENTATION

Channel	Detector	Sensitivity	Range	Information to Operator	Information to Logic Element (Scram)	Information to Servo System	Recorded Information
Start-up	Retractable gas filled B-10 filled proportional	Neutrons-approximately 12 counts/nv	Source level to full power	Neutron Flux	None	None	Relative power level on log scale
Log N	Fixed fission counter	Neutrons-approximately .7 cps/nv	Source level to $3x10^6$ watts	Power level Period	Period scram	None	Power level log scale and period
Linear level safety	Compensated ion chamber	Neutrons-approximately $4x10^{-14}$amp/nv	1 watt to $3x10^6$ watts	Power level	Level scram	Power level	Power level linear scale (either (channel)
Linear level safety	Compensated ion chamber	Neutrons-approximately $4x10^{-14}$amp/nv	1 watt to $3x10^6$ watts	Power level	Level scram	None	

G. WASTE DISPOSAL AND FACILITY MONITORING SYSTEMS

1. Waste Disposal Systems Design Features

a. Liquid Radioactive Waste Disposal System

All liquid waste (except sanitary waste) from the reactor building shall flow to retention tanks. These tanks shall be located either underground with a dirt cover or in a locked room(s) in the reactor building.

b. Gaseous Radioactive Waste Disposal System

All gaseous radioactive waste from the beam ports, thermal column, pneumatic irradiation system and all other radioactive gas exhaust points associated with the reactor itself shall be collected in a manifold and discharged to the reactor stack through an absolute filter, blower and damper.

c. Solid Radioactive Waste Storage

Solid Radioactive wastes shall either be stored in radioactive waste storage containers located within the reactor building or removed from the site by a commercial licensed organization.

2. Area and Exhaust Gas Monitor Design Features

a. Three fixed gamma monitors employing suitable detectors shall be employed in the reactor building. Each of these shall have the following characteristics:

1) A range consistent with the expected radiation levels in the area to be monitored (0.01 to 10 mr/hr, 0.1 to 100 mr/hr, or 1 to 1,000 mr/hr).

2) A radiation dose rate output indicated in the control room.

3) An adjustable high radiation alarm which shall be annunciated in the control room.

4) The three fixed gamma monitors shall be located to detect radiation as follows: At the pool biological shield between a beam port and the thermal column, above the storage container for new fuel elements, and at the reactor bridge.

b. A gamma monitor shall be provided near the primary coolant system, and an additional one shall be provided near the secondary coolant system for use in determining the presence of abnormally high concentrations of radioactivity in these systems. The characteristics of these monitors shall be as stated in a. above.

c. Six additional direct reading area monitors employing Geiger tube detectors shall be provided to monitor the pneumatic system receiver stations, the beam port areas, and other areas as required. Each of these shall have the following characteristics:

1) A range consistent with expected radiation levels in the area being monitored (0 to 10 mr/hr or 0 to 50 mr/hr).

2) A radiation dose rate output at the instrument.

3) An adjustable high radiation alarm to alarm at the instrument and create both an audible and visual signal.

d. A stack exhaust gas monitor system shall be provided which draws a representative sample of air from the exhaust gas. The monitor with indicators and alarms in the control room, shall have the following characteristics:

1) A beta particulate monitor with an alarm.

2) A gas monitor incorporating a scintillation detector with high level alarm and a minimum detectability level for an Argon-41 concentration in air of $10^{-6}\mu c/cc$. The monitor shall have a range of at least four decades.

3. <u>Other Radiation Monitoring Equipment</u>

a. Portable survey instruments for measuring beta-gamma dose rates in the range from .01 mr/hr to 250 r/hr shall be available at the facility. Portable instruments for measuring fast and thermal neutron fluxes in the range from 1 n/cm^2-sec to 25,000 n/cm^2-sec shall also be available to the facility.

b. Reactor excursion monitors shall be placed in the facility for measuring gamma and neutron doses in the event of an accident.

c. A radiation monitor shall be provided to monitor all persons leaving the reactor room for beta-gamma contamination.

4. <u>High Radiation Area</u>

During reactor operation, the dose rate from the delay tank may be in excess of 100 millirem per hour. On three sides, the tank shall be shielded. On the fourth side, the tank is shielded using a "maze" so that access to the tank is possible through a door equipped with a lock.

H. FUEL STORAGE

1. New Fuel Storage

New fuel shall be stored in a security container in "egg crate" boxes. Sheet cadmium at least 0.020 inches thick shall be fastened around the outside of the boxes in the region which contains the fuel. The number of fuel elements which can be placed in each box shall not exceed two. An adjacent box shall be empty and no two elements can be in adjacent spaces in adjacent boxes. For all conditions of moderation possible at the site K_{eff} shall be less than 0.8.

2. Irradiated Fuel Storage

Two types of irradiated fuel element storage racks shall be provided. One type of rack shall contain spaces for nine fuel assemblies and shall have approximate over-all dimensions of 35.5 in. wide by 26 in. high by 6.25 in. thick, and shall be fixed to the pool wall. At least two of these racks shall be provided. The second type of rack shall consist of two of the nine fuel assembly racks described above attached together with a minimum space between the center lines of fuel assemblies in adjacent racks of 12 inches. This 18 fuel assembly rack shall be covered on the two 35.5 x 26 in. outside faces with a neutron absorbing material. At least one 18 fuel assembly rack shall be provided, and the rack may be moved within the pool. The fuel storage racks may also be used to store core components other than fuel assemblies. The irradiated fuel storage racks shall have a maximum K_{eff} of 0.8 for all conditions of moderation possible at the site. Storage spaces shall be provided for at least 36 fuel assemblies.

I. EXPERIMENTAL FACILITIES

The permanent experimental facilities shall consist of the following:

1. Thermal column.
2. Beam ports; two 8 inch dia, and four 6 dia.
3. A six inch diameter through port.
4. Radiation baskets.
5. A two-tube pneumatic tube system.
6. Dry gamma cave.

J. ADMINISTRATIVE AND PROCEDURAL SAFEGUARDS

1. Organization

The Rhode Island Atomic Energy Commission (RIAEC) shall have the responsibility for the safe operation of the reactor. The RIAEC shall appoint a Director of Operations and a Reactor Utilization Committee consisting of a minimum of five members, as follows:

(1) The Director of Operations

(2) The Reactor Facility Health Physicist

(3) A qualified representative from the faculty of Brown University

(4) A qualified representative from the faculty of Providence College

(5) A qualified representative from the faculty of the University of Rhode Island.

A qualified alternate may serve in lieu of one of the above.

The Director and Health Physicist are not eligible for chairmanship of the Committee. The Reactor Utilization Committee shall have the following functions:

a. Review proposals for the use of the reactor considering the suitability of the reactor for the proposed use and the safety factors involved.

b. Approve or disapprove proposed use of the reactor.

c. Review at least annually the operating and emergency procedures and the overall radiation safety aspects of the facility.

The Reactor Utilization Committee shall maintain a written record of its findings regarding the above.

2. Qualifications of Personnel

a. The Director of Operations shall have at least a bachelors degree in one of the physical sciences or engineering, and he shall be trained in reactor technology and be a licensed senior operator.

b. The staff Health Physicist shall be professionally trained and shall have at least a bachelors degree in one of the physical or biological sciences or engineering. He shall have experience such as may have been gained through employment in a responsible technical position in the field of health physics.

c. The reactor operators and senior operators shall be licensed in accordance with the provisions of 10 CFR 55.

d. In the event of temporary vacancy in the position of Director of Operations or the Health Physicist, the functions of that position shall be assumed by qualified alternates appointed by the RIAEC.

3. Responsibilities of Personnel

a. Director

(1) The Director shall have responsibility for all activities in the reactor facility which may affect reactor operations or involve radiation hazards, including controlling the admission of personnel to the building. This responsibility shall encompass administrative control of all experiments being performed in the facility including those of outside agencies.

(2) It shall be the responsibility of the Director to insure that all proposed experiments, design modifications, or changes in operating and emergency procedures are performed in accordance with the license. Where uncertainty exists, the Director shall refer the decision to the Reactor Utilization Committee.

b. Senior Reactor Operators

(1) A licensed senior reactor operator shall be assigned each shift and be responsible for all activities during his shift which may affect reactor operation or involve radiation hazards. The reactor operators on duty shall be responsible directly to the senior operator.

(2) The reactor operations which affect core reactivity shall not be performed without the senior operator on duty or readily available on call. The senior operator shall be present at the facility during initial startup and approach to power, recovery from an unplanned or unscheduled shutdown or significant reduction in power, and refueling. The name of the person serving as senior operator as well as the time he assumes the duty shall be entered in the reactor log. When the senior operator is relieved, he shall turn the operation duties over to another licensed senior operator. In such instances, the change of duty shall be logged and shall be definite, clear, and explicit. The senior operator being relieved of his duty shall insure that all pertinent information is logged. The senior operator assuming duty shall check the log for information or instructions.

c. Reactor Operators

(1) The responsible senior operator shall designate for his shift a licensed operator (hereafter called "operator") who shall have primary responsibility under the senior operator for the operation of the reactor and all associated control and safety devices, the proper functioning of which is essential to the safety of the reactor or personnel in the facility. The operator shall be responsible directly to the senior operator.

(2) Only one operator shall have the above duty at any given time. Each operator shall enter in the reactor log the date and time he assumed duty.

(3) When operations are performed which may affect core reactivity a licensed operator shall be stationed in the control room. When it is necessary for him to leave the control room during such an operation, he shall turn the reactor and the reactor controls over to a designated relief, who shall also be a licensed operator. In such instances, the change of duty shall be definite, clear, and explicit. The relief shall acknowledge his entry on duty by proper notation in the reactor log.

(4) The operator, under the senior operator on duty, shall be responsible for the operation of the reactor according to the approved operating schedule.

(5) The operator shall be authorized at any time to reduce the power of the reactor or to scram the reactor without reference to higher authority, when in his judgement such action appears advisable or necessary for the safety of the reactor, related equipment, or personnel. Any person working on the reactor bridge shall be similarly authorized to scram the reactor by pressing a scram button located on the bridge.

d. Health Physicist

The Health Physicist shall be responsible for assuring that adequate radiation monitoring and control are in effect to prevent undue exposure of individuals to radiation.

4. Written Instructions and Procedures

Detailed written operating instructions and procedures shall be prepared for all normal operations and maintenance and for emergencies. These procedures shall be reviewed and approved by qualified personnel before use. Each member of the staff shall be familiar with those procedures and instructions for which he has responsibility.

5. Site Emergency Plans

The Rhode Island Nuclear Science Center shall have available the services of other state agencies for dealing with certain types of emergencies. The RIAEC shall enter into an agreement with the Rhode Island Civil Defense Agency whereby the Civil Defense Agency will maintain an emergency monitoring and communications vehicle which they shall make available to the Nuclear Science Center in the event of an emergency involving release of fission products or other radioactive isotopes to the atmosphere. The emergency vehicle shall contain equipment such as portable radiation monitors, respirators, and a particulate air sampler. Communications using the statewide emergency network shall be available.

Personnel of the Civil Defense Agency and of local fire departments shall have received training from the Civil Defense Training Officer in the use of certain radiological instruments. Future training shall be augmented by including orientation on the reactor facility.

K. OPERATING LIMITATIONS

1. General

The following administrative controls shall be employed to assure the safe operation of the facility:

a. The reactor shall not be operated whenever there are any significant defects in fuel elements, control rods, or control circuitry.

b. The reactor control and safety system must be turned on and functioning properly and an appropriate neutron source must be in the core during any change which can affect core reactivity.

c. During operations which could affect core reactivity, a licensed operator shall be stationed in the control room. Communications between the control room and the senior reactor operator directing the operation shall be maintained.

d. The operator shall not attempt to start up the reactor following an automatic scram or unexplained power decrease until the senior operator has determined the cause of the scram or power decrease and has authorized a start-up.

e. The reactivity of all core loadings to be utilized in operating the reactor shall be determined using unirradiated fuel elements or elements containing fission products in which the effect of xenon poisoning on total core reactivity has decayed to 0.05% delta k/k or less.

f Critical experiments shall be performed under the supervision of the Director or other competent supervisory scientist licensed as a senior reactor operator. During the experiment there shall be present, in addition to this licensed supervisor, at least one other technically qualified person who shall act as an independent observer. Each step in the procedure shall be considered in advance by both persons, each calculation shall be checked by both persons, and no step shall be taken without the concurrence of both. A written record shall be made at the time of each fuel element addition or other core change which could significantly affect core reactivity.

g. The basic operating principles for the assembly and reloading of cores whose nuclear properties have been previously determined from critical experiments shall be as follows:

All core loading changes shall be performed under the supervision of a person having a senior operator's license. During the operation there shall be present in addition to the designated senior reactor operator at least one other technically qualified person who shall act as an observer.

The exact procedure to be followed for a particular reloading operation will be determined by the observer and the senior reactor operator in charge of the operation before the operation begins. Each step in the procedures shall be considered by both persons, and no step shall be taken without the concurrence of both.

2. Experiments

a. "Experiments" as used in this section shall be construed as any apparatus or device installed in the core region which is not a component of the core.

b. The Reactor Utilization Committee shall review and approve all experiments before initial performance at the facility. New types of experiments or experiments of a type significantly different from those previously performed shall be described and documented for the study of the Reactor Utilization Committee. The documentation shall include at least:

(1) The purpose of the experiment,

(2) A description of the experiment, and

(3) An analysis of the possible hazards associated with the performance of the experiment.

c. All use of experimental facilities shall be approved by the Director of Operations.

d. The absolute value of the reactivity worth of any single independent experiment shall not exceed 0.006. If such experiments are connected or otherwise related so that their combined reactivity could be added to the core simultaneously, their combined reactivity shall not exceed 0.006.

e. The calculated reactivity worth of any single independent experiment not rigidly fixed in place shall not exceed 0.0008. If such experiments are connected or otherwise related so that their combined reactivity could be added to the core simultaneously, their combined reactivity worth shall not exceed 0.0008.

f. No experiment shall be installed in the reactor in such a manner that it could shadow the nuclear instrumentation system monitors and thereby give erroneous or unreliable information to the control system safety circuits.

g. No experiment shall be installed in the reactor in such a manner that it could fail so as to interfere with the insertion of a reactor control element.

h. No experiment shall be performed involving materials used in such a way that they might credibly result in an explosion.

i. No experiment shall be performed involving materials which could credibly contaminate the reactor pool causing corrosive action on the reactor components.

j. Experiments shall not be performed involving equipment whose failure could credibly result in fuel element damage.

k. There shall be no more than one vacant fuel element position within the periphery of the active section of the core.

3. Operations

a. Site

Control of access to the reactor facility shall be the responsibility of the Director of Operations.

b. Containment

(1) During any operation in which the control rods are withdrawn from the core containing fuel, the following conditions shall be satisfied:

a. Confinement building penetrations which are not designed and set to close automatically on actuation of the evacuation button shall be sealed, except that doors other than the truck door may be opened during reactor operation. If a door is to remain open, an individual from the reactor operations staff is continuously in attendance at the door.

b. The building clean-up system is operable.

(2) Requirements for Retest of Confinement

(a) Method of Retest

The building cleanup system shall be retested by pressing an evacuation button and observing that the following functions occur automatically:

1. Evacuation horn blows.

2. air conditioning and normal ventilation has turned off.

3. Dampers on all ventilating ducts leading to the outside have closed.

4. Building cleanup system-air scrubber and basement chem lab blower come on.

5. The negative differential pressure between the inside and outside of the building is at least 0.5 inches of water. This shall be determined by reading the differential magna helic gauge located in the control room.

(b) Frequency of Retest

The building cleanup system including the auxiliary electrical power system shall be retested at least weekly.

(3) The exhaust rate through the cleanup system shall not exceed 4500 cfm with not more than 1500 cfm coming from the reactor building and passing through the charcoal scrubber. The remaining air will be provided by a separate blower from an uncontaminated source. This shall create a pressure in the building which is equivalent to at least 0.5 inch of water below atmospheric pressure.

c. Primary Coolant System

(1) The minimum depth of water above the top of the active core shall be 23 feet.

(2) No piping shall be placed in the pool which could cause or fail so as to cause a siphon of the pool water to below the level of the ten inch coolant line penetrations.

(3) Makeup System

The effluent water of the primary coolant water makeup system shall be of a quality to insure compliance with K.3.c.(5) and (6) below.

(4) Cleanup System

The effluent water of the primary coolant water clean up system shall be of a quality to insure compliance with K.3.c.(5) and (6) below.

(5) The primary coolant shall be sampled at a minimum frequency of once per week and the samples analyzed for gross radioactivity, pH, and conductivity in accordance with written procedures. Corrective action shall be taken to avoid exceeding the limits listed below:

pH	5.5 to 7.5
conductivity	2 μmho/cm

(6) The radioactive materials contained in the pool water and in the primary coolant water shall be such that the radiation level one meter above the surface of the pool shall be less than 10 mrem/hr.

(7) During the forced circulation mode of operation, the primary coolant flow rate shall not be less than 1580 gpm.

d. Secondary Cooling System

(1) The secondary coolant shall be sampled at a minimum frequency of once per week and the samples analyzed for pH in accordance with written procedures. Corrective action shall be taken to avoid exceeding the pH limit given below:

pH	5.5 to 9

(2) The concentration of radionuclides in the secondary water shall be determined at least once each day the reactor operates using forced convection cooling. The concentration shall be determined at least once per week when not being operated using forced convection cooling.

(3) If the radioactive materials contained in the secondary coolant exceed a radionuclide concentration in excess of the values in 10 CFR 20, Appendix B, Table I, Column II, above background, the reactor shall be shutdown and the condition corrected before operation using the secondary cooling system resumes.

(4) The secondary coolant system shall be placed in operation as required during power operation utilizing forced convection in order to maintain a primary coolant core outlet temperature of 1250°F or below.

e. Reactor Core and Control Elements

(1) The reactor shall not contain in excess of 35 fuel elements. There shall be a minimum of four operable control elements.

(2) The limiting thermal and hydraulic core characteristics based on a 14 element, graphite and beryllium reflected core are specified below:

(a)	Maximum Heat Flux	.424 MW/M^2
(c)	Maximum Fuel Surface Temperature	110°C
(d)	Coolant Velocity during Forced Convection Cooling	1.48 M/sec
(e)	Coolant Inlet Temperature	115°F max.
(f)	Average Coolant Temperature Rise	10°F max.
(g)	Primary System Bulk Outlet Coolant Temperature	125°F max.
(h)	Temperature Margin in Primary Coolant ($T_{sat}-T_{surf}$)	5.8°C
(i)	Number of Coolant Passes Through Core	1

(3) <u>Principal Nuclear Characteristics of the Core</u>

(a) <u>Core and Control System Reactivity Worth</u>

1. The reactor shall be subcritical by at least 1% $\Delta k/k$ from the cold, Xe-free, critical condition with the most reactive control element and the servo regulating element fully withdrawn.

2. The maximum worth of the servo regulating element shall be 0.7% $\Delta k/k$.

(b) <u>Maximum Reactivity Addition Rate - $\Delta k/k$/sec</u>

1.	By servo regulating element	maximum of 0.0002
2.	Manual by control element	maximum of 0.0002

(c) Reactivity Coefficients

1.	Temperature coefficient (coolant + Doppler)	approximately $-1.8 \times 10^{-4}/^{o}C$ (calculated)
2.	Void coefficient (core average)	approximately $-2.7 \times 10^{-3}/\%$ void (calculated)

(4) Principal Core Operating Limitations

(a) Excess Reactivity

The cold, clean excess reactivity for any core used in the reactor shall not exceed 0.047.

(b) Minimum Shutdown Margin

All reactor cores used shall be such that they would be subcritical if any single control element and the servo regulating element were withdrawn.

(c) Reactivity Coefficient Limitation

The reactor power coefficient (as inferred by the control rod movements required to compensate for changes in power) shall be negative.

(d) Control Element Drive Performance Requirements

All control element drives shall meet the following specifications:

1. The control drive withdrawal rate shall not be more than 3.6 inches per minute.

2. For the electronic scram system, the time from initiation of a scram condition until control element release shall not exceed 100 milliseconds.

3. The time from initiation of a scram condition until the control element is fully inserted shall not exceed 900 milliseconds.

4. It shall be demonstrated at least every 3 months that the above specifications are met.

(e) Servo Regulating Element Drive Performance Requirements

If in use during operation, the servo regulating element drive shall meet the following specifications:

1. The drive withdrawal rate shall not be more than 78 inches per minute.

2. It shall be demonstrated at least once per month that the above specification is met.

(f) Beryllium Reflector Lifetime

1. Maximum allowed accumulated neutron exposure is 1×10^{22} NVT.

2. To prevent physical damage to the beryllium reflectors and flux trap, an inspection of the components and a calculation of total exposure will be conducted annually.

f. Reactor Safety Systems

(1) The reactor safety system shall be operable during all reactor operation. The safety system shall be checked out before each start-up and functionally tested for calibration at least monthly.

(2) It shall be permissible to continue operations with one or more of the safety system functions that produce only an alarm temporarily disabled providing that additional procedural controls are instituted to replace the lost safety system alarm function(s).

(3) The control element withdrawal interlocks and the servo system control interlocks shall be functionally tested at least once per month.

(4) During reactor startup or during mechanical changes that could affect core reactivity, the startup range neutron monitoring channel shall be operable and shall provide a neutron count rate of at least 3 counts per second with a signal to noise ratio at least 3 to 1.

(5) The linear level safety channels shall not read less than 15% of full scale when the reactor is operating at power levels above 1 watt.

(6) Following a reduction in power level, the operator shall adjust the servo power schedule to the new power level before switching to automatic operation.

(7) An alarm condition from any one of the items listed in Section F.2.b. after working hours shall transmit coded information to a continuously manned central station in Providence, Rhode Island. The central station shall be provided with written instructions on the steps to be taken following an alarm.

g. Waste Disposal and Reactor Monitoring Systems

(1) The liquid waste retention tank discharge shall flow to a monitor station in the reactor building where the effluent shall be batch sampled and the gross activity per unit volume determined before release. All off-site releases shall be directly into the municipal sewer system.

(2) Gaseous radioactive waste shall be disposed of using the reactor stack. Disposal limits shall conform to the following table. In this table, the MPC stated is for individual isotopes and mixtures contained in Column 1, Table II, Appendix B of 10 CFR 20.

	1	2
Type of Activity	Maximum Curies per second to be released	Curies per second to be released averaged over one year
Particulate Matter and Halogens with half-lives longer than 8 days	140 X MPC (uc/cc)	14 X MPC (uc/cc)
All other Radioactive Isotopes	10^5 X MPC (uc/cc)	10^4 X MPC (uc/cc)

(3) All radioactive liquid and solid wastes disposed of off-site shall be within the limits established by 10 CFR 20 or shall be removed from the site by a commercial licensed organization.

(4) The exhaust gas monitor shall be calibrated to alarm at an instantaneous release rate which instantaneously exceeds the limits stated in Column 2 for the annual average release rate. If the maximum permissible stack release rate stated in Column 1 is exceeded, the reactor shall immediately be placed in the shutdown mode of operation and the situation investigated.

(5) The area, primary and secondary coolant system and the exhaust gas monitors shall be in operation at all times when control elements or the servo regulating elements are withdrawn; however, individual area coolant system monitors may be taken out of service for maintenance and repair if replaced with portable radiation detection equipment. Adequate spare parts shall be on hand to allow necessary repairs to be made during the maintenance or calibration outages of the monitors.

(6) The area and the primary and secondary coolant system monitors shall be adjusted to alarm at a maximum reading of 2 mr/hr or 200% of the normal radiation levels in their area, whichever is larger.

(7) The door which controls entrance to the "maze" leading to the delay tank shall be locked with the key in the possession of the Director or a licensed senior operator. Entrance to the delay tank high radiation area shall require the presence of the Health Physicist or a licensed senior operator and the use of direct reading portable radiation monitoring equipment.

h. Fuel Storage

(1) New fuel shall be stored in egg crate boxes located in a security container. Access to the security container shall be restricted, through use of a lock, to the Director of Operations and the licensed senior reactor operators.

(2) Irradiated fuel, not in use in the reactor core, shall be stored in the criticality safe storage racks described in Section H. Only one fuel assembly may be inserted or moved from a storage rack at a time.

(3) Safety against inadvertent criticality shall be provided by limiting the number of fuel assemblies per rack to nine and then positively securing such racks at least 30 cm. apart, or by limiting the number of fuel assemblies to 18 per rack and then covering the two large faces of each rack with a sheet of aluminum covered cadmium.

4. Maintenance

(a) The electronic control and the process control system shall be checked for proper operation and calibration before each reactor start-up. If maintenance or recalibration is required, it shall be performed before reactor start-up proceeds.

(b) Maintenance shall be performed with the approval of the Director. Equipment and system maintenance records shall be kept to facilitate scheduling and completion of all necessary maintenance.

(c) Routine maintenance on all control and process system components shall be performed in accordance with written schedules and with written procedures.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING THE ORDER TO CONVERT FROM

HIGH-ENRICHED TO LOW-ENRICHED URANIUM FUEL

FACILITY OPERATING LICENSE NO. R-95

RHODE ISLAND ATOMIC ENERGY COMMISSION

DOCKET NO. 50-193

1.0 INTRODUCTION

Section 50.64 of Title 10 of the Code of Federal Regulations (10 CFR 50.64) requires, except under certain conditions, that licensees of non-power reactors convert to low-enriched uranium (LEU) fuel. The Rhode Island Atomic Energy Commission (the licensee) has proposed to convert the fuel in the Rhode Island Nuclear Science Center Research Reactor (the reactor) from high-enriched uranium (HEU) to LEU. In the letter of November 18, 1991, as supplemented on July 23, December 22, 1992, and January 13, 1993, the licensee submitted a safety analysis report (SAR) and revised Technical Specifications (TS) including the changes needed to convert to LEU fuel. Upon evaluating the SAR and proposed changes to the TS, the U. S. Nuclear Regulatory Commission (NRC) staff issues this Safety Evaluation (SE) for an order to convert the reactor from HEU to LEU fuel.

2.0 EVALUATION

2.1 General Facility Description

The reactor is licensed to operate at thermal power levels not to exceed 2 megawatts thermal (Mw(t)), using thin plate-type fuel. The reactor is cooled by forced convection at a nominal flow rate of about 1730 gallons per minute for power levels greater than 0.1 Mw(t), but may be cooled by natural convection for power levels of 0.1 Mw(t) or less. The primary coolant is cooled by a heat exchanger to the secondary coolant system. The secondary coolant system then rejects the heat to the atmosphere at cooling towers.

2.2 Fuel Construction and Geometry

The HEU fuel elements used at the reactor are of a typical materials test reactor (MTR)-type design, each consisting of 18 fueled plates. Each HEU fuel plate is a sandwich consisting of a nominally 0.508-millimeter (mm) thick layer of a dispersion of aluminum-uranium oxide (aluminum alloy, Al_x, U_3O_8) completely clad in a cover of aluminum approximately 0.508 mm thick. The uranium in the HEU fuel meat is enriched to about 93 percent uranium-235, and each plate contains approximately 6.9 grams (gm) of this isotope.

The LEU fuel elements will be of a similar design with essentially the same outer dimensions as the HEU fuel element, but will contain 22 aluminum (Al)-clad fuel plates. Each of these plates will consist of uranium silicide dispersed in aluminum (U_3Si_2-Al) and completely clad in Al alloy. In the LEU plates, the fuel meat will be about 0.508 mm thick, and the clad will be about 0.381 mm thick. The cladding alloy on the LEU fuel will be 6061 Al instead of 1100 Al for the HEU fuel (6061 Al is also specified in the current Technical Specifications as an allowable clad material). The uranium in the LEU fuel meat is enriched to less than 20 percent uranium-235, and each plate contains about 12.5 gm.

The overall width of the LEU fuel plate will be about 7.14 centimeters (cm) compared to 7.11 cm for the HEU fuel, and the width of the active fuel will be approximately 6.1 cm maximum compared to 5.59 cm for HEU. The length of active LEU fuel plate will be 59.69 cm versus 60.96 cm for HEU.

The Argonne National Laboratory developed these fuel element plates for conversion to LEU fuel at the U.S. non-power reactors. These fuel element plates were tested extensively under hostile environmental conditions for non-power reactors in the Oak Ridge Research Reactor and demonstrated acceptable fuel performance. The NRC staff reviewed and approved use of this type of fuel in NUREG-1313, "SER Related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-power Reactors," July 1988. The characteristics of the fuel proposed for this LEU conversion are consistent with those previously accepted and, therefore, the fuel construction and geometry is acceptable.

2.3 Core Configuration

A typical HEU core consists of approximately 30 fuel elements in a five-by-six array (core configurations can range from 28 to 35 fuel elements). The core is moderated and cooled by light water. The HEU core uses graphite reflectors. Control rods consist of four safety blades and one regulating rod. The safety blades are boron carbide (B_4C) and aluminum with aluminum cladding. The safety blades fall vertically between two rows of fuel elements on reactor trip. The regulating rod is a square tube of boral and is next to the core in position "D1" for the HEU core.

The proposed LEU core consists of 14 fuel elements in a three-by-five array with the central element consisting of a non-fuel beryllium component with a 38-mm hole incorporated as a neutron flux trap. The moderating and cooling functions for the LEU core are the same as those for the HEU core. The proposed LEU core reflectors include not only the graphite reflectors as for the HEU core, but also beryllium reflectors to enhance neutron utilization. These new beryllium reflectors are approximately the same size as the graphite reflectors: about 7.47 cm by 7.47 cm by 73.66 cm.

For the first few LEU core modifications, the licensee will accomplish reactivity changes through rearrangement of the reflectors to establish an equilibrium core configuration. The safety blades remain the same as those for the HEU core; however the regulating rod will be moved adjacent to the

core fuel elements to position "D2." To accommodate this change in position, the regulating rod also needs to be redesigned to maintain an acceptable reactivity worth. This redesigned regulating rod will maintain approximately the same physical dimensions, but will be made of stainless steel for less reactivity worth. These design differences have been analyzed by the licensee and evaluated by the staff and are acceptable.

2.4 Fuel Storage

The licensee analyzed the method of storing the LEU fuel to ensure that acceptable reactivity condition (K_{eff} <0.8) is maintained. The licensee proposed Technical Specification changes to limit the number of fuel elements per new fuel storage box and to limit fuel elements in adjacent new fuel storage boxes. The method of storing the fuel is acceptable because the mechanical design of the two fuels are similar and because the proposed Technical Specifications require the licensee to store the fuel only in specified configurations and areas (security container for new fuel and criticality safe storage racks for irradiated fuel) to limit reactivity conditions.

2.5 Critical Operating Masses of Uranium-235

Each HEU fuel element contains about 124 gm of uranium-235 and each LEU fuel element contains about 275 gm of uranium-235. This corresponds to a uranium-235 operating mass of approximately 3,850 gm for the LEU core and up to about 4,340 gm for the HEU core. This resulting LEU fuel loading is reasonable for the intended purpose of the reactor and is consistent with other LEU conversions considering configuration and power level. Therefore, the fuel loading of the proposed LEU cores is acceptable.

2.6 Basic Nuclear Parameters

Nuclear input parameters for reactivity calculations, such as the prompt neutron lifetime and the effective delayed neutron fraction changed as expected from the HEU to LEU fuel. The equilibrium core prompt neutron lifetime decreased slightly from 76 to 68 microseconds, primarily because of the increased leakage from the smaller core. The effective delayed neutron fraction for the LEU fuel is virtually unchanged from that of the HEU fuel. These effects are as expected for conversion to LEU.

2.7 Excess Reactivity

The licensee calculated the amount of excess reactivity for the proposed LEU cores that is needed to control the reactor and compensate for various operational losses, e.g., burnup, xenon, samarium, and temperature variations. The calculations indicated that the LEU core design requirement of a maximum of 4.1 percent delta k/k is slightly less than that for the HEU core and will not exceed the excess reactivity permitted by the current (HEU) Technical Specifications. The licensee will verify the excess reactivity during the LEU reactor core startup testing. Therefore, the excess reactivity for the proposed LEU conversion is acceptable.

2.8 Control Rod Worth

The proposed LEU core uses the same safety blades as have been used for the HEU cores. The licensee evaluated the worth of these safety blades with Argonne National Laboratory standard neutron kinetics models and computer codes and verified that they will acceptably meet the Technical Specification requirements with the proposed LEU cores. The licensee will verify the worth of these blades during the LEU reactor core startup testing.

The position and design of the regulating rod (a non-scrammable rod) is to be changed in the proposed LEU core. Calculations showed that the reactivity worth of the regulating rod in the current position, D1, would decrease from about 0.48 percent delta k/k to 0.2 to 0.3 percent delta k/k. To ensure an adequate reactivity for the regulating rod, the licensee proposed moving the rod to the D2 position, which is the grid position directly adjacent to the proposed LEU core. However, this movement would result in an increased worth for the regulating rod beyond that allowed by Technical Specifications to limit reactivity insertion from rod withdrawal. Therefore, the licensee proposed to redesign the regulating rod of stainless steel material to maintain a worth of about 0.4 to 0.5 percent delta k/k. This design acceptably ensures control for normal plant operations and limits the potential reactivity insertion transients. Further, the licensee will determine the worth of this rod during the startup testing of the LEU reactor core and verify that the control rods perform as designed.

Therefore, control rod worth design is acceptable.

2.9 Shutdown Margin

The NRC staff requires reasonable assurance that a reactor can be shut down from any operating condition, even if the safety blade of maximum worth and any non-scrammable rod are in the most reactive position. Using the calculated safety blade worth, proposed LEU core configurations, and Argonne National Laboratory standard neutron kinetics methods, the licensee calculated that shutdown margin would not be lower than 5.69 percent delta k/k. This shutdown margin is considerably greater than the 1 percent delta k/k required in Technical Specifications, and will be verified during startup testing of the LEU reactor core. Therefore, the proposed shutdown margins for the LEU cores are acceptable.

2.10 Beryllium Core Element and Reflectors

The licensee plans to use a beryllium core element and beryllium reflectors in the LEU cores to enhance flux shape. The licensee included the use of these components in the analysis of the proposed LEU core nuclear characteristics, as discussed elsewhere herein. The licensee also presented data and analyzed these components for material degradation caused by neutron embrittlement, and determined that the beryllium components would not need to be replaced for 45.8 years. To limit the effect of neutron embrittlement, the licensee proposed a Technical Specification that conservatively limits the fluence on the beryllium components. The licensee also proposed annual inspection of the

beryllium components and determination of neutron fluence on those components in the Technical Specifications. The licensee stated that beryllium has also been successfully used at other non-power reactors. The licensee also referenced additional evidence that the beryllium would not be degraded by the conditions in the proposed LEU reactor core. The staff reviewed this information and concluded that the beryllium core elements and reflectors as described in submittal and supplements provided by the licensee are acceptable.

2.11 Core Power Characteristics

The licensee performed analyses of core power characteristics for the proposed LEU cores. The analyses of power density and power peaking were cooperative efforts by the licensee and the staff at the Argonne National Laboratory. These analyses used Argonne National Laboratory standard nuclear kinetics models, and standard computer programs.

The analyses calculated a maximum heat flux of about 0.424 megawatts per square meter (Mw/m^2) for the LEU fuel and core. This heat flux is consistent with the proposed core design. The licensee proposed a corresponding change to the Technical Specification K.3.e(2) for this parameter. The staff found this change acceptable.

The results of the core analyses also showed that the maximum total power peaking factor was about 2.64 for all control blades withdrawn and was 3.06 for control blades 50 percent inserted. The licensee calculated these maximum power peaking factors for the initial ("startup") core configuration and calculated a decrease to a power peaking factor of about 2.36 at equilibrium core configurations. The staff reviewed the analysis inputs, methods, and results, and concluded that the licensee acceptably determined the power conditions to be used in analyzing thermal-hydraulic conditions, and transient and accident conditions as discussed later herein.

2.12 Thermal-Hydraulics

The licensee performed a thermal-hydraulic analysis of the LEU elements and core. The licensee acceptably modeled power peaking factors, thermal conductivity, fluid flow conditions, and fuel and core configurations for the proposed LEU fuel elements and core. These analyses demonstrated that the LEU fuel elements and core would be cooled and maintained within acceptable limits for forced or natural convection conditions during normal operation. That is, these analyses demonstrated that calculated thermal-hydraulic conditions for the LEU fuel, under both forced convection and natural convection flow and associated power conditions, would maintain a substantial margin to fluid saturation (boiling conditions) and assumed fuel failure.

These analyses for anticipated conditions demonstrated that for a maximum fuel surface temperature of 110 degrees Celsius, there would be a margin of 5.8 degrees Celsius to the saturation temperature of the water at core depth. Therefore, no fuel damage would occur at allowed operating conditions. These analyses also demonstrated that the LEU fuel element and core configurations

would not result in flow instabilities and thus ensured that fuel element and core configurations would not be adversely effected by the range of expected flow rates for steady state operational conditions.

The licensee also analyzed the LEU fuel element and core thermal-hydraulic design for off-normal conditions. The analysis for natural convection conditions demonstrated that the 0.1 Mw limit on power operations gives significant margin to incipient boiling and assumed fuel damage; that is, the licensee calculated that incipient boiling under natural convection conditions would occur at greater than 0.2 Mw.

The analyses of off-normal conditions for forced convection flow demonstrated that the LEU fuel elements and core would be protected from high power operations and low flow rates with reactor safety system trip set points of 2.4 Mw (which is about 120 percent of the normal full power level) and 1580 gallons per minute (gpm) (which is approximately 91.3 percent of the normal full flow rate), respectively. The licensee proposed Technical Specification trip setpoints more conservative than these values, proposing trip setpoints at 115 percent high neutron flux (power) and 1600 gpm primary coolant low flow rate. The licensee also proposed to change the low flow rate of the primary coolant alarm set point to 1650 gpm for LEU core from 1340 gpm for the HEU core, to ensure margin between the alarm and the trip set point. The staff finds these reactor safety system trip and alarm set points changes to the Technical Specifications to be acceptable for the conversion to LEU fuel.

These analyses and proposed changes to Technical Specifications are acceptable for the thermal-hydraulic performance of the LEU core.

2.13 Reactivity Feedback Coefficients

The licensee computed the temperature coefficient of reactivity and the void coefficient of reactivity for the HEU and LEU cores. The licensee calculated the temperature coefficient to be -1.8×10^{-4} delta k/k/degree Celsius for the LEU core compared to -2×10^{-4} delta k/k/degree Celsius for the HEU core. The licensee calculated the void coefficient (core average) to be approximately -2.7×10^{-3} delta k/k/percent void for the LEU core compared to -1.5×10^{-3} delta k/k/percent void for the HEU core. All coefficients have been conservatively considered in nuclear, transient and accident analyses, and are specified in the Technical Specifications. The licensee will also verify the temperature coefficient to be negative and of similar magnitude to that of the HEU core during the startup tests. Therefore, the licensee acceptably addressed the reactivity feedback coefficients for conversion to the LEU fuel.

2.14 Fission Product Containment and Inventory

The cladding is the primary barrier to fission product release for both the HEU and LEU fuels. The cladding of the HEU and LEU fuel plates differ in thickness and material composition. The cladding thickness of the LEU fuel is approximately 0.381 mm compared to 0.508 for the HEU fuel cladding. The cladding material is composed of 6061 Al (an aluminum alloy) for the LEU fuel and 1100 Al (another aluminum alloy) or 6061 Al for the HEU fuel.

The Department of Energy and the Argonne National Laboratory designed and fabricated the LEU fuel. This LEU fuel has been tested extensively at the Oak Ridge Research Reactor. These tests demonstrated excellent performance for the proposed LEU fuel comparable to that for HEU fuel. Further, use of similar fuel elements and plates in other non-power reactors has continued to demonstrate the excellent fission product retention capability of the LEU fuel.

The total inventory of fission products from operating the proposed LEU core at 2 Mw(t) will not differ significantly from that for the HEU core. Therefore, the results of the previously assumed release of fission products remains valid in that it conservatively assumed a 10 percent release of the total inventory.

However, the fission product inventory in each LEU fuel element and plate will be greater than that for the HEU fuel, because a core of LEU fuel contains fewer fuel elements and plates for the same power level. The licensee estimated this increased fission product inventory per fuel element and the potential effect of a plate failure. These analyses and evaluations demonstrate that the consequences of the fission product release does not exceed previously established acceptance criteria.

The licensee and the NRC staff have found no new or significant safety considerations on fission product containment and inventory for the LEU fuel. Therefore, the proposed operations of the LEU fuel is acceptable to contain the expected fission product inventory.

2.15 Potential Accident Scenarios

In this section the staff evaluates the startup accident and the loss of coolant accident, which are the two most limiting accidents for this type of reactor.

2.15.1 Startup Accident

This potential accident analysis assumes the maximum reactivity insertion with the reactor at cold clean conditions, reactor power at the source level, and the regulating rod withdrawn. The maximum reactivity insertion is the sequential withdrawal of all safety blades at the maximum rate. The licensee assumes that the period scram fails, and that the reactor trips on the high neutron flux scram (120 percent of full power). It also conservatively assumes a delay before the safety blades are free to drop off 0.5 seconds. This analysis results in a maximum fuel temperature of 88.1 degrees Celsius for the LEU fuel (compared to 67.3 degrees Celsius for the HEU fuel).

The licensee also presented a more conservative analysis with the same assumptions except it assumed that the reactor did not trip on the high neutron flux scram. This scenario would result in the reactor power continuing to rise until the negative reactivity from the void and temperature coefficients of reactivity compensate for the positive reactivity from the withdrawing safety blades. This analysis resulted in a peak clad temperature

of 148.5 degrees Celsius (compared to 149.1 degrees Celsius for HEU fuel). The licensee indicated that these maximum fuel temperatures are well below the melting temperature (582 degrees Celsius) of the LEU fuel cladding material (6061 Al, an aluminum alloy). The licensee also determined that if the reactor did not trip automatically, the fuel would operate in nucleate boiling with no damage until a trip was manually initiated.

The staff finds that the startup accident, as analyzed, continues to be the most limiting credible reactivity insertion accident for the LEU core. Based on this analysis, the staff concluded that the reactor has substantial margin from temperatures at which some cladding deterioration has been observed, as discussed in NUREG-1313, "SER Related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-power Reactors," July 1988. Also, the conversion process would not introduce a new, and previously unanalyzed reactivity accident. Therefore, the staff concludes the licensee has acceptably demonstrated that the health and safety of the public will be protected from the consequences of any credible reactivity insertion event.

2.15.2 Loss of Coolant Accident

The licensee assumed that the loss of coolant accident (LOCA) for the core results from a maximum size break in an 8 inch diameter beam port containing no plugs. This is the only penetration to the reactor pool which could result in a LOCA that could drain the pool to a point where fuel would be uncovered. The licensee proposed additional restraints to reduce the rate of loss of coolant for the LEU cores. The licensee ensured that the actual maximum size opening for the failure of any experimental installation will be less than the area of a 0.5 inch (or 1.27 centimeters) diameter hole. The licensee proposed that the experiment installation be designed to withstand the backpressure equivalent of the hydraulic head of the pool, or a minimum of about 25 feet of water. The licensee calculated that without makeup flow for a beam port break equivalent to the 0.5 inch diameter hole, the core would remain completely covered for more than about 10.5 hours after the low water level scram set point was reached. The licensee demonstrated that the fission product decay heat for this event would not result in core damage.

As an additional conservatism for the LEU LOCA, the licensee stated that, (1) each beam port has a shutter that may be closed to substantially reduce the loss of coolant from the reactor pool, (2) water loss to the core could be considered to be further mitigated since the core sits in a grid box and draining of this box is through a 1.25 centimeter hole drilled in the bottom of the grid box, and (3) the bottom 21 centimeters of the core will still be covered by water at the end of the drainage, because of the relative elevations of the beam ports and the core. These conservatisms were not used in the LOCA analyses.

The licensee also noted that the reactor has a automatic pool make-up system with a flow rate of about 20 gallons per minute, which is approximately equivalent to the flow rate through a break in the beam port with an experiment opening equal to a 0.5 inch diameter hole. The licensee indicated that it has the capability and procedural guidance to manually lineup piping

to supply additional flow to the reactor pool. The licensee submitted information on the source of makeup water to the Narrangansett Bay Campus with backup generators and cross connections to ensure supply to the reactor. The licensee indicated that a minimum demand of 5 gpm or greater can be met for 24 hours after electrical power is lost. These makeup flow capabilities were not taken credit for in the LOCA analysis.

Additionally, each year the licensee inspects the reactor pool surfaces, including beam ports. This inspection could identify problems and correct them before any significant leakage occurs, and so reduces the probability of a LOCA.

The licensee conservatively analyzed the LOCA and demonstrated acceptable results. The staff also finds significant additional conservatism to prevent and mitigate a LOCA that were not specifically credited in the analyses, which provides additional assurance that the public health and safety will be protected.

3.0 Changes to Technical Specifications

3.1 Technical Specification D, "Secondary Coolant System" and Table F.2 "Reactor Nuclear Instrumentation"

The licensee proposed to delete "Change 4" which is no longer applicable, and is therefore, acceptable.

3.2 Technical Specification E, "REACTOR CORE AND CONTROL ELEMENTS"

This specification lists the characteristics and nominal dimensions of the reactor core and control elements. Changes to reflect the LEU fuel and core are discussed in the following related sections.

3.2.1 Technical Specification E.1, "Principal Core Materials"

The licensee proposed revising this specification to indicate the specific composition of the proposed LEU core. The changes include that, (1) the LEU fuel matrix will be U_3Si_2-Al instead of the HEU fuel matrix alloy Al_x, U_3O_8, (2) the uranium-235 enrichment will be approximately 20 percent for the LEU fuel instead of the approximately 93 percent enrichment of the HEU, (3) the fuel cladding will be 6061 aluminum instead of 1100 and/or 6061 aluminum, (4) the aluminum clad beryllium reflectors will be added, and (5) the servo element (the control rod) will be made of 314 stainless steel instead of boral clad with aluminum. The licensee also proposed adding "graphite" to the specification to distinguish between the two types of reflectors, because the LEU core will include beryllium reflectors. These changes are as required by the LEU fuel design which as previously discussed have been demonstrated to be acceptable. Therefore, the changes are acceptable.

3.2.2 Technical Specification E.2, "Fuel Elements"

The licensee proposed revising this specification to indicate the specific dimensions, number of plates and uranium-235 content for the proposed LEU fuel elements. The changes include that, (1) the LEU fuel plate width overall will be 2.81 inches instead of 2.8 inches for the HEU fuel, (2) the active plate width will be 2.4 inches maximum instead of 2.2 inches, (3) the active plate length will be 23.5 inches instead of 24 inches, (4) the clad thickness will be about 0.02 inch instead of 0.024 inch, (5) the fuel matrix thickness will be about 0.02 inch instead of 0.012 inch, (6) the water gap between plates will be 0.08 inch instead of 0.1 inch, (7) the number of plates in each fuel element will be 22 instead of 18, and (8) the amount of uranium-235 in each fuel element will be 275 grams instead of 124 grams. These changes are as required by the LEU fuel design, and, therefore, are acceptable.

3.2.3 Technical Specification E.3, "Reflector Elements - Graphite and Beryllium"

The licensee proposed revising this specification to indicate the specific configuration and composition of the proposed LEU core. The licensee proposed adding "Graphite and Beryllium" to the title and corresponding place in the Table of Contents to account for the fact that the LEU core design will add the use of the beryllium reflectors. The licensee proposed adding to this specification the standard beryllium element dimensions of 2.94 inches by 2.94 inches by 29 inches and the plug-type beryllium element dimensions of 2.94 inches by 2.94 inches by 29 inches with a 1.5 inch diameter through hole. Based on the previous discussion and acceptance of the use of beryllium reflectors for the LEU core, these changes are acceptable.

3.2.4 Technical Specification E.5, "Servo Regulating Element"

The licensee proposed changes to this specification to indicate the specific composition of the proposed regulating rod (servo regulating element). The licensee proposed changing the material from boral tube to stainless steel. This change is acceptable based on the previously discussed analysis of the proposed regulating rod.

3.3 Technical Specification "Table F.1 - REACTOR SAFETY SYSTEMS"

This specification lists the reactor safety system settings for the LEU core. The changes include, (1) revision of the high neutron flux trip set point to 115 percent of full scale (2.3 Mw) from the 130 percent (2.6 Mw) value for HEU, and (2) revision of the low flow rate of primary coolant set point and alarm to 1600 gallons per minute (gpm) and 1650 gpm, respectively, from HEU values of 1200 gpm and 1350 gpm, respectively. These changes give continued assurance that the LEU fuel would be acceptably protected from high power operations and low forced convection flow with reactor safety system trip set points and alarms. Therefore, these changes are acceptable.

3.4 Technical Specification G.2, "Area and Exhaust Gas Monitor Design Features"

The licensee proposed to change Technical Specification G.2.d.2 to replace "sensitivity" with "minimum detectability level" for the range of Argon-41 concentrations that a gas monitor can measure. This change clarifies the specification, does not change the intent of the specification, and, therefore, is acceptable.

3.5 Technical Specification H.1, "New Fuel Storage"

The licensee proposed a change to the limits on the new fuel storage requirements to limit reactivity in new fuel storage. This change would limit the number of elements to two in each new fuel storage box and ensure that the adjacent box would be empty. This change conservatively represents the assumptions in licensee reactivity calculations for new fuel storage, as previously discussed, and is, therefore, acceptable.

3.6 Technical Specification K, "OPERATING LIMITATIONS"

3.6.1 Technical Specification K.3.b(2)(a), "Methods for Retest of Confinement"

The licensee proposed a change to Technical Specifications K.3.b.(2)(a)4 and K.3.b.(2)(a)5. The change to K.3.b.(2)(a)4 would replace "fresh air blower" with "basement chem lab blower." The change to K.3.b.(2)(a)5 would replace "manometer" with "magna helic gauge." These changes reflect the actual function and designation of the equipment at the facility. Therefore, this change clarifies the specification, does not change the intent, and is acceptable.

3.6.2 Technical Specification K.3.c(7), "Primary Coolant System Minimum Flow Rate"

The proposed changes include increasing the minimum primary coolant flow rate to 1580 gpm for the LEU conversion from 1200 gpm for the HEU fuel. This change ensures that the flow does not reduce to the point that incipient boiling may occur. Therefore, the change is acceptable on that bases as previously discussed in the thermal-hydraulics section.

The licensee also proposed to delete an exemption from the minimum flow requirements during reactor power determinations by coolant heat balance. The change would no longer allow the licensee to reduce reactor coolant flow to 600 gallons per minute if all other Technical Specifications were met during heat balance activities. This change will remove an exemption that no longer applies and will ensure that the thermal-hydraulic conditions are maintained in accordance with the assumptions of the licensee safety analyses.

3.6.3 Technical Specification K.3.e(2), "Reactor Core and Control Elements"

The licensee proposed changes to this specification to indicate the specific configuration and characteristics of the proposed LEU core. The changes include that, (1) the specification will be based on a LEU core that will have 14 fuel elements instead of the 28 elements that were conservatively assumed for the HEU core in this specification, (2) the specification will include the use of beryllium reflectors in addition to the graphite reflectors, (3) the maximum heat flux for the LEU core will be 0.424 Mw/m^2 instead of about 0.149 Mw/m^2 for the HEU fuel, (4) the maximum core specific power will be deleted because the value (1,120 watts/gm uranium-235 for the HEU fuel) was a calculated value, can not be directly measured and has no direct bearing on the operation of the LEU core, (5) the maximum fuel surface temperature will be 110 degrees Celsius for the LEU fuel instead of about 92 degrees Celsius for the HEU fuel, (6) the minimum coolant velocity during forced convection cooling will be 1.48 meters per second for the LEU fuel instead of approximately 0.81 meters per second, and (7) the temperature margin in the primary coolant, which is the difference between the water saturation temperature and the surface temperature, will be at least 5.8 degrees Celsius for the LEU core instead of about 24 degrees Celsius for the HEU core. These changes are consistent with the configuration of the LEU core and the limiting thermal and hydraulic core characteristics from analysis of the LEU fuel and core, and are, therefore, acceptable.

3.6.4 Technical Specification K.3.e(3)(c), "Reactivity Coefficients"

The licensee calculated the reactivity coefficients for the LEU core and proposed associated Technical Specification changes for the LEU core. Specifically, (1) the temperature coefficient specification will be approximately -1.8×10^{-4} delta k/k/degree Celsius (calculated) for the LEU core instead of -2×10^{-4} delta k/k/degree Celsius for the HEU core and, (2) the void coefficient (core average) specification will be approximately -2.7×10^{-3} delta k/k/percent void (calculated) for the LEU core instead of -1.5×10^{-3} delta k/k/percent void for the HEU core. These changes represent the design of the LEU core and have been acceptably calculated and considered by the licensee in their safety analyses. Therefore, the changes are acceptable.

3.6.5 Technical Specification K.3.e(4)(f), "Beryllium Lifetime"

The licensee proposed changing this specification to delete a specification on fission density that applied to alloy, uranium aluminide, and uranium oxide HEU fuel and to add a specification on beryllium lifetime. This new specification is consistent with the beryllium irradiation performance as previously discussed and, therefore, is acceptable.

3.6.6 Technical Specification K.3.g(1), "Waste Disposal and Reactor Monitoring Systems"

The licensee proposed changing the offsite release point to the municipal sewer system from Narragansett Bay. This is consistent with the more conservative actual release point, and, therefore, is acceptable.

4.0 CONCLUSIONS

The NRC staff reviewed and evaluated all of the operational and safety factors affected by the use of LEU fuel in the place of HEU fuel in the reactor. The staff concludes that 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 health and safety of the public. Therefore, the conversion to LEU fuel is acceptable.

Principal Contributors: Marvin M. Mendonca, NRC
William R. Carpenter, EG&G
Robert E. Carter, EG&G

Date: March 17, 1993